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Proposed Inservice Inspection Alternative VEGP-ISI-ALT-04-04

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ELECTRIC POWER
RESEARCH INSTITUTE

Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections

2019 TECHNICAL REPORT

Technical Bases for Inspection Requirements for PWR Steam Generator Feedwater and Main Steam Nozzle-to-Shell Welds and Nozzle Inside Radius Sections

3002014590

Final Report, April 2019

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ABSTRACT

Certain welds in pressurized water reactor (PWR) steam generators (SGs) are classified as Class 2, Category C-B, pressure retaining welds in pressure vessels. This report focuses on the nozzle-to-shell welds and the inside radius sections of PWR SG feedwater and main steam nozzles, which are listed in ASME Code Section XI, Table IWC-2500-1 under the following item numbers:

- Item No. C2.21: Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds
- Item No. C2.32: Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds when inside of vessel is accessible
- Item No. C2.22: Nozzle inside radius section

The requirements for Item No. C2.21 call for both surface and volumetric examinations, while Item Nos. C2.22 and C2.32 require only volumetric examination. These items require examination of all nozzles at the terminal ends of piping runs during each inspection interval, which means that all nozzle-to-shell welds and nozzle inside radius sections are examined every interval.

The objectives of this report are to evaluate the current examination requirements for PWR SG feedwater and main steam nozzle-to-shell welds and nozzle inside radius sections and establish the technical bases for various alternative inspection scenarios. To accomplish these objectives, various topics are addressed in this report, including a review of previous related projects, a review of inspection history and results, a survey of components in the industry, selection of representative components and operating transients for stress analysis, evaluation of potential degradation mechanisms, and a flaw tolerance evaluation consisting of probabilistic and deterministic fracture mechanics analyses.

Based on the evaluations performed in this report, the technical bases are provided for various ASME Code, Section XI inspection schedules for the PWR SG main steam and feedwater nozzle-to-shell welds and nozzle inside radius sections listed under Item Nos. C2.21, C.2.22, and C2.32 in Table IWC-2500-1. The evaluations show that, after the pre-service inspection (PSI), the probability of rupture is below the U.S. Nuclear Regulatory Commission (NRC) safety goal of 10^{-6} failures per year after 80 years of plant operation, while the probability of leakage is about an order of magnitude below the safety goal for 80 years. Therefore, from a safety viewpoint, no other inspections are required through 80 years of plant operation. The evaluation also considered limited coverage during subsequent in-service inspections to address components for which full examination coverage cannot be obtained because of physical obstructions that are present for some components in some plants. For an inspection scenario consisting of PSI

followed by 20-year in-service inspection (ISI) inspections, the failure probabilities (in terms of both rupture and leakage) are significantly below the acceptance criteria of 10^{-6} failures per year after 80 years of operation. The results are even more favorable if previous 10-year inspections are combined with the 20-year inspection interval.

Keywords

ASME Boiler & Pressure Vessel Code (BPVC)
Nozzle inside radius section
Nozzle-to-shell weld
Section XI
Steam generator feedwater nozzle
Steam generator main steam nozzle
Volumetric examination

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EXECUTIVE SUMMARY

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PRIMARY AUDIENCE: In-service inspection (ISI) program engineers for nuclear utilities

SECONDARY AUDIENCE: Technical staff for nuclear utilities and regulators

KEY RESEARCH QUESTION

Based on operating history of pressurized water reactor (PWR) steam generator (SG) feedwater and main steam nozzles combined with flaw tolerance evaluations, can technical bases be established for various ASME Code, Section XI inspection scenarios for these components for use in optimizing component examinations without compromising plant safety or reliability?

RESEARCH OVERVIEW

Factors considered in establishing the technical bases for a variety of inspection scenarios for PWR SG main steam and feedwater nozzles consisted of operational history, inspection results to-date, and flaw tolerance evaluations involving probabilistic and deterministic fracture mechanics analyses considering potential applicable degradation mechanisms (corrosion fatigue, mechanical fatigue, and thermal fatigue).

KEY FINDINGS

- A survey was conducted through the entire industry to collect the number of examinations performed and associated examination results for Item Nos. C2.21, C2.32, and C2.22, including the PWR SG feedwater and main steam nozzle components evaluated in this report. The survey results showed that of a combined total of 727 examinations performed on Item Nos. C2.21, C2.32, and C.22 components, only 1 (PWR) examination (for Item No. C2.21) identified a flaw that exceeded the acceptance criteria of ASME Code Section XI (the indication was dispositioned by light grinding).
- A degradation mechanism evaluation was performed for all PWR SG Item No. C2.22 nozzle inside radius section components and all PWR SG Item Nos. C2.21 and C2.32 nozzle-to-shell welds. The only potential degradation mechanism for these components is mechanical/corrosion fatigue (along with thermal transients for the feedwater nozzle). These results were considered in the fracture mechanics evaluation for these components.
- Probabilistic fracture mechanics evaluations were performed and showed that the U.S. Nuclear Regulatory Commission (NRC) safety goal of 10^{-6} failures per year would be met for various inspection scenarios. This was supplemented by deterministic fracture mechanics evaluations, which showed that the components are very flaw-tolerant.
- The current ASME Code, Section XI inspection schedules for the main steam and feedwater nozzle-to-shell welds and nozzle inside radius sections listed under Item Nos. C2.21, C.2.22, and C2.32 in Table IWC-2500-1 can be optimized without compromising plant safety.

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WHY THIS MATTERS

Establishing the technical bases for various inspection scenarios for PWR SG feedwater and main steam nozzles—based on operating history, inspection results to-date, and evaluation of the potential applicable degradation mechanisms—provides the benefit of possible optimization of these examinations in the future, potentially reducing health and safety risk of personnel, promoting improved ALARA practices, and decreasing overall inspection burdens, all without adversely impacting the safe operations of nuclear facilities.

HOW TO APPLY RESULTS

This report develops technical bases for various inspection scenarios using inputs designed to evaluate the applicability of the range of conditions experienced at operating reactors. Section 9 presents an approach to define the applicability of these technical bases for a given plant based on key criteria that determine whether the results of these analyses bound actual plant operation.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- Industry advisors have contributed to the review of this report.

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ACRONYMS

ASME	American Society of Mechanical Engineers
BTP	Branch Technical Position
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
BWRVIP	BWR Vessels and Internals Project
CE	Combustion Engineering
CC	Code Case
CRD	Control Rod Drive
DFM	Deterministic Fracture Mechanics
ECSCC	External Chloride Stress Corrosion Cracking
FAC	Flow-Accelerated Corrosion
FEA	Finite Element Analysis
FEM	Finite Element Model
FSAR	Final Safety Analysis Report
FW	Feedwater
HX	Heat Exchanger
HAZ	heat affected zone
IGSCC	Intergranular Stress Corrosion Cracking
ISI	In-Service Inspection
LEFM	Linear Elastic Fracture Mechanics
MIC	Microbiologically Influenced Corrosion
MRP	Materials Research Project
MS	Main Steam
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System

OTSG	Once-Through Steam Generator
PDI	Performance Demonstration Initiative
PFM	Probabilistic Fracture Mechanics
P&ID	Piping and Instrumentation Diagram
POD	Probability of Detection
PWHT	Post-Weld Heat Treatment
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
PVRUF	Pressure Vessel Research Facility User's Facility
RPV	Reactor Pressure Vessel
RSG	Replacement Steam Generator
R/t	Radius-to-Thickness Ratio
SG	Steam Generator
SCC	Stress Corrosion Cracking
TASCS	Thermal Stratification, Cycling, and Striping
TGSCC	Transgranular Stress Corrosion Cracking
V&V	Verification and Validation
W	Westinghouse

UNIT CONVERSION FACTORS

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1 inch = 2.54 cm = 25.4 mm

1°F = 1.8°C + 32

1°FΔ = 1.8°CΔ

1 psi = 6,895 Pa = 6.895x10-3 MPa

1 BTU = 1,055 joules

1 pound = 0.454 kg

1 foot = 30.48 cm = 304.8 mm

1 ksi = 6.895 MPa

1 ksi √inch = 1.099 MPa √m

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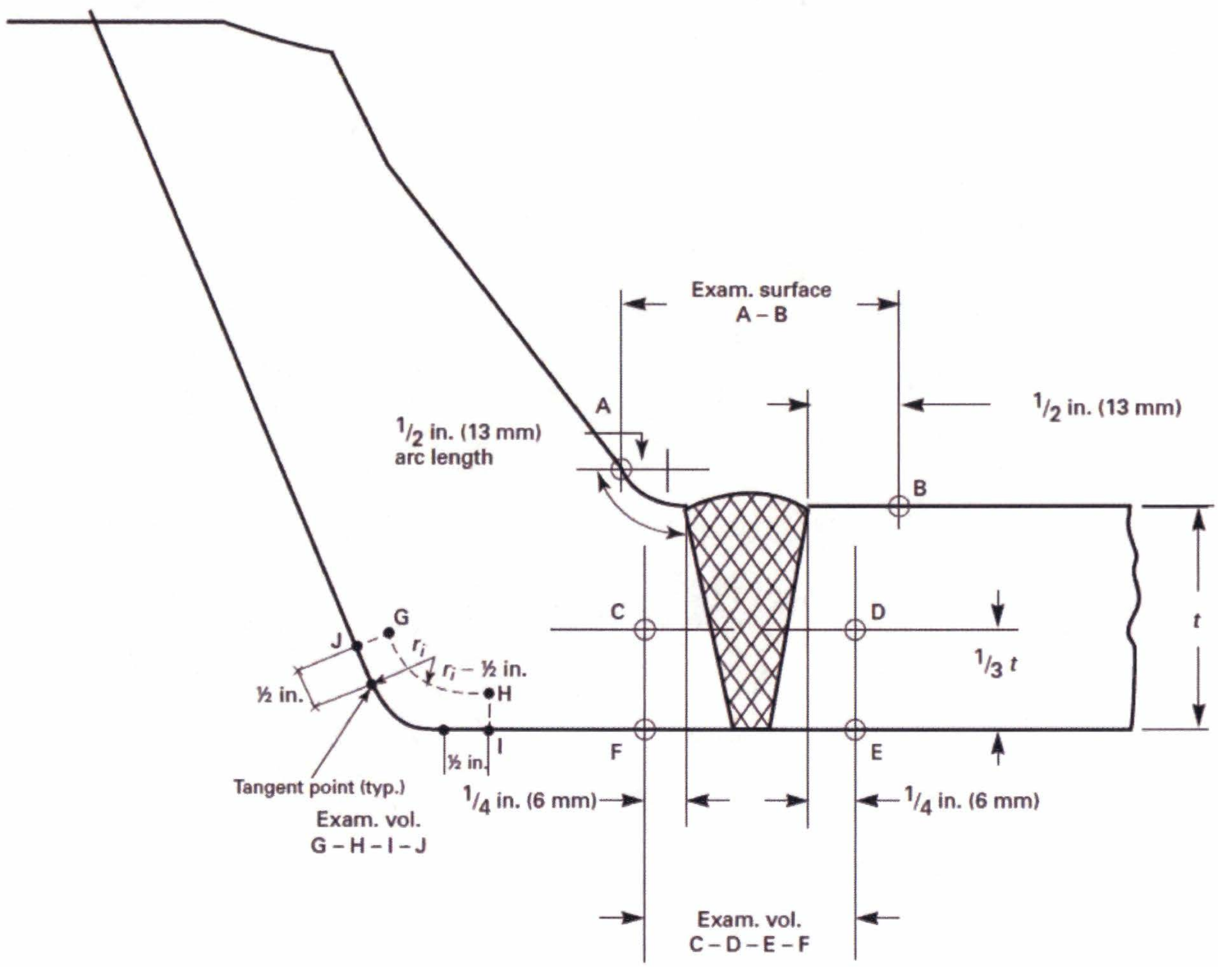
1 INTRODUCTION

1.1 Background

The welds on the secondary side in pressurized water reactor (PWR) steam generators (SGs) are classified as Class 2, Category C-B, pressure retaining welds in pressure vessels. This report focuses on the nozzle-to-shell welds and the inside radius sections of PWR SG feedwater and main steam nozzles, which are listed in ASME Code, Section XI, Table IWC-2500-1 [1] under the following item numbers:

- Item No. C2.21: Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds
- Item No. C2.32: Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds when inside of vessel is accessible
- Item No. C2.22: Nozzle inside radius section

The examination requirements for these components are also provided in Table IWC-2500-1 of the ASME Code, 2017 Edition [1]. The requirements for Item No. C2.21 require both surface and volumetric examinations of essentially 100% of the examination volume of the selected welds, while Item Nos. C2.22 and C2.32 require only volumetric examinations of essentially 100% of the examination volume of the selected welds. All these items require examination of all nozzles at the terminal ends of piping runs during each inspection interval. In the case of multiple vessels of similar design, size, and service, the examinations may be limited to a single vessel. In this case, all nozzle-to-shell welds and nozzle inside radius sections are examined every 10 years for a single SG representing the population. Typical nozzle-to-shell weld and nozzle inside radius section configurations and associated examination surfaces and volumes are provided in ASME Code Section XI, Figures IWC-2500-4(a), (b), and (d), which are reproduced in Figures 1-1, 1-2, and 1-3, respectively. Note that Figure 1-3 shows a configuration in which the nozzle is welded outside of the shell or head; this configuration is not considered in this report because it was not one of the configurations identified during the review of main steam and feedwater nozzle designs currently in use.



(a) See Notes (1), (2), and (3)

Figure 1-1
ASME Code, Section XI, Figure IWC-2500-4(a)

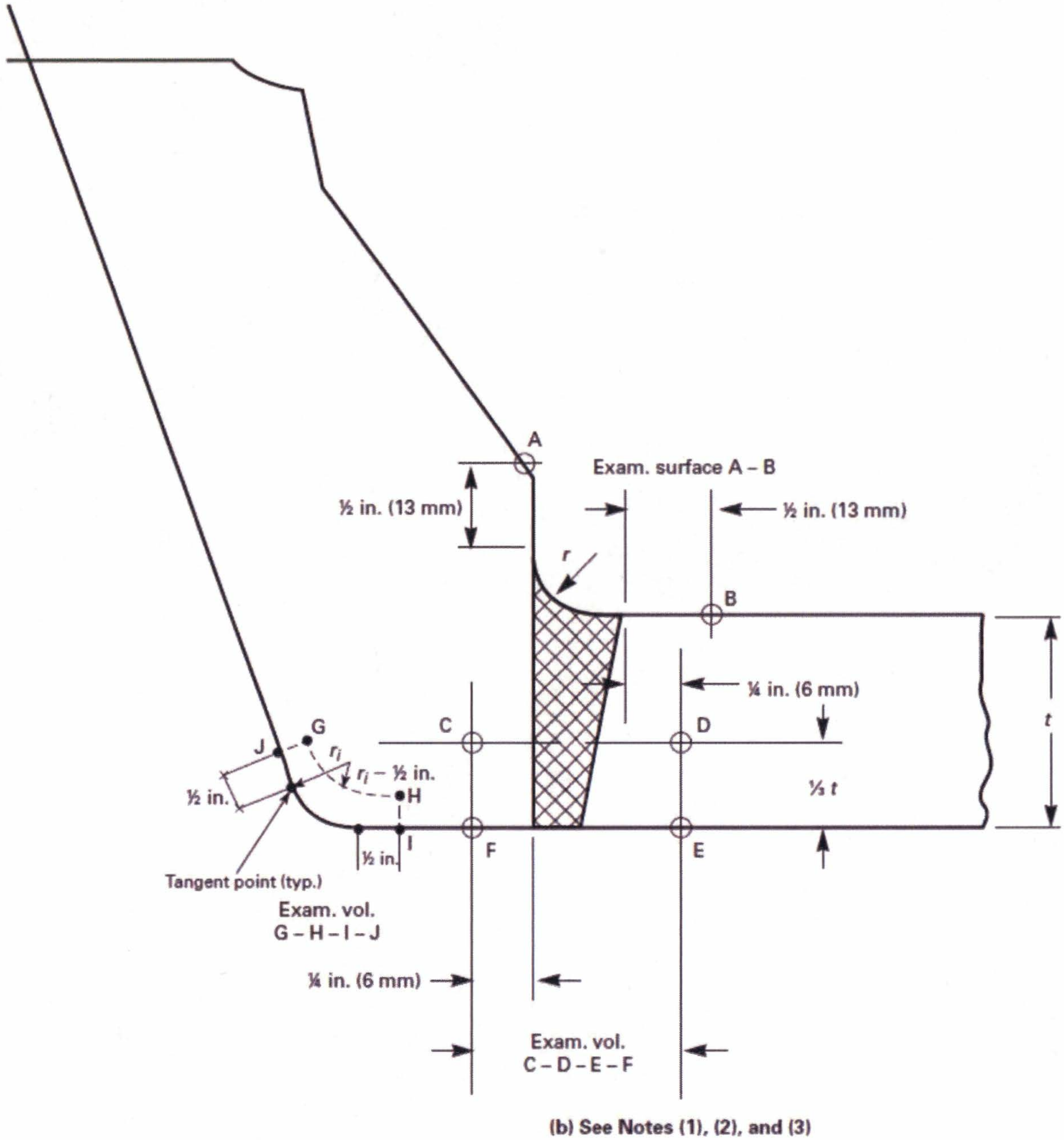
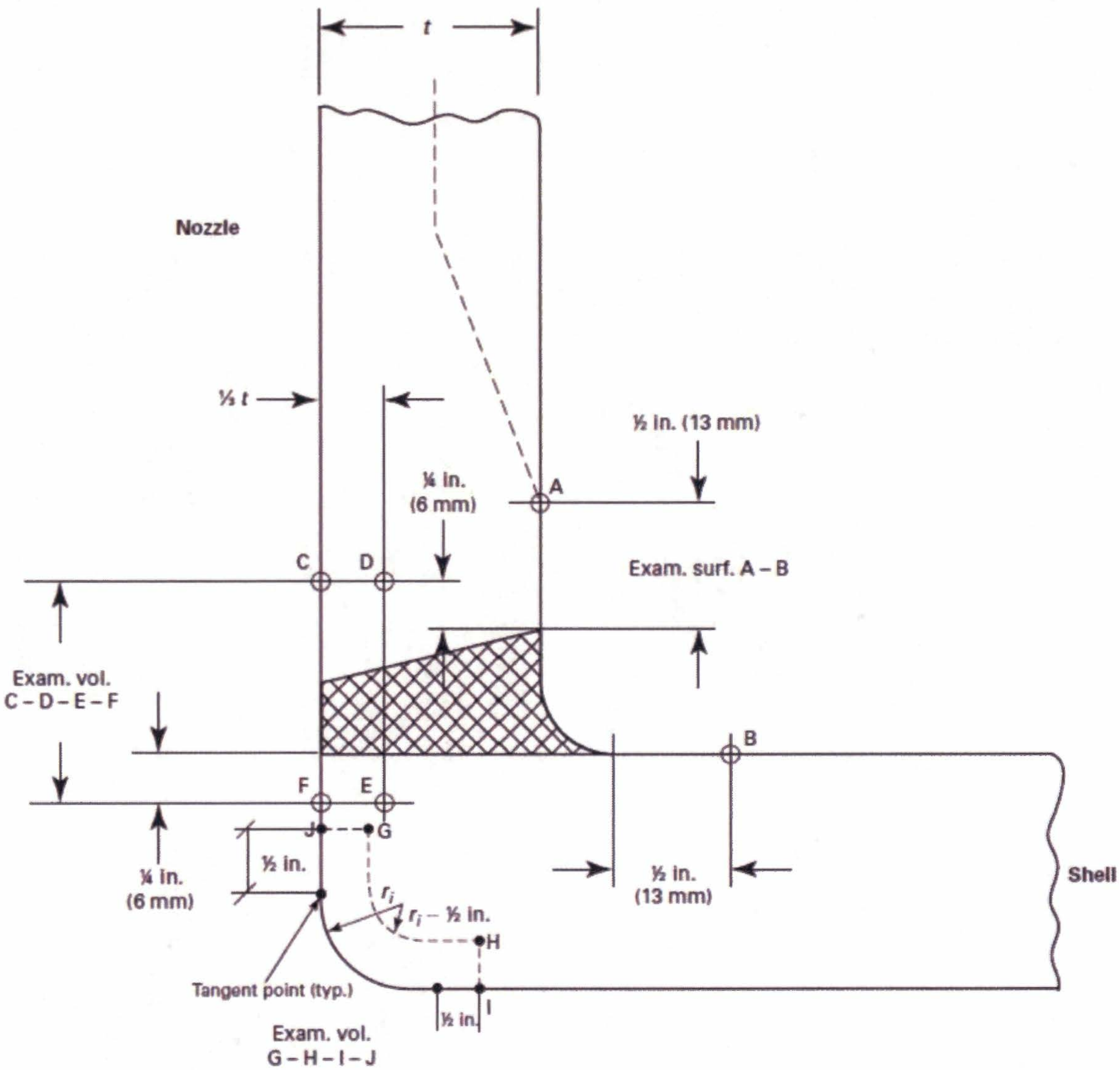


Figure 1-2
ASME Code, Section XI, Figure IWC-2500-4(b)



(d) See Notes (1) – (4)

GENERAL NOTES:
(a) $^1/_4$ in. = 6 mm
(b) $^1/_2$ in. = 13 mm
(c) NPS 12 = DN 300

NOTES:
(1) Nozzle sizes over NPS 4 (DN 100); vessel thickness over $^1/_2$ in. (13 mm).
(2) The dimensions for the examination volume shall be determined from the edge of the weld bevel if the weld toe extends beyond the bevel.
(3) Examination of inner radius volume is required only for nozzles greater than NPS 12 and may be limited to inner radius at vessel I.D.
(4) Configurations may include nozzle-to-shell or reinforcing-plate-to-nozzle welds that are other than full-penetration welds.

Figure 1-3
ASME Code Section XI, Figure IWC-2500-4(d)

Numerous examinations of these components have been performed by all plants in the U.S. fleet and other international plants that follow ASME Code Section XI, and many years of operating history now exist. This field experience serves as the impetus behind researching the potential for establishing the technical bases for the inspection requirements of these components.

1.2 Objective

The objective of this report is to evaluate the examination requirements for PWR SG feedwater and main steam nozzle-to-shell welds and nozzle inside radius sections and provide the technical bases for various examination scenarios for these components. To accomplish this objective, this report addresses various topics and is organized in the following sequence:

Section 2, Review of Previous Related Projects

Section 3, Review of Inspection History and Examination Effectiveness

Section 4, Survey of Components and Selection of Representative Components for Analysis

Section 5, Material Properties, Operating Loads, and Transients

Section 6, Evaluation of Potential Degradation Mechanisms

Section 7, Component Stress Analysis

Section 8, Probabilistic and Deterministic Fracture Mechanics Evaluation

Section 9, Plant-Specific Applicability

Section 10, Summary and Conclusions

2 REVIEW OF PREVIOUS RELATED PROJECTS

There have been several industry initiatives to provide alternative examination requirements for nozzles in lieu of the requirements in ASME Code Section XI. Some have been related to nozzle-to-shell welds and nozzle inside radius sections, while others have been related to other components. Most previous initiatives have focused on Class 1 nozzles, where the basis for alternative examination requirements has relied on plant operating experience and the results of nondestructive examinations performed on the relevant Code Item, as well as supplementary deterministic and/or probabilistic flaw tolerance evaluations. Although these previous initiatives are associated with Class 1 nozzles, they are reviewed and discussed in this section to provide guidance for the Class 2 nozzles currently being evaluated.

2.1 Nozzle-To-Shell Welds

Because most previous initiatives were related to Class 1 nozzle-to-shell welds, it is pertinent to review the Class 1 nozzle-to-shell welds in ASME Code Section XI. Currently, Class 1 nozzle-to-shell welds include the following:

- Item No. B3.90: Reactor vessel nozzle-to-vessel welds
- Item No. B3.110: Pressurizer nozzle-to-vessel welds
- Item No. B3.130: Steam generator (primary side) nozzle-to-vessel welds
- Item No. B3.150: Heat exchanger (primary side) nozzle-to-vessel welds

In 2002, the EPRI BWR Vessel and Internals Project (BWRVIP) undertook a study to optimize the inspection requirements for BWR reactor vessel nozzle-to-shell welds (Item No. B3.90 for BWRs). This work is documented in BWRVIP-108 [2] and provided the justification for the reduction of nozzle-to-shell weld examinations from 100% to a 25% sample of each nozzle type every 10 years. The feedwater and control rod drive (CRD) return line nozzles were excluded from that study. The justification was based on an industry survey of examination results to-date, which found no records of indications in the inspection results. A selection process was then used to identify a sampling of nozzles to use in deterministic and probabilistic fracture mechanics evaluations to evaluate the safety implications of reducing the examination population from 100% to 25%. The results of the analyses supported the alternative inspection criteria of a 25% sample of each nozzle type. In the safety evaluation of BWRVIP-108 [3], the NRC stated that licensees who plan to request relief from the current ASME Code Section XI requirements should demonstrate plant-specific applicability of the conclusions of BWRVIP-108 to their unit(s) by demonstrating that several general and nozzle-specific criteria are met.

As a result of the NRC conditions placed on BWRVIP-108, the BWRVIP initiated a follow-on study that resulted in BWRVIP-241 [4]. The intent of this study was to determine the extent to which the NRC conditions could be addressed to permit greater applicability to the BWR fleet without the need for a relief request. The results of this follow-on study concluded that not all BWR nozzles satisfy the NRC conditions and, as such, some require a plant-specific evaluation. Two of the criteria set forth by the NRC in Reference [3] were modified as a result of this study. The NRC issued a safety evaluation for BWRVIP-241 [5] accepting the modifications in Reference [4] to the original criteria established in Reference [3] but still requiring that plant-specific applicability be demonstrated against the modified criteria. The work performed in BWRVIP-108 and BWRVIP-241 provided the technical basis for ASME Code Case N-702 [6]. This Code Case was conditionally approved for use in Regulatory Guide 1.147 [7], where the NRC requested that applicability of the Code Case be demonstrated to the criteria in Section 5.0 of the NRC safety evaluation of BWRVIP-108 [3] and Section 5.0 of the NRC safety evaluation of BWRVIP-241 [5].

The PWR Owners Group also undertook a study to optimize the inspection requirements for PWR reactor vessel nozzle-to-shell welds (Item No. B3.90 for PWRs) [8]. A different approach was used in this study compared to the approach used by the BWRVIP for BWR nozzles, in that extension of the inspection interval was sought rather than a reduction in the inspection sample size. Based on operating experience, results of a survey of examination findings to-date, and the results of deterministic and probabilistic fracture mechanics evaluations, the NRC granted extension of the inspection interval from 10 to 20 years for PWR reactor vessel nozzle-to-shell welds. The probabilistic approach used elements of Code Case N-691 [9], which provides guidelines for risk-informed application to extend PWR inspection intervals. Three pilot plants (representing NSSS designs from Westinghouse, Combustion Engineering [CE], and Babcock and Wilcox [B&W]) were used in the study. In the safety evaluation [10], the NRC concluded that the methodology in Reference [8] can be applied to other PWR plants by confirming the applicability of parameters in Appendix A of Reference [8] on a plant-specific basis. The study in Reference [8] did not result in any changes to ASME Code Section XI requirements. However, because the 10-year inspection interval is required by Section XI, IWB-2412, as regulated in 10 CFR 50.55(a), licensees must submit an Exemption Request for NRC approval to extend the inspection interval of PWR reactor vessel nozzle-to-shell welds from 10 to 20 years.

Recognizing the difficulties involved in volumetric examination of regenerative and residual heat exchangers (because these components were not designed for such examinations), ASME published Code Case N-706 in November 2005 to allow visual (VT-2) examinations in lieu of the volumetric examinations for these heat exchangers. This Code Case applies to PWR heat exchangers fabricated from stainless steel for Examination Categories B-B, B-D, and B-J in Table IWB-2500-1 of ASME Code Section XI and Examination Categories C-A, C-B, and C-F-1 in Table IWC-2500-1 of ASME Code Section XI. This includes Class 1 Item No. B3.150 and Class 2 Item Nos. C2.21 and C2.32. The Code Case was revised in 2007 to N-706-1 [11], and the technical basis is provided in Reference [12], which relied on deterministic and probabilistic fracture mechanics evaluations. Code Case N-706-1 is unconditionally approved for use according to the latest revision of RG 1.147 and is currently being used by most U.S. utilities for the examination of stainless steel heat exchangers in PWRs.

2.2 Nozzle Inside Radius Sections

Previous initiatives related to nozzle inside radius sections are mostly associated with Class 1 components. Prior to the 1999 Addenda, ASME Code Section XI, IWB-2500 required examination of the inside radius sections of the following Class 1 vessel nozzles:

- Item No. B3.100: Reactor vessel nozzle inside radius section
- Item No. B3.120: Pressurizer nozzle inside radius section
- Item No. B3.140: Steam generator (primary side) nozzle inside radius section
- Item No. B3.160: Heat exchanger (primary side) nozzle inside radius section

ASME Code Section XI required volumetric examinations of all nozzles during the first interval and all successive inspection intervals with no deferral of inspection to the end of the interval. This means that essentially 100% of all nozzle inside radius sections were examined every 10 years.

The impracticality and difficulties associated with examining nozzle inside radius sections are documented in numerous Relief Requests to the NRC (for example, References [13, 14, 15]) for use of alternative methods to the Code-required examinations. Recognizing the hardship and associated dose in performing these examinations and coupled with the additional efforts required by the NRC for evaluation of numerous Relief Requests, cognizant ASME Code Section XI committees published a white paper attached to the Reference [13] Relief Request, which showed that examination of Item Nos. B3.120 and B3.140 could be eliminated with no safety implication for the plants. The white paper considered operating experience, examinations performed to-date for these Code Items, and the results of deterministic and probabilistic fracture mechanics evaluations to show that these examinations could be safely eliminated from the Code. This white paper provided the technical basis for Code Case N-619 [16], published in February 1999. This Code Case is conditioned by the NRC in Regulatory Guide 1.147 [7] to include a requirement for a VT-1 examination in lieu of the volumetric examination required by the Code. The elements of this Code Case were incorporated into Section XI in the Summer 1999 Addenda (the NRC condition notwithstanding). The NRC has since conditioned this item in 10 CFR 50.55(a) to require the use of the 1998 Edition of the ASME Code, Section XI. The 1998 Edition requires a volumetric examination; however, the condition allows for a VT-1 examination in lieu of the volumetric examination.

Following ASME approval of the alternative examination requirements for Item Nos. B3.120 and B3.140, the industry initiated development of alternative examination requirements for reactor vessel nozzle inside radius sections (Item No. B3.100). The results of this work were published in Reference [17], which was based on an industry survey of examination results to-date along with deterministic and probabilistic fracture mechanics evaluations and concluded that the reactor vessel nozzles are unlikely to fail under any anticipated service conditions. This conclusion applies to all reactor vessel nozzles regardless of plant type, with the exceptions of BWR feedwater and CRD return line nozzles. Therefore, the required inspection can be eliminated without any risk to the structural integrity of the vessel. The results of this study formed the technical basis for Code Case N-648-1 [18]. In lieu of the volumetric examination mandated by the ASME Code, Code Case N-648-1 requires VT-1 examination of the surface of the inside radius sections of all reactor vessel nozzles except for BWR feedwater and CRD return line nozzles. This Code Case has been conditionally approved for use by the NRC in Regulatory Guide 1.147 [7], with the NRC condition requiring a VT-1 examination in lieu of the volumetric examination using the allowable flaw length criteria of Table IWB-3512-1 with limiting assumptions on the flaw aspect ratio.

The work undertaken by the EPRI BWRVIP in 2001 and documented in Reference [2] also included a study to further optimize the inspection requirements for the inside radius sections of BWR reactor vessel nozzles. This work provided the justification for the reduction of the nozzle blend radius section examination population from a 100% to a 25% sample of each nozzle type every 10 years.

The study documented in Reference [12], which formed the basis for Code Case N-706-1 [11], also applies to the inside radius sections of Class 1 and Class 2 heat exchangers nozzles. Specifically, Code Case N-706-1 permits the use of VT-2 for examination of Class 1 Item No. B3.160 and Class 2 Item No. C2.22 in lieu of the ASME Code required volumetric examinations for stainless steel heat exchangers in PWRs.

2.3 Other Related Work

In addition to the initiatives discussed in Sections 2.1 and 2.2 to optimize examinations of nozzle-to-shell welds and nozzle inside radius sections, several other efforts have been conducted by the industry and cognizant ASME Section XI Code Committees to support optimizing the inspection of related plant components. These efforts are summarized as follows:

- In September 1995, BWRVIP-05 [19] provided the technical basis for elimination of examinations for BWR reactor vessel circumferential seam welds. This report was approved by the NRC in 1995. Since its approval, the alternative inspection requirements have been implemented by the entire U.S. BWR fleet.
- Code Case N-560-2 [20] was approved by ASME in March 2000 and provides alternative inspection requirements for Class 1, Category B-J piping welds. This Code Case permits a reduction in the examination population of Category B-J welds from 25% to 10% using risk-informed principles. The technical basis for this Code Case is provided in EPRI TR-112657, Rev. B-A [21]. This Code Case is not approved for use by the NRC and is listed in Regulatory Guide 1.193 [41] but has been used extensively by the industry through Relief Requests to the NRC.
- Code Cases N-577-1 [22] and N-578-1 [23] were approved by ASME in March 2000, and they provide alternative examination requirements for Class 1, 2, and 3 piping welds based on risk-informed principles. The use of these Code Cases has led to significant reductions in examinations of such piping welds. The technical bases for these Code Cases are provided in WCAP-14572 [24] and EPRI TR-112657, Rev. B-A [21], respectively. These Code Cases are not approved for use by the NRC and are listed in Regulatory Guide 1.193 [41] but have been used extensively by the industry through Relief Requests to the NRC.
- Code Case N-716-1 [25] was approved by ASME in January 2013 and provides alternative piping classification and examination requirements for Class 1, 2, 3, and non-Class piping welds as well as for Category C-A, C-B, C-D, and C-G components. The technical basis for this Code Case is provided in References [26] and [27]. This Code Case was unconditionally approved for use and incorporated into Revision 17 of NRC Regulatory Guide 1.147 [7].

- Code Case N-613-2 [28] was approved by ASME in December 2010 and provides alternatives to the examination volume requirements in Figures 2500-7(a), (b), (c), and (d) of ASME Code Section XI for the ultrasonic examination of reactor vessel nozzle-to-vessel shell welds and nozzle inside radius sections. It is unconditionally approved for use in NRC Regulatory Guide 1.147 [7]. This Code Case results in a significant reduction in the examination volume for Category B-D nozzle welds by reducing the inspection volume of adjacent material from half the shell thickness to ½ inch. This has resulted in a significant reduction in both qualification and scanning times.
- Code Case N-552 [29] was approved by ASME in April 2002 and provides alternative requirements to ASME Code Section XI, Appendix VIII, Supplement 5 for procedure qualification for inspecting nozzle inside radius sections from the outside surface. It is conditionally approved in NRC Regulatory Guide 1.147 [7]. This Code Case permits computational modeling techniques for the qualification of the nozzle inside radius section examination in lieu of qualification on multiple configurations.
- EPRI 3002007626 [30], published in April 2016, provides the technical basis for alternative inspection requirements for the reactor vessel threads in flange examination (ASME Code Section XI, Examination Category B-G-1, Item No. B6.40). Since then, at least four utilities have sought relief and received NRC approval for using this alternative [31, 32, 33, 34].
- EPRI 3002012966 [35], published in April 2018, provides the technical basis for alternative inspection requirements for the accessible areas of reactor vessel interior (ASME Section XI Examination Category B-N-1).

2.4 Conclusions

Tables 2-1 and 2-2 provide Code and alternative requirements for relevant nozzle-to-shell welds and nozzle inside radius sections, respectively. Table 2-1 shows that, of the six Code Items related to nozzle-to-shell welds in ASME Code Section XI, IWB-2500 and IWC-2500, only the Class 1 reactor vessel nozzle-to-shell welds (Item No. B3.90) and PWR stainless steel heat exchanger components (Item Nos. B3.150, C2.21, and C2.32) have been addressed or optimized. The technical bases for inspection scenarios associated with Item Nos. C2.21 and C2.32 nozzle-to-shell welds in PWR SG feedwater and main steam nozzles have not yet been established.

Table 2-2 shows that, of the four Code Items related to the nozzle inside radius sections currently in ASME Code Section XI, IWB-2500, the Class 1 pressurizer nozzle inside radius section (Item No. B3.120) and steam generator (primary side) nozzle inside radius section (Item No. B3.140) were eliminated from Section XI, while the Class 1 reactor vessel nozzle inside radius section (Item No. B3.100) and PWR stainless steel heat exchanger components (Item Nos. B3.160 and C2.22) have been addressed or optimized. The technical bases for inspection scenarios associated with Item No. C2.22 nozzle inside radius sections in PWR SG feedwater and main steam nozzles have not yet been established.

Table 2-1
Code Examination Requirements and Alternative Requirements for Nozzle-to-Shell Welds (shaded entries are those in the scope of this evaluation)

Code Item No.	Parts Examined	Code Requirements			Alternative Requirements				
		Exam Method	Exam Sample	Exam Frequency	Alternative to Code		Exam Method	Exam Sample	Exam Frequency
B3.90	Reactor Vessel Nozzle-to-Vessel Welds	Volumetric	All welds	Every interval	PWRs	WCAP-16168-NP-A	Volumetric	All welds	20 years
					BWRs	CC N-702	Volumetric or VT-1	25% of each nozzle type	Every interval
B3.110	Pressurizer Nozzle-to-Vessel Welds	Volumetric	All welds	Every interval	None		N/A	N/A	N/A
B3.130	Steam Generator (Primary Side) Nozzle-to-Vessel Welds	Volumetric	All welds	Every interval	None		N/A	N/A	N/A
B3.150	Heat Exchanger (Primary Side) Nozzle-to-Vessel Welds	Volumetric	All welds	Every interval	PWR SS HXs	CC N-706-1	VT-2	All nozzles	Every interval
					All others	None	N/A	N/A	N/A
C2.21	Nozzle-to-Shell Weld	Surface and Volumetric	All nozzles at terminal ends of piping runs (1)	Every interval	PWR SS HXs	CC N-706-1	VT-2	All nozzles	Every interval
					All others	None	N/A	N/A	N/A
C2.32	Nozzle-to-Shell Welds (When Inside of Vessel Is Accessible)	Volumetric	All nozzles at terminal ends of piping runs (1)	Every interval	PWR SS HXs	CC N-706-1	VT-2	All nozzles	Every interval
					All others	None	N/A	N/A	N/A

Note:

1. In the case of multiple vessels of similar design, size, and service (such as steam generators), the required examinations may be limited to one vessel or distributed among the vessels.

Table 2-2
Code Examination Requirements and Alternative Requirements for Nozzle Inside Radius Sections (shaded entry is within the scope of this report)

Code Item No.	Parts Examined	Code Examination Requirements: Method	Code Examination Requirements: Sample	Code Examination Requirements: Frequency	Alternative Examination Requirements: Alternative to Code		Alternative Examination Requirements: Method	Alternative Examination Requirements: Sample	Alternative Examination Requirements: Frequency
B3.100	Reactor Vessel Nozzle Inside Radius Section	Volumetric	All nozzles	Every interval	PWRs	CC N-638-1	VT-1	All nozzles	Every interval
					BWRs	CC N-638-1	VT-1	All nozzles	Every interval
						CC N-702	Volumetric or VT-1	25% of each nozzle type	Every interval
B3.120 (1, 2)	Pressurizer Nozzle Inside Radius Section	N/A (2)	N/A (2)	N/A (2)	CC N-619 (2)		None	None	None
B3.140 (1, 2)	Steam Generator (Primary Side) Inside Radius Section	N/A (2)	N/A (2)	N/A (2)	CC N-619 (2)		None	None	None
B3.160	Heat Exchanger (Primary Side) Inside Radius Section	Volumetric	All nozzles	Every interval	PWR SS HXs	CC N-706-1	VT-2	All nozzles	Every Interval
					All Others	None	N/A	N/A	N/A
C2.22	Nozzle Inside Radius Section	Volumetric	All nozzles at terminal ends of piping runs (3)	Every interval	PWR SS HXs	CC N-706-1	VT-2	All nozzles	Every Interval
					All Others	None	N/A	N/A	N/A

Notes:

1. These items were eliminated from Section XI starting with the 1999 Addenda.
2. These items were conditioned by the NRC in Regulatory Guide 1.147 [7] to require the use of the 1998 Edition of Section XI. This edition requires a volumetric examination; however, the condition allows for a VT-1 examination in lieu of the volumetric examination.
3. In the case of multiple vessels of similar design, size, and service (such as steam generators), the required examinations may be limited to one vessel or distributed among the vessels.

3 REVIEW OF INSPECTION HISTORY AND EXAMINATION EFFECTIVENESS

As part of this report, a survey was conducted to collect the number of examinations performed and associated examination results for Item Nos. C2.21, C2.32, and C2.22 for both U.S. and international nuclear units. The PWR SG feedwater and main steam nozzle components evaluated in this report are addressed by these item numbers. The survey was conducted between April and September of 2017. Responses were obtained from a total of 69 U.S. and 5 international units. The data gathered from this survey covered plant designs fabricated by B&W, CE, Westinghouse, and General Electric (GE). The following information was requested:

- Plant name
- Total number of applicable components for the ASME Code Section XI Item and associated vessel description (for example, heat exchanger)
- Total number of ISI examinations performed on the subject component to-date
- Number of examinations containing flaws that exceed ASME Code Section XI acceptance standards (that is, IWB-3500)
- Total number of flaws (across all examinations for this item)
- How were the flaws that exceeded ASME Code Section XI acceptance standards dispositioned?
- Estimated dose accumulated per examination (including any pre- and post-examination activities such as insulation removal, scaffolding erection, and so on)
- Does this examination have any impact on outage critical path?
- Have any Relief Requests been submitted and/or approved for this item?
- Comments or any additional information

3.1 Summary of Survey Results

The survey results for the subject items are provided in Reference [36] and are summarized in Table 3-1, Table 3-2, and Table 3-3. The results for all components are provided by plant design type and are discussed next. Results related to PWR SG feedwater and main steam nozzle components are highlighted where appropriate.

- Item No. C2.21 (nozzle-to-shell [nozzle-to-head or nozzle-to-nozzle] welds): The summary of results for Item No. C2.21 is shown in Table 3-1. This item applies to both PWRs and BWRs, and the table shows that it is present in all PWR and BWR plant designs. Among the PWRs, most of these components (at least 90%) are SG feedwater and main steam nozzle-to-shell welds. Of the 483 examinations identified by plants that responded to the survey that have been performed on this item to-date—including 377 examinations of PWR components—only one (PWR) examination found indications that exceeded the acceptance criteria of ASME Code Section XI. A magnetic particle examination identified linear indications of 0.3 in. and 0.5 in. in the outside surface of the nozzle-to-shell weld of the main steam nozzle. After blending (light grinding) and reexamination, the indications were found to be below the acceptance criteria and returned to service [79]. ASME Code Section XI, Table IWC-2500-1 [1] requires surface and volumetric examinations of all nozzles at terminal ends of piping runs each inspection interval. In the case of multiple vessels of similar design, size, and service, the examinations may be limited to only one vessel.

- Item No. C2.32 (nozzle-to-shell [nozzle-to-head or nozzle-to-nozzle] welds when inside of vessel is accessible): The summary of results for Item No. C2.32 is shown in Table 3-2. This item applies to both PWRs and BWRs; however, the table shows that it is present in only three B&W PWR units and one Westinghouse 2-loop PWR unit. None of these items is associated with PWR SG feedwater or main steam nozzles.
- Item No. C2.22 (nozzle inside radius section): The summary of results for Item No. C2.22 is presented in Table 3-3. This item applies to both PWRs and BWRs, and the table shows that it is present in all PWR and BWR plant designs. Among the PWRs, most of these components (at least 90%) are SG feedwater and main steam nozzle inside radius section components. Of the 232 examinations identified by plants that responded to the survey that have been performed on this item to-date—including 174 examinations of PWR components—no examinations identified flaws that exceed the ASME Code Section XI flaw acceptance criteria. ASME Code Section XI, Table IWC-2500-1 [1] requires surface and volumetric examinations of all nozzles at terminal ends of piping runs each inspection interval. In the case of multiple vessels of similar design, size, and service, the examinations may be limited to only one vessel.

Table 3-1
Summary of Survey Results for Item No. C2.21

Plant Type	Total No. of Units in Survey	No. of Units with this Item	Total No. of Applicable Components	Total No. of Examinations	Total No. of Exams that Exceed Section XI Acceptance Criteria
BWR	27	21	70	106	0
PWR W-2 Loop	4	4	8	15	0
PWR W-3 Loop	11	11	78	126	0
PWR W-4 Loop	20	20	213	143	1
PWR-CE	6	6	56	62	0
PWR B&W	6	6	20	31	0
Total	74	68	445	483	1

Table 3-2
Summary of Survey Results for Item No. C2.32

Plant Type	Total No. of Units in Survey	No. of Units with this Item	Total No. of Applicable Components	Total No. of Examinations	Total No of Exams that Exceed Section XI Acceptance Criteria
BWR	27	0	0	0	0
PWR W-2 Loop	4	1	4	12	0
PWR W-3 Loop	11	0	0	0	0
PWR W-4 Loop	20	0	0	0	0
PWR-CE	6	0	0	0	0
PWR B&W	6	3	4	0	0
Total	74	4	8	12	0

Table 3-3
Summary of Survey Results for Item No. C2.22

Plant Type	Total No. of Units in Survey	No. of Units with this Item	Total No. of Applicable Components	Total No. of Examinations	Total No. of Exams Containing Flaws that Exceed Section XI Acceptance Criteria
BWR	27	15	43	58	0
PWR W-2 Loop	4	4	7	19	0
PWR W-3 Loop	11	11	51	36	0
PWR W-4 Loop	20	20	103	82	0
PWR-CE	6	6	18	27	0
PWR B&W	6	2	8	10	0
Total	74	58	230	232	0

3.2 Conclusions

A survey was conducted to collect the number of examinations performed and associated examination results for Item Nos. C2.21, C2.32, and C2.22, including the PWR SG feedwater and main steam nozzle components evaluated in this report. The survey results are summarized in Table 3-1, Table 3-2, and Table 3-3. Of a total of 727 examinations identified by the plants that responded to the survey that have been performed on Item Nos. C2.21, C2.32, and C.22 components, only one (PWR) examination (for Item No. C2.21) identified linear indications that exceeded the acceptance criteria of ASME Code Section XI (they were dispositioned by light grinding). The vast majority of the 563 PWR examinations performed were for the SG feedwater and main steam nozzle components evaluated in this report.

4 SURVEY OF COMPONENTS AND SELECTION OF REPRESENTATIVE COMPONENTS FOR ANALYSIS

The previous section reviewed the examination history for Item Nos. C2.21, C2.32, and C2.22, including the PWR SG feedwater and main steam nozzle components evaluated in this report. In this section, representative components are selected for analysis. Selection of representative components considered the following factors:

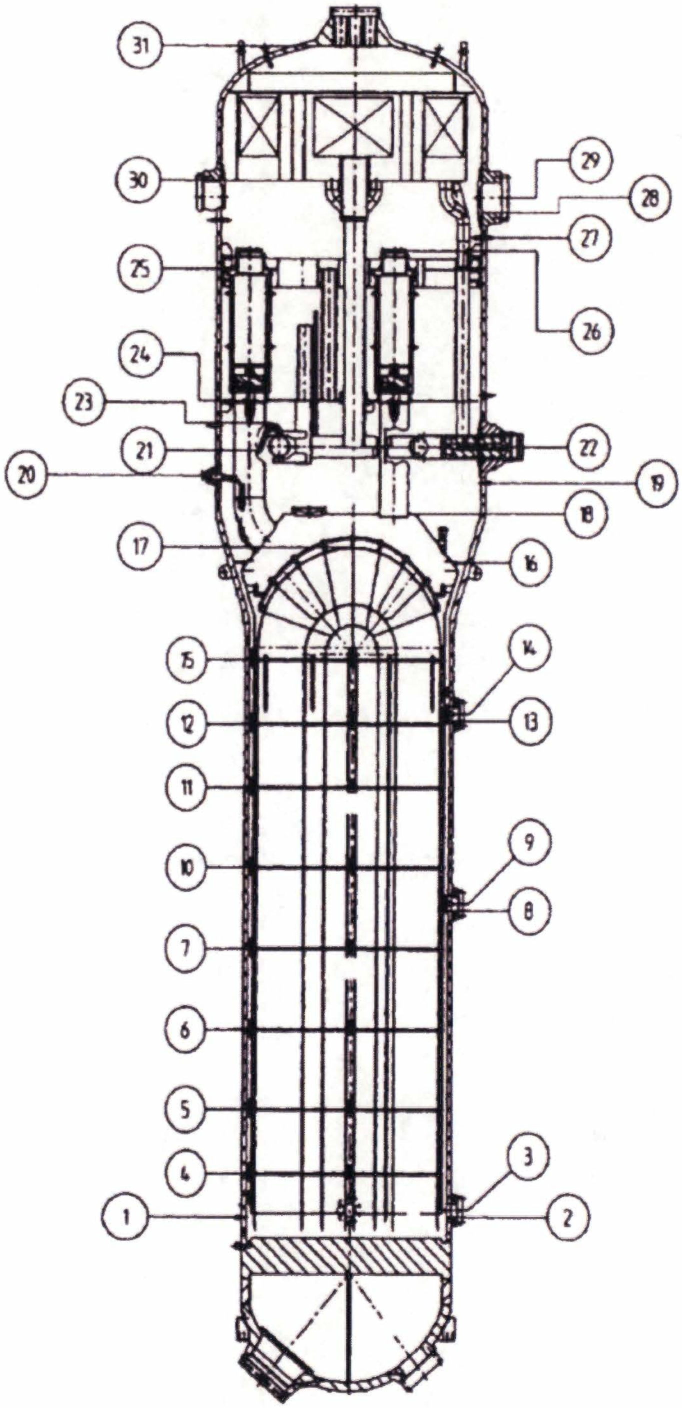
- Whether a single PWR component could represent all plant designs types (that is, 2-loop Westinghouse, 3-loop Westinghouse, 4-loop Westinghouse, B&W, and CE)
- Component geometry
- Component operating characteristics (loads)[1]
- Component materials
- Field experience with regard to service-induced cracking[2]
- The availability and quality of component-specific information

4.1 Steam Generators and NSSS Suppliers

All PWR plants contain SGs to convert primary side heat into steam to generate power. As part of the secondary side, all SGs have two main types of nozzles: main steam nozzles (the SG outlet to carry steam to the turbine) and feedwater nozzles (the inlet to return condensate to the SG). These two nozzle types are the subject of this report.

Three NSSS vendors were investigated. Westinghouse has three designs classified by the number of reactor coolant loops (two, three, or four). In Westinghouse designs, each loop contains one steam generator. CE plants have two reactor coolant loops with a total of two hot legs (each running to one SG) and four cold legs. Similar to CE plants, B&W plants also have two reactor coolant loops with a total of two hot legs (each running to one SG) and four cold legs. Both Westinghouse and CE plants use a U-tube SG design, while B&W plants use a once-through steam generator (OTSG) design. Schematics of Westinghouse 2-loop, CE, and B&W SGs are shown in Figures 4-1, 4-2, and 4-3, respectively.

[1] General operating characteristics (loads) were considered in this section; for a detailed discussion of applicable loads, see Section 5. Similarly, only thermal and pressure transients were considered in this section with regard to degradation; for a detailed discussion of degradation mechanisms, see Section 6.
[2] As discussed in Section 3.1, the Reference [36] survey results indicate that only one indication was found for an Item No. C2.21 component, which was dispositioned by light grinding.



ITEM	DESIGNATION
1	Lower WR level tap CL
2	Bottom of hand hole (ID)
3	Hand hole CL
4	TSP 1 CL
5	TSP 2 CL
6	TSP 3 CL
7	TSP 4 CL
8	Bottom of hand hole (ID)
9	Hand hole CL
10	TSP 5 CL
11	TSP 6 CL
12	TSP 7 CL
13	Bottom of hand hole (ID)
14	Hand hole CL
15	TSP 8 CL
16	Foreign object catcher CL
17	Top of tube bundle (Apex)
18	Top of wrapper roof
19	Lower NR level tap CL
20	Recirculation nozzle CL
21	Bottom of J-nozzles (outlet)
22	Feedwater nozzle CL
23	Top of J-nozzles (Apex)
24	Lower deck plate (Bottom)
25	Upper deck plate (Top)
26	Top of cyclones (Foreign object catcher CL)
27	Upper NR and WR level tap CL
28	Bottom of secondary manway (ID)
29	Secondary manway CL
30	Bottom of dryer block
31	Bottom of steam outlet nozzle

Figure 4-1
Westinghouse U-Tube Steam Generator

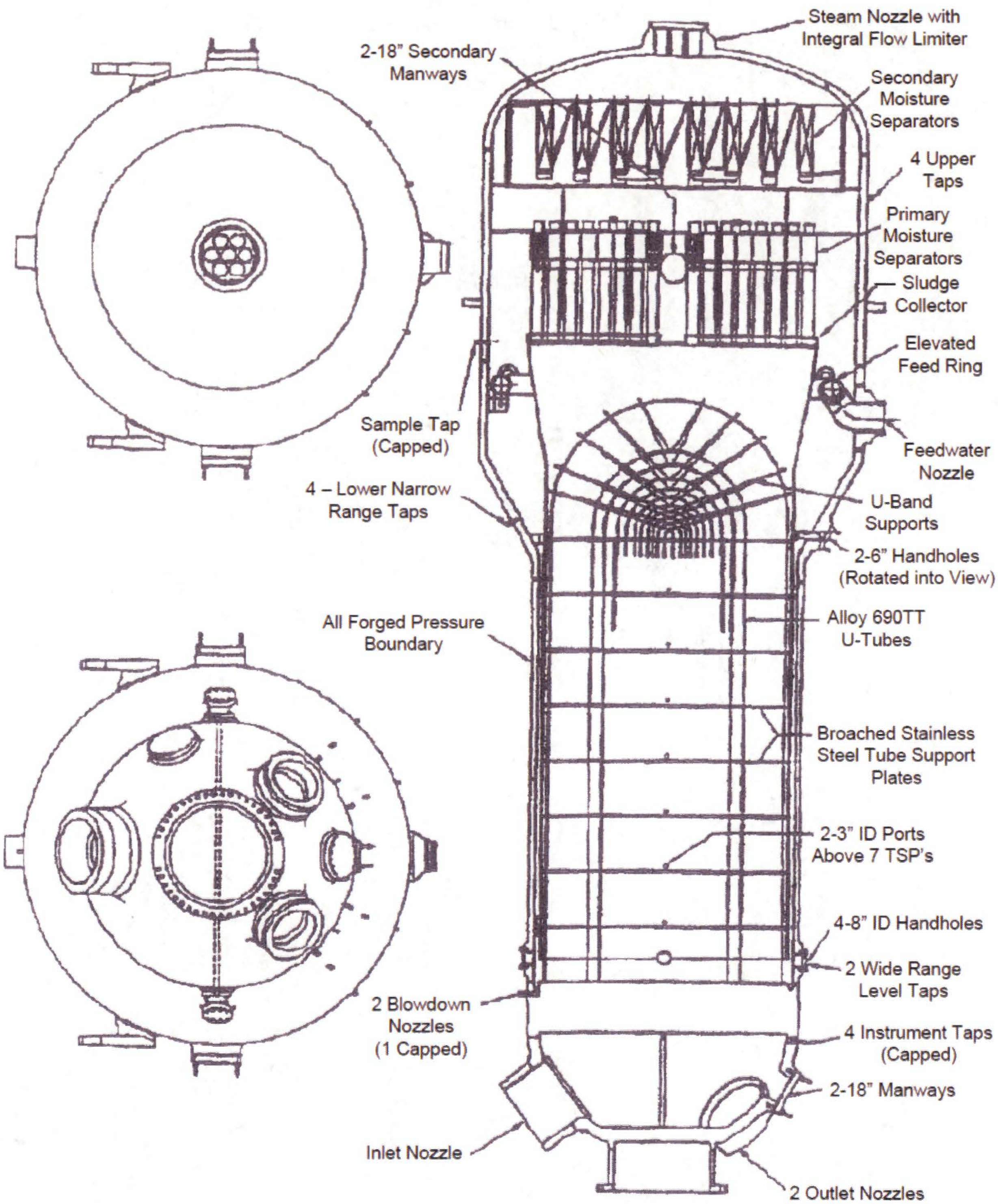


Figure 4-2
CE U-Tube Steam Generator

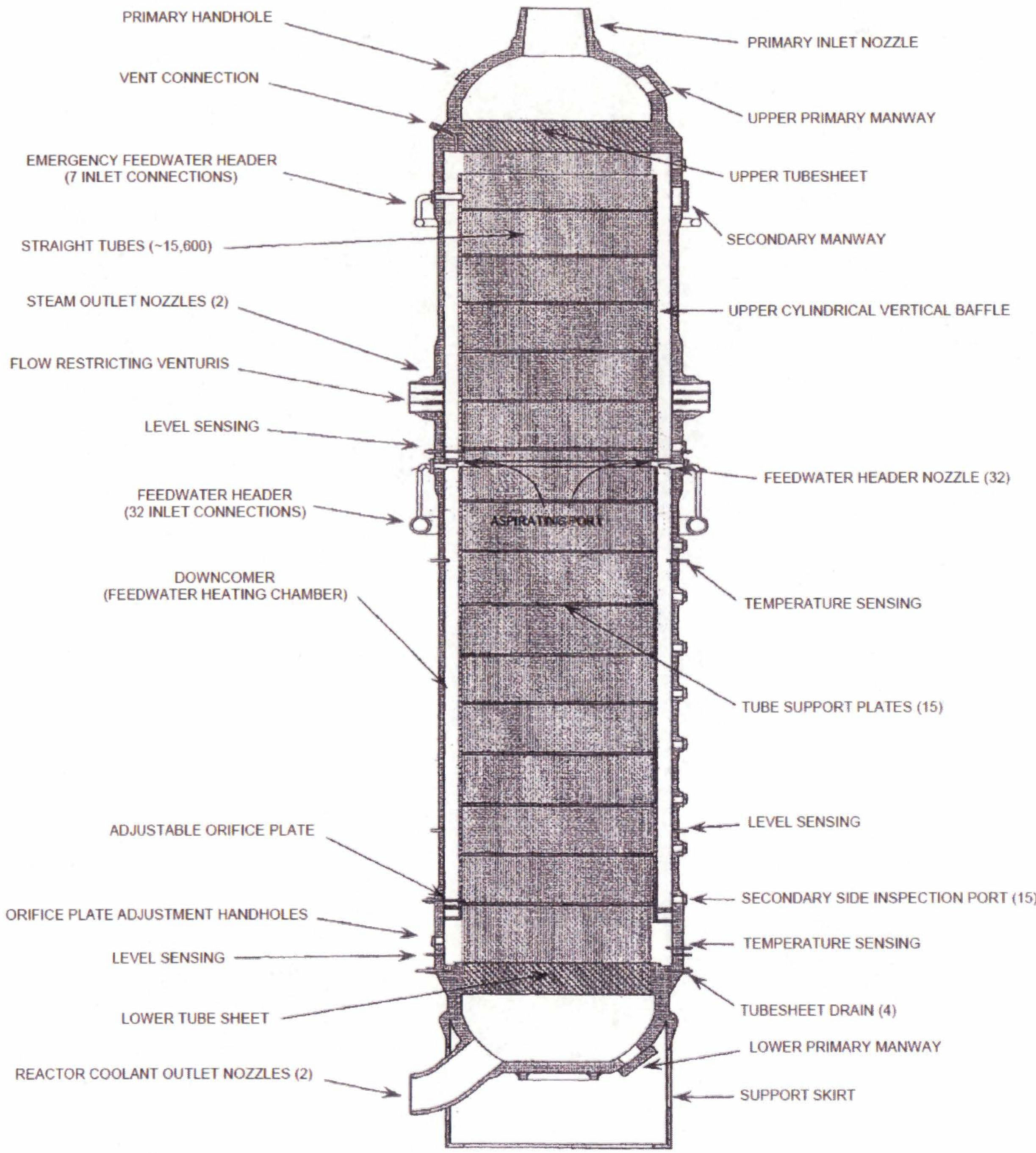


Figure 4-3
B&W Once-Through Steam Generator

As Figures 4-1 and 4-2 show, the Westinghouse and CE SG designs are similar, though the CE design is larger. The B&W OTSG design is substantially different. One notable difference is that the main steam nozzle is located at the top of the SG dome for the Westinghouse and CE designs, while the B&W SG design has the main steam nozzle located on the side of the shell.

4.2 Steam Generator Operating Experience

PWR SGs and their associated components have been subject to degradation throughout their operating history. The first generation of SGs had Alloy 600 tubing that was subject to stress corrosion cracking. Due to excessive cracking and the need for plugging tubes, many SGs were replaced throughout the fleet (50% as of 2004 [37]). This activity has increased as the plants continue to age. Therefore, much of the information used for this report comes from replacement SGs (RSGs). Although RSGs have not been in service for the full life of the plant, they did not substantially change the operating characteristics of the SG secondary side.

SG feedwater nozzles (and connected piping) are subject to thermal fatigue and have experienced cracking incidents, mainly associated with certain design and operating characteristics. For designs in which auxiliary feedwater enters the SGs through the main feedwater nozzles, the potential for fluid stratification cycling exists under low-flow conditions during plant heat-up and cooldown that can result in fatigue crack initiation. The feedwater piping connected to the nozzle has experienced failures dating back to 1979, when DC Cook found a through-wall leak. This incident led to all plants being required to inspect their feedwater piping, which identified more cracking and through-wall leaks. The degradation has mainly been found in the pipe adjacent to the feedwater nozzles (at a distance away from the nozzle-to-shell welds and nozzle inside radius sections that had no relevant impact on the evaluation in this report).

Thermal fatigue due to cycling and stratification in the feedwater piping has led to many piping replacements and design modifications to eliminate this phenomenon [38]. Many RSGs were designed with an integral thermal sleeve with the nozzle safe end, which prevents bypass flow from reaching the feedwater nozzle body. The thermal sleeve therefore protects the feedwater nozzle components evaluated in this report (nozzle-to-shell welds and nozzle inside radius sections) from thermal stratification (see Section 6 for more detail).

In contrast to PWR SG feedwater nozzles, main steam nozzles do not have operating experience of degradation or cracking due to thermal fatigue.

4.3 Variation Among Steam Generator Designs

As noted, many SGs were replaced due to plugging in a large percentage of tubes. The RSGs were not always replaced by the original NSSS supplier. Because there are numerous RSG manufacturers and designers, SG designs can vary—making it challenging to identify a bounding or representative design. To better understand the extent of the variations among SG designs, information was tabulated across the SG population using both public (for example, Final Safety Analysis Report [FSAR]) and plant-specific information. Although the tabulated information is not comprehensive, a best effort was made to compile consistent and accurate information on the following:

1. Plant name
2. NSSS vendor
3. Number of reactor coolant loops
4. Steam generator manufacturer
5. Steam generator designer
6. Steam generator model designation
7. Upper and lower steam generator shell material
8. Upper and lower steam generator shell diameter and thickness (for U-tube designs)
9. Design code
10. Feedwater nozzle material, diameter, thickness, and inner radius
11. Main steam nozzle material, diameter, thickness, and inner radius
12. Plant design/operating pressure and temperature
13. Reactor coolant flow rate, steam flow rate, feedwater temperature, and steam pressure

A review of the information compiled showed that, despite the numerous designers and manufacturers, RSG designs are fairly consistent. This makes sense considering constraints such as the RSG having to fit in the same space and work to similar operating parameters as the original SG. Variations are most prominent in nozzle designs. Variations in SG dimensions, design temperatures and pressures, and ASME Code design considerations are discussed in Sections 4.3.1 through 4.3.3. Configurations for which relief has been sought due to difficulties in performing ASME Code examinations are discussed in Section 4.4. The selection of the main steam and feedwater nozzle components to be used in this evaluation is discussed in Section 4.5.

4.3.1 Dimensions

Based on the dimensional information compiled, the following trends were identified:

4.3.1.1 Westinghouse Designs

1. The size (diameter and thickness) of the SG is related to the model and manufacturer rather than the number of loops. The upper shell that contains the feedwater and main steam nozzles varies from an outside diameter of 166 in. to 177 in., which is a variation of less than 10%. Similarly, the radius-to-thickness (R/t) ratio is largely influenced by the diameter of the vessel. The R/t ratio varies from 23 to 26 for the upper shell portion.
2. The main steam nominal pipe size (NPS) is approximately 30 in. Most plants have an NPS of around 32 in. with a minimum of 28 in. Again, this variation is approximately 10%. From the available specific nozzle information, the R/t ratio for the nozzle body ranges from a value of 2 to 3.
3. The feedwater NPS is mostly 16 in. The range in pipe size varied between 14 in. and 18 in. This variation in pipe size is about 20%. From the available specific nozzle information, the R/t ratio for the nozzle body ranges from a value of 1 to 2.

4.3.1.2 CE Designs

1. CE SG shell diameters are all very similar with negligible variation among plants. The upper shell outside diameter for CE designs is approximately 240 in., which is much larger than any of the Westinghouse designs. The one exception is the CE System 80 SG design. The R/t ratio for the System 80 design is approximately 21, while the other CE designs have a ratio of approximately 24.5. Due to the uniqueness of the System 80 design, and the fact that this design is represented at only one plant reviewed, this report will not consider the System 80 design.
2. The main steam nominal piping has a 34-in. (ID) diameter. The one exception is the System 80 design that contains two main steam nozzles, each with an outside diameter of 28 in. Due to the uniqueness of the System 80 design, and the fact that this design is represented at only one plant reviewed, this report will not consider the System 80 design. Sufficient specific information on other (non-System 80) main steam nozzles was not available to develop an R/t ratio for the nozzle body.
3. The feedwater NPS varies between 14 in. and 18 in. This is similar to the Westinghouse plant designs and is a variation in pipe size of about 20%. From the available specific nozzle information, the R/t ratio for the nozzle body ranges from a value of 1 to 2.

4.3.1.3 B&W Designs

1. The OTSGs have a shell outside diameter of approximately 150 in. with negligible variation among plants. The R/t ratios among plants will be similar but are difficult to specify because of varying thicknesses throughout the vessel. There is a nominal thickness for the vessel, but there are also increased thicknesses at the nozzle locations. The available specific information does not always provide the vessel thickness at the same location, so a direct comparison is not feasible. At the SG nozzle locations, the R/t ratio will be approximately 11, while other SG locations will have R/t ratios of 24.5.
2. The main steam NPS is 24 inches for all plants for which information was obtained (no variation in pipe size). Based on available information, the R/t ratio is 2.5.
3. From plant-specific information obtained, B&W plants do not have feedwater nozzles welded into the SG shells (the nozzle is actually a bolted joint) and have multiple penetrations in the shell that riser pipes enter to provide feedwater flow to the feedwater ring inside the SG. Regardless, the NPS of the feedwater system is 14 in. (no variation in pipe size).

4.3.2 Design Pressures and Temperatures

Based on the information compiled, the design pressure for the steam side of the SG varies from 1,000 psi to 1,285 psi. The majority of SGs have a design pressure of approximately 1,100 psi (the average pressure among tabulated values is approximately 1,115 psi). The design temperature for the steam side of the SG varies from 550°F to 630°F. The majority of SGs have a design temperature of approximately 600°F (the average temperature among tabulated values is approximately 590°F).

4.3.3 ASME Code Design Considerations

Based on the information compiled, all SGs were designed to Section III of the ASME Code, although numerous editions were used. Some SGs were designed to the 1966 Edition while others were designed to the 1998 Edition. Despite the different ASME Code editions, the general stress criteria used to design the SGs have remained consistent since early versions of the ASME Code.

There are two items to consider for the SG shell and nozzles. First is the required thickness due to pressure. The general design rules for vessels are contained in NB-3300 [39]; the required pressure thickness (t) contained in NB-3324 is:

$$t = \frac{PR_o}{S_m + 0.5P} \qquad \textbf{Eq. 4-1}$$

where P is the design pressure, R_o is the outside radius and S_m is the design stress intensity.

Equation 4-1 is based on the calculation of pressure stress intensity, which is dominated by the pressure hoop stress (PR/t) where R is the mean radius. Because these vessels are all made of low-alloy steel (refer to Section 5), S_m is similar for all vessels. With the allowable stress consistent, the ratio between R and t needs to remain consistent to account for variations in the design pressure. As such, the component selected for this application will provide a ratio that can be used to determine whether a given plant will be bounded by the evaluation contained in this report.

The second item to consider for the SG nozzles is the design requirement for openings and reinforcement. NB-3332 [39] defines the rules for area reinforcement in vessels and formed heads. The SG main steam and feedwater nozzles would have been designed to these requirements, which have remained consistent throughout ASME Code history. These rules define two main criteria: 1) the amount of metal required for reinforcement and 2) the distance limits for the location for reinforcement. These values are based on the size of the hole, thickness of the shell/head, and diameter of the shell/head. For the main steam and feedwater nozzles specifically, the variations in hole size will affect the detailed nozzle geometry and reinforcement. Because most hole sizes for these nozzles are similar, the rules in the ASME Code will drive most nozzle geometries toward a consistent design. As noted earlier, the variation in the attached pipe size was around 20% for the feedwater piping, but most plants have a single pipe size. The average feedwater pipe size is approximately 16 in. NPS.

Because the design parameters and materials are similar, the design rules will drive toward consistency in the *R/t* ratio used to compare the relative stresses in the SGs. This is consistent with previous work: four of the five criteria set forth by the NRC in Reference [3] on plant-specific applicability of the BWRVIP-108 report [2] relate to the *R/t* ratio. This ratio also helps normalize differences between plant designs when multiple parameters are different. As noted in Section 4.3.1, *R/t* ratios for Westinghouse and CE SGs vary based on SG size; however, Westinghouse SGs have a ratio that varies from 23 to 26, while CE generators have a ratio of approximately 24.5. These values are all consistent, even though the SG diameters for CE plants are much larger. This fact allows the CE and Westinghouse generators to be represented by a single configuration (as will be discussed in Section 4.4). The *R/t* ratio will control the pressure loading because it is dependent on the geometry. Based on the data obtained, the *R/t* variations are expected to be only slightly greater than 10%. Such variations are handled by sensitivity analyses in the probabilistic fracture mechanics (PFM) portion of the evaluation (see Section 8), similar to the approach used in BWRVIP-241 [4] to address the NRC plant-specific applicability criteria in Reference [3]. Figure 4-4 shows a plot of *R/t* ratios (SG upper shell and nozzle bore) for multiple plant designs. The plot shows that feedwater nozzle values are clustered around the same *R/t* ratio while there is slightly more variation of *R/t* values for the main steam nozzles.

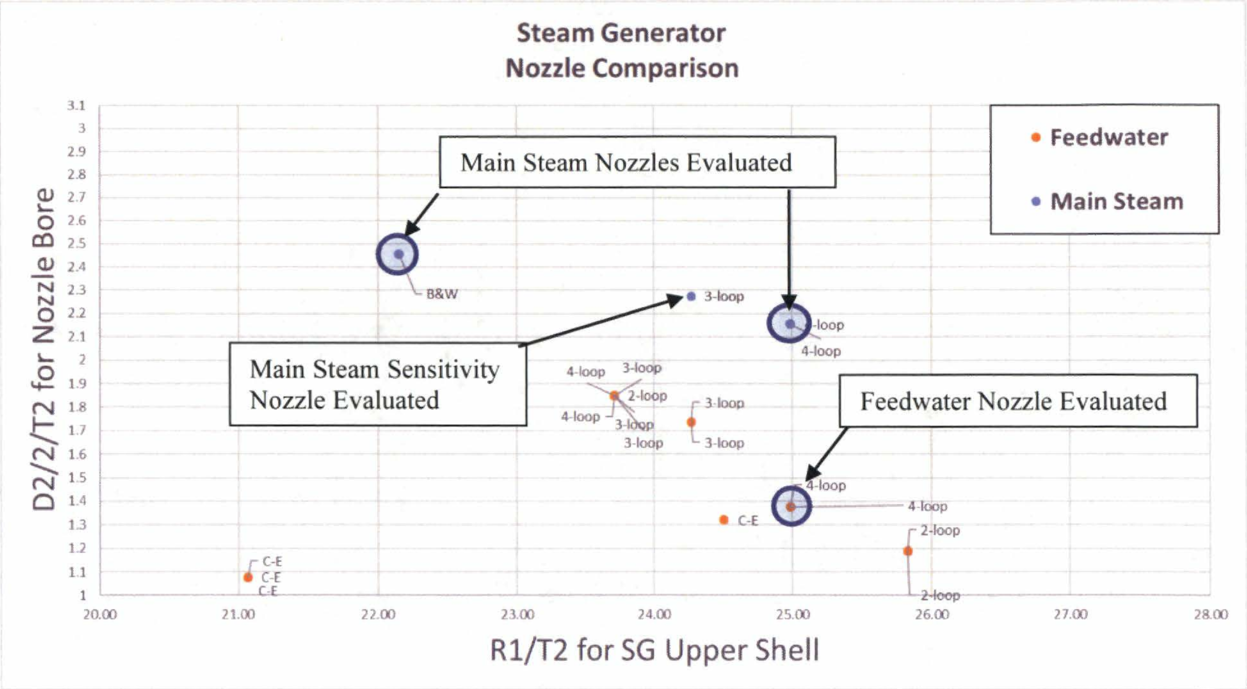


Figure 4-4
PWR Steam Generator Nozzle Comparison

4.4 Configurations with Impractical or Limited Exams

The survey responses discussed in Section 3 include cases in which components have been the subject of a relief request. These relief requests are related to either the impracticality of performing an ISI exam or the ability to obtain only limited coverage due to plant-specific component configuration. Several relief requests relate to a specific Westinghouse main steam nozzle design that is difficult to examine due to the presence of a sharp corner at the inside radius, which prevents the ISI examination being performed as intended. A sketch of this configuration was obtained from one such relief request for the South Texas Project [40] and is shown in Figure 4-5. References [15], [42], and [43] are additional relief requests addressing this same configuration. Because the relief requests indicate an inability to perform the ISI examinations as intended, it is unlikely that the nozzle inside radius section of this main steam nozzle configuration has received a full coverage (essentially 100% of the examination volume) examination since initial construction. The configuration shown in Figure 4-5 is the type of Westinghouse main steam nozzle selected for evaluation in Section 4.5.1. Because this nozzle has not received a complete in-service exam, the base case evaluated in the fracture mechanics evaluations in this report is treated as pre-service inspection (PSI) with no ISI. See Section 8 for more details on the base case evaluated.

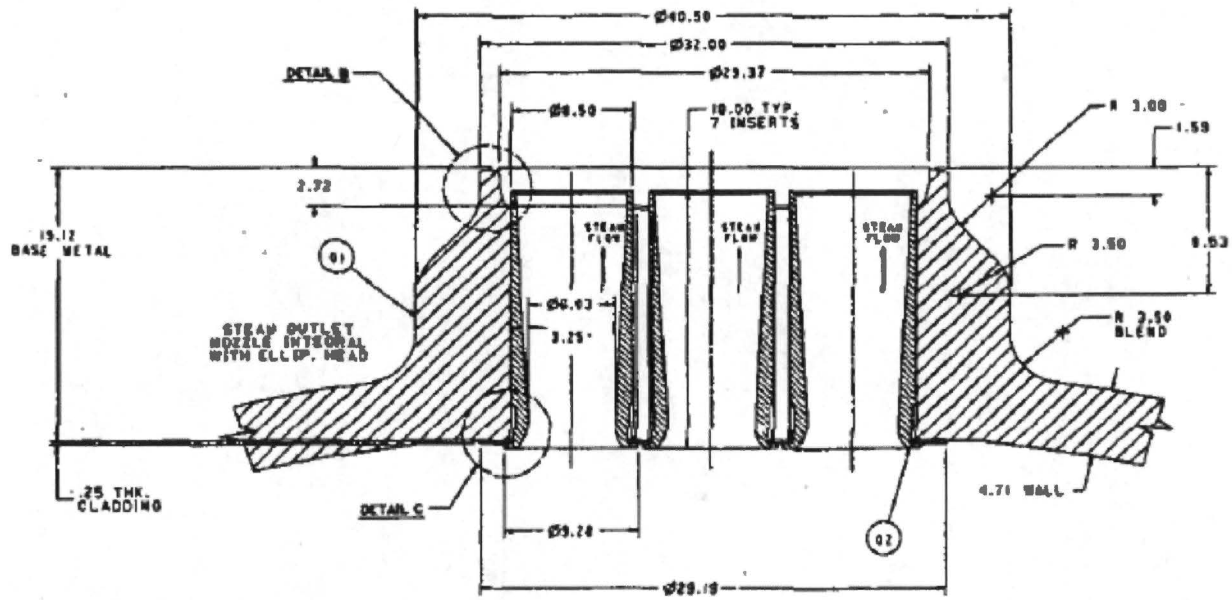
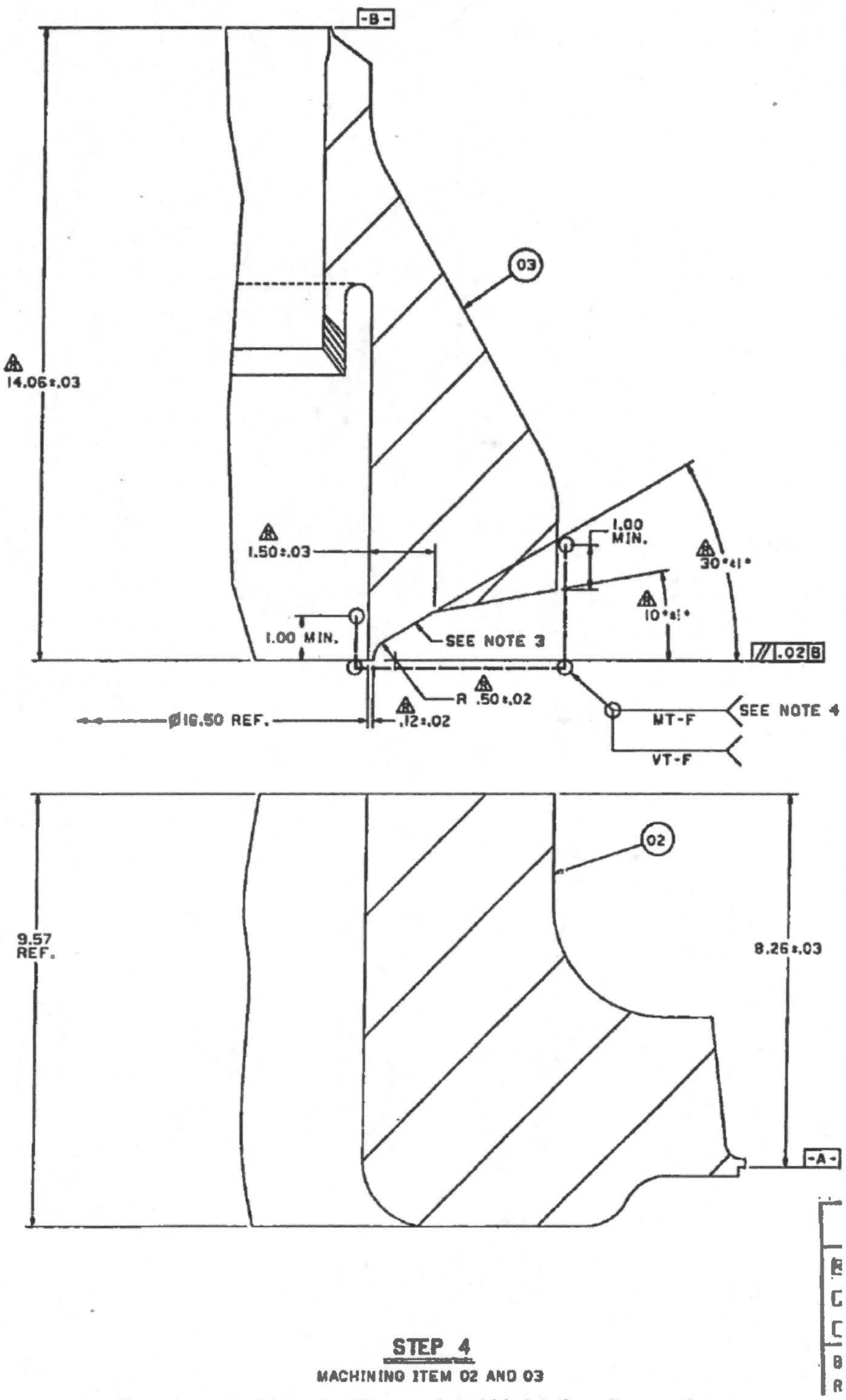


Figure 4-5
Westinghouse Main Steam Nozzle with Corner [40]

Another example of an impractical examination is the Point Beach feedwater nozzle design shown in Figure 4-6. Reference [44] documents a relief request for performing an alternative examination of this configuration, which is unique because the nozzle was cut off and a new weld added to fit the SG through the equipment hatch. The configuration shown in Figure 4-6 is not addressed in this report.



Feedwater Nozzle Extension Weld Configuration

Figure 4-6
Point Beach Additional Feedwater Nozzle Weld [44]

4.5 Selection of Components for Evaluation

The Section 4.3 review was performed to determine what type of variation could be expected between different PWR SG designs. The main conclusion was that, where variations exist, they are not considered significant and can be addressed by sensitivity analyses. Therefore, instead of determining (or defining) bounding components, representative components were selected based on their availability through the EPRI survey results, the factors discussed in Section 4.3, and a set of related criteria.

4.5.1 Main Steam Nozzle

A detailed review of nozzle configurations was performed for the various SG designs. CE and Westinghouse main steam nozzles each have two configurations. For CE plants, one configuration is similar to the Westinghouse design, with a main steam nozzle that exits the top dead center of the SG head, while the other design (CE System 80, shown in Figure 4-7) has two main steam nozzles that symmetrically exit the top of the SG. The System 80 design also has a portion of the nozzle that extends into the SG. Because of these unique aspects, the System 80 design is not addressed in this report.

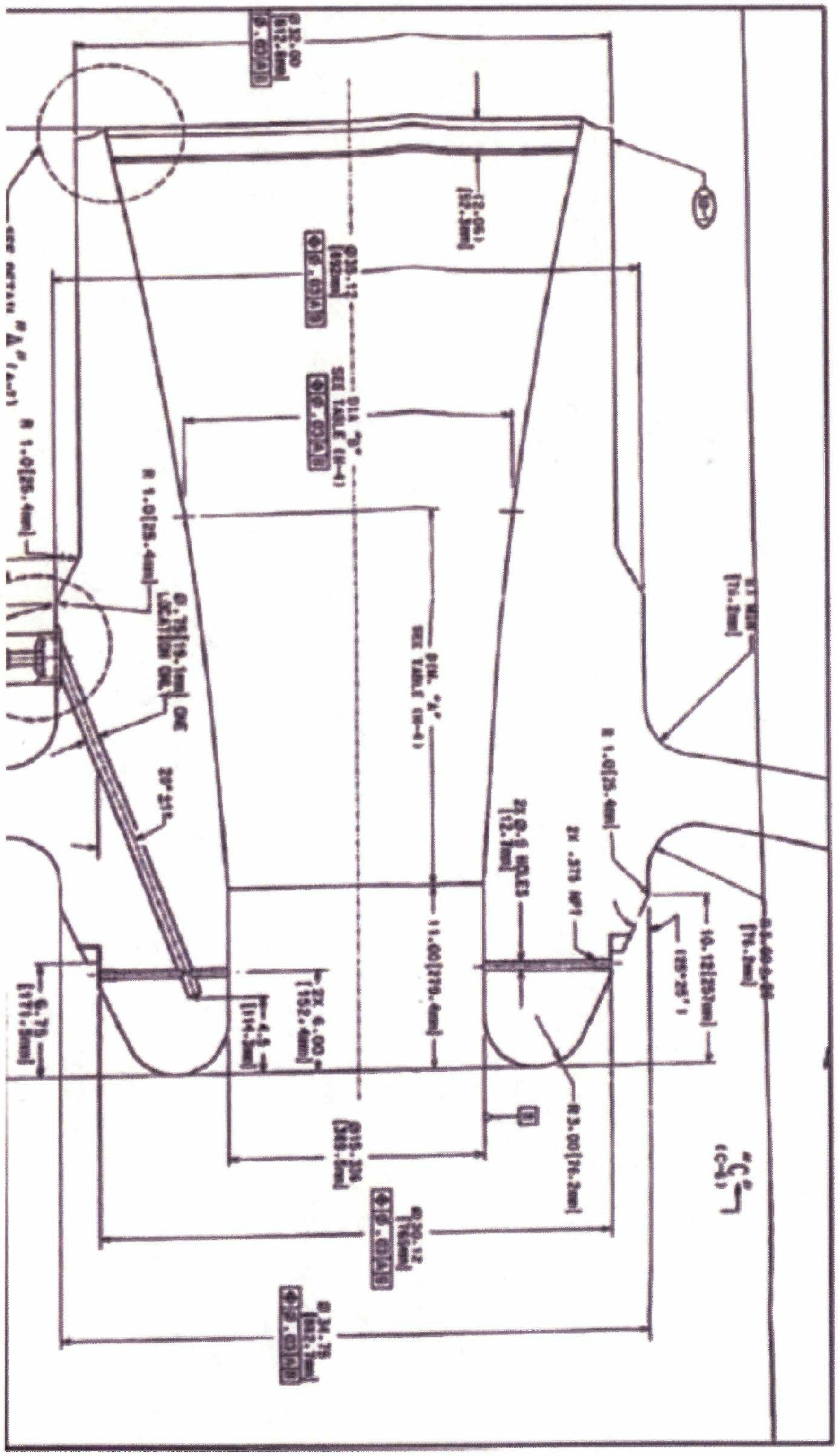


Figure 4-7
System 80 Main Steam Nozzle Configuration [45]

For the Westinghouse/CE design in which a single main steam nozzle exits the top of the SG head, the symmetric configuration creates a much more consistent stress state at the nozzle-to-shell/head and nozzle inside radius section locations. For this configuration, the nozzle design type shown in Figure 4-5 is selected and was assumed to cover both the Westinghouse and CE designs. Because this nozzle type has not received a complete in-service examination since initial construction, assessing the failure risk of the design is prudent. The dimensions for the main steam nozzle selected for analysis are contained in Figure 4-8.

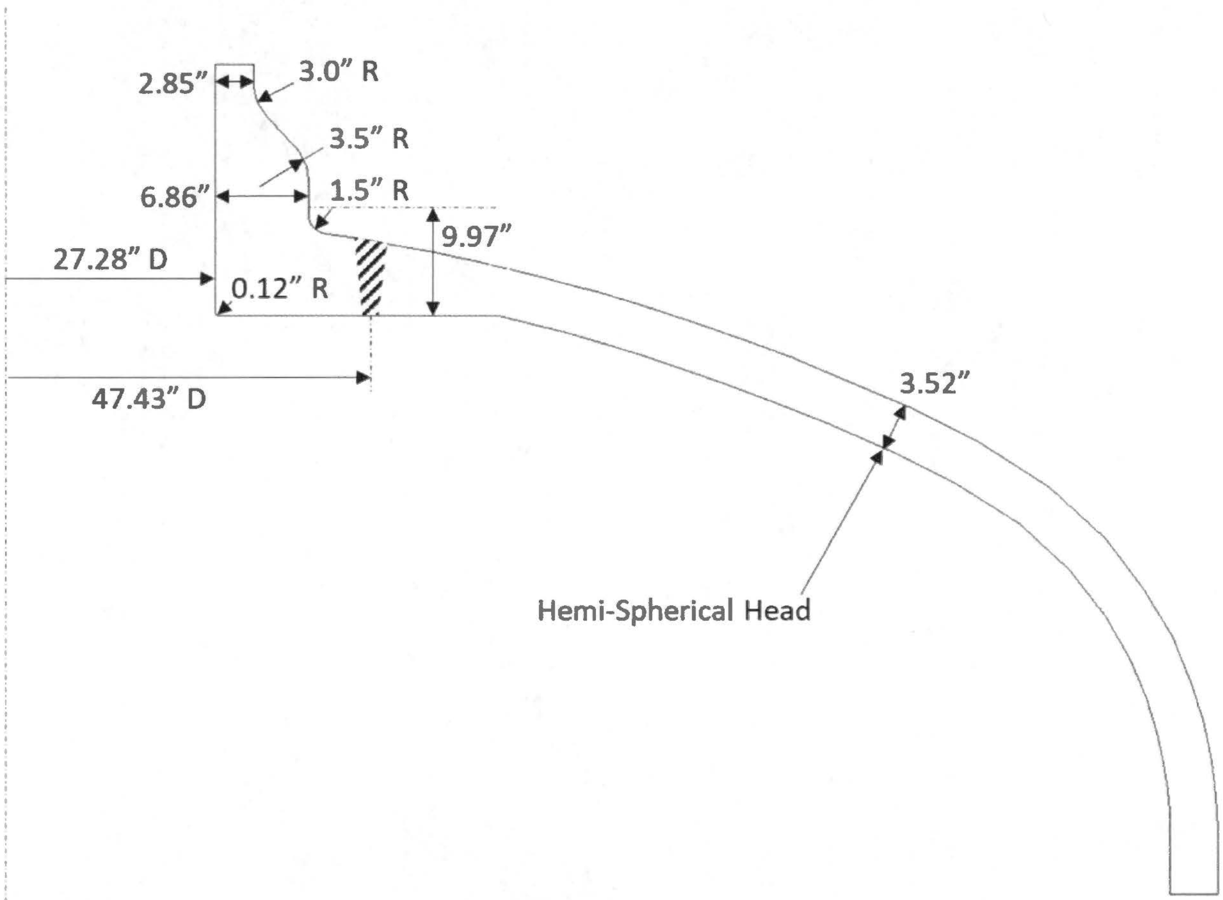


Figure 4-8
PWR Main Steam Nozzle Selected for Evaluation: Westinghouse 4-loop Design, Top of Steam Generator

As mentioned, Westinghouse plants have two main steam nozzle configurations. One is a traditional nozzle that is a constant reinforced cylinder that tapers to the pipe; the other is a tapered nozzle with a concave taper that resembles a triangle. Due to the differences between these two types of Westinghouse main steam nozzles, additional stress modeling was performed in Section 7 to evaluate how the stresses varied through the wall thicknesses compared to the nozzle selected for evaluation.

The B&W plant OTSG has a main steam nozzle that exits the side of the SG. This creates a varied stress level around the nozzle-to-shell weld and nozzle inside radius sections. Because the basic geometry between the B&W and the Westinghouse/CE design are sufficiently distinct, the B&W design is evaluated separately. The B&W main steam nozzles are essentially uniform across the B&W fleet. The dimensions for the B&W main steam nozzle selected for analysis are shown in Figure 4-9.

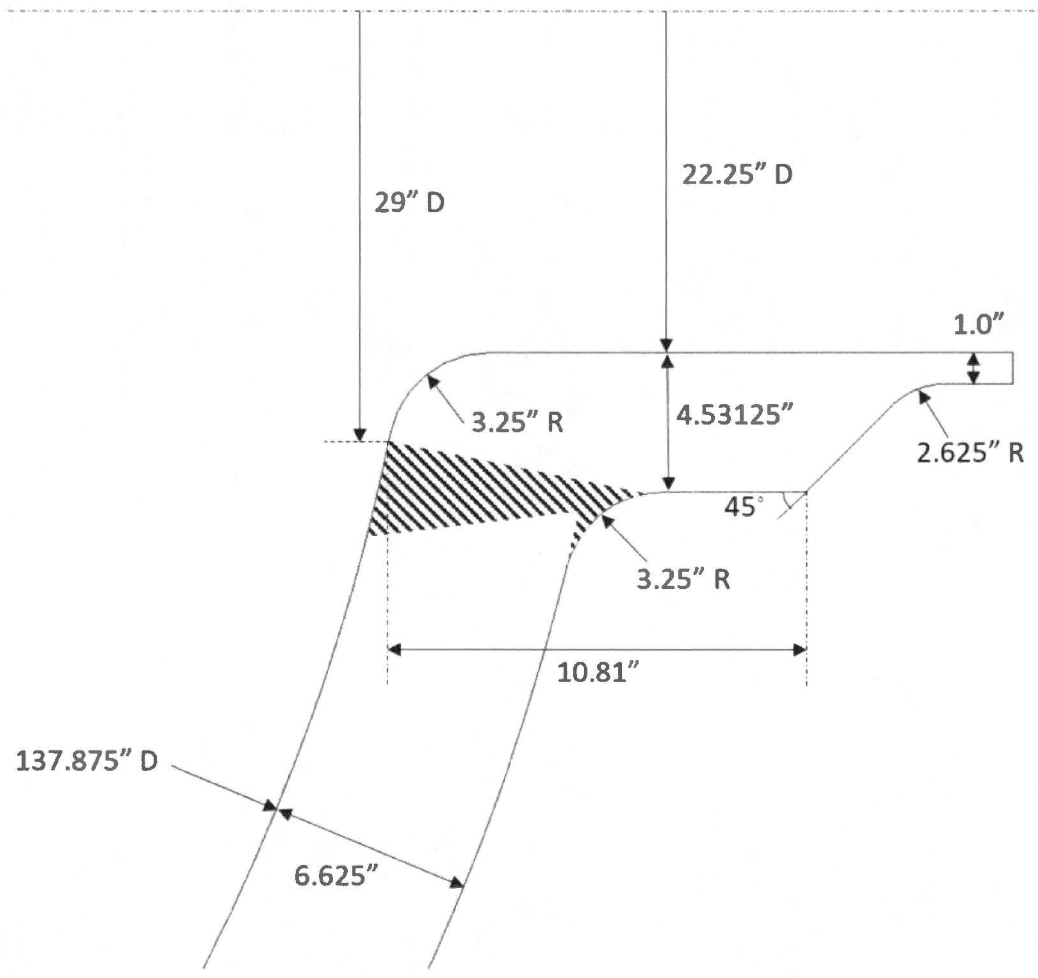


Figure 4-9
PWR Main Steam Nozzle Selected for Evaluation: B&W Design, Side of Steam Generator

Westinghouse configuration selected for analysis:

- R/t ratio for the vessel: 87.96/3.52 = 25
- R/t ratio for the nozzle: 14/6.5 = 2.2

B&W configuration selected for analysis:

- R/t ratio for the vessel (at the nozzle location): 75.56/6.63 = 11.4
- R/t ratio for the nozzle: 11.13/4.53 = 2.5

4.5.2 Feedwater Nozzle

Feedwater nozzles enter through the side of the SG in all plant designs. For B&W plants, the nozzle is a bolted joint, so no examinations of the feedwater nozzle-to-shell weld or nozzle inside radius section are required. For the Westinghouse and CE plant designs, based on a review of available designs, the variations are related only to the size of the nozzle/shell diameter and the thickness of the nozzle reinforcement. Most Westinghouse and CE plants contain the same basic configuration in which a thicker nozzle tapers down to the feedwater piping. The CE System 80 design also contains an auxiliary feedwater nozzle that enters the steam generator in a separate location from the regular feedwater nozzle. Because this feedwater nozzle is much smaller than the other designs, it is not addressed in this report. A feedwater nozzle for a Westinghouse 4-loop plant was selected for evaluation. The dimensions for the feedwater nozzle configuration selected for analysis are shown in Figure 4-10.

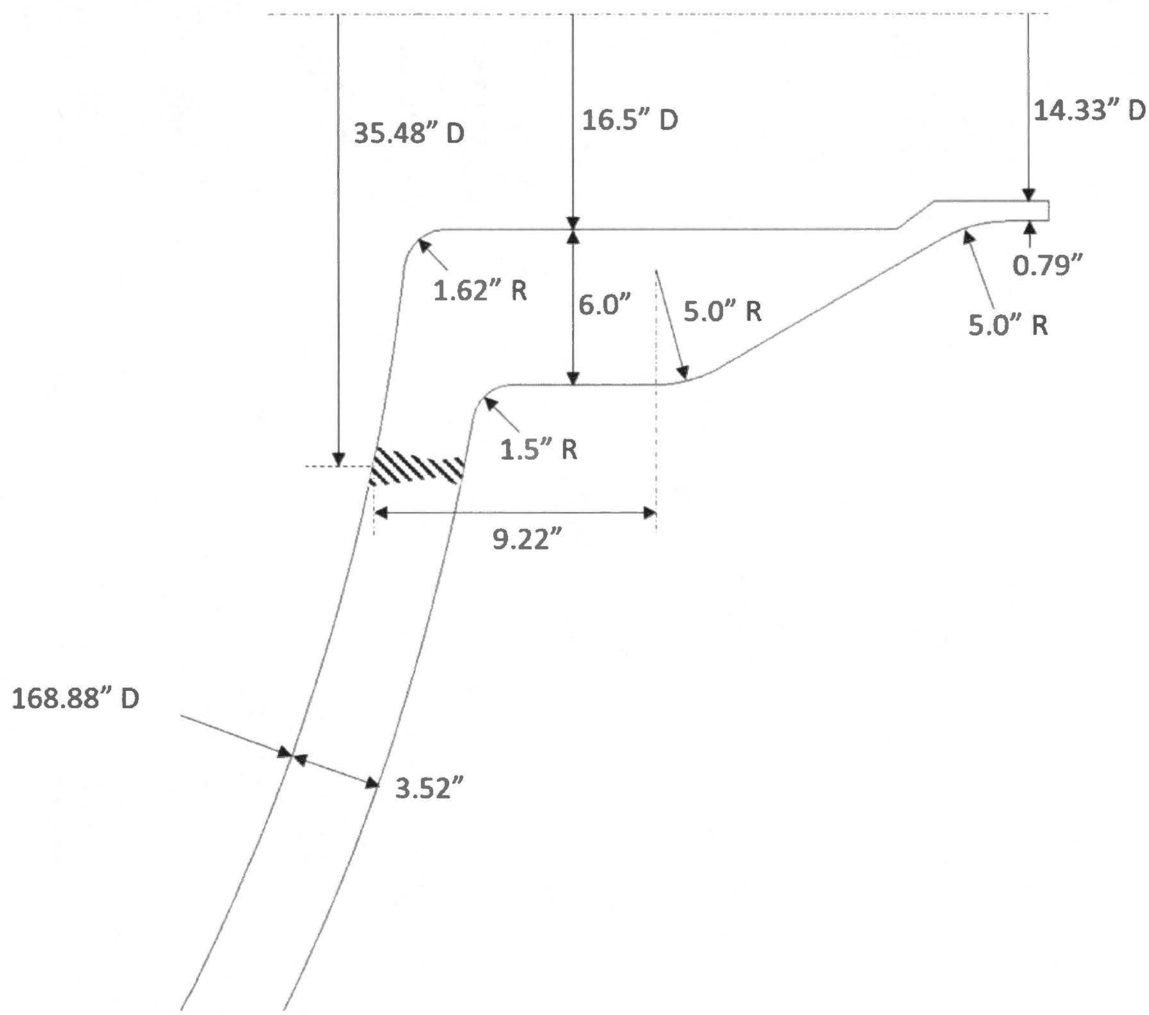


Figure 4-10
PWR Feedwater Nozzle Selected for Evaluation: Westinghouse 4-Loop Design

Westinghouse configuration selected for analysis:

- *R/t* ratio for the vessel: 87.96/3.52 = 25
- *R/t* ratio for the nozzle: 8.25/6 = 1.4

4.6 Conclusions

PWR SG designs and operating experience were reviewed. Information was also reviewed regarding variability among SG designs in terms of dimensions, design pressures and temperatures, and ASME Code design considerations. Geometrical variations among SG designs are shown in Table 4-1.

Table 4-1
Summary of SG Geometrical Parameters for Various Designs

Component	Parameter	Westinghouse	CE	B&W
SG Upper Shell	Diameter (in.)	166 to 177	240	150
	R/t Ratio	23 to 26	24.5	SG Nozzle: 11 Others: 24 5
MS Piping	Nominal Pipe Size (in)	28 to 32	34	24
	R/t Ratio	2 to 3	NA	2.5
FW Piping	Nominal Pipe Size (in.)	14 to 18	14 to 18	N/A
	R/t Ratio	1 to 2	1 to 2	N/A

The following observations are made:

- The most important parameter from an ASME Section III design viewpoint is the *R/t* ratio. Table 4-1 shows that variations in this parameter are not very significant (especially between the Westinghouse and CE plants) and can be addressed by sensitivity studies to be performed in Section 8. Therefore, instead of determining (or defining) bounding components, representative components were selected based on the plants that responded to the EPRI survey, the factors discussed in Section 4.3, and a set of related criteria. As shown in Figure 4-4, the two selected configurations for the main steam nozzles lie at the extremities of the *R/t* data, which means that they experience the highest pressure stresses. The feedwater nozzle values are clustered closer together, so in this case, a representative geometry was chosen. Any items not explicitly bounded by the selected geometries are addressed by sensitivity analysis in Section 8.

- To cover the PWR SG feedwater and main steam nozzle-to-shell weld and nozzle inside radius section components evaluated in this report, three representative configurations were selected:
  - One Westinghouse 4-loop main steam nozzle as shown in Figure 4-8
  - One B&W main steam nozzle as shown in Figure 4-9
  - One Westinghouse 4-loop feedwater nozzle as shown in Figure 4-10

  The selected components are representative for CE, Westinghouse, and B&W SG designs (with the exceptions noted above). Section 9 covers the parameters that need to be checked to determine whether a specific plant is covered by the evaluation performed in this report.

- Items not covered in this report are the auxiliary feedwater nozzles and the CE System 80 feedwater and main steam nozzles.

5 MATERIAL PROPERTIES, OPERATING LOADS, AND TRANSIENTS

5.1 Material Selection and Properties

This section discusses the selection of the materials and related properties that will be used in the stress analyses for the components selected in Section 4. Material properties important to the fracture mechanics evaluations, such as toughness and crack growth parameters, are addressed in later sections.

According to Section 4.5, three PWR SG nozzles (main steam nozzles for two different PWR designs and a feedwater nozzle for one PWR design) were selected for evaluation. As discussed in Section 4.3, available SG data were tabulated to evaluate the variation among SG designs. One item tabulated was the material of the steam generator shell, head, and nozzles. The tabulated materials show that many of the materials are the same. For all SG designs, the nozzles are fabricated from a forging material. This forging material specification is a low-alloy steel such as SA-508 Class 2. The head or shell will also be a low-alloy steel material, but the specification will vary based on the fabrication practices. If the material was forged, the material specification will be a low-alloy steel such as SA-508 Class 2. If the material was formed from a plate, it will be made from a low-alloy steel such as SA-533 Grade A.

The exact material specification used will be based on the fabrication practices used for the specific SG. For instance, an SG shell section can be forged from a solid ingot, or a plate can be rolled and then seam-welded. Similarly, the head can also be manufactured from a forging or use a formed plate(s). Regardless of the fabrication process used, the material specification is for a low-alloy steel material. The weld material used to join the nozzle to the head or shell would have been a compatible low-alloy steel filler metal. The actual weld specification used would be determined based on the welding process used. For instance, large components can be fabricated using rotating gimbles and therefore use submerged arc welding methods. If manual welding processes were employed, shielded metal arc welding (SMAW) material specifications would be used. The exact process and specification are not important for this evaluation. The weld material properties are assumed to be similar low-alloy steel to match the nozzle and head/shell.

Based on the tabulated data, it was decided that for the Westinghouse 4-loop design, SA-533 Grade A Class 2 low-alloy steel will be used for the SG nozzles, heads, and nozzle-to-head/nozzle-to-shell welds for both the main steam and feedwater nozzles. For the B&W design, some information indicates the use of low-alloy steel for all components, whereas other input notes that plain carbon steel may be included as part of the secondary shell. Because carbon steel may be present, SA-516 Grade 70 carbon steel will be used for the nozzle, shell, and nozzle-to-shell weld for the B&W main steam nozzle. Temperature-dependent material properties were obtained from the relevant tables in the 2013 Edition of ASME Code, Section II, Part D [46] and are listed in Tables 5-1 and 5-2. For the B&W design, a mix of the two materials may be present throughout the fleet. Although the SA-516 Grade 70 material was selected for analysis, that affects only the thermal transient stresses used in the fracture mechanics evaluation. As noted in the sensitivity analysis performed in Section 8, the thermal transients do not contribute significantly to crack growth or probability of failure. Therefore, the use of SA-516 Grade 70 covers the potential use of a different material in the B&W design to ensure that both low-alloy and carbon steel are addressed in the report.

Table 5-1
Material Properties for Low-Alloy Steel (SA-533 Grade A Class 2) [46]

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) (BTU/lb-°F)
70	29.0	7.0	5.49	0 107
100	28.9	7.1	5.46	0 108
150	28 7	7.2	5.44	0.111
200	28 5	7.3	5.44	0.115
250	28 3	7.3	5.42	0.117
300	28.0	7.4	5.42	0.121
350	27 8	7.5	5 39	0.123
400	27.6	7.6	5.35	0.126
450	27.3	7.6	5.32	0.129
500	27 0	7.7	5.25	0.131
550	26.7	7.8	5.21	0.134
600	26.3	7.8	5.14	0.137
650	25 8	7.9	5.07	0 139
700	25.3	7.9	5.00	0.142

Notes:

1. Density (ρ) = 0.280 lb/in^3, assumed temperature independent.
2. Poisson's Ratio (ν) = 0.3, assumed temperature independent.

Table 5-2
Material Properties for Carbon Steel (SA-516 Grade 70) [46]

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) (BTU/lb-°F)
70	29.2	6.4	8.08	0.103
100	29.1	6.5	8.03	0.106
150	28.8	6.6	7.92	0.110
200	28.6	6.7	7.80	0.114
250	28.4	6.8	7.64	0.117
300	28.1	6.9	7.48	0.119
350	27.9	7.0	7.31	0.122
400	27.7	7.1	7.15	0.124
450	27.4	7.2	6.97	0.126
500	27.1	7.3	6.81	0.128
550	26.8	7.3	6.64	0.131
600	26.4	7.4	6.48	0.134
650	25.9	7.5	6.32	0.136
700	25.3	7.6	6.16	0.140

Notes:

1. Density (ρ) = 0.280 lb/in^3, assumed temperature independent.
2. Poisson's Ratio (ν) = 0.3, assumed temperature independent.

5.2 Operating Loads and Transients

Operating loads and transients applicable to the PWR SG feedwater and main steam nozzles are discussed in this section. The loads considered for these components are those due to thermal and pressure transients. As with other RCS components, the SG was designed to the ASME Code and considered all service levels: design, normal (A), upset (B), emergency (C), faulted (D), and test conditions. ASME Code Section XI requires that the component be able to withstand the conditions during operation. For the fracture mechanics evaluation of failure (where $K > K_{IC}$), the maximum load on the component from normal, upset, emergency, and faulted conditions needs to be considered. Because the plant is not expected to operate under emergency or faulted conditions, crack growth will be considered only for normal and upset events. Test conditions beyond a system leak test are not considered. The original hydrostatic test was performed during fabrication. IWA-4540 requires a hydrostatic or system leakage test in accordance with IWA-5000; IWA-5212 points to IWC-5000 for hydrostatic and leakage tests of Class 2 components. IWC-5220 requires the leakage test to be performed at the system pressure during normal service. Because any pressure tests will be performed at the operating pressure, no separate test conditions need to be included in the evaluation.

Attached piping loads are not considered in the evaluation. The piping system was designed so that the piping loads did not exceed criteria in ASME Code Section III, NB-3652. The combined pressure plus bending stress is required to be less than 1.5 times the design stress intensity ($1.5S_m$). Because the nozzle section is much thicker than the piping, any stresses due to bending are much smaller in the nozzle compared to those calculated in the piping attached to the nozzle. Therefore, the stresses due to loads from the attached piping were not included in the analysis—they are overwhelmed by the pressure and thermal stresses at the nozzle blend radius and nozzle-to-shell weld. Using Equation 9 of ASME Section III, NB-3652, if the stress in the piping was assumed to be equal to $1.0S_m$, the portion of the piping stress from pressure could be a maximum of $0.5S_m$. For the feedwater and main steam nozzles evaluated in this report, the ratio of the section modulus of the piping to that of the nozzle ranges from 8% to 18%. This translates to a stress value of 4% to 9% of S_m. Whereas these stress comparisons are based on the section thickness of the piping and nozzle, the nozzle inside radius section is affected by the nozzle connection to the SG shell/head—reducing the stress further because of the increased stiffness from the SG shell/head. The resulting stress is small, so it was not explicitly included in the evaluation. Rather, this relatively small contribution to the total stress is addressed by sensitivity studies on stress described in Section 8.

Thermal and pressure transients for each of the three NSSS vendors were considered in the evaluation of the main steam and feedwater nozzles. The thermal and pressure transients were developed by using plant data and information from system descriptions, process and instrumentation diagrams (P&IDs), other plant documents, and relevant industry literature. The SG data tabulated to evaluate SG design variability (as discussed in Section 4.3) included the available operating temperatures and pressures of the SG secondary side and the feedwater system. As expected, there are some variations in the operating characteristics of the various SG designs. The SG steam normal power operating temperature ranges from approximately 510°F to 570°F, with an average value of approximately 530°F. The normal operating pressure has a larger variation, from 735 psi to 1,042 psi, with an average of approximately 870 psi. Normal power feedwater temperatures have a narrower range, from 430°F to 470°F, with an average value of approximately 445°F. Such variations are expected, and the transients defined for evaluation were modified to ensure that they are bounding for all investigated plant design types. Modifications typically include increasing temperature ramp rates, increasing the magnitude of temperature and pressure changes, and/or increasing the number of design transient cycles.

During normal PWR SG operation, the primary coolant enters the inlet nozzle (primary side), flows through the SG tubes, and leaves through the outlet nozzles (primary side). Feedwater enters the SG through the feedwater nozzle (secondary side) where it is distributed via the feedwater distribution ring and mixes with the recirculation flow. The fixed recirculation flow descends through the annular downcomer, which is an annular passage formed by the inner surface of the SG shell and the cylindrical shell wrapper. At the bottom of the downcomer, the secondary water is directed upward past the vertical tubes where heat transfer from the primary side produces a water-steam mixture. After the water-steam mixture passes through separators and dryers, a dry steam continues through the main steam nozzle (secondary side). As noted in Section 4, the U-tube steam generators for the Westinghouse and CE plants are different from the OTSG design for the B&W plants. One difference is that the OTSG design allows for superheating of the steam in the SG. However, as noted in the discussion of the operating temperatures and pressures, the B&W plants still operate in nominal ranges when compared to the PWR fleet. No B&W-specific modifications to transients are needed, and the modified transients (discussed above) will provide margin to bound the entire PWR fleet.

The main steam nozzle is affected by transients that occur in the steam space. Because the main steam nozzle is an outlet nozzle that directs steam to the turbine, it experiences transients only from the SG steam region. This is true for both the nozzle inside radius section and nozzle-to-shell weld. The feedwater nozzle is affected by the transients that occur below the SG steam space. The feedwater nozzle-to-shell weld is subjected to transients only from the SG region below the steam space. The feedwater nozzle inside radius section also experiences some transients caused by the incoming feedwater flow. As discussed in Section 4, the feedwater piping has been subject to prior thermal fatigue failures. Many RSGs were designed with modifications, including an integral thermal sleeve with the nozzle safe end, which prevents bypass feedwater flow from impinging on the feedwater nozzle body. The thermal sleeve therefore protects the feedwater nozzle locations of interest (nozzle-to-vessel weld and nozzle inside radius section) from the effects of thermal stratification and high-cycle thermal striping (see Section 6 for more detail). Because of the presence of the thermal sleeve, fatigue due to incoming feedwater thermal transients is not expected for the nozzle-to-shell weld location. However, the stress state of the nozzle inside radius section will be affected by incoming feedwater thermal transients. Therefore, a severe thermal event will be considered for both locations to evaluate the impact of the higher stresses over the life of the nozzle. Non-integral thermal sleeve designs are not addressed in this report.

FSARs often contain a summary of reactor coolant system design transients. As an example, Table 5-3 shows a summary of the normal, upset, emergency, and faulted transients and associated design cycles for Farley [47]. (Note: Design cycles are not necessarily indicative of how a plant actually operates.) EPRI previously performed a compilation of PWR fleet transients in MRP-393 [48]. Table 5-3 contains a comparison of the design cycles from the Farley FSAR and the MRP-393 projected cycles based on data collected from transient monitoring systems.

Table 5-3
Summary of Reactor Coolant System Design Transients for Farley [47] Compared to MRP-393 [48] Cycles

Transient	Classification	Farley 40-Year Design Cycles	MRP-393 60-Year Projections
Heat-Up	Normal	200	200
Cooldown	Normal	200	200
Plant Loading	Normal	18,300	<1,000
Plant Unloading	Normal	18,300	<1,000
Step Load Increase	Normal	2,000	Not Typical
Step Load Decrease	Normal	2,000	Not Typical
Large Step Load Decrease	Normal	200	20
Loss of Load	Upset	80	Not Typical
Loss of Power	Upset	40	Not Typical
Loss of Flow	Upset	80	Not Typical
Reactor Trip	Upset	400	~200
Inadvertent Auxiliary Spray	Upset	10	Not Typical
Pipe Break	Faulted	1	N/A

For 60 years of plant life, PWRs are not expected to have more than 200 heat-up and cooldown cycles. Because many plants do not load-follow, many events related to loading and unloading do not occur at the frequency stipulated in the FSAR. Rather than approximately 20,000 cycles over the plant lifetime, these loading and unloading cycles are projected to be less than 1,000 for 60 years of operation. Other events—such as loss of load, loss or power, and loss of flow—are rare and have occurred infrequently (if at all) at most operating plants. These transients are noted as "not typical" for their frequency. This means that their number is very small and representative transients for these conditions were not captured by fatigue monitoring systems. As such, these transients would have a negligible impact on the fatigue crack growth. Other events, such as reactor trips, have occurred and are projected to occur at about half of the design limit, or 200 cycles per 60 years of operation. Because many design events have not occurred, bounding cycle limits are selected based on the projected number of cycles for PWRs.

Because the SGs are made of ferritic steels, the failure mode will be brittle fracture ($K > K_{IC}$). As discussed in Section 6, thermal fatigue is a degradation mechanism. Therefore, for the fracture mechanics evaluation, transients (pressure and thermal) that will significantly contribute to fatigue crack growth were considered. In addition, the maximum stress state due to all applied loads needs to be evaluated for failure. For fatigue crack growth, any cycle will contribute to fatigue crack growth, but small changes in temperature and pressure may have an insignificant effect on the growth because the change in stress intensity factor is small. On the other hand, heat-up and cooldown have large temperature and pressure changes because they cycle from ambient to operating conditions. Therefore, the heat-up and cooldown transients will have the largest contribution to crack growth.

Different plant designs have different transient characteristics. Table 5-4 contains a sample compilation of design transients and their expected temperature and pressure variations. This table includes temperatures in the steam space (applicable to the main steam nozzle), downcomer region (near the feedwater nozzle), and feedwater temperatures (feedwater nozzle). Table 5-4 is not an exhaustive list for each plant type but assumes similarities between plant designs to ensure that a set of transients is developed that covers the various plant designs. As mentioned previously, the transients with the largest pressure and temperature variations will lead to the most significant changes in predicted crack growth.

Table 5-4
Compilation of Design Transients and Expected Temperature and Pressure Variations

Transients[1]	ΔT (°F)			ΔP (psi)[2]
	Steam Space	Secondary Side (Downcomer)	Feedwater	
Westinghouse Type				
Heat-Up	**477**	**477**		**1020**
Cooldown	**477**	**477**		**1020**
Plant Loading	**28**	**48**	**370**	**223**
Plant Unloading	**28**	**46**	**370**	**223**
Large Step Load Decrease	**10**	**18**	**310**	**150**
Loss of Load	**49**	**93**	**403**	**342**
Loss of Load from Coastline	90	93	403	475
Loss of Power	47			342
Combustion Engineering Type				
Plant Heat-Up	**N/A**	**462**		**885**
Plant Cooldown	**N/A**	**462**		**885**
Plant Loading	**N/A**	**3**	**170**	**20**
Plant Unloading	**N/A**	**3**	**170**	**20**
10% Step Load Increase	N/A	1	16	8
10% Step Load Decrease	N/A	1	16	8
Reactor Trip Loss of Load	**N/A**	**10**	**366**	**76**
Loss of Primary Flow	N/A	10	366	31
Loss of Load	N/A	16	366	126

Table 5-4 (continued)
Compilation of Design Transients and Expected Temperature and Pressure Variations

Transients[1]	ΔT (°F)			ΔP (psi)[2]
	Steam Space	Secondary Side (Downcomer)	Feedwater	
B&W Type				
Heat-Up	**N/A**	**500**	**0**	**900**
Cooldown	**N/A**	**520**	**0**	**1050**
Plant Loading from 8% to 100 FP	**N/A**	**32**	**200**	**25**
Plant Unloading 100% to 8% FP	**N/A**	**34**	**200**	**40**
Step Load Reduction	N/A	50	315	230
Reactor Trip	**N/A**	**60**	**0**	**150**
Loss of Feedwater Reactor Trip	N/A	65	418	200
Change of Flow	N/A	5	25	30
Rod Withdrawal Accident	N/A	60	100	120
Drop of One Control Rod	N/A	60	200	100
Loss of Station Power	**N/A**	**70**	**400**	

Notes:

1. Transients selected for analysis are **highlighted**.

2. The pressure in the SG is assumed to be constant. Therefore, only one pressure value is given.

Heat-up and cooldown will be included for the evaluation. This will include a full pressure range and temperature range; the specific event is described later in this section. Plant loading and unloading contains some temperature variation and a pressure change. The values chosen for the analysis are discussed later under the loading and unloading transient.

The remaining transients shown in Table 5-3 that typically occur include large step load decrease and reactor trip. Table 5-4 shows that these events do not include any significant temperature variations in the SG steam space or secondary side, but the feedwater nozzle does see larger temperature variations. To account for the reactor trip and large step load decrease transients, a loss of load event is selected. The loss of load event has similar temperature and pressure changes to the reactor trip transient and bounds the large step load decrease temperature and pressure changes. Discussion on the event modeled is discussed under the loss of load transient next. Additional cycles of the loss of load event were considered to address other transients not explicitly modeled.

The transients applicable to the selected PWR SG feedwater and main steam nozzles are described as follows:

- **Heat-Up and Cooldown [Normal].** Heat-up occurs from cold shutdown to rated temperature and pressure conditions. Cooldown occurs from the rated temperature and pressure condition to cold shutdown. Typical rated temperature and pressure conditions for the SG secondary side are 550°F and 1,000 psig. Based on plant technical specification limits, most heat-up and cooldown events are restricted to a ramp rate of 100°F per hour or less; however, to bound the variation expected among plant design types, an assumed bounding rate of 200°F per hour is used. Therefore, heat-up begins at an ambient temperature of 70°F, and the temperature increases to 550°F at a rate of 200°F per hour while the pressure increases following saturated conditions from 0 psig to 1,000 psig. Similarly, cooldown begins at 550°F, and the temperature decreases to 70°F at a rate of 200°F per hour while the pressure decreases following saturated conditions from 1,000 psig to 0 psig. Typical flow rates for normal operating conditions are 3.73×10^6 lb/hr for a Westinghouse-type main steam nozzle, 1.35×10^6 lb/hr for a B&W-type main steam nozzle, and 3.76×10^6 lb/hr for a Westinghouse-type feedwater nozzle. These flow rates are common for all transients discussed in this section. Typical design cycles for heat-up and cooldowns are 200 cycles over 40 years, which most PWR plants showed remained adequate for 60 years of operation. This evaluation conservatively considers 300 heat-up and cooldown cycles over a 60-year period (or five cycles per year).

- **Plant Loading and Unloading [Normal].** The discussion for these transients is applicable to the loading transient. The unloading transient is a reverse of the loading transient. This transient initiates at 0% power and increases to 100% power at a rate of 5% per minute. Typical 0% power temperature and pressure are 500°F and 1,000 psig, respectively, for the steam; 550°F and 1,000 psig, respectively, for the downcomer flow near feedwater nozzle; and 70°F and 1,000 psig, respectively, for the incoming feedwater. Typical 100% power temperature and pressure are 520°F and 800 psig, respectively, for the steam; 500°F and 1,000 psig, respectively, for the downcomer flow near feedwater nozzle; and 430°F and 800 psig, respectively, for the incoming feedwater. Because the colder feedwater of 430°F at 100% power enters into the SG secondary side, the bulk temperature and pressure inside the SG decrease. Plants are typically designed to accommodate 18,300 cycles of this transient over 40 years. Most plants do not approach 1,000 cycles for this event over 60 years of operation. For a conservative estimate on the number of cycles, 5,000 cycles are considered over a 60-year period (or 84 cycles per year).

- **Loss of Load [Upset].** This transient initiates at 100% power and causes the temperature and pressure to increase in the primary and secondary systems. It is assumed that the steam temperature increases by 65°F (starting at 520°F) and pressure increases by 300 psi (starting at 800 psig) in 10 seconds, while the incoming 430°F feedwater is stopped after this 10-second period. Then, the temperature for the downcomer flow near the feedwater nozzle is increased by 85°F (starting at 500°F). This is followed by 32°F feedwater injection at 70 seconds, and the temperatures for the steam and downcomer flow decrease to 550°F. The nominal operating conditions for this event vary from the nominal conditions used for the heat-up and cooldown transients. The values used for the heat-up and cooldown events are larger and represent rated pressure and temperature, whereas the values used for this event represent lower values representative of operating transients.

Typical design cycles for the loss of load event are 100 cycles over 40 years, which most PWR plants showed remained adequate for 60 years of operation. This evaluation conservatively considers 360 loss of load cycles over a 60-year period (or six cycles per year). This larger number of cycles is expected to represent the other events that are not specifically modeled (reactor trip, large step load decrease, loss of power, and so on).

An additional transient for feedwater nozzle–specific events is defined as follows:

- **Loss of Power [Upset].** This transient initiates at 100% power and causes the initiation of cold (minimum of 32°F) auxiliary feedwater flow into the hot (560°F) SG via the feedwater nozzle. This is assumed to occur once per year for 60 years of operation. This transient is applied only in the feedwater nozzle stress analysis and is intended to bound any feedwater nozzle–specific transients.

Design transients for faulted conditions are also defined for the SG secondary side. During these events, the safety injection system is postulated to activate, which will lead to depressurization of the SG primary side and an associated decrease in temperature in the RCS. It is also assumed to cause a decrease of the temperature and pressure in the SG secondary side. Two of the more significant design transients for faulted conditions in the SG secondary side are a steam line break and a feedwater line break. For a steam line break, the temperature in the SG secondary side is postulated to increase instantaneously after the rupture; however, the magnitude of the increase is assumed to be small. For a feedwater line break, the temperature in the SG secondary side is postulated to increase in the active feedwater loops, while the inactive feedwater loop temperature is assumed to decrease after the rupture; however, the magnitude of the temperature decrease is postulated to be small. Because these types of events are projected to occur only once in the life of a plant, a postulated flaw only needs to remain stable during these events based on the stresses due to pressure and thermal conditions. Because a break in a line would be present, the system will drop in pressure during these events. The pressure drop will unload the system and reduce the stresses in affected components. The normal and upset conditions bound the faulted transient condition; therefore, the faulted condition was not evaluated separately.

Based on this, Table 5-5 contains the final list of events that are evaluated for the feedwater and main steam nozzles. As discussed, there are variations in the SG designs and the typical operating pressures of each design. Because typical operating pressures are used in this evaluation, increased stresses are evaluated in the sensitivity evaluations to account for these variations. See Section 8 for a discussion of these additional evaluations.

Table 5-5
Thermal Transients Applicable to PWR SG Feedwater and Main Steam Nozzles (1, 2)

Transient	Max. T_{ST}, °F (3)	Min. T_{ST}, °F (3)	Max. T_{DC}, °F (4)	Min. T_{DC}, °F (4)	Max. T_{FW}, °F (5)	Min. T_{FW}, °F (5)	Max. Press., psig	Min. Press., psig	60-Year Cycles
Heat-Up and Cooldown	550	70	550	70	N/A	N/A	1,000	0	300
Plant Loading	550	520	550	500	430	70	1,000	800	5,000
Plant Unloading	550	520	550	500	430	70	1,000	800	5,000
Loss of Load	585	520	585	500	430	32	1,100	800	360
Loss of Power (6)	N/A	N/A	560	560	32	32	1,120	1,120	60

Notes:

1. The table is applicable to both main steam and feedwater nozzles.
2. The typical flow rates for normal operating conditions were used for all transients and are 3.73×10^6 lb/hr for the Westinghouse-type main steam nozzle, 1.35×10^6 lb/hr for the B&W-type main steam nozzle, and 3.76×10^6 lb/hr for the Westinghouse-type feedwater nozzle.
3. T_{ST} is the fluid temperature for steam, which was used in main steam nozzle stress analysis.
4. T_{DC} is the fluid temperature in the downcomer region.
5. T_{FW} is the feedwater temperature, which was used in the feedwater nozzle stress analysis.
6. Loss of power affects only the feedwater nozzle, so it was applied only in the feedwater nozzle stress analysis.

6 EVALUATION OF POTENTIAL DEGRADATION MECHANISMS

This section evaluates the potential degradation mechanisms for the components selected in Section 4. According to Section 4.5, a total of three PWR SG nozzles (main steam nozzles for two different PWR designs and a feedwater nozzle for one PWR design) were selected for evaluation. The materials, operating loads, and transients (including pressures and temperatures) applicable to these nozzles are discussed in Section 5. All nozzles experience a constant, high flow of fluid (water or steam) during normal operations. The fluid is chemistry-controlled to limit the concentration of dissolved oxygen and initiating contaminants (for example, chloride, fluoride, and sulfate).

6.1 List of Mechanisms Evaluated

Potential degradation mechanisms affecting nuclear power plant components are discussed in References [21] and [49]. A list of the mechanisms relevant to the selected components is as follows:

- Environmentally-assisted cracking
  - Intergranular stress corrosion cracking (IGSCC)
  - Transgranular stress corrosion cracking (TGSCC)
  - External chloride stress corrosion cracking (ECSCC)
  - Primary water stress corrosion cracking (PWSCC)
  - Corrosion fatigue
- Localized corrosion
  - Microbiologically influenced corrosion (MIC)
  - Pitting
  - Crevice corrosion
- Flow sensitive
  - Erosion-cavitation
  - Erosion (that is, abrasive wear)
  - Flow-accelerated corrosion (FAC)

- General corrosion
  - Corrosion/wastage
  - Galvanic corrosion
- Fatigue
  - Thermal stratification, cycling, and striping (TASCS)
  - Thermal transients
  - Mechanical fatigue (that is, vibration)

In the following subsections, the selected components are evaluated for potential susceptibility to each of these degradation mechanisms.

6.2 Degradation Mechanism Evaluation

6.2.1 Intergranular Stress Corrosion Cracking (IGSCC)

IGSCC results from a combination of sensitized stainless steel materials (caused by a depletion of chromium in regions adjacent to the grain boundaries in weld heat-affected zones [HAZs]), high stress caused by applied loads or welding residual stress, and a corrosive environment (high level of oxygen or other contaminants). For PWRs, welds and HAZs in wrought austenitic steel piping exposed to high dissolved oxygen levels and stagnant flow (for example, stagnant, oxygenated borated water systems) are susceptible to IGSCC.

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to IGSCC because they are fabricated from carbon steel or low-alloy steel.

6.2.2 Transgranular Stress Corrosion Cracking (TGSCC)

TGSCC is stress corrosion cracking that occurs through the grains of the material and usually occurs in the presence of halogens and sulfides. It is not necessarily associated with a particular metallurgical condition, such as grain boundary sensitization, but is affected by high local residual stresses—such as those caused by welding or local cold work. In PWRs, austenitic stainless steels are generally susceptible to TGSCC in the presence of chlorides and oxygen.

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to TGSCC because they are fabricated from carbon steel or low-alloy steel.

6.2.3 External Chloride Stress Corrosion Cracking (ECSCC)

ECSCC is the electrochemical reaction caused by a corrosive medium upon the external surfaces of a piping system. Austenitic steel piping and welds are considered susceptible to ECSCC when exposed to chloride contamination (from insulation, brackish water, or concentration of fluids containing chlorides), temperatures greater than 150°F, and tensile stress.

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to ECSCC since they are fabricated from carbon steel or low alloy steel.

6.2.4 Primary Water Stress Corrosion Cracking (PWSCC)

PWSCC occurs in PWRs when high-temperature primary water is present in combination with a susceptible material and high tensile stress. Component susceptibility is established under the plant's existing Alloy 600 program.

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to PWSCC because they are not fabricated using Alloy 82/182/600 materials.

6.2.5 Corrosion Fatigue

Corrosion fatigue (also referred to as *environmental assisted fatigue* [EAF]), is the reduction in the fatigue life of a component due to the synergistic combination of mechanical fatigue and corrosion in a corrosive environment. The reactor water environment is sufficiently corrosive to promote corrosion fatigue, depending on the nature of the fluid chemistry control. The presence of contaminants, such as sulfur/sulfates or chlorides, in combination with cyclic loading, is required for this mechanism to be active.

Even though all components are exposed to chemistry-controlled fluid, which limits the presence of contaminants such as sulfur/sulfates and chlorides, PWR SG feedwater nozzle components may still be susceptible to corrosion fatigue (components in a steam environment, such as the main steam nozzles, are not affected). Corrosion fatigue will be considered where applicable when performing the fracture mechanics analyses in Section 8. It will be addressed through the use of the ASME Code Section XI water fatigue crack growth law.

6.2.6 Microbiologically Influenced Corrosion (MIC)

Microbes, primarily bacteria, may cause widespread damage to low-alloy and carbon steels, stainless steels, and other alloys. Areas considered susceptible to degradation from MIC are piping components with fluids containing organic material or with organic material deposits. The most vulnerable components are raw water systems, storage tanks, and transport systems. Systems with low-to-intermittent flow conditions, temperatures less than 150°F, and pH below 10 are primary candidates.

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to MIC due to the elevated operating temperatures, constant high flow rates, and chemistry-controlled fluids.

6.2.7 Pitting

Pitting corrosion is a form of localized attack on exposed surfaces, with much greater corrosion rates at some locations than at others. High local concentrations of impurity ions, such as chlorides or sulfates, tend to concentrate in oxygen-depleted pits—giving rise to a potentially concentrated aggressive solution in this zone. All structural materials are potentially susceptible to pitting. Pitting can occur in low-flow or stagnant regions in components or within crevices. Susceptibility to pitting is a strong function of the material, oxygen level, and chloride level concentration.

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to pitting due to the constant high flow and chemistry-controlled fluids in these components, which limits the presence of oxygen, oxidizing species, and initiating contaminants.

6.2.8 Crevice Corrosion

Crevice corrosion is the electrochemical process caused by differences in anodic and cathodic reactions that are produced by geometric crevices in an oxygenated medium within a piping system. Regions containing crevices (narrow gaps)—such as those caused by the presence of thermal sleeves—that can result in oxygen depletion and, subsequently, a relatively high concentration of chloride ions or other impurities are considered susceptible to crevice corrosion. Crevices produced by other geometric effects (for example, at backing rings) can also provide sites for crevice corrosion.

The only component evaluated in this report potentially having a crevice caused by the presence of a thermal sleeve is the PWR SG feedwater nozzle. However, even if the thermal sleeve configuration represented a geometric crevice, it would not affect either the nozzle-to-shell weld or the nozzle inside radius sections. Therefore, no components evaluated in this report are susceptible to crevice corrosion.

6.2.9 Erosion-Cavitation

This degradation mechanism represents degradation caused by turbulent flow conditions, which erode the pipe wall by cavitation. Cavitation damage is the result of the formation and instantaneous collapse of small voids within a fluid subjected to rapid pressure and velocity changes as it passes through a region where the flow is restricted (for example, a valve, pump, or orifice).

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to erosion-cavitation because there are no cavitation sources immediately upstream of the components.

6.2.10 Erosion

This degradation mechanism is applicable to all metals and alloys and can occur when the operating fluid contains particulates (more severe at higher concentrations). For each environment-material combination, there is a threshold velocity above which impacting objects may produce metal loss.

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to erosion because they are exposed to chemistry-controlled water (or steam), which eliminates the presence of particulates.

6.2.11 Flow-Accelerated Corrosion

FAC is a complex phenomenon that generally occurs in plain carbon steels and exhibits attributes of erosion and corrosion under both single-phase (water) and two-phase (water/steam) conditions. Factors that influence FAC include the following:

- Flow path geometry and velocity (FAC rates are highest in the vicinity of sharp discontinuities such as branch connections, elbows, and in areas of shop and field welds, particularly at locations where backing rings were used and/or weld repairs were performed).
- pH and dissolved oxygen (results have shown that FAC rates decrease as pH and dissolved oxygen are increased).
- Moisture content of steam (higher moisture content results in higher rates of FAC).
- Temperature (FAC is most severe at a temperature of approximately 180°C).
- Material chromium content (FAC rates are highest in plain carbon steels; small amounts of alloying elements such as chromium can provide excellent resistance to FAC).

Component susceptibility is typically established under the plant's existing FAC program.

The only component evaluated in this report that is potentially included in the plant FAC program is the PWR SG feedwater nozzle (main steam nozzles are typically excluded from the FAC program due to the low moisture content of the steam). However, the nozzle-to-shell weld is sufficiently remote from the region of flow that it is not affected. Furthermore, the feedwater nozzle inside radius section has a smooth transition from the feedwater piping to the SG shell, so there is no sharp discontinuity that might increase this location's susceptibility to FAC. In addition, industry operating experience has not identified FAC as an issue at the inside radius section of nozzles. Therefore, neither the PWR SG feedwater nozzle components nor main steam components evaluated in this report are considered susceptible to FAC.

6.2.12 Corrosion/Wastage

General corrosion is characterized by an electrochemical reaction that occurs relatively uniformly over the entire surface area exposed to a corrosive environment. For carbon and alloy steels, normal reactor water can serve as that corrosive environment, depending on the nature of the fluid chemistry control. In contrast, austenitic stainless steels are not susceptible to general corrosion in the reactor environment.

The PWR SG feedwater and main steam nozzle components evaluated in this report are fabricated from either carbon or low-alloy steels and are subjected to reactor fluid environments. However, these components are exposed to fluids subjected to strict chemistry controls and are therefore not susceptible to general corrosion/wastage.

6.2.13 Galvanic Corrosion

Galvanic corrosion results when two electrochemically dissimilar materials are in contact with one another in the presence of an electrolyte. In the light water reactor environment, reactor water and other fluid sources can serve as an electrolyte. More corrosion-resistant alloys will not suffer from galvanic corrosion, but they may affect the galvanic corrosion of other materials.

The PWR SG feedwater and main steam nozzle components evaluated in this report are not susceptible to galvanic corrosion because they do not feature two electrochemically dissimilar metals in contact with one another in the presence of an electrolyte. Even in the potential scenario in which carbon steel was in contact with low-alloy steel, there is essentially no corrosion potential difference between these two metals. Therefore, none of the components evaluated in this report is susceptible to galvanic corrosion.

6.2.14 Thermal Stratification, Cycling, and Striping (TASCS)

Areas where there can be leakage past valves separating hot and cold fluids and regions where there might be intermittent mixing of hot and cold fluids caused by fluid injection are susceptible to TASCS. Alternating stresses caused by thermal cycling of a component result in accumulated fatigue damage and can lead to crack initiation and growth.

The only location with potential for high-frequency thermal cycling is the PWR SG feedwater nozzle. For PWR designs in which auxiliary feedwater enters the SGs through the main feedwater nozzles, the potential for fluid stratification exists under low-flow conditions during plant heat-up and cooldown. As discussed in Section 4.2, the feedwater piping in some plants has been subjected to prior thermal fatigue failures. Many RSGs were designed with modifications including an integral thermal sleeve with the nozzle safe end, which prevents potential bypass flow from reaching the feedwater nozzle body. The thermal sleeve therefore protects the feedwater nozzle components of interest (nozzle-to-vessel weld and inside radius section) from TASCS. In addition, the nozzle-to-shell weld does not encounter the incoming feedwater flow and is located remote from the portions of the nozzle affected by incoming feedwater flow—so it is therefore not affected by TASCS. Therefore, the PWR SG feedwater nozzle and main steam nozzle components evaluated in this report are not considered susceptible to TASCS.

6.2.15 Thermal Transients

Areas considered susceptible to thermal transients include components in which there are significant pressure and/or thermal excursions. In piping, significant temperature excursions consist of a relatively rapid cold water injection that results in a temperature change greater than 150°F for carbon steel piping or 200°F for austenitic steel piping. When these temperature changes are exceeded, additional evaluations are required to determine whether the temperature change is greater than what is allowed.

The only component evaluated in this report that is potentially susceptible to significant thermal transients is the PWR SG feedwater nozzle. When feedwater is initiated during plant heat-up, sufficient temperature differences exist between the incoming (colder) feedwater fluid and the (hotter) SG bulk fluid to cause a significant thermal transient at the feedwater nozzle. The fact that the feedwater nozzles typically have thermal sleeves and the feedwater injects into the SG via a feedwater sparger, combined with the fact that the nozzle-to-shell welds are sufficiently remote from the incoming feedwater flow, means that the locations evaluated in this report are likely protected from such significant transients. However, for the purposes of this evaluation, both the PWR SG feedwater and main steam nozzles are assumed to be susceptible to thermal transients. Fatigue due to the thermal transients identified in Table 5-5 will be considered, where applicable, when performing the fracture mechanics analyses in Section 8.

6.2.16 Mechanical Fatigue

Mechanical fatigue (vibration) can occur in locations subjected to high-frequency reversible loads such as pressure fluctuations caused by pumps. Therefore, mechanical fatigue potentially affects all PWR SG feedwater and main steam nozzle components in the scope of this evaluation. Mechanical fatigue will be considered, where applicable, when performing the fracture mechanics analyses in Section 8.

6.3 Conclusions

All PWR SG Item No. C2.22 nozzle inside radius section components, as well as all PWR SG Item Nos. C2.21 and C2.32 nozzle-to-shell welds, were evaluated for their susceptibility to the degradation mechanisms listed in Section 6.1. The results conclude the following:

- The PWR SG feedwater nozzle inside radius section component is potentially susceptible to corrosion fatigue, thermal transients, and mechanical fatigue.
- The PWR SG feedwater nozzle-to-shell welds are potentially susceptible to corrosion fatigue and mechanical fatigue.
- The PWR SG main steam nozzle components are susceptible to mechanical fatigue.

Therefore, these fatigue-related mechanisms will be considered when performing the probabilistic and deterministic fracture mechanics evaluations in Section 8.

7 COMPONENT STRESS ANALYSIS

This section discusses the stress analyses for the three nozzles selected in Section 4 as well as some alternative configurations to assess geometric differences. Due to the complex behavior of the stress distributions near the nozzle-to-shell welds and nozzle inside radius sections, finite element analyses (FEAs) were performed for all three nozzles. The material properties, operating loads, and transients listed in Section 5 were used as inputs for the stress analyses. Finite element models (FEMs) were developed for these components using the ANSYS finite element analysis software package [51]. Two-dimensional (2-D) axisymmetric or 3-D quarter models were used for the nozzles, as appropriate.

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 history was then used as input to perform a stress analysis for each transient. For internal pressure, arbitrary unit internal pressure was applied to the FEMs. The stress results from the unit pressure were scaled to correspond to the actual pressure values. The stress results were used in fracture mechanics evaluations conducted in Section 8.

In performing the analyses, the following assumptions were made during development of the FEMs and thermal/mechanical stress evaluations:

- The nozzle-to-shell 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 is 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 calculated at a stress-free temperature of 70°F and do not include any thermal stress effects.
- For all thermal heat transfer analyses, an additional time of 3,600 seconds was added to the end of each transient 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).

7.1 Stress Analysis for PWR SG Main Steam Nozzle (Westinghouse 4-Loop Design)

7.1.1 Finite Element Model

An FEM of the Westinghouse 4-loop SG main steam nozzle was developed using the ANSYS finite element analysis software package [51], with dimensions shown in Figure 4-8. This configuration exits the center of the SG top head, resulting in an axisymmetric arrangement; therefore, a 2-D model was used in the development of the FEM. The 2-D axisymmetric model was constructed using 2-D structural solid, PLANE42, elements. The thermal equivalent element for the thermal transient analyses is PLANE55. The FEM is shown in Figure 7-1 and includes a local portion of the SG hemispherical head, the main steam nozzle, and nozzle-to-shell weld. The designation of the materials involved in the model and associated material properties are discussed in Section 5.1.

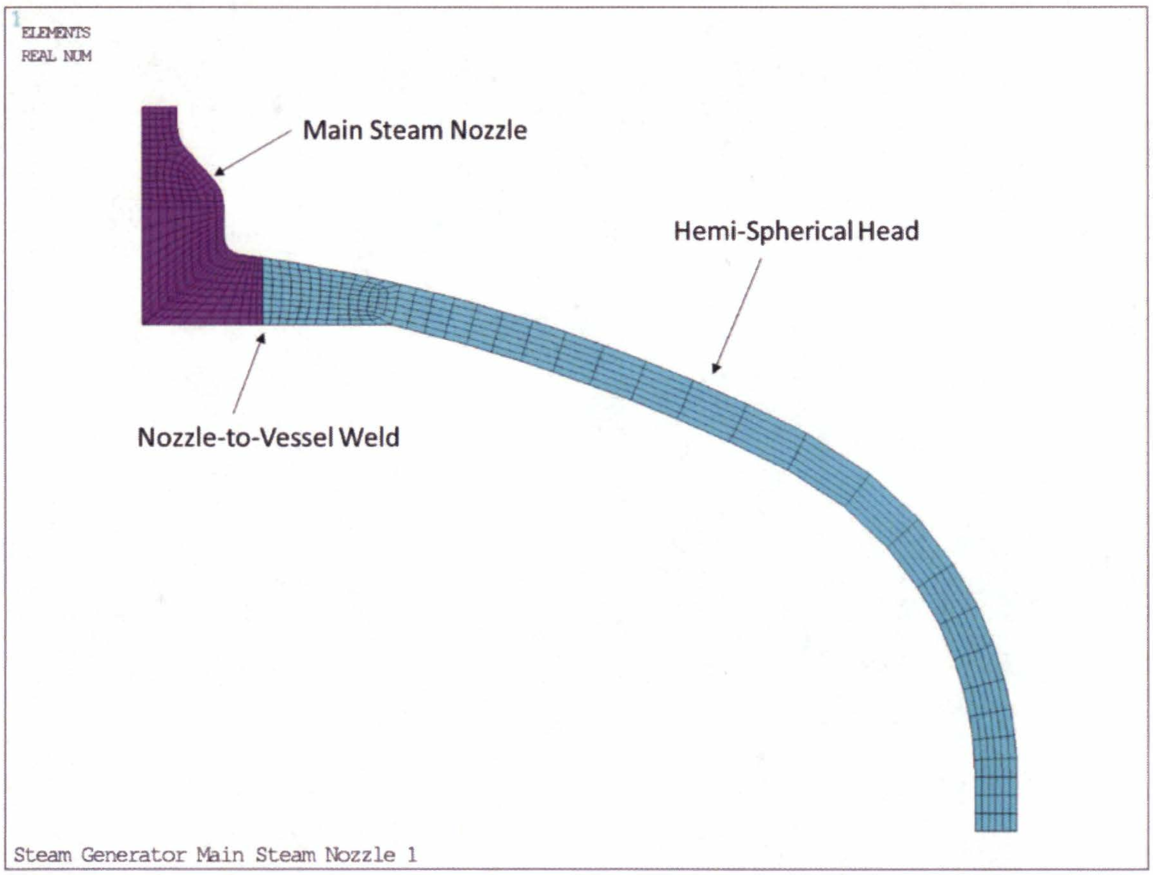


Figure 7-1
2-D Finite Element Model and Mesh for PWR SG Main Steam Nozzle: Westinghouse 4-Loop Design

7.1.2 Pressure/Thermal Stress Analysis

7.1.2.1 Internal Pressure Loading Analysis

A unit internal pressure of 1,000 psi was applied to the interior surfaces of the model. The resulting stresses will be scaled up to the transient pressures for use in the fracture mechanics evaluations. An induced end-cap load was applied to the free end of the FEM nozzle end in the form of tensile axial pressure calculated using Equation 7-1:

$$P_{end\text{-}cap} = \frac{P \cdot ID^2}{OD^2 - ID^2} \qquad \textbf{Eq. 7-1}$$

where,

$P_{end\text{-}cap}$ = End-cap pressure on nozzle/shell free end (psi)

P = Internal pressure (psi)

ID = Inside diameter of nozzle/shell (in.)

OD = Outside diameter of nozzle/shell (in.)

Symmetric boundary conditions were applied to the one axial free end of the SG hemispherical head, while axial displacement couples were applied on the free end of the nozzle. The applied pressure load and boundary conditions for this case are shown in Figure 7-2.

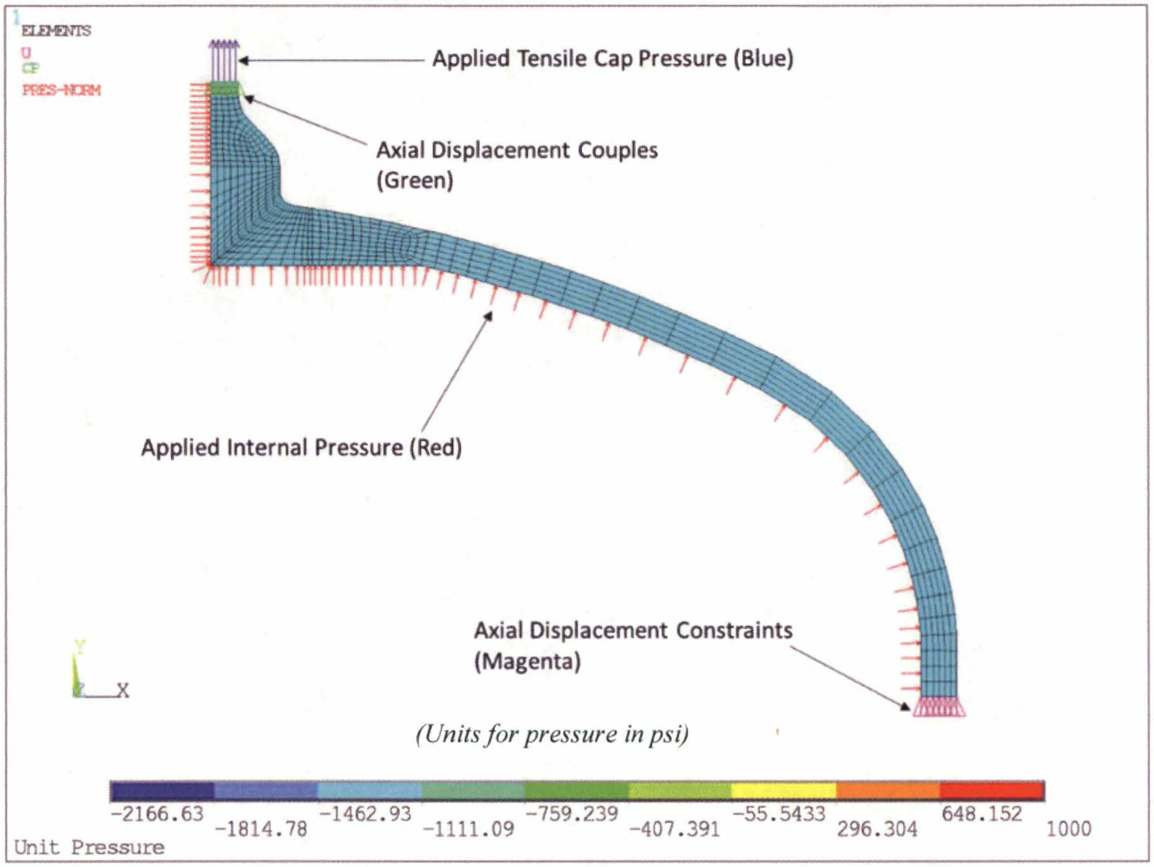
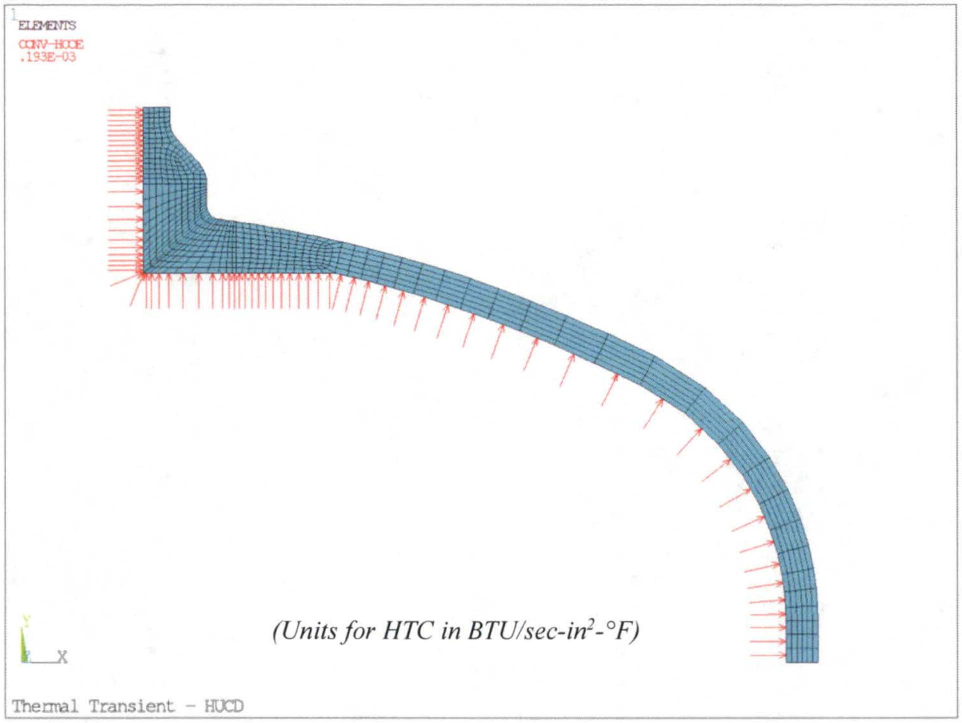


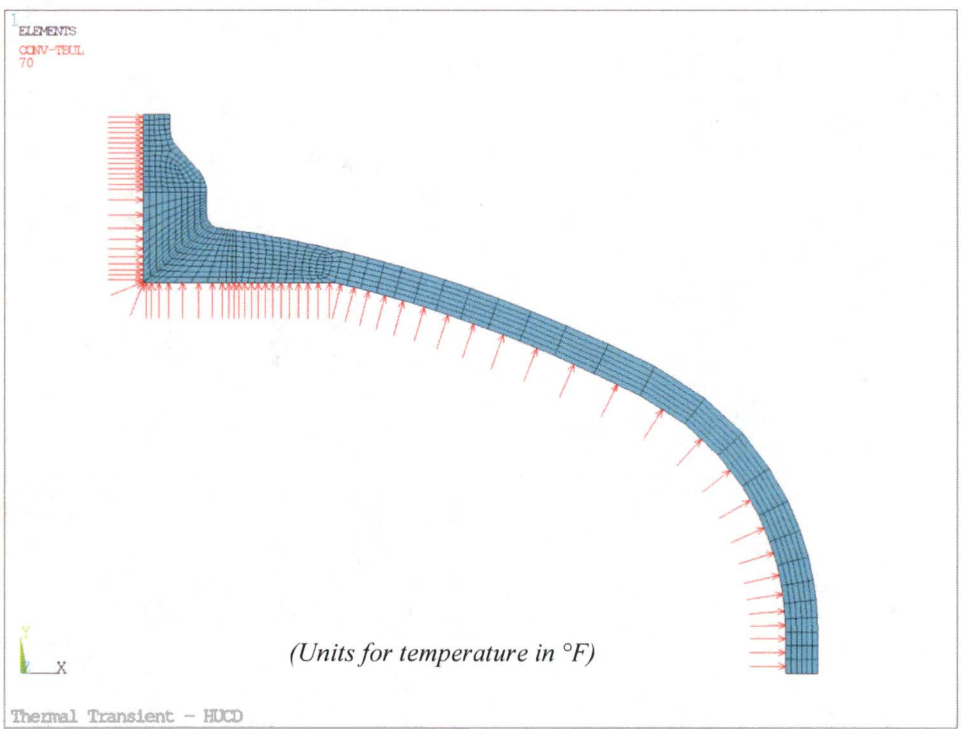
Figure 7-2
Applied Boundary Conditions and Unit Internal Pressure for PWR SG Main Steam Nozzle: Westinghouse 4-Loop Design

7.1.2.2 Thermal Heat Transfer Analyses

The thermal transients identified in Table 5-5 were applied to the interior surface nodes of the nozzle and shell. A nominal heat transfer coefficient of 100 BTU/hr-ft^2-°F was applied to the inside surface of the nozzle and shell for all evaluated transients. The main steam nozzle is located in a steam environment. Steam (especially dry steam) is not as good of a conductor as water; therefore, the heat transfer coefficient value is expected to be smaller. A check of the heat transfer coefficient value for forced convection was performed based on the saturated steam properties, the steam flow rate for the vessel region (3.73×10^6 lb/hr), and the dimension of the vessel configuration. The heat transfer coefficient calculated was approximately 100 BTU/hr-ft^2-°F. Therefore, using the value of 100 BTU/hr-ft^2-°F is representative or conservative for all evaluated transients. Heat transfer coefficients or temperatures were not applied to the insulated outside surfaces. Figure 7-3 shows representative plots of the thermal loads for the heat-up/cooldown transient applied to the SG main steam nozzle (Westinghouse 4-loop design). Note that the heat-up and cooldown transients were evaluated as a single transient (heat-up followed by cooldown). Therefore, discussion in this section and others refers to the one composite transient as *heat-up/cooldown*.



Heat Transfer Coefficient



Bulk Temperature

Figure 7-3
Applied Thermal Boundary Conditions for Thermal Transient Analyses for PWR SG Main Steam Nozzle: Westinghouse 4-Loop Design (heat-up/cooldown transient shown; loads applied at time = 46,080 seconds [during cooldown portion])

7.1.2.3 Thermal Stress Analyses

Symmetric boundary conditions were applied to the bottom free end of the head, while axial displacement couples were applied on the free end of the nozzle. The reference temperature for the thermal strain calculation was assumed to be 70°F. Figure 7-4 shows a plot of the boundary conditions applied for the thermal stress analyses.

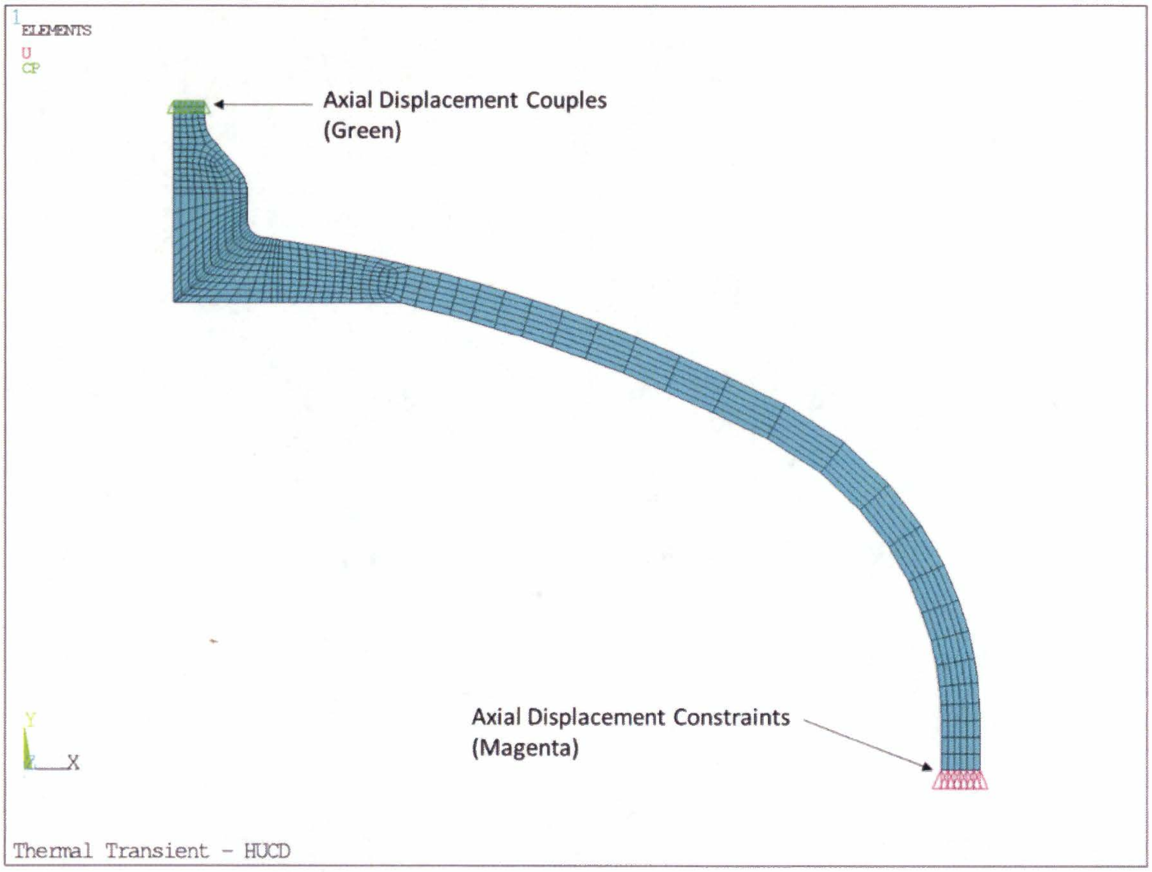
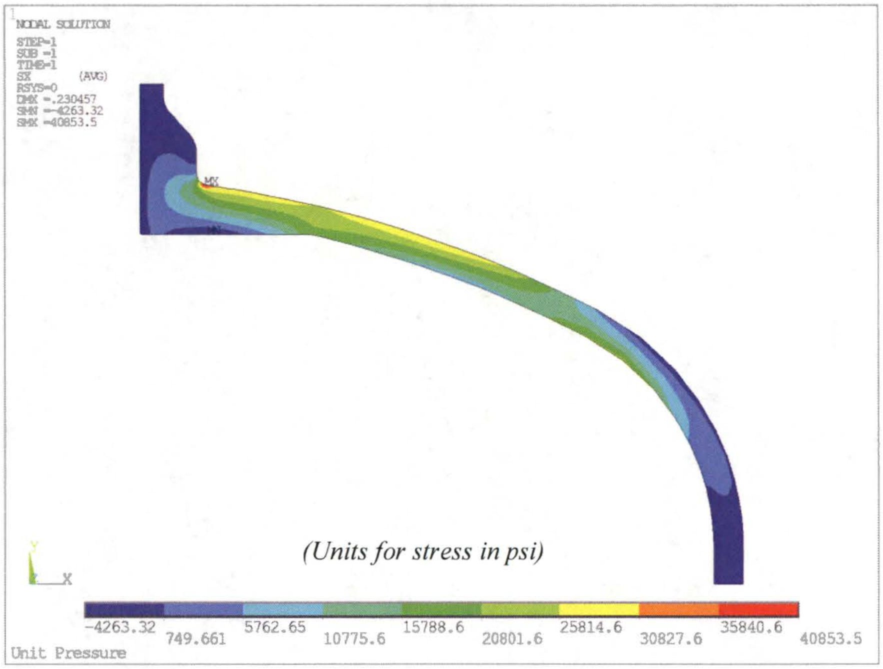


Figure 7-4
Applied Mechanical Boundary Conditions for Thermal Stress Analyses for PWR SG Main Steam Nozzle: Westinghouse 4-Loop Design

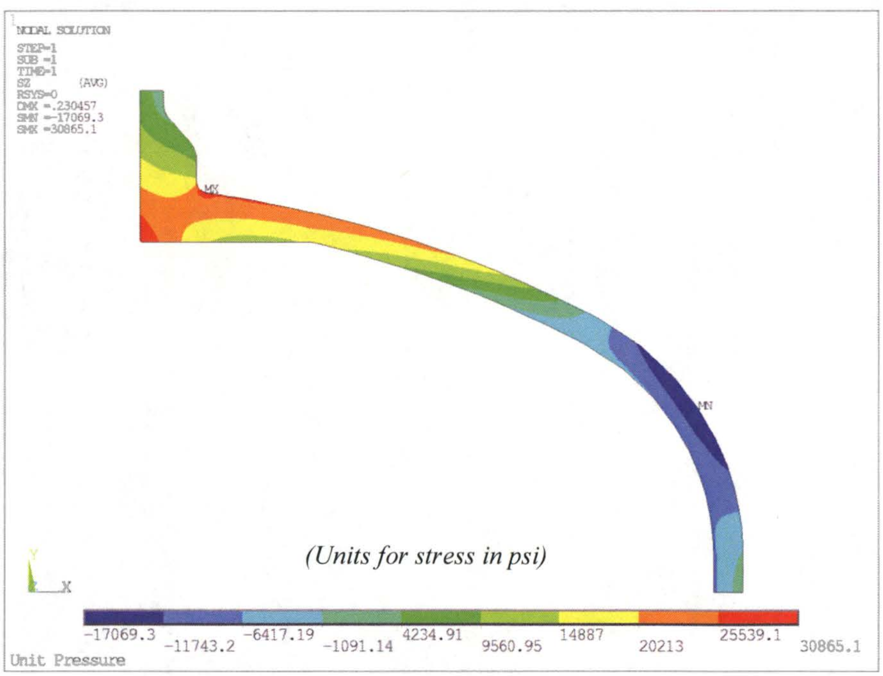
7.1.3 Stress Analysis Results

The component stress contour plots are plotted in the global Cartesian coordinate system, where the Z-direction aligns with the nozzle hoop direction and the X-direction aligns with the nozzle radial direction (meridional for the nozzle-to-head weld). The stresses for a unit internal pressure of 1,000 psig are shown in Figure 7-5. Representative temperature and stress contour plots for the composite heat-up/cooldown transient are shown in Figures 7-6 and 7-7, respectively. The times shown in Figures 7-6 and 7-7 are when the maximum stress intensity occurs. Figure 7-8 shows the path locations where stresses were extracted for use in the fracture mechanics evaluations. Path P1 was chosen for evaluating the nozzle inside radius location, and Path P2 was chosen for evaluating the nozzle-to-shell weld location. All stresses were extracted in a Cartesian coordinate system (global to the SG vessel), which is the same coordinate triad shown in Figures 7-5 and 7-7. Through-wall stress distributions for Paths P1 and P2 are shown in

Figures 7-9 and 7-10, respectively. In these figures, thermal stresses are shown at the times when the stresses peaked during each of the evaluated transients. Figures 7-9 and 7-10 show that the pressure stress is the largest stress as a function of depth into the thickness, which confirms the use of the parameter (PR/t) in the subsequent sensitivity analyses discussed in Section 4.3.3.



X-Direction Stress (Radial to Nozzle, Meridional for Nozzle-to-Head Weld)



Z-Direction Stress (Hoop for Nozzle)

Figure 7-5
Stress Contours due to Unit Internal Pressure for PWR SG Main Steam Nozzle: Westinghouse 4-Loop Design

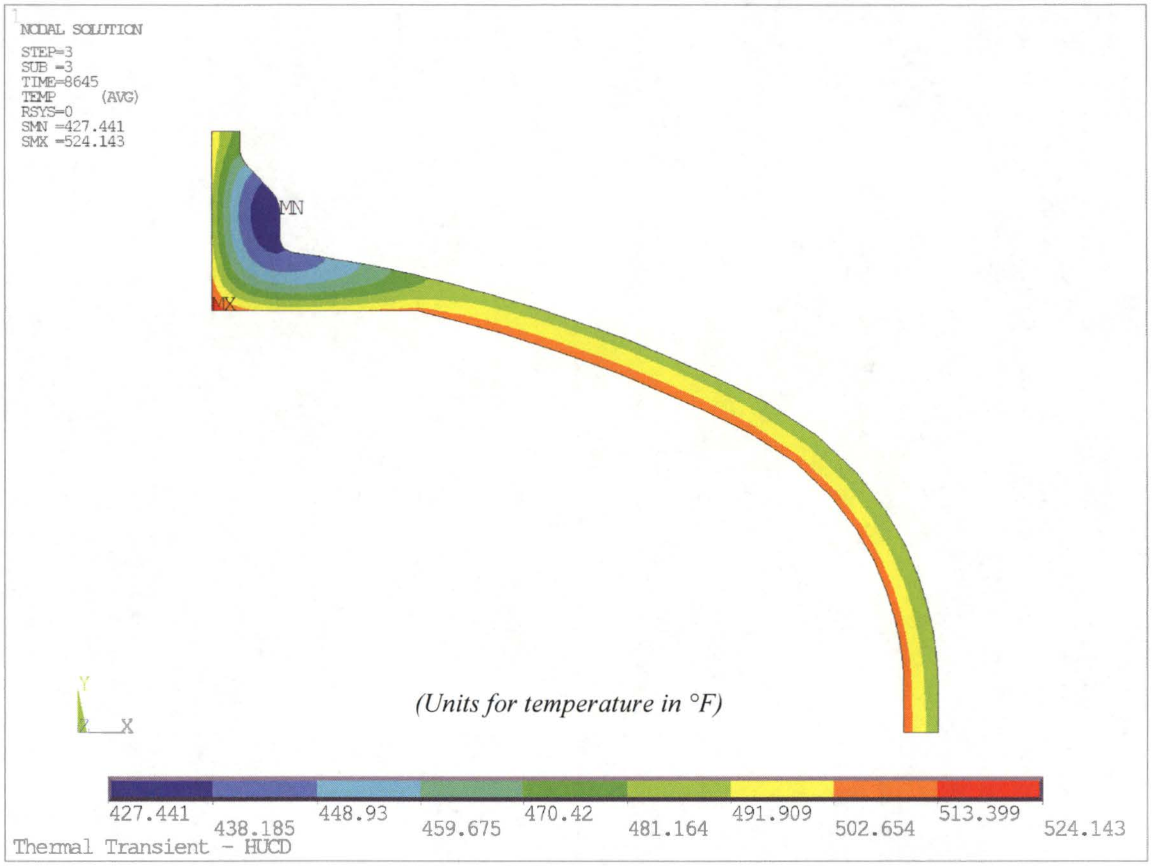
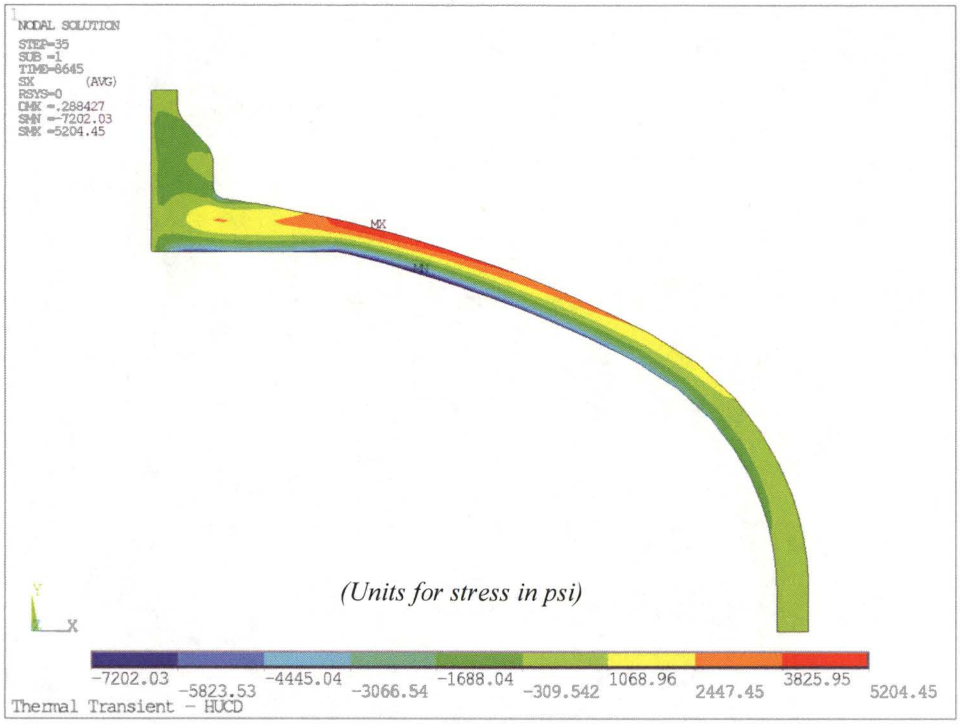
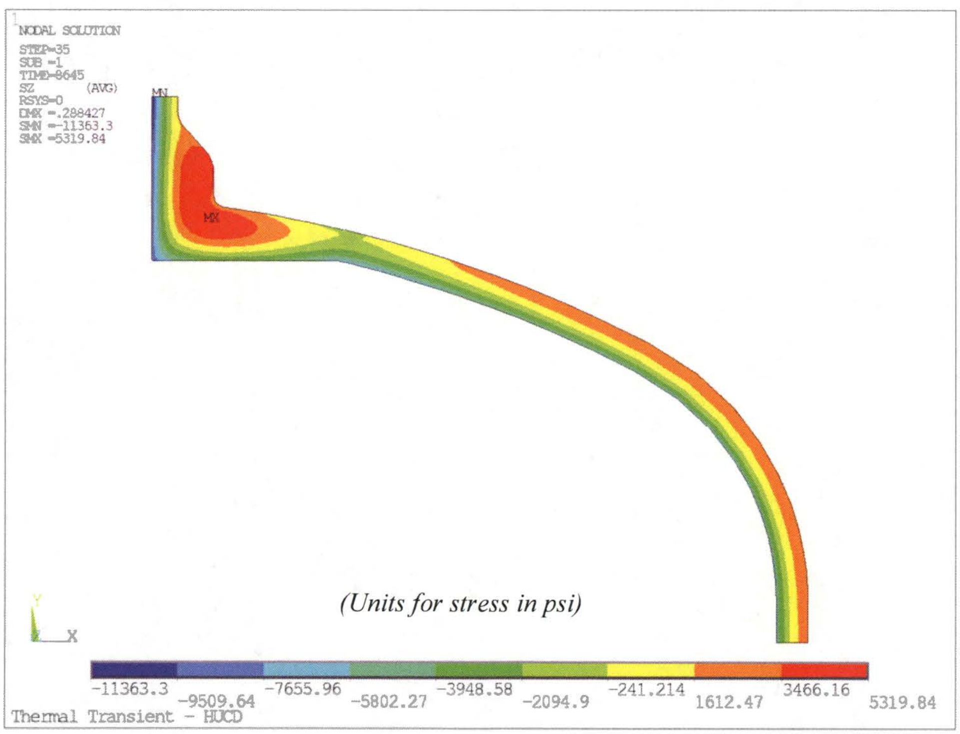


Figure 7-6
Temperature Contour of Heat-Up/Cooldown Transient (time = 8,645 seconds, end of heat-up) for PWR SG Main Steam Nozzle: Westinghouse 4-Loop Design



X-Direction Stress (Radial to Nozzle, Meridional for Nozzle-to-Head Weld)



Z-Direction Stress (Hoop for Nozzle)

Figure 7-7
Stress Contours of Heat-Up/Cooldown

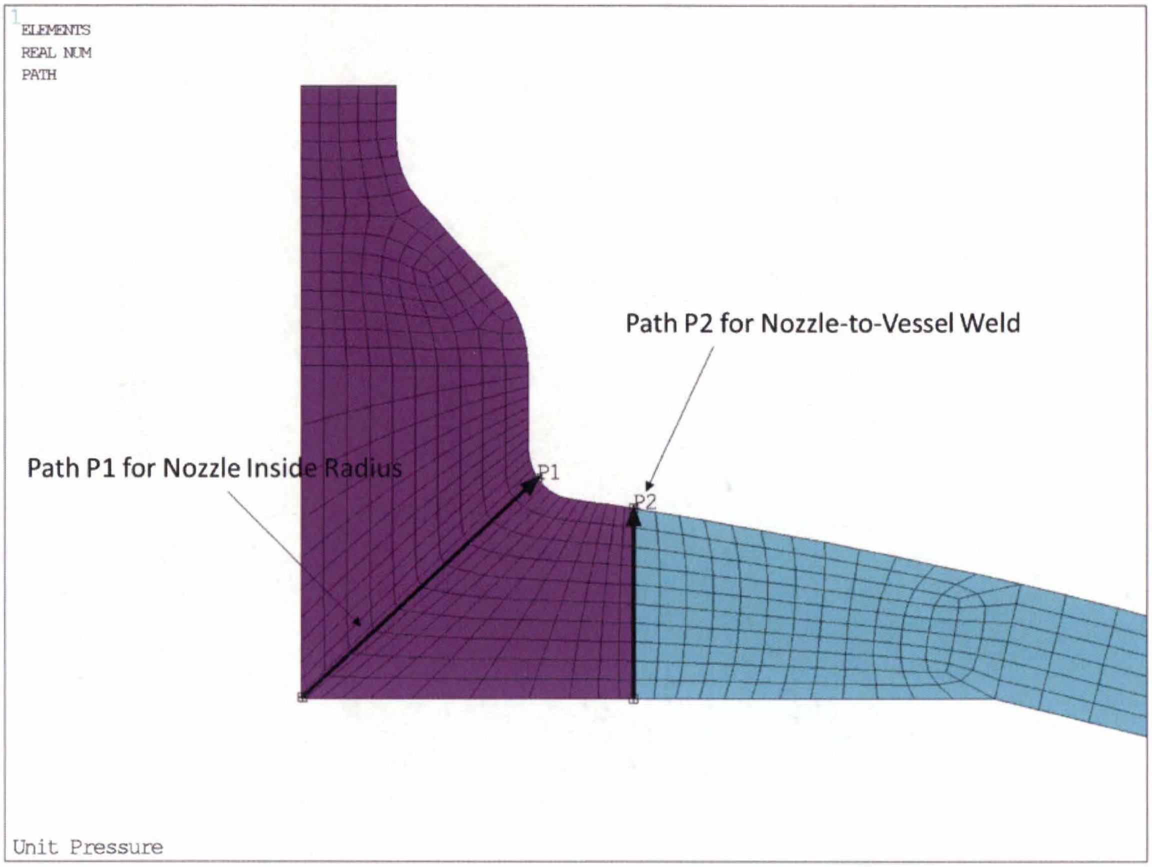
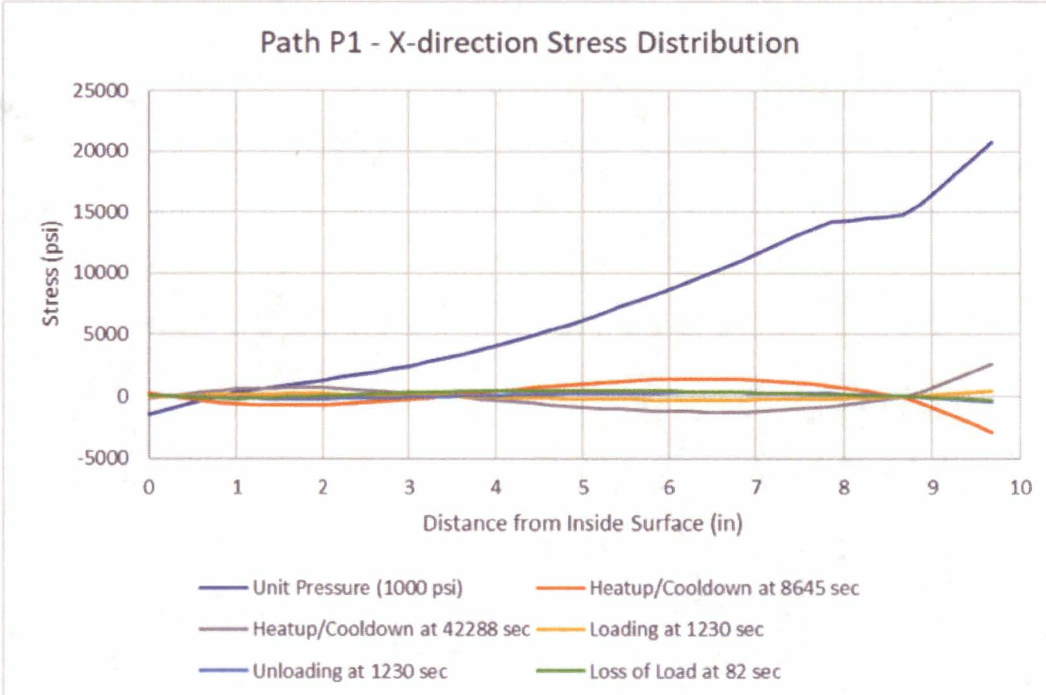
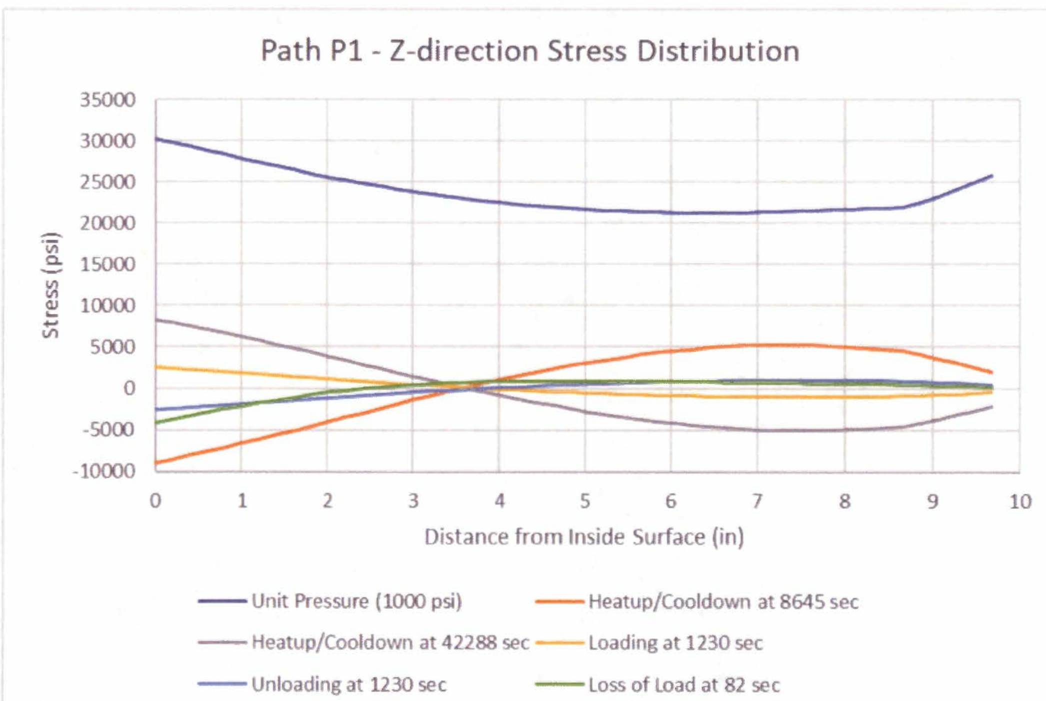


Figure 7-8
Path Locations for PWR SG Main Steam Nozzle (Westinghouse 4-Loop Design)

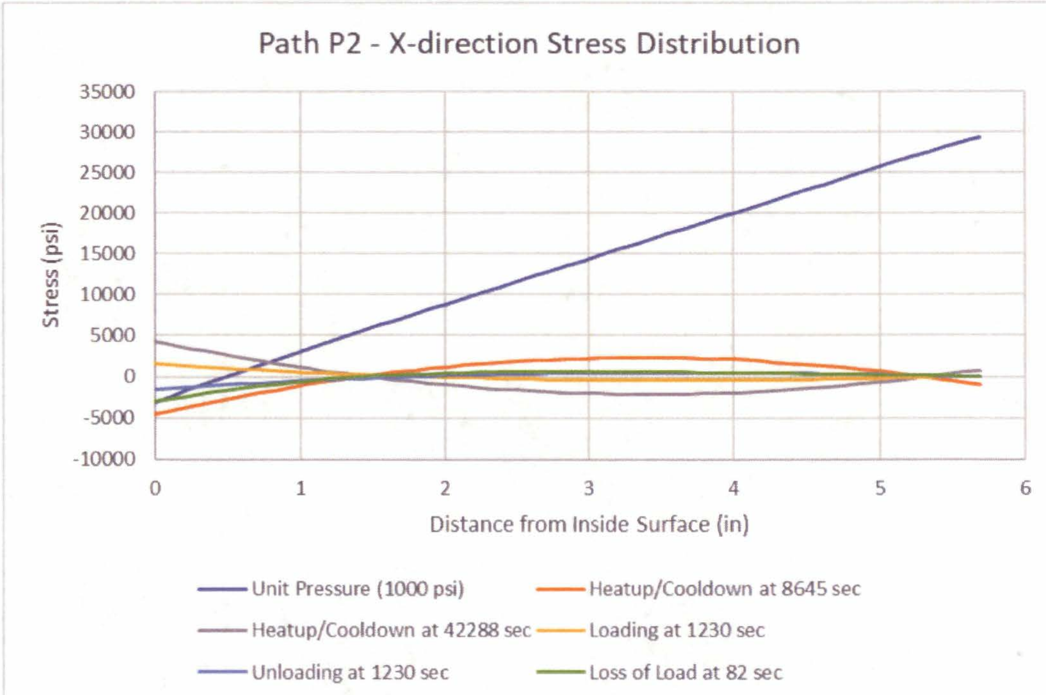


X-Direction Stress (Radial to Nozzle, Meridional for Nozzle-to-Head Weld)

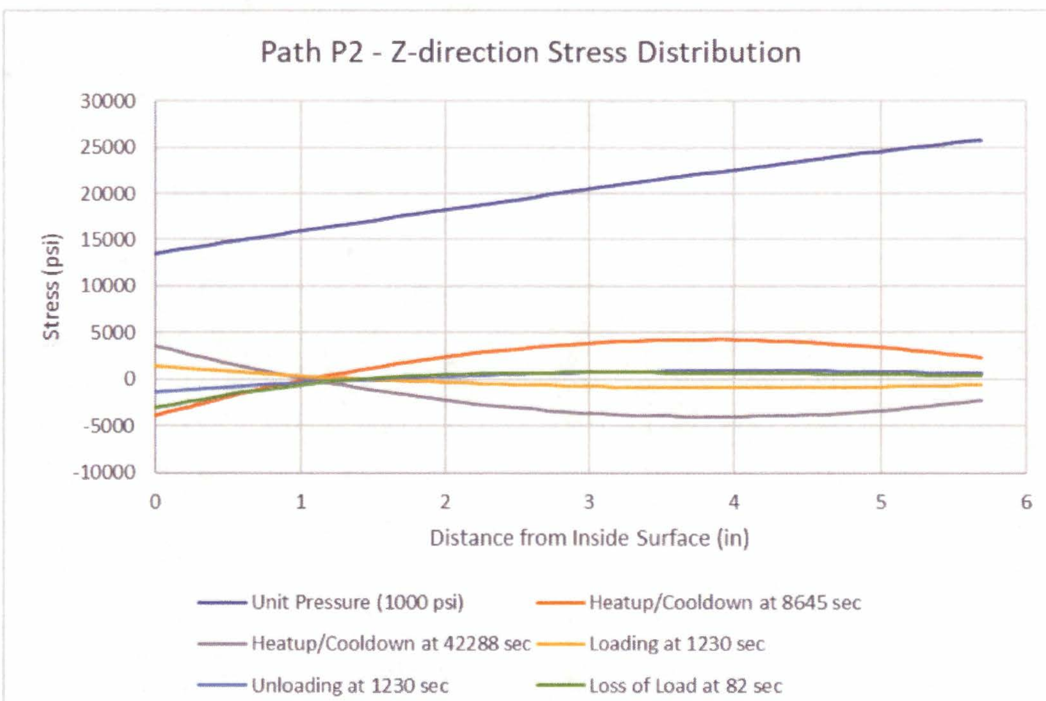


Z-Direction Stress (Hoop for Nozzle)

Figure 7-9
Through-Wall Stress Distribution at Path P1 for PWR SG Main Steam Nozzle: Westinghouse 4-Loop Design



X-Direction Stress (Radial to Nozzle, Meridional for Nozzle-to-Head Weld)



Z-Direction Stress (Hoop for Nozzle)

Figure 7-10
Through-Wall Stress Distribution at Path P2 for PWR SG Main Steam Nozzle: Westinghouse 4-Loop Design

7.2 Stress Analysis for PWR SG Main Steam Nozzle: B&W Design

7.2.1 Finite Element Model

An FEM of the B&W SG main steam nozzle was developed using the ANSYS finite element analysis software package [51], with dimensions shown in Figure 4-9. The nozzle for this configuration is attached to the side of the shell, which renders it non-axisymmetric—therefore, a 3-D model was used in the development of the FEM. The 3-D quarter model was constructed using 8-node structural solid, SOLID45 elements. The thermal equivalent element for the thermal transient analyses is SOLID70. The model is shown in Figure 7-11, and includes a local portion of the SG shell, the main steam nozzle, and the nozzle-to-shell weld. The designation of the materials involved in the model and associated material properties are discussed in Section 5.1.

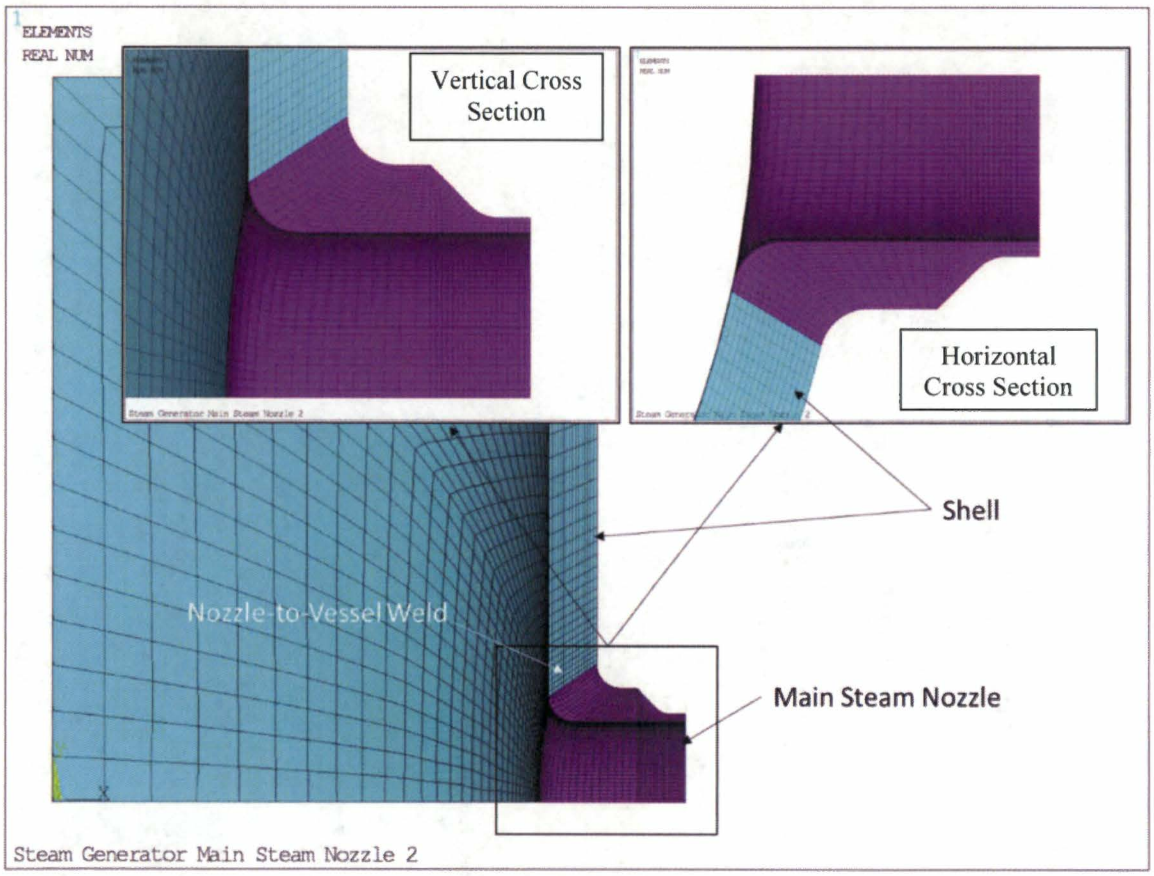


Figure 7-11
3-D Finite Element Model and Mesh for PWR SG Main Steam Nozzle: B&W Design

7.2.2 Pressure/Thermal Stress Analysis

7.2.2.1 Internal Pressure Loading Analysis

A unit internal pressure of 1,000 psi was applied to the interior surfaces of the model. The resulting stresses will be scaled up to the transient pressures for use in the fracture mechanics evaluations. An induced end-cap load was applied to the free end of the main steam nozzle and the upper free end of the shell in the form of tensile axial pressures calculated using Equation 7-1. Symmetric boundary conditions were applied to the bottom axial free end of the model and both circumferential free ends of the model, while axial displacement couples were applied on the free end of the nozzle and the other (upper) axial free end of the shell. The applied pressure load and boundary conditions for this case are shown in Figure 7-12.

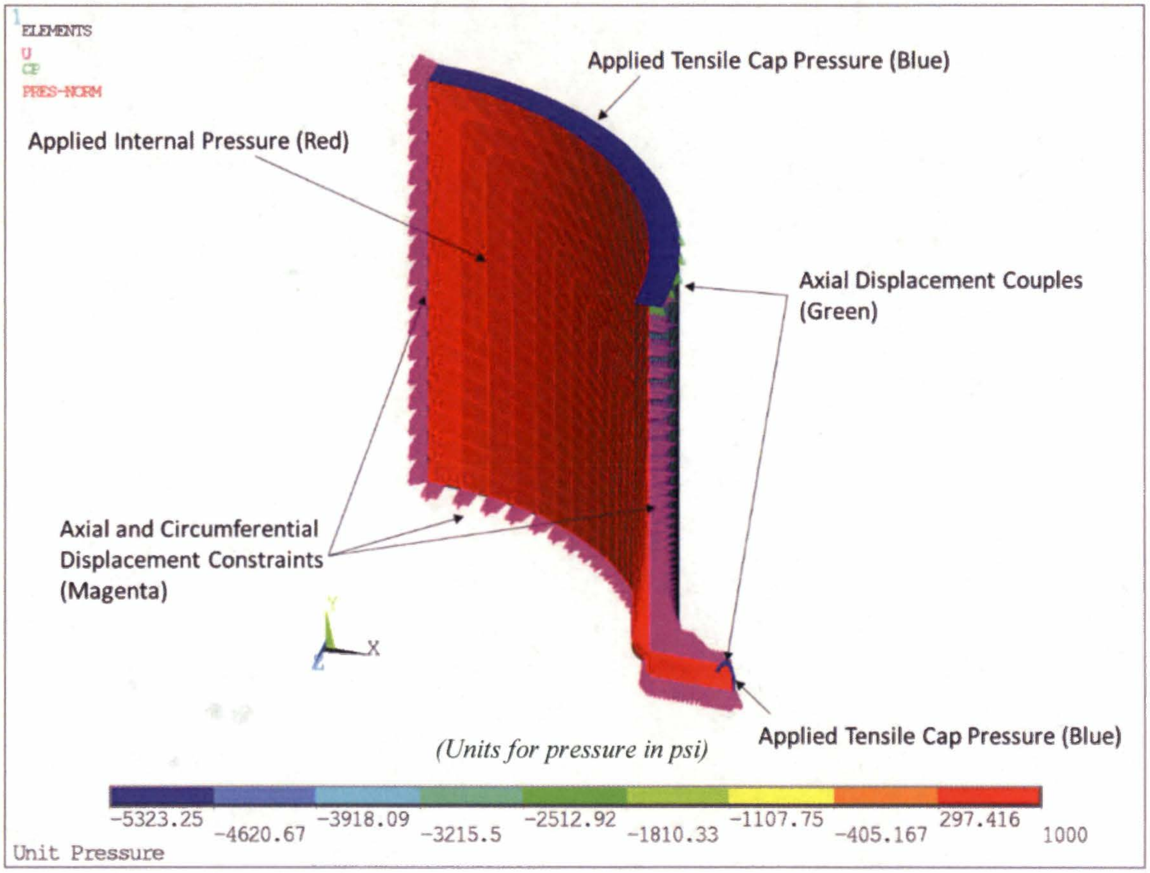
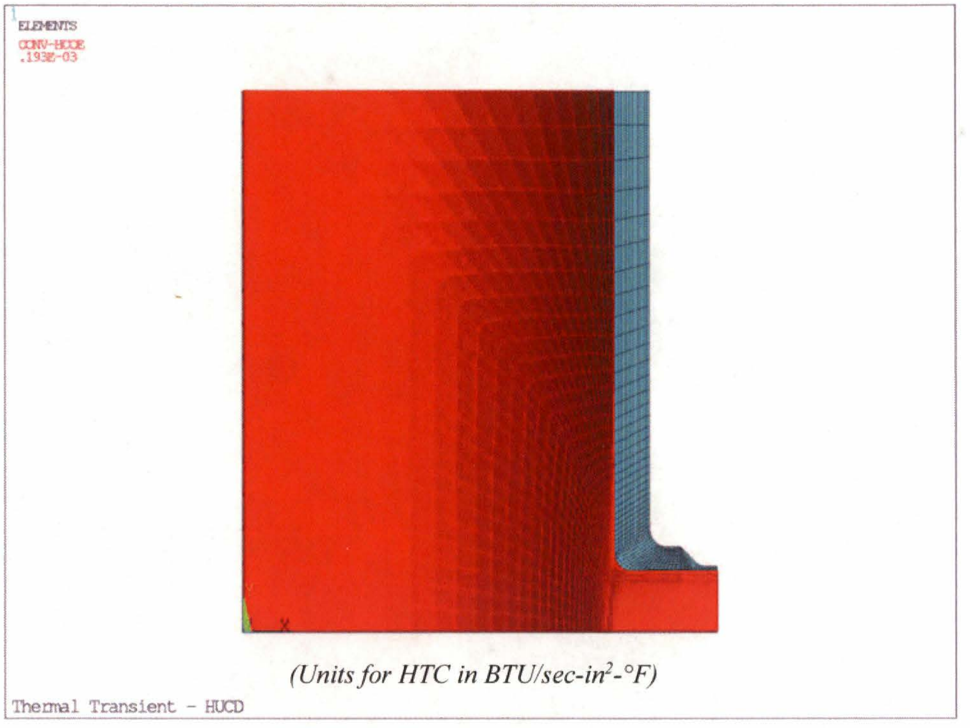


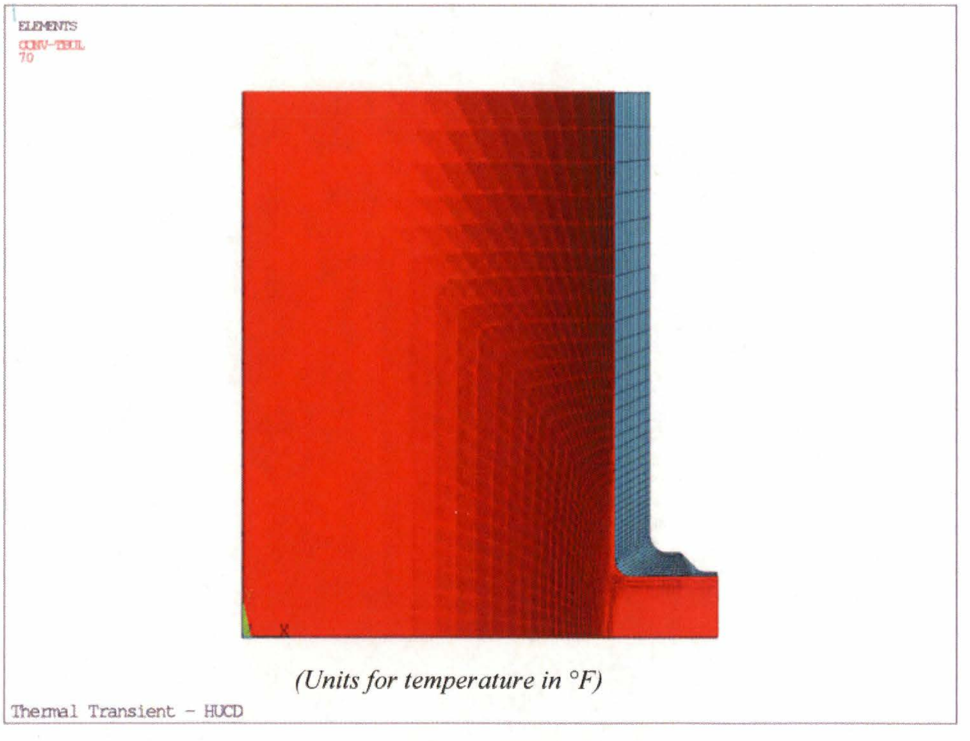
Figure 7-12
Applied Boundary Conditions and Unit Internal Pressure for PWR SG Main Steam Nozzle: B&W Design

7.2.2.2 Thermal Heat Transfer Analyses

The thermal transients discussed in Section 5.2 were applied to the interior surfaces of the nozzle and shell. A nominal heat transfer coefficient of 100 BTU/hr-ft^2-°F was applied to the inside surface of the nozzle and shell for all evaluated transients. The main steam nozzle is located in a steam environment. Steam (especially dry steam) is not as good of a thermal conductor as water; therefore, the heat transfer coefficient value is expected to be smaller. A check of the heat transfer coefficient value for forced convection was performed based on the saturated steam properties, the steam flow rate for the vessel region (1.35×10^6 lb/hr), and the dimension of the vessel configuration. The heat transfer coefficient calculated was approximately 100 BTU/hr-ft^2-°F. Therefore, using the value of 100 BTU/hr-ft^2-°F is representative or conservative for all evaluated transients. Heat transfer coefficients or temperatures were not applied to the insulated outside surfaces. Figure 7-13 shows representative plots of the thermal loads for the heat-up/cooldown transient applied to the SG main steam nozzle (B&W design). The heat-up and cooldown transients were modeled as one single transient with heat-up followed by cooldown, so both events are referred to by the single transient analyzed (heat-up/cooldown).



Heat Transfer Coefficient



Bulk Temperature

Figure 7-13
Applied Thermal Boundary Conditions for Thermal Transient Analyses for PWR SG Main Steam Nozzle: B&W Design (heat-up/cooldown transient shown; loads applied at time = 46,080 seconds [during cooldown portion])

7.2.2.3 Thermal Stress Analyses

Symmetric boundary conditions were applied to the bottom and circumferential free end of the shell; axial displacement couples were applied on the free end of the nozzle and the upper free end of the shell. The reference temperature for the thermal strain calculation was assumed to be 70°F. Figure 7-14 shows a plot of the boundary conditions applied for the thermal stress analyses.

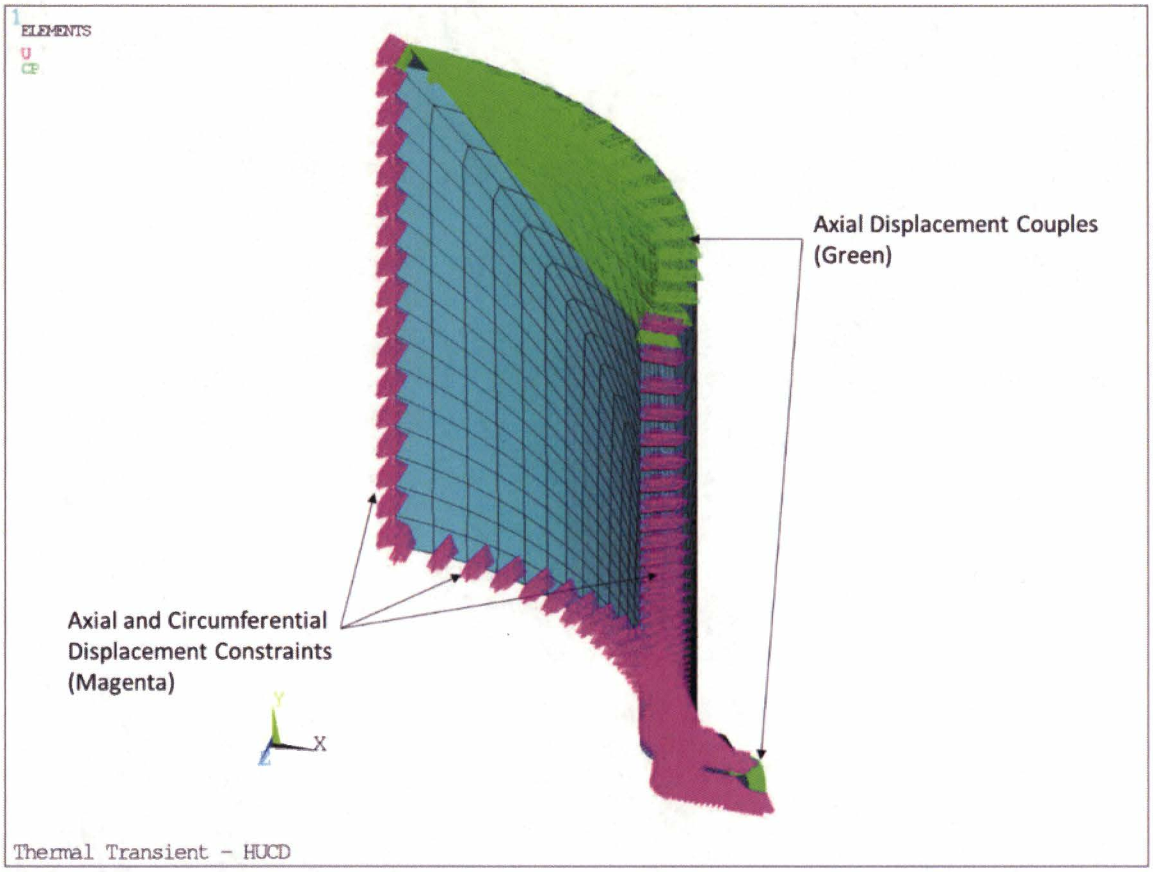
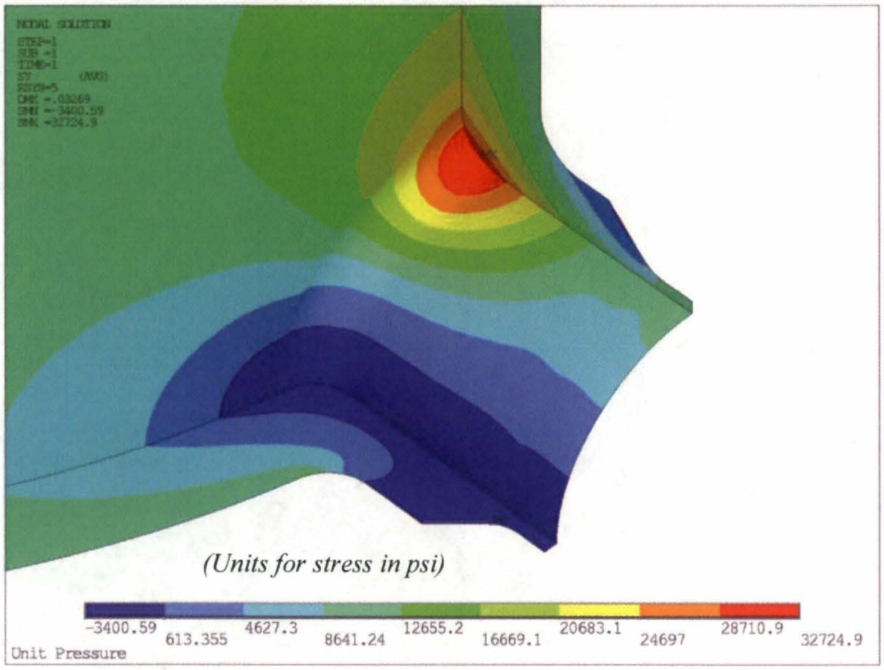


Figure 7-14
Applied Mechanical Boundary Conditions for Thermal Stress Analyses for PWR SG Main Steam Nozzle: B&W Design

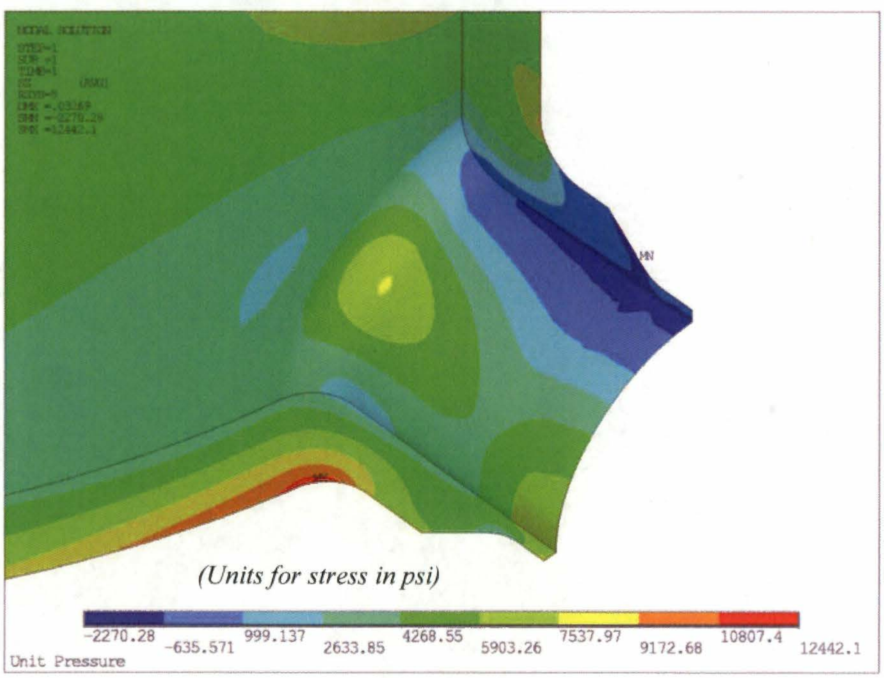
7.2.3 Stress Analysis Results

The hoop stress and axial stress contour plots for the unit internal pressure case are shown in Figure 7-15. Representative temperature and stress contour plots for the heat-up/cooldown transient are shown in Figures 7-16 and 7-17, respectively. The time shown in Figures 7-16 and 7-17 is when the maximum stress intensity occurs. Figure 7-18 shows the path locations where stresses were extracted. Paths P1 and P2 were chosen for the nozzle inside radius location, and Paths P3 and P4 were chosen for the nozzle-to-shell weld location. All stresses were extracted in a cylindrical coordinate system with the Y-direction corresponding to the hoop direction of the

SG shell and the Z-direction corresponding to the axial direction of the SG shell. Through-wall stress distributions for Paths P1 through P4 are shown in Figures 7-19 through 7-22, respectively. Again, the plots indicate that the pressure stress is generally the highest stress overall through the thickness, confirming the use of (PR/t) as a parameter in the subsequent sensitivity analyses.



SG Hoop Stress



SG Axial Stress

Figure 7-15
Stress Contours due to Unit Internal Pressure for PWR SG Main Steam Nozzle: B&W Design

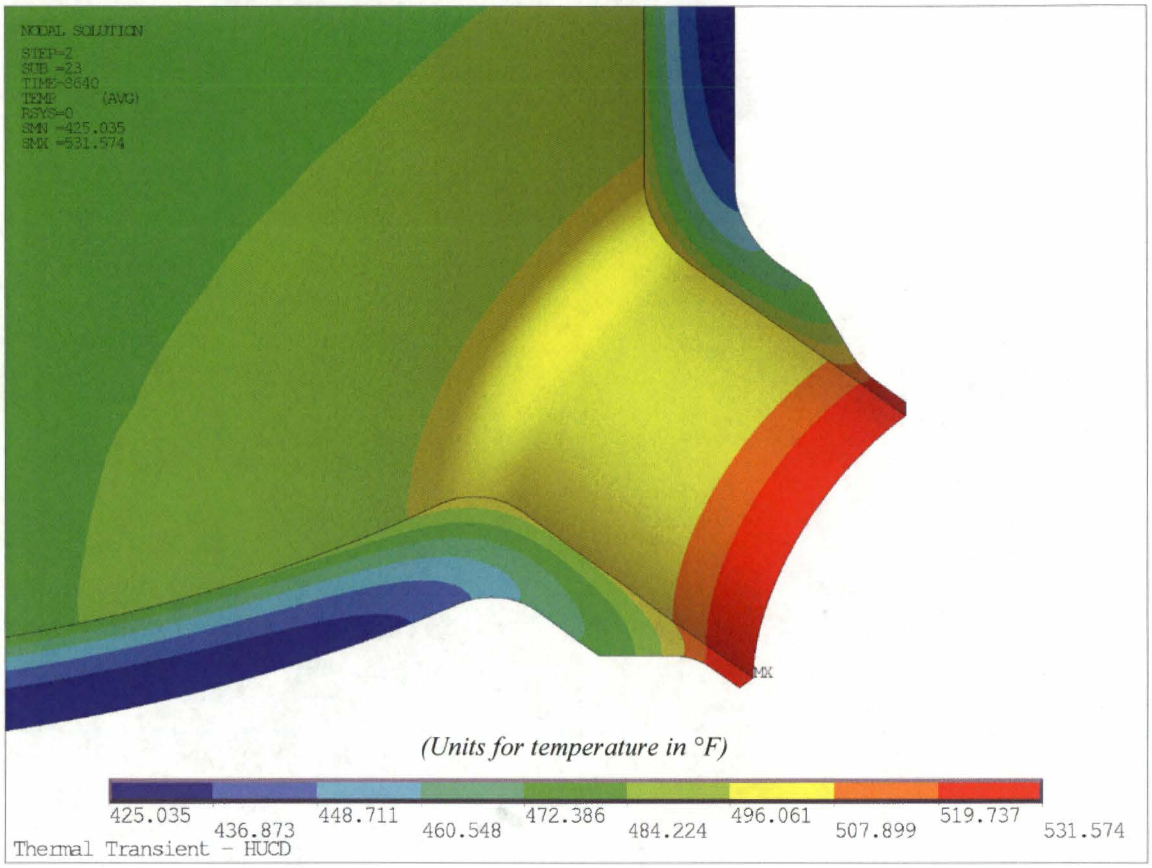
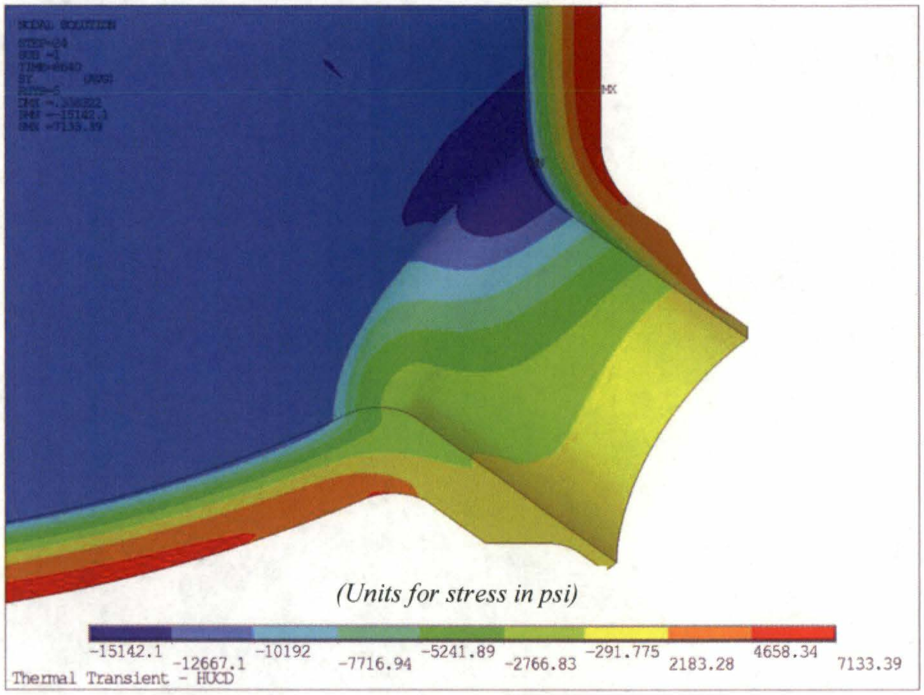
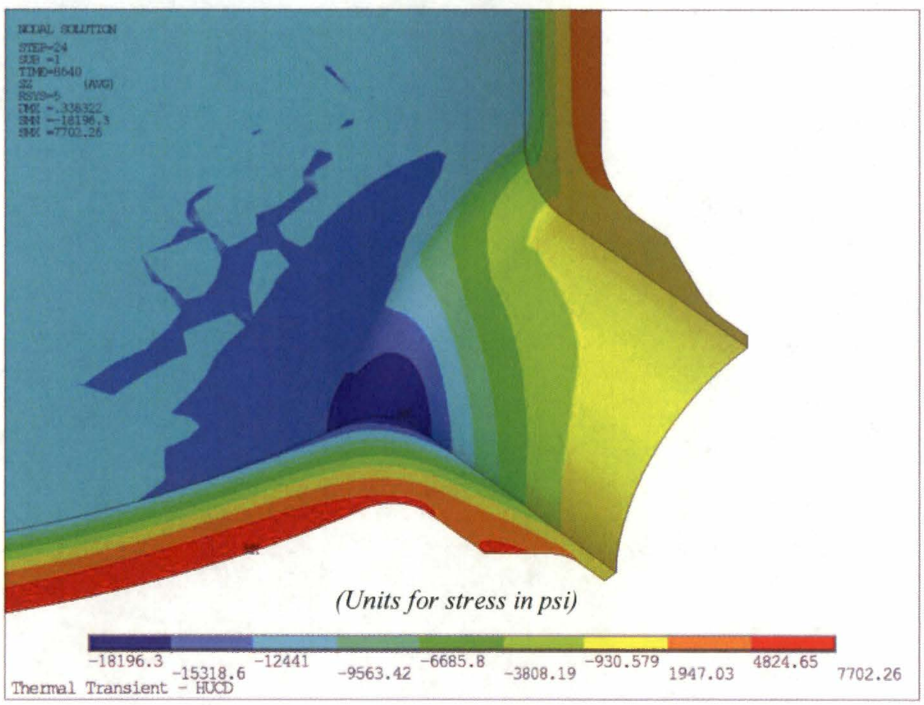


Figure 7-16
Temperature Contour of Heat-Up/Cooldown Transient (time = 8,640 seconds, end of heat-up) for PWR SG Main Steam Nozzle: B&W Design



SG Hoop Stress



SG Axial Stress

Figure 7-17
Stress Contours of Heat-Up/Cooldown Transient (time = 8,640 seconds, end of heat-up) for PWR SG Main Steam Nozzle: B&W Design

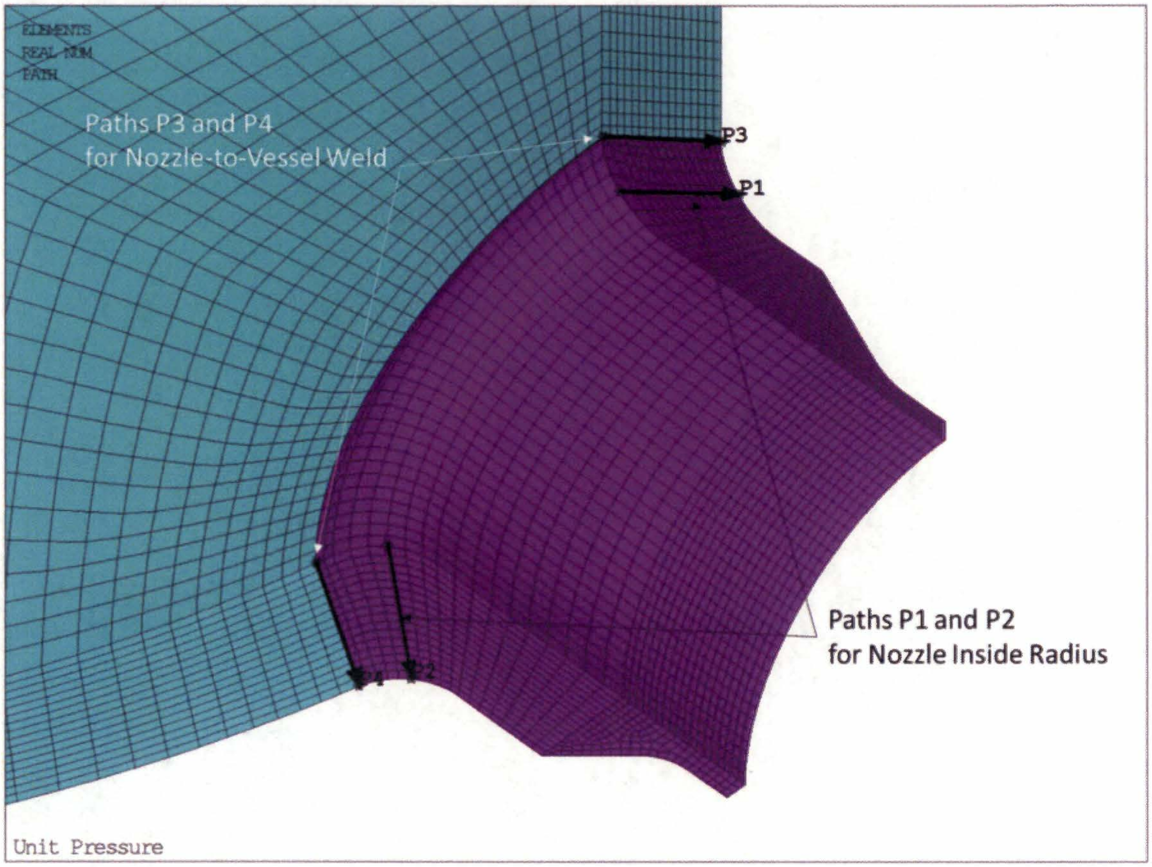
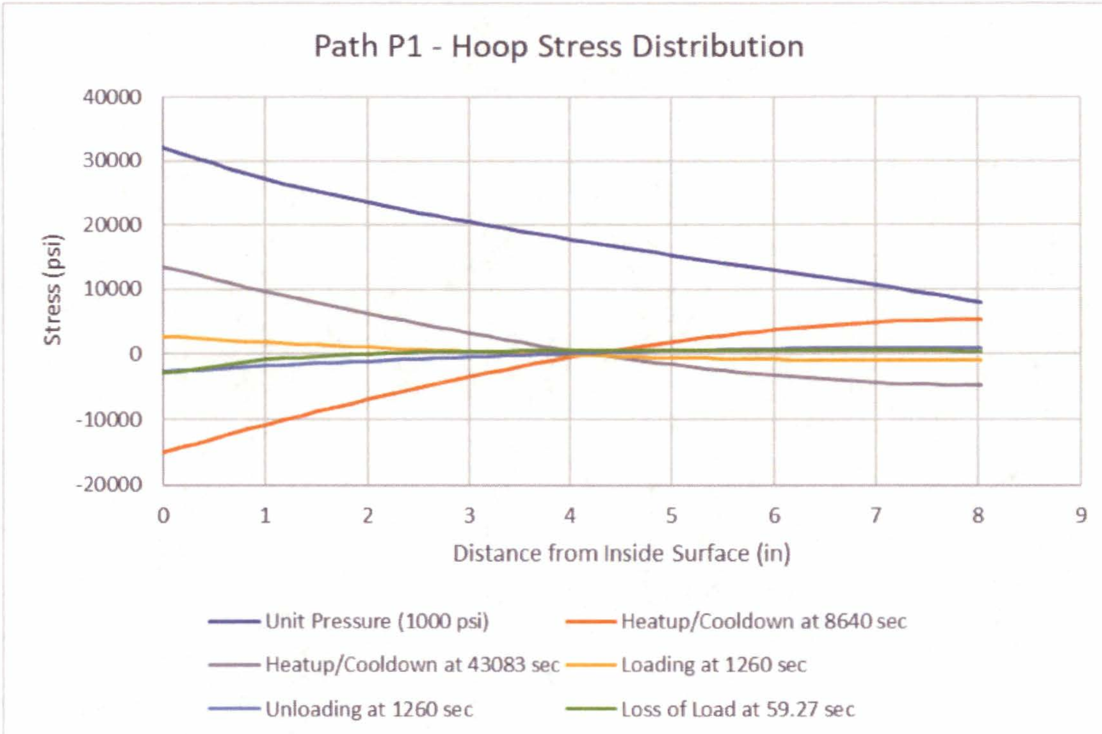
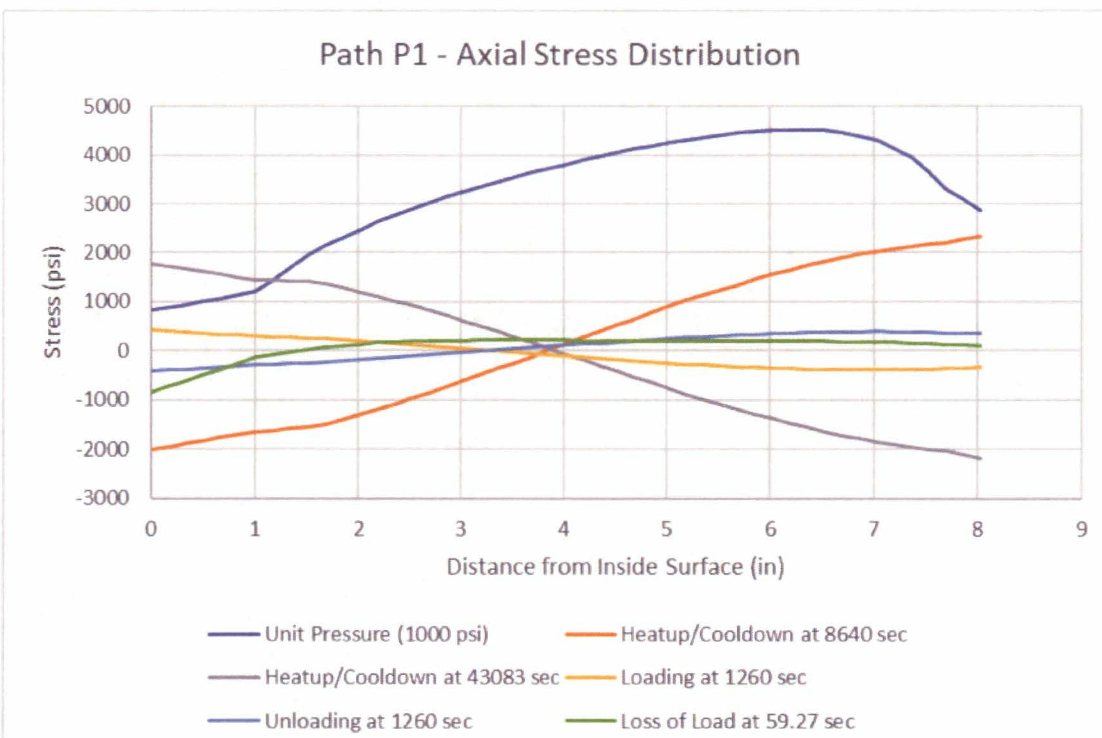


Figure 7-18
Path Locations for PWR SG Main Steam Nozzle: B&W Design

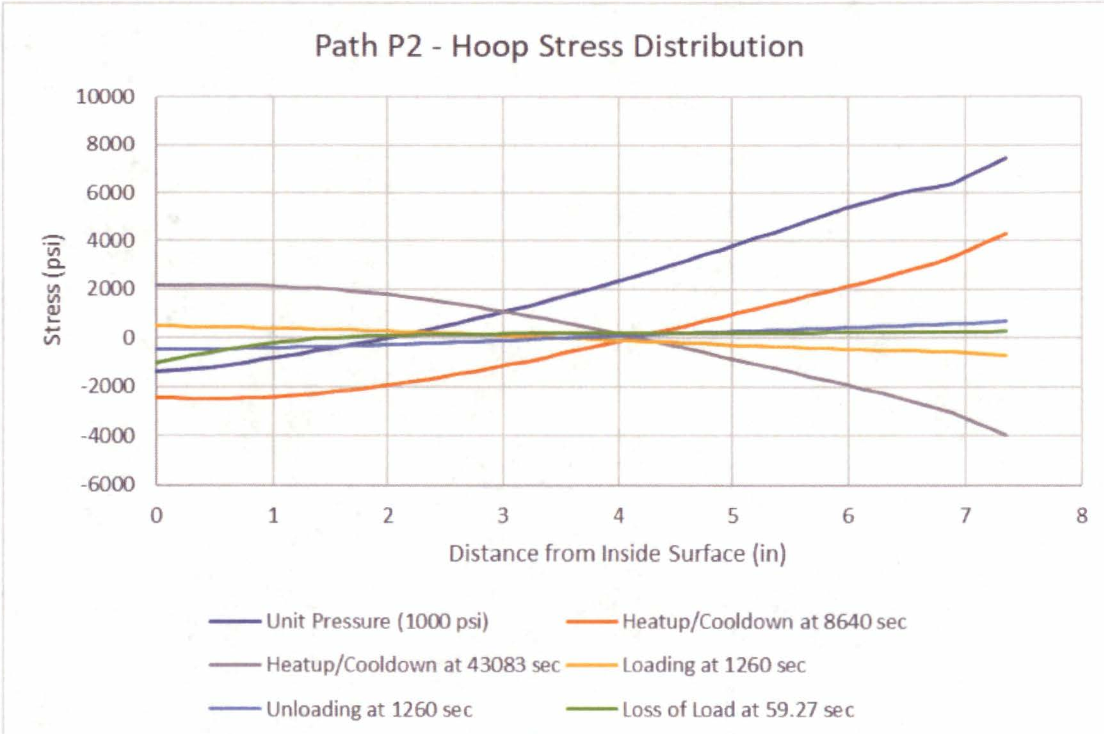


SG Hoop Stress

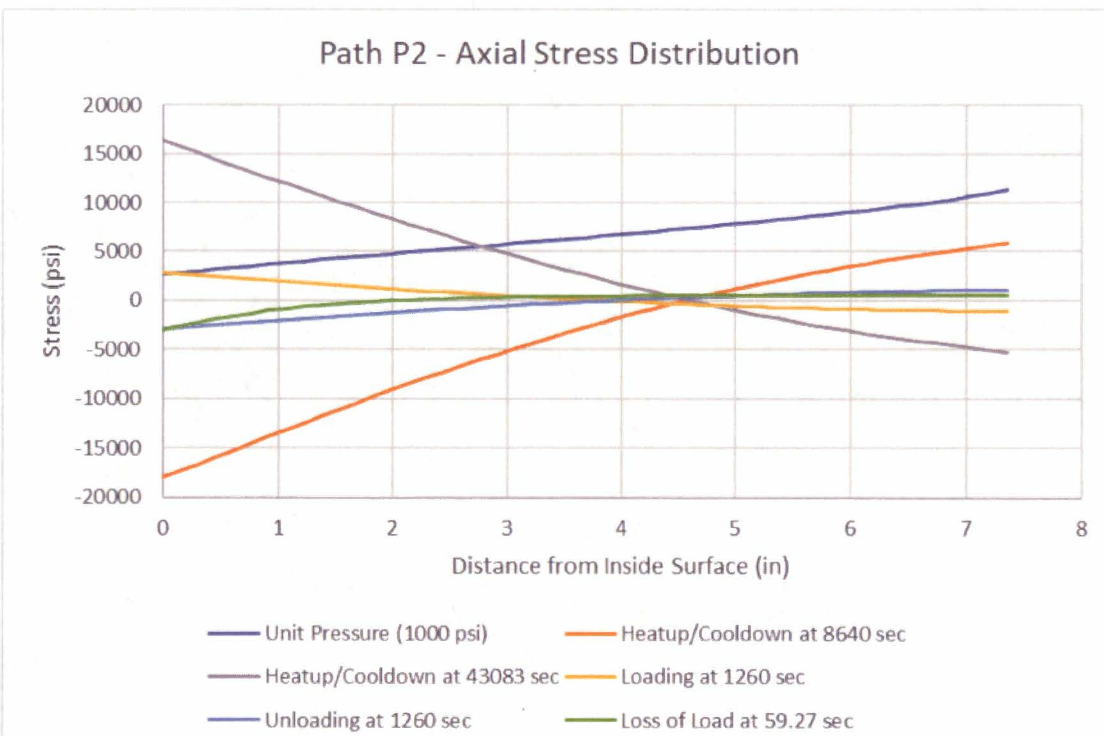


SG Axial Stress

Figure 7-19
Through-Wall Stress Distribution at Path P1 for PWR SG Main Steam Nozzle: B&W Design

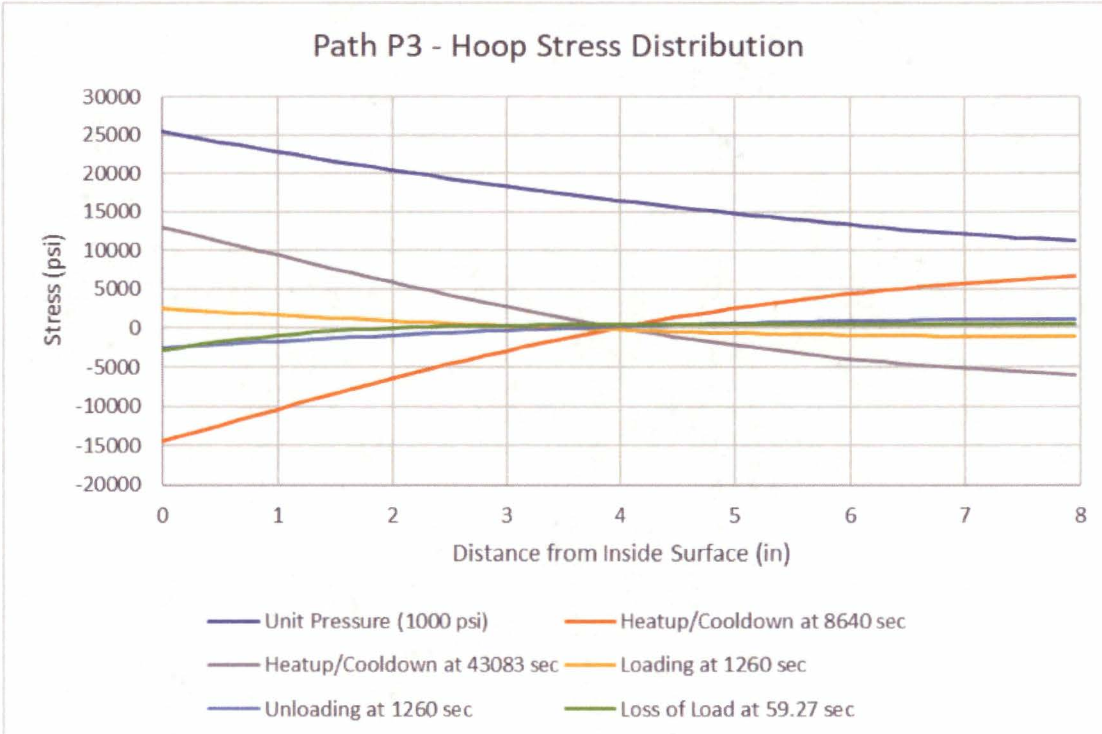


SG Hoop Stress

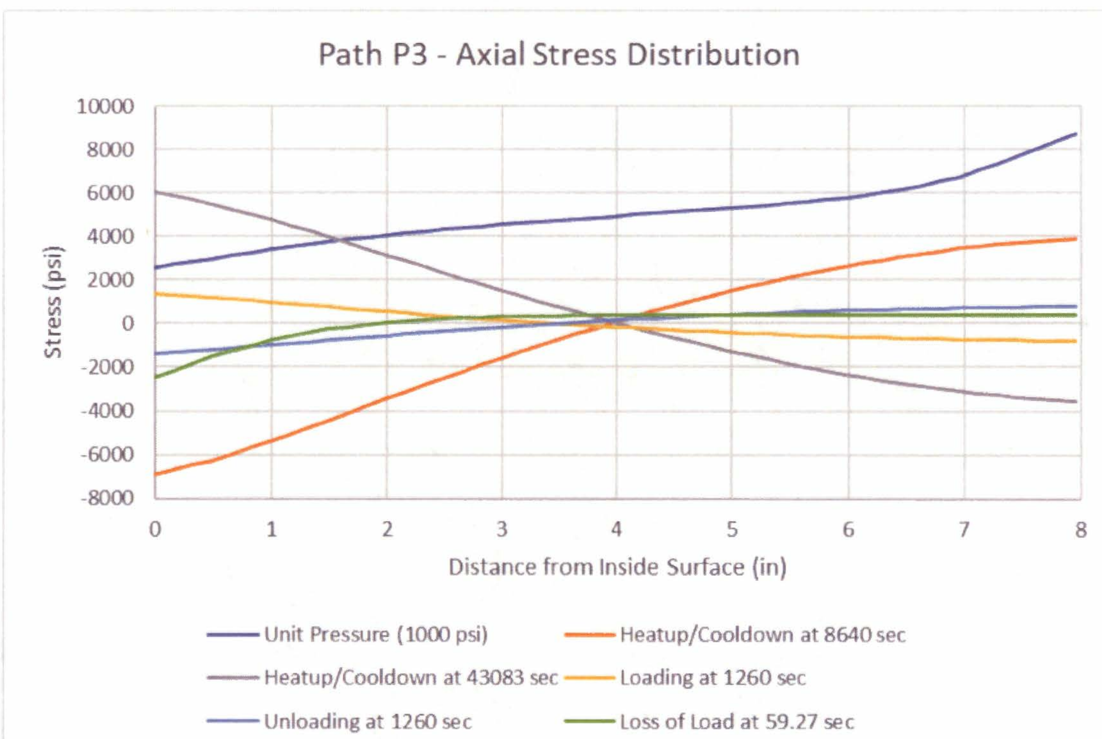


SG Axial Stress

Figure 7-20
Through-Wall Stress Distribution at Path P2 for PWR SG Main Steam Nozzle: B&W Design

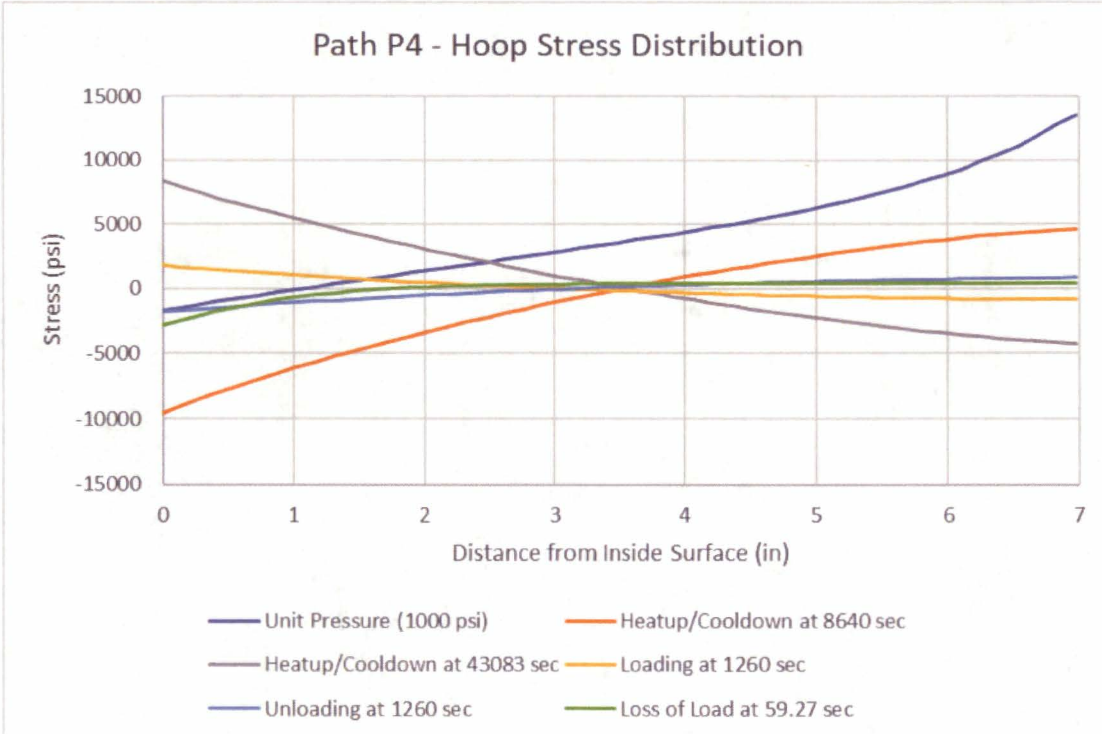


SG Hoop Stress

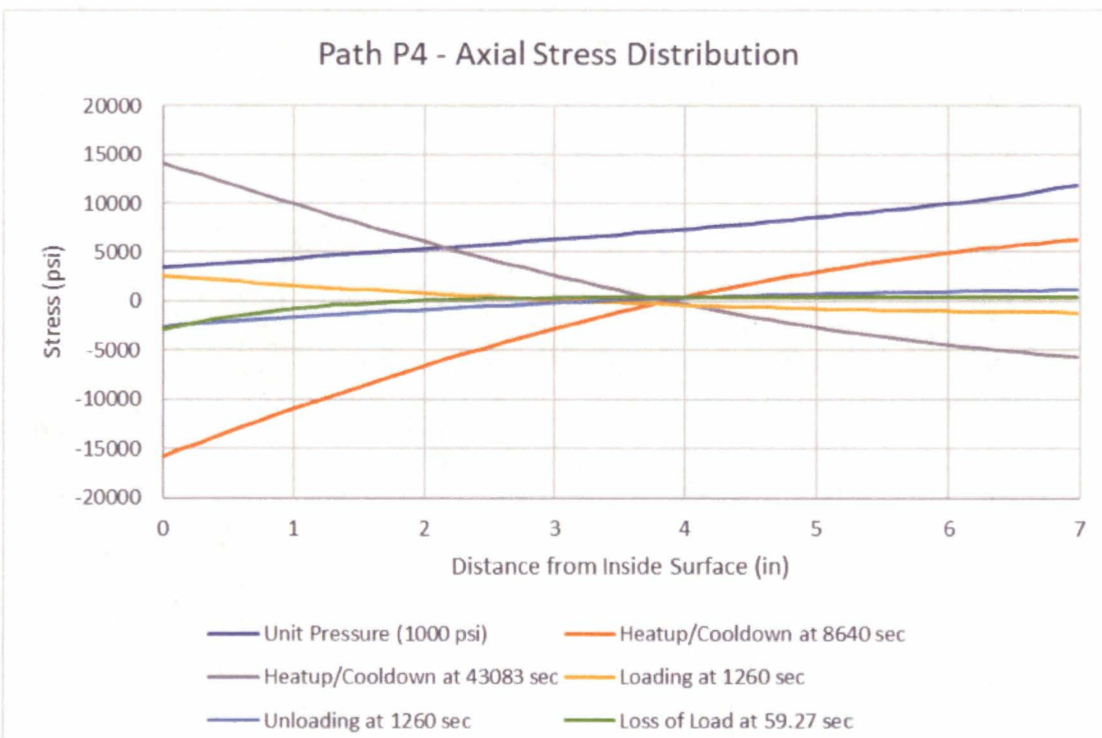


SG Axial Stress

Figure 7-21
Through-Wall Stress Distribution at Path P3 for PWR SG Main Steam Nozzle: B&W Design



SG Hoop Stress



SG Axial Stress

Figure 7-22
Through-Wall Stress Distribution at Path P4 for PWR SG Main Steam Nozzle: B&W Design

7.3 Stress Analysis for PWR SG Feedwater Nozzle

7.3.1 Finite Element Model

A FEM of the PWR SG feedwater nozzle configuration selected for analysis was developed using the ANSYS finite element analysis software package [51], with dimensions shown in Figure 4-10. Because this configuration is not axisymmetric, a 3-D FEM was used in the development of the FEM. The 3-D quarter model was constructed using 8-node structural solid, SOLID45 elements. The thermal equivalent element for the thermal transient analyses is SOLID70. The model is shown in Figure 7-23, and includes a local portion of the SG shell, the feedwater nozzle, and nozzle-to-shell weld. Note that the feedwater nozzle thermal sleeve and the resulting annular region between the thermal sleeve and the feedwater nozzle are not modeled, but their effect on thermal transient behavior is represented by an equivalent heat transfer coefficient. The designation of the materials involved in the model and the associated material properties are discussed in Section 5.1.

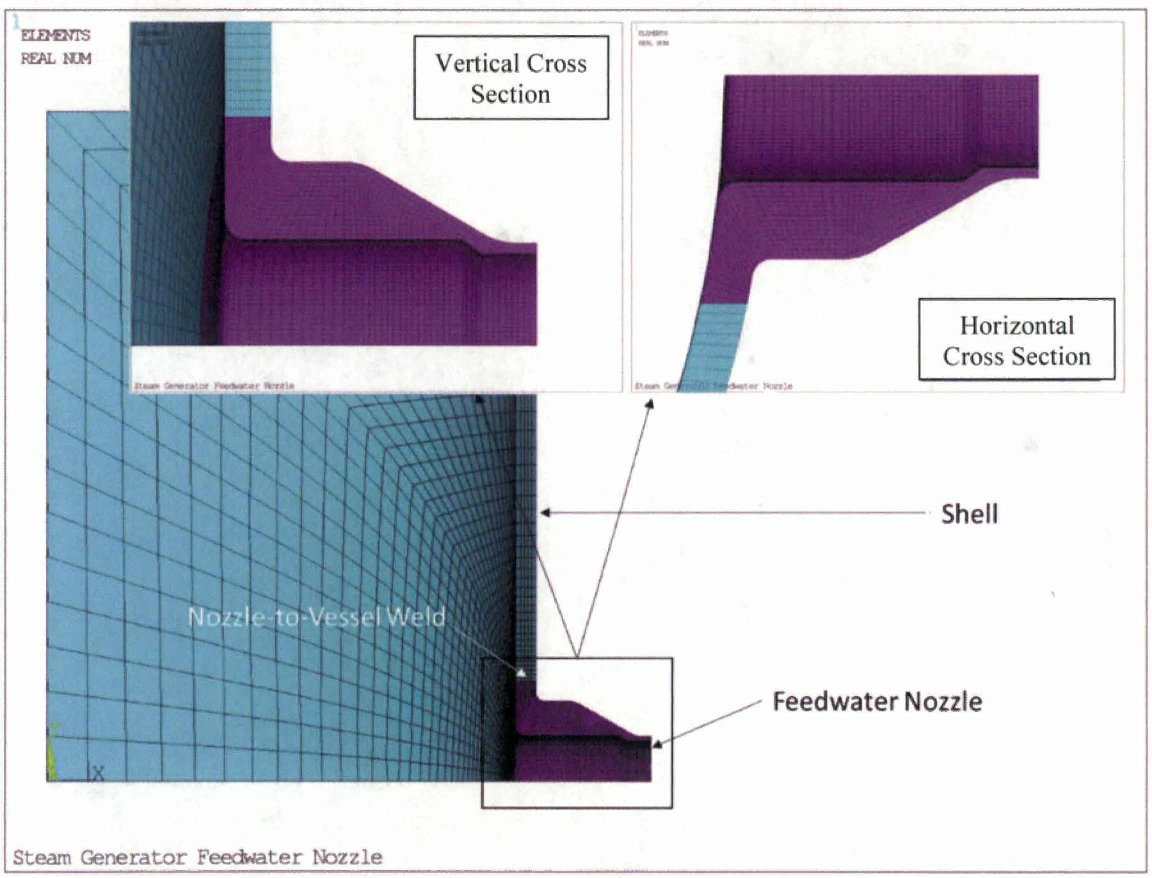


Figure 7-23
3-D Finite Element Model and Mesh for PWR SG Feedwater Nozzle

7.3.2 Pressure/Thermal Stress Analysis

7.3.2.1 Internal Pressure Loading Analysis

A unit internal pressure of 1,000 psi was applied to the interior surfaces of the FEM. The resulting stresses will be scaled up to the transient pressures for use in the fracture mechanics evaluations. An induced end-cap load was applied to the free end of the feedwater nozzle and the upper free end of the shell in the form of tensile axial pressures calculated using Equation 7-1. The applied pressure load and boundary conditions for this case are shown in Figure 7-24.

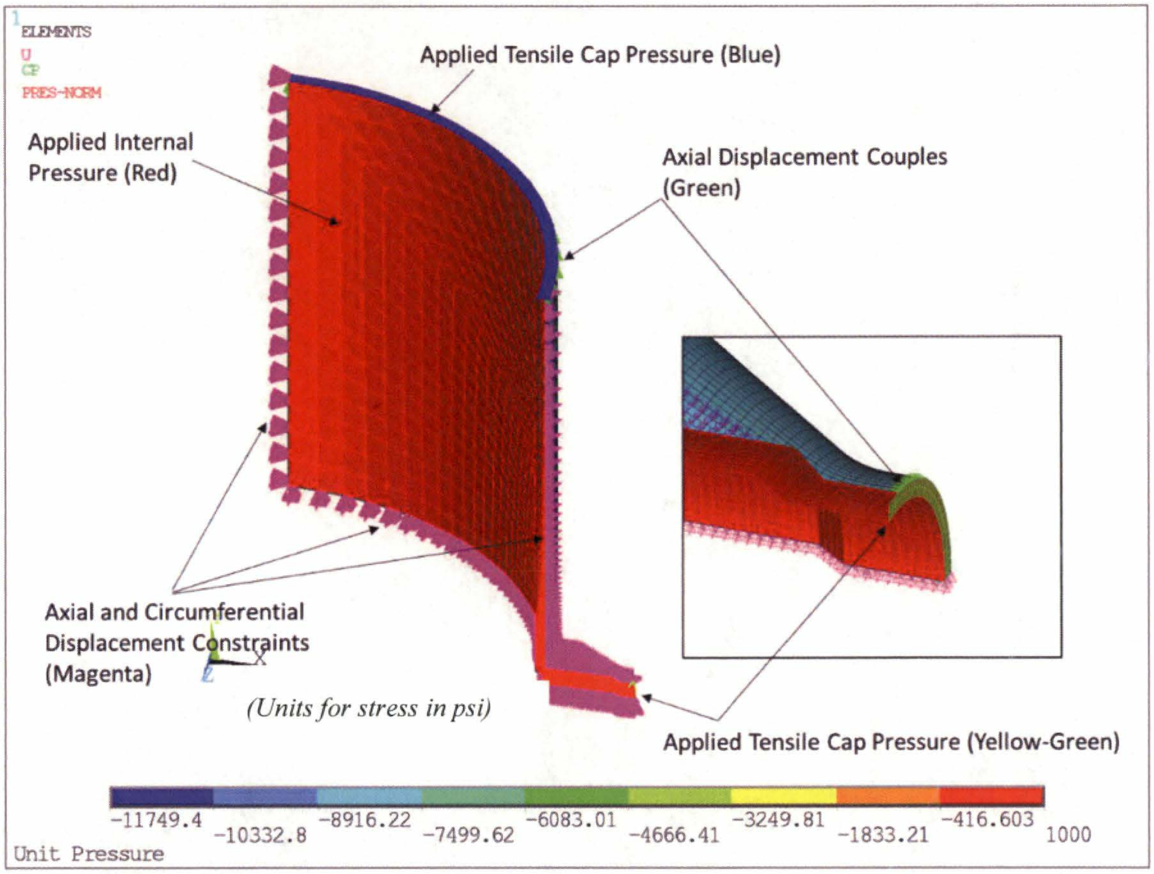
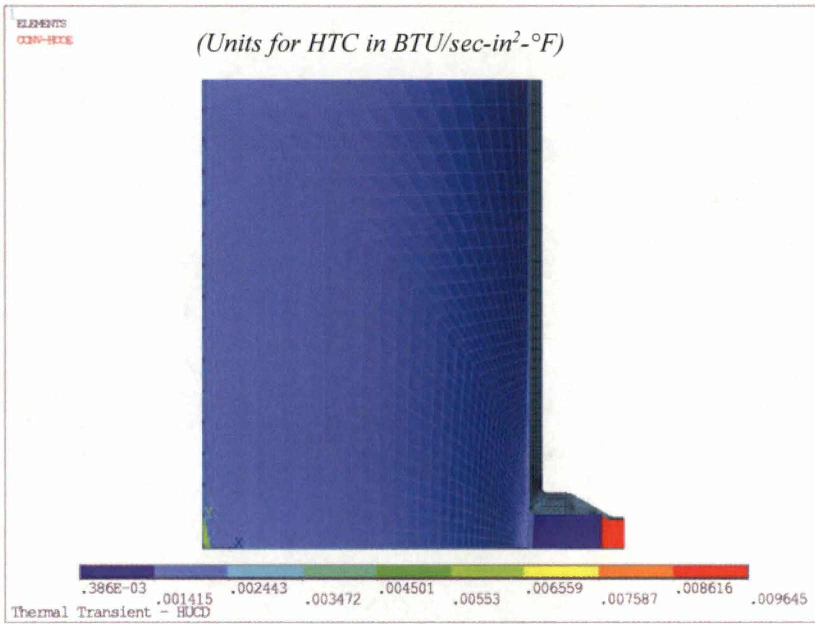


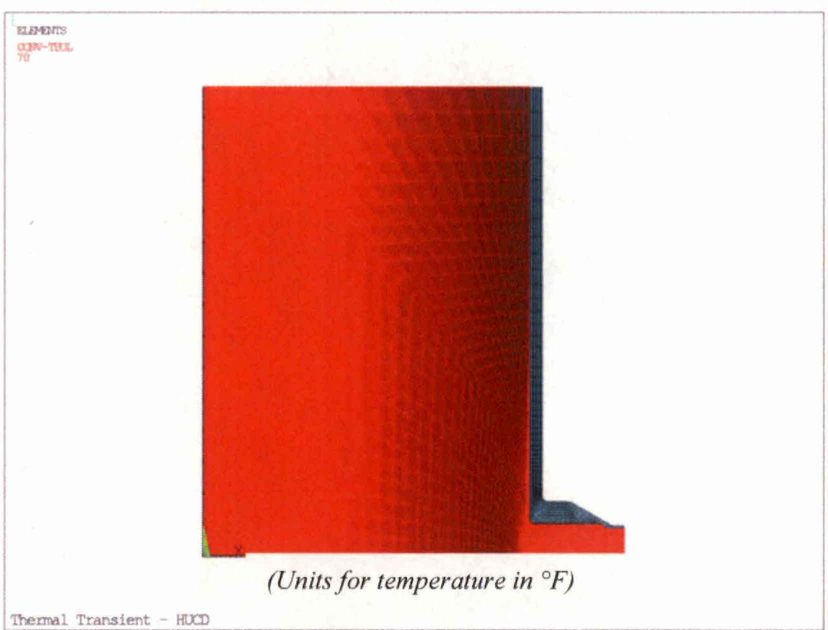
Figure 7-24
Applied Boundary Conditions and Unit Internal Pressure for PWR SG Feedwater Nozzle

7.3.2.2 Thermal Heat Transfer Analyses

The thermal transients discussed in Section 5.2 were applied to the interior surfaces of the nozzle and shell. A nominal heat transfer coefficient of 1,000 BTU/hr-ft^2-°F was applied to the inside surface of the shell, and a conservative coefficient of 5,000 BTU/hr-ft^2-°F was applied to the inside surface of the nozzle. An equivalent overall heat transfer coefficient of 200 BTU/hr-ft^2-°F was applied to the nozzle surface protected by the thermal sleeve, which accounts for the gap between the nozzle and the thermal sleeve, the conductivity of the water in the gap, and the conductivity of a typical 0.495-in. thick thermal sleeve. The nominal shell heat transfer coefficient value was confirmed based on water properties for the incoming nozzle flow and is limited by the material conductivity; the value of 1,000 BTU/hr-ft^2-°F was determined to be conservative. Heat transfer coefficients and temperatures were not applied to the insulated outside surfaces. Figure 7-25 shows representative plots of the thermal loads for the heat-up/cooldown transient applied to the SG feedwater nozzle. The heat-up and cooldown transients were modeled as one single transient with heat-up followed by cooldown, so both events are referred to by the single transient analyzed (heat-up/cooldown).



Heat Transfer Coefficient



Bulk Temperature

Figure 7-25
Applied Thermal Boundary Conditions for Thermal Transient Analyses for PWR SG Feedwater Nozzle (heat-up/cooldown transient shown; loads applied at time = 46,080 seconds [end of cooldown])

7.3.2.3 Thermal Stress Analyses

Symmetric boundary conditions were applied to the bottom and circumferential free ends of the shell; axial displacement couples were applied on the free end of the nozzle and the upper free end of the shell. The reference temperature for the thermal strain calculation is assumed to be 70°F. Figure 7-26 shows a plot of the boundary conditions applied for the thermal stress analyses.

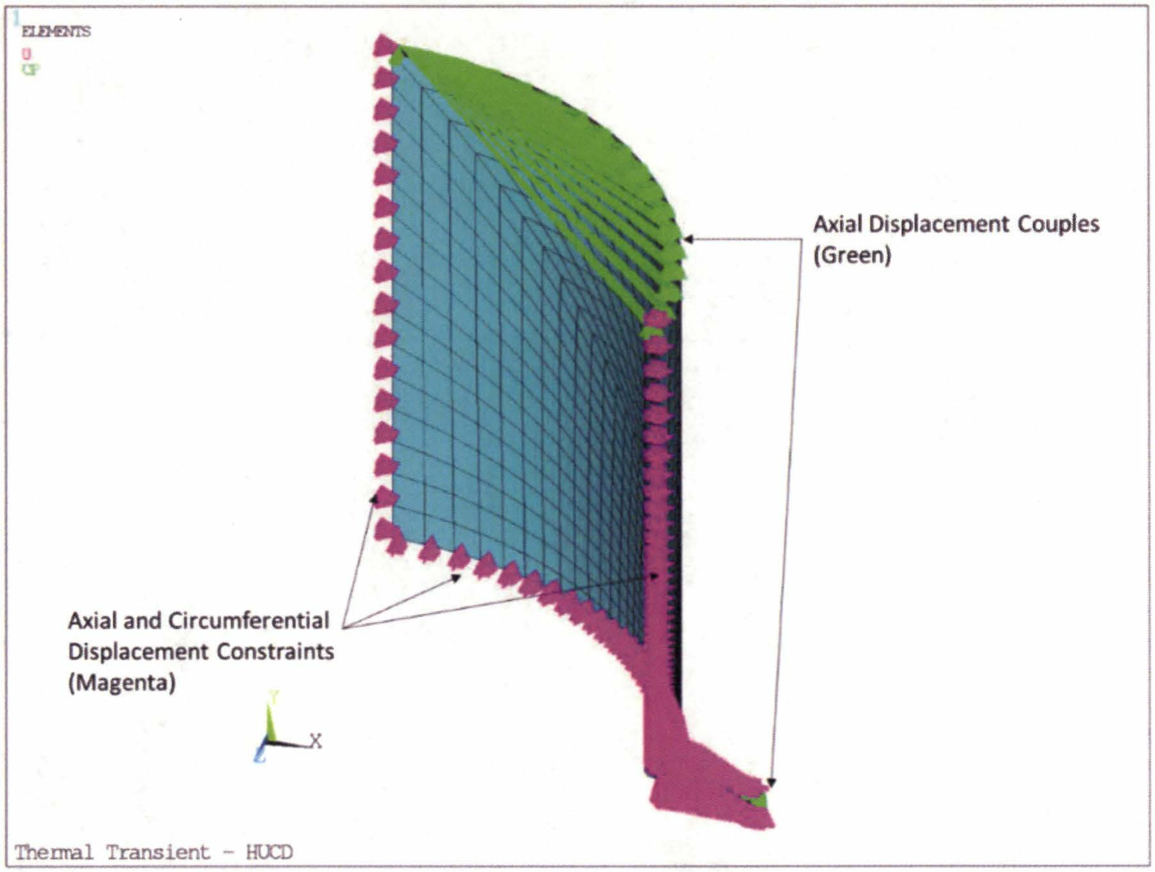
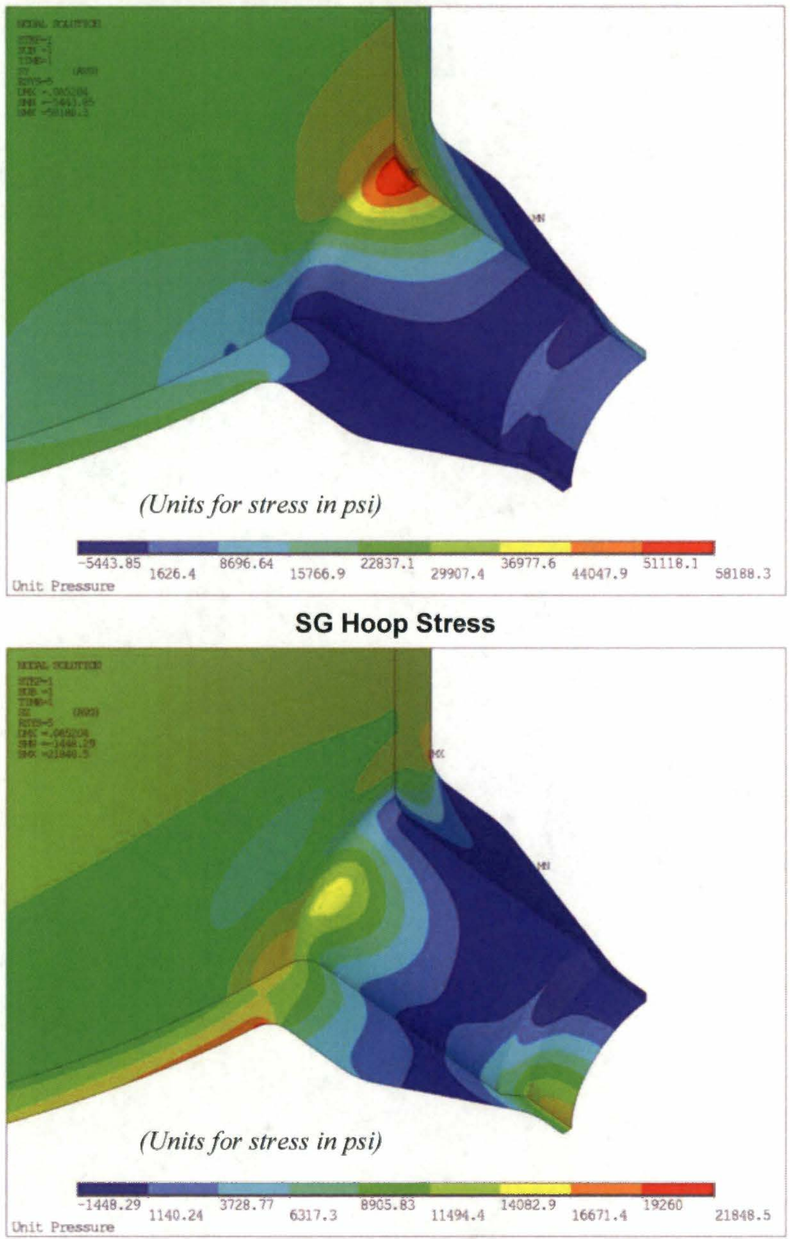


Figure 7-26
Applied Mechanical Boundary Conditions for Thermal Stress Analyses for PWR SG Feedwater Nozzle

7.3.3 Stress Analysis Results

The hoop and axial stress contour plots for internal pressure are shown in Figure 7-27. Representative temperature and stress contour plots for the heat-up/cooldown transient are shown in Figures 7-28 and 7-29, respectively. The times shown in Figures 7-28 and 7-29 are when the maximum stress intensity occurs during each transient. Figure 7-30 is a temperature contour plot for the loss of power event at a time of 37,440 seconds. As seen in the figure, the temperature at the nozzle corner section is greater than 300°F. Because this plot is at the final time of the transient when temperature is lowest, it follows that the nozzle corner section always remains above 300°F during the entire transient. Therefore, this temperature is used in Section 8.2.2.7 to justify the use of a higher critical stress intensity factor for this transient. Figure 7-31 shows the path locations where stresses were extracted. Paths P1 and P2 were chosen for the nozzle inside radius location, and Paths P3 and P4 were chosen for the nozzle-to-shell weld location. All stresses were extracted in a cylindrical coordinate system with the Y-direction coinciding with the hoop direction of the SG shell and the Z-direction coinciding with the axial direction of the SG shell. Through-wall stress distributions for Paths P1 through P4 are shown in Figures 7-32 through 7-35, respectively. In these figures, the thermal stresses shown are for the time during each transient when the thermal stress was a maximum. As can be seen from Figures 7-32 through 7-35, in general, the pressure stress is usually the larger stress. As shown in the plots for Paths P1 and P3, the pressure stress is much larger than the thermal stresses—confirming the use of (PR/t) as a parameter in the subsequent sensitivity analyses.



SG Hoop Stress

SG Axial Stress

Figure 7-27
Stress Contours due to Unit Internal Pressure for PWR SG Feedwater Nozzle

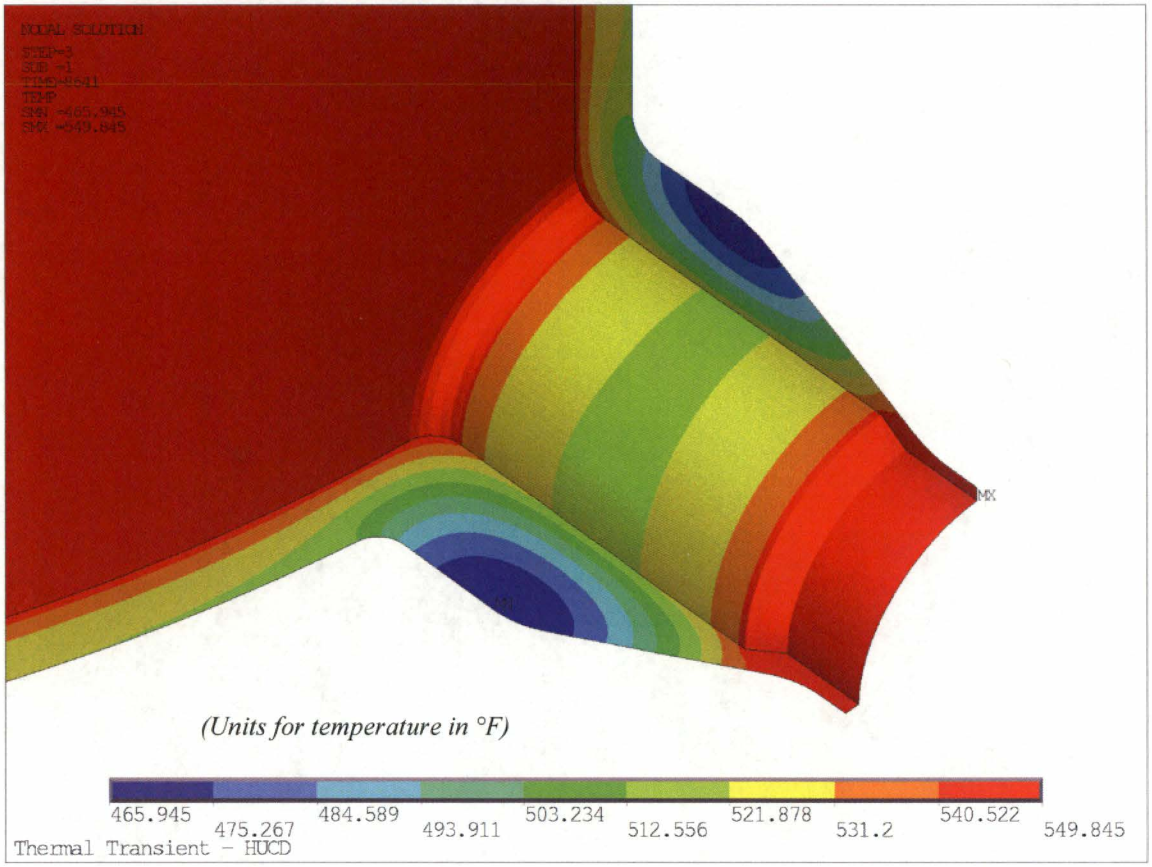
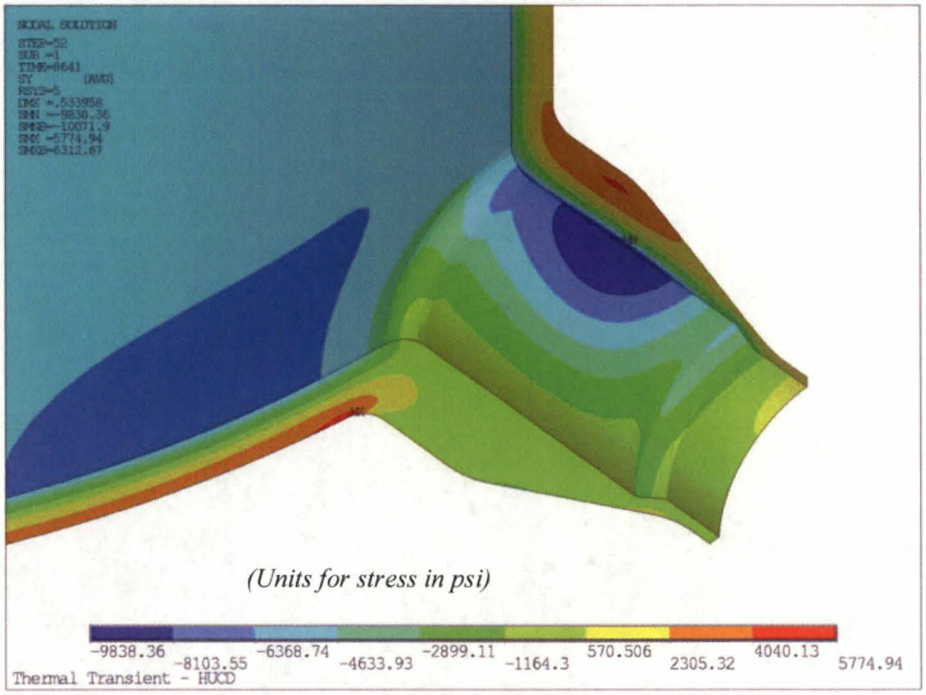
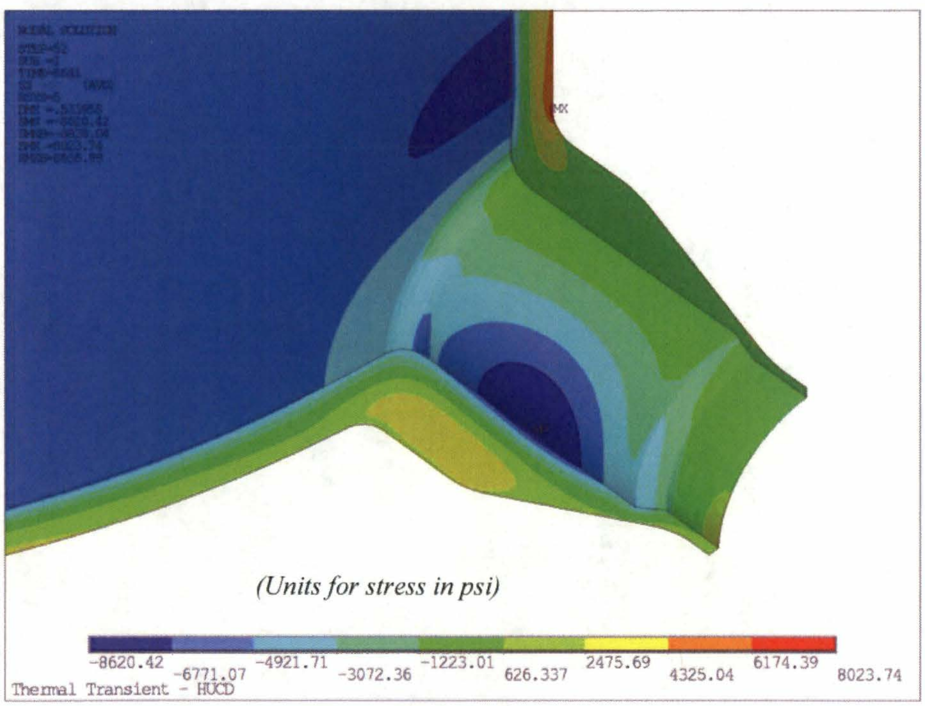


Figure 7-28
Temperature Contour of Heat-Up/Cooldown Transient (time = 8,641 seconds, end of heat-up) for PWR SG Feedwater Nozzle



SG Hoop Stress



SG Axial Stress

Figure 7-29
Stress Contours of Heat-Up/Cooldown Transient (time = 8,641 seconds, end of heat-up) for PWR SG Feedwater Nozzle

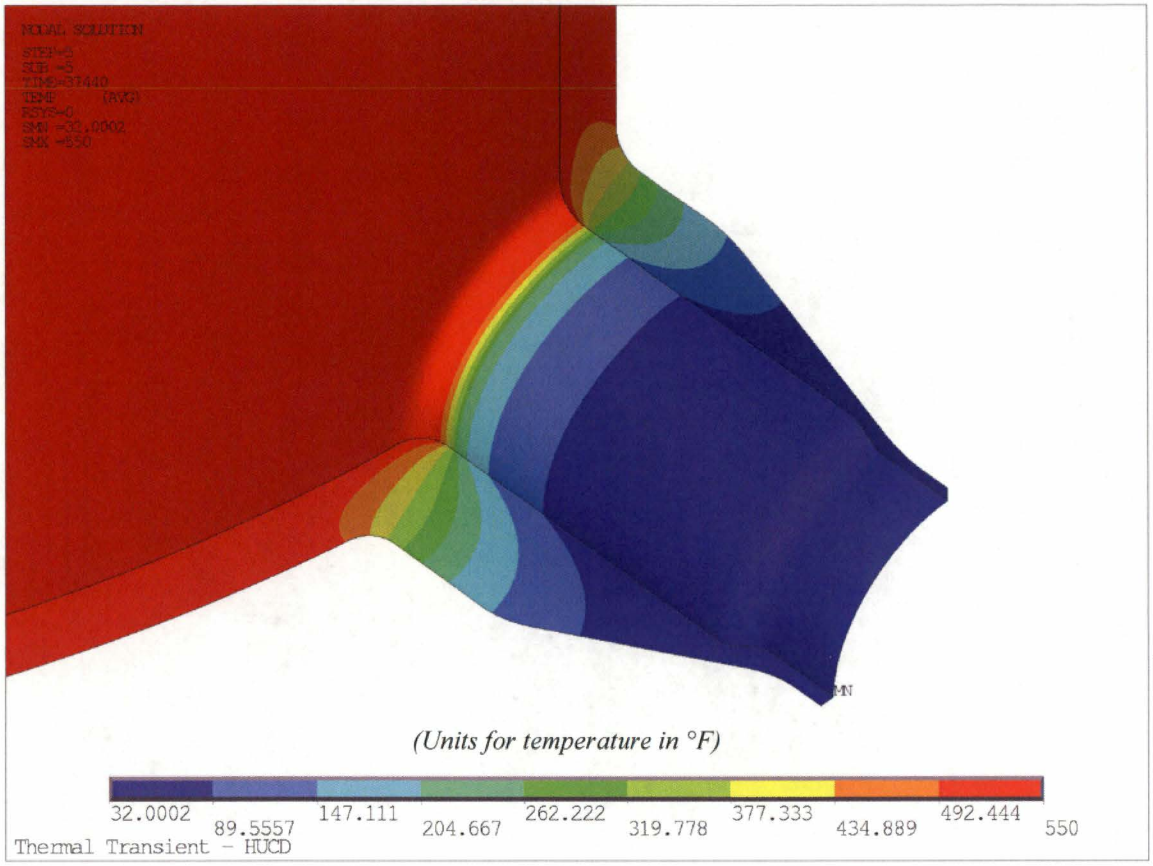


Figure 7-30
Temperature Contour of Loss of Power Transient at 37,440 seconds for PWR SG Feedwater Nozzle

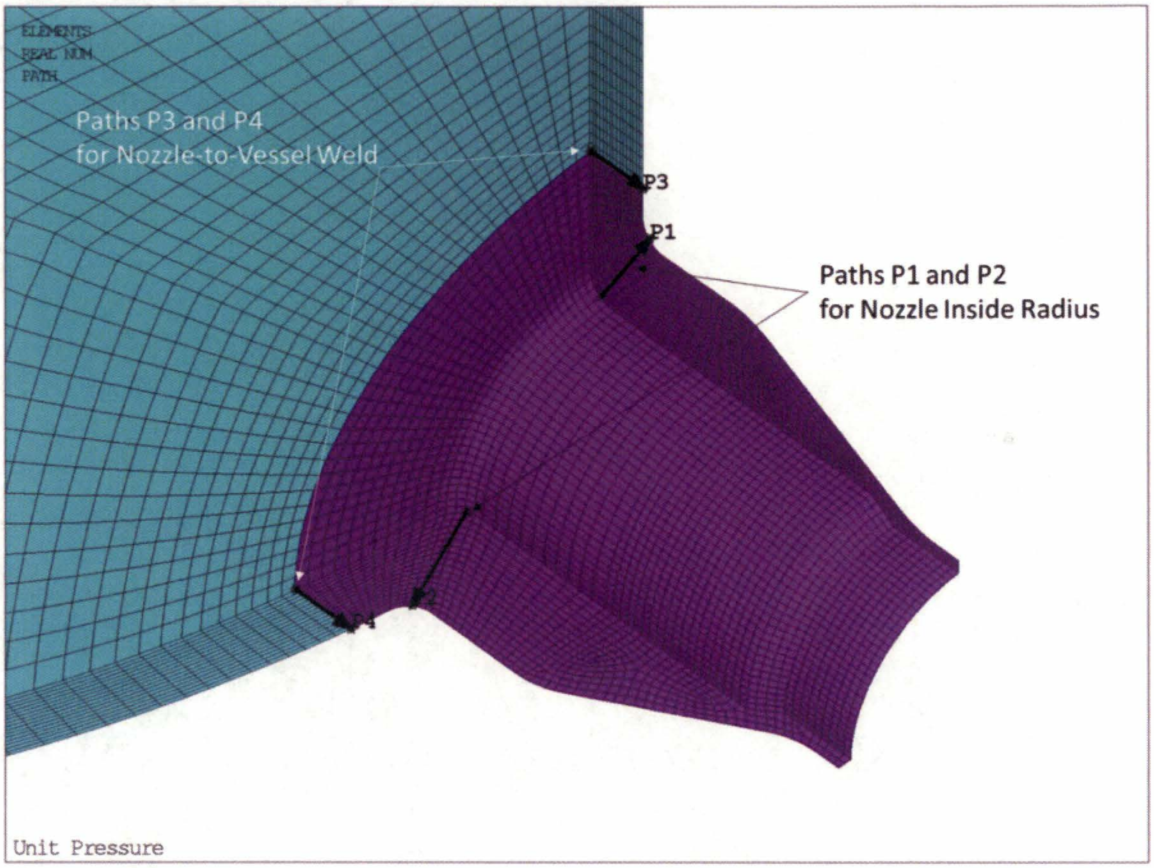
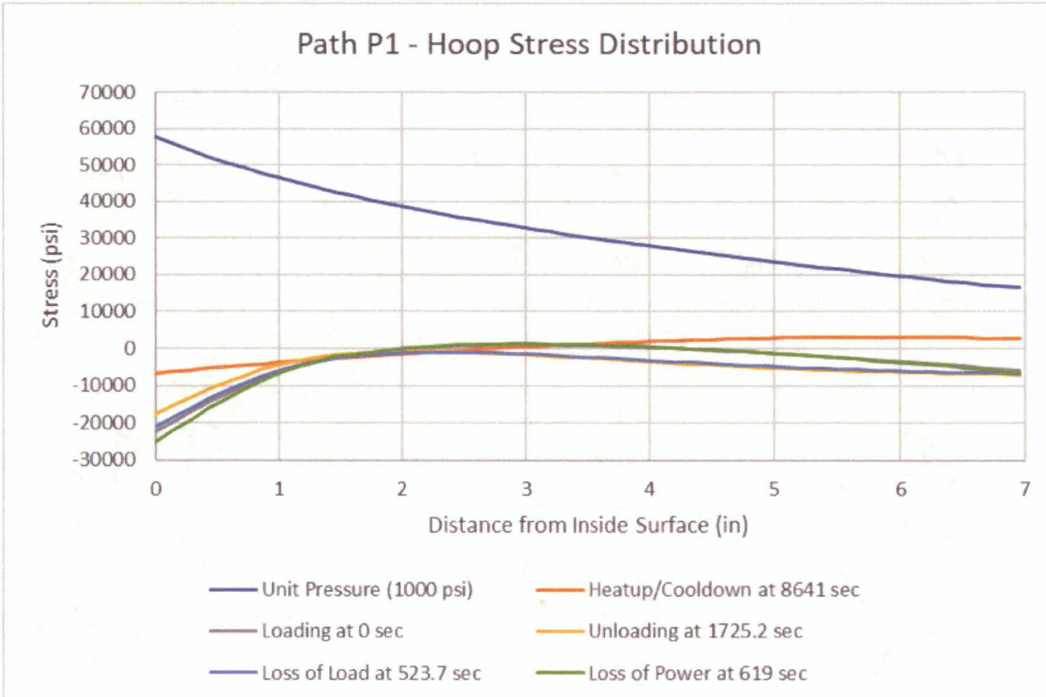
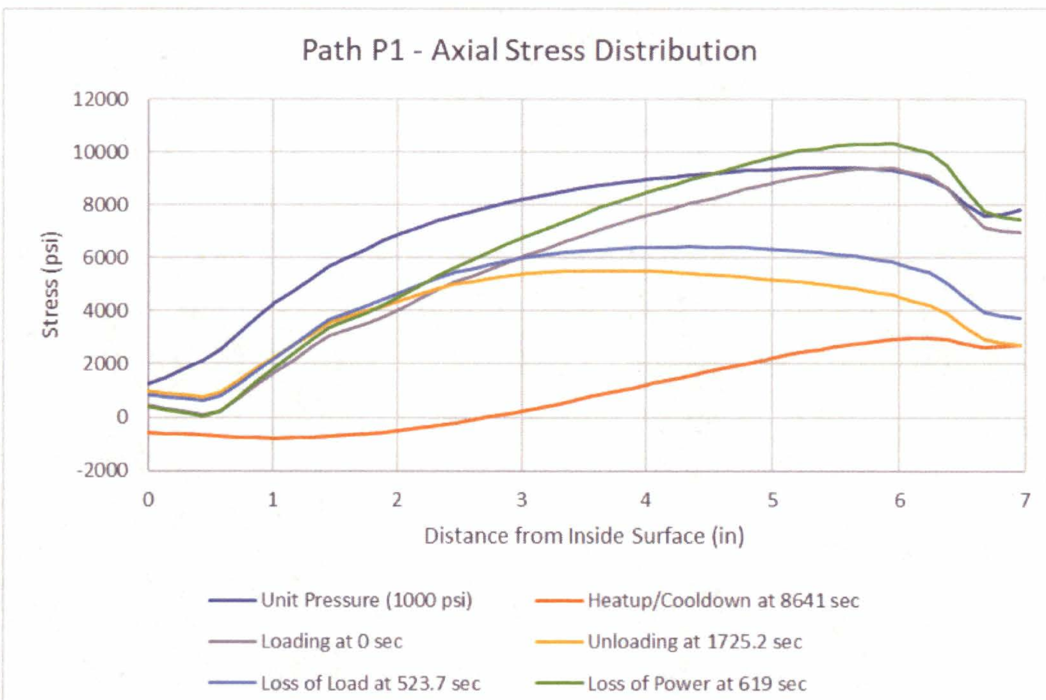


Figure 7-31
Path Locations for PWR SG Feedwater Nozzle

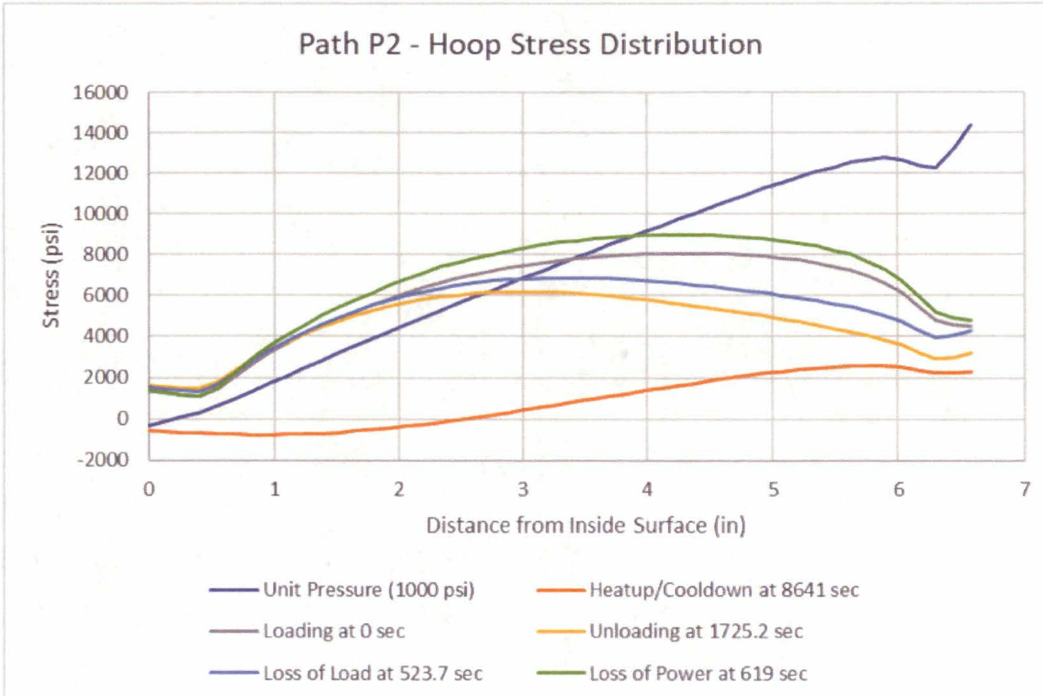


SG Hoop Stress

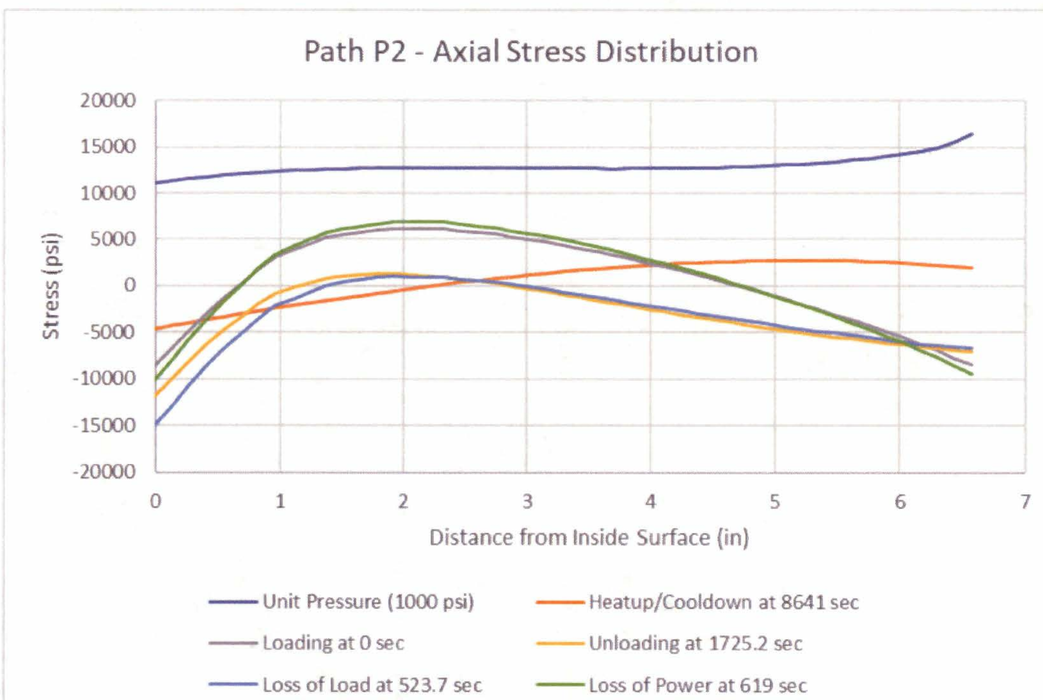


SG Axial Stress

Figure 7-32
Through-Wall Stress Distribution at Path P1 for PWR SG Feedwater Nozzle

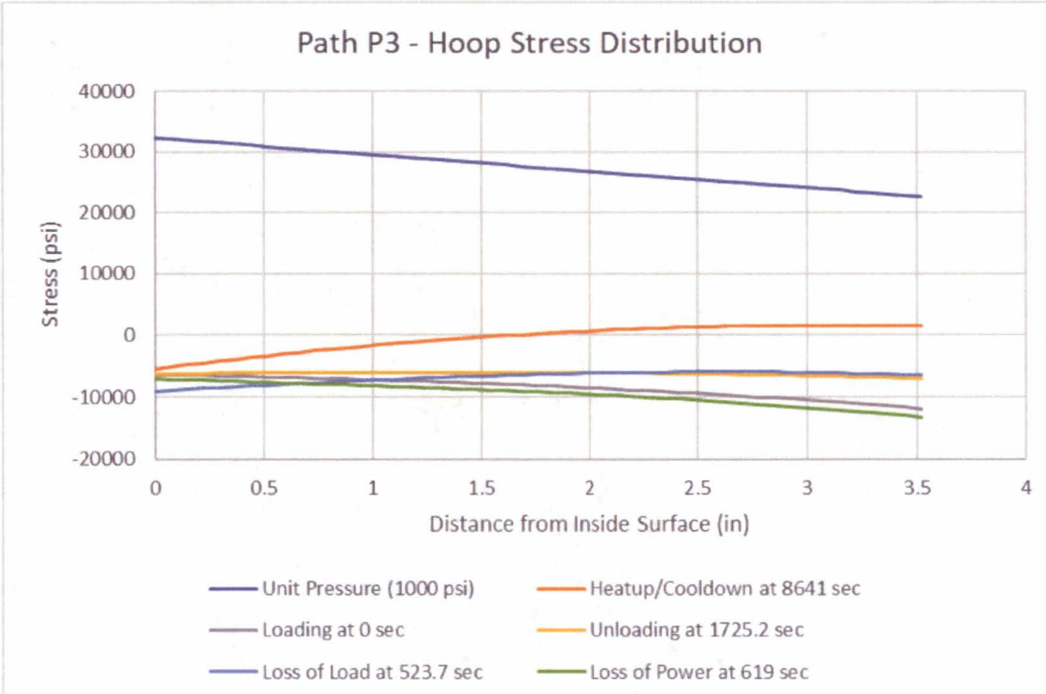


SG Hoop Stress

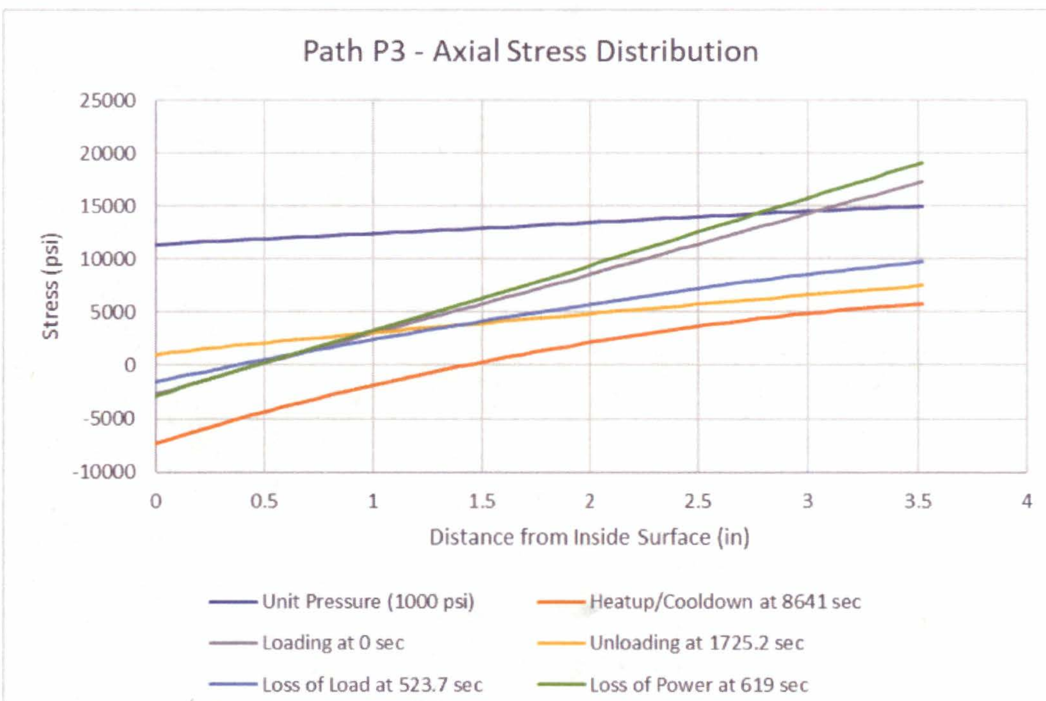


SG Axial Stress

Figure 7-33
Through-Wall Stress Distribution at Path P2 for PWR SG Feedwater Nozzle

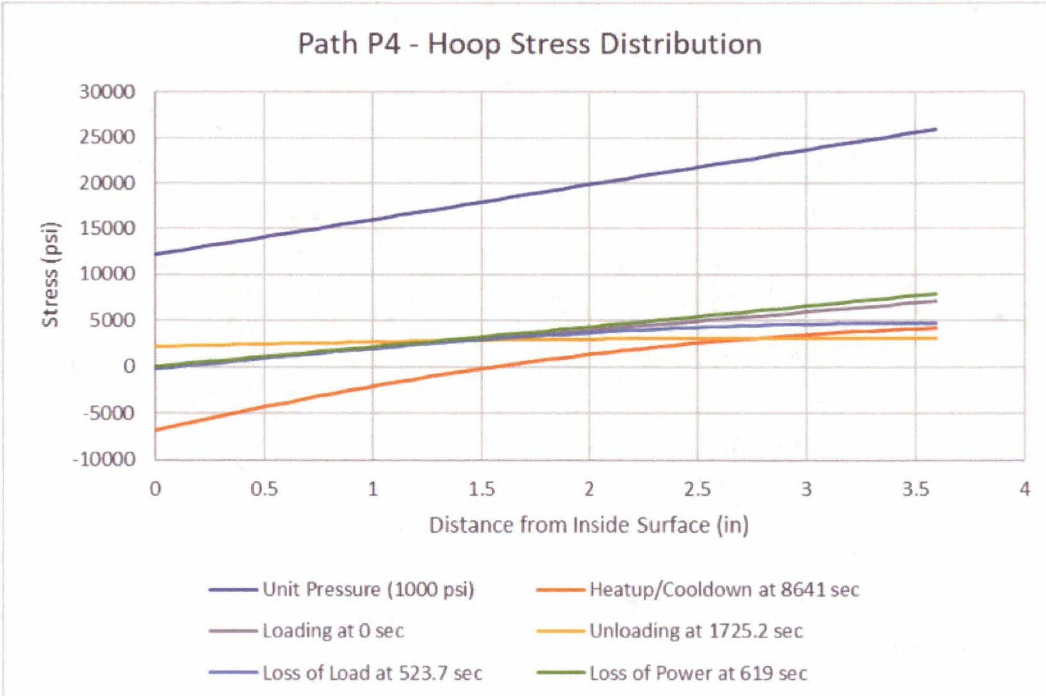


SG Hoop Stress

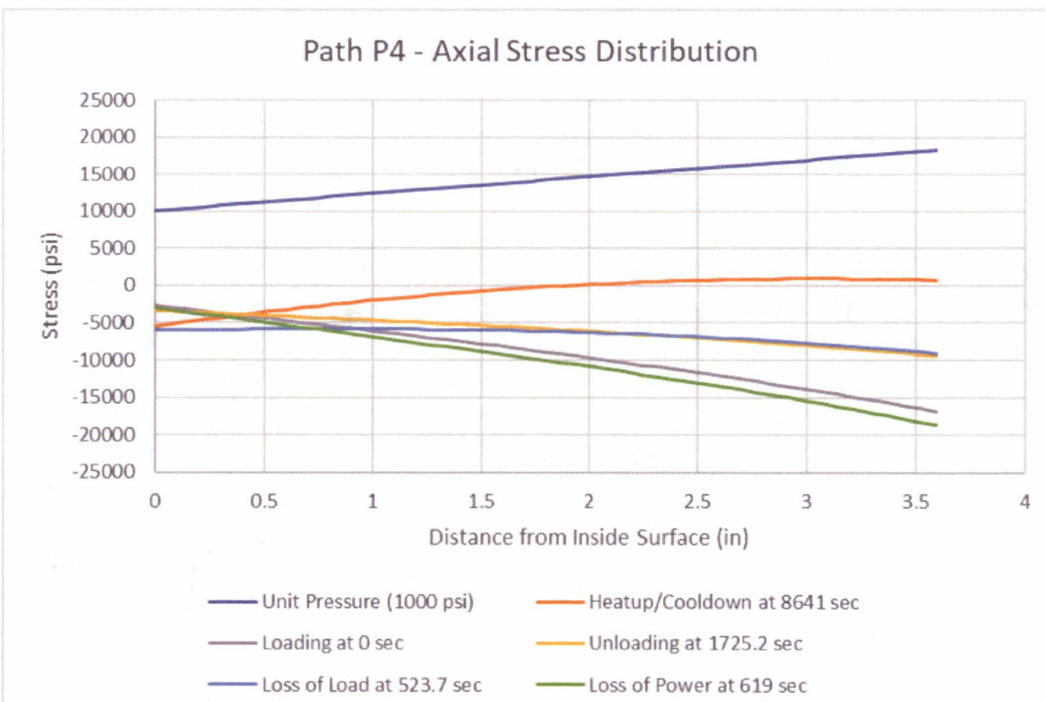


SG Axial Stress

Figure 7-34
Through-Wall Stress Distribution at Path P3 for PWR SG Feedwater Nozzle



SG Hoop Stress



SG Axial Stress

Figure 7-35
Through-Wall Stress Distribution at Path P4 for PWR SG Feedwater Nozzle

7.4 Stress Comparisons for Main Steam Nozzle Designs

In Section 4.5.1, two main steam nozzles were selected for evaluation (one for the B&W design and one for the CE and Westinghouse designs). As noted in that section, the B&W designs are all similar; however, there are variations in Westinghouse main steam nozzle geometries. Therefore, two additional designs were modeled to evaluate differences in through-wall stresses compared to the Westinghouse and CE nozzle selected for analysis, hereinafter referred to as the *original analysis nozzle*. The first geometry featured a tapered nozzle reinforcement and is referred to here as the *tapered nozzle*. The 2-D FEM for this design is shown in Figure 7-36. The second nozzle geometry evaluated has a reduced amount of nozzle reinforcement and is referred to as the *reduced reinforcement nozzle*. The 2-D FEM for this design is shown in Figure 7-38. As identified in Section 7.1 for the original analysis nozzle, the dominant load is pressure. Therefore, to quantify the stress variation for the path stresses of these two additional nozzle designs, a unit pressure load of 1,000 psig was evaluated consistent with the pressure load applied to the original analysis nozzle in Section 7.1. As with the prior 2-D pressure stress analysis, two paths were extracted, where Path 1 is the nozzle blend radius path and Path 2 is the nozzle-to-shell weld path. The stress contour plots for the two different nozzle geometries evaluated in this section are shown in Figures 7-37 and 7-39.

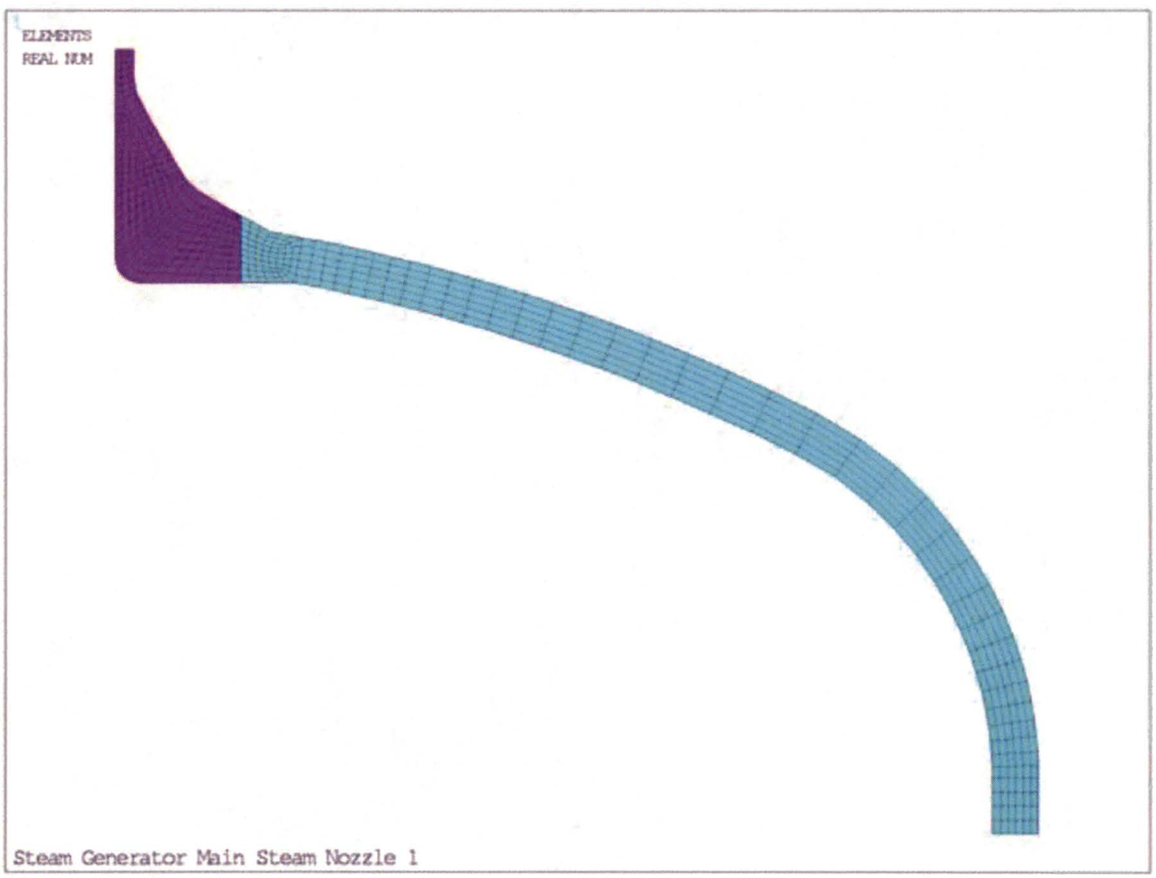


Figure 7-36
Tapered Main Steam Nozzle Finite Element Model

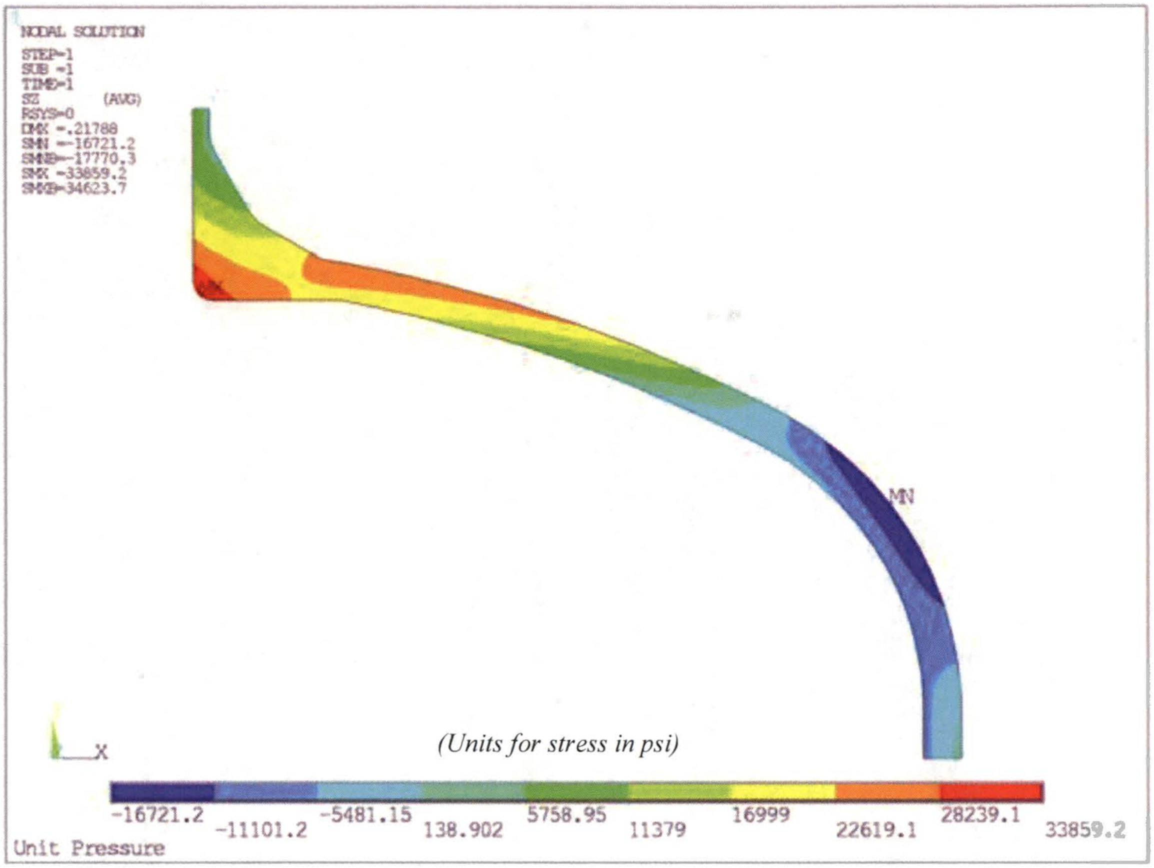


Figure 7-37
Tapered Main Steam Nozzle Pressure Stress Results (Z-direction, nozzle hoop)

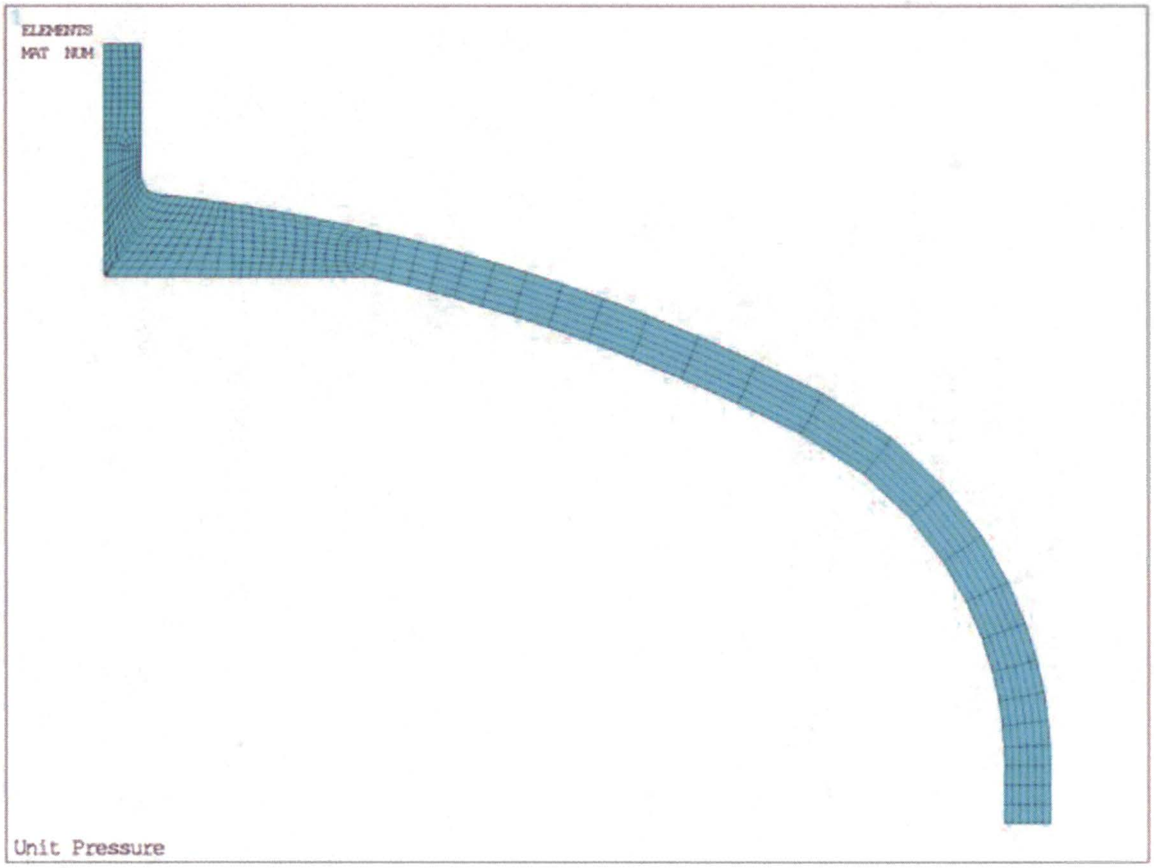


Figure 7-38
Reduced Reinforcement Main Steam Nozzle Finite Element Model

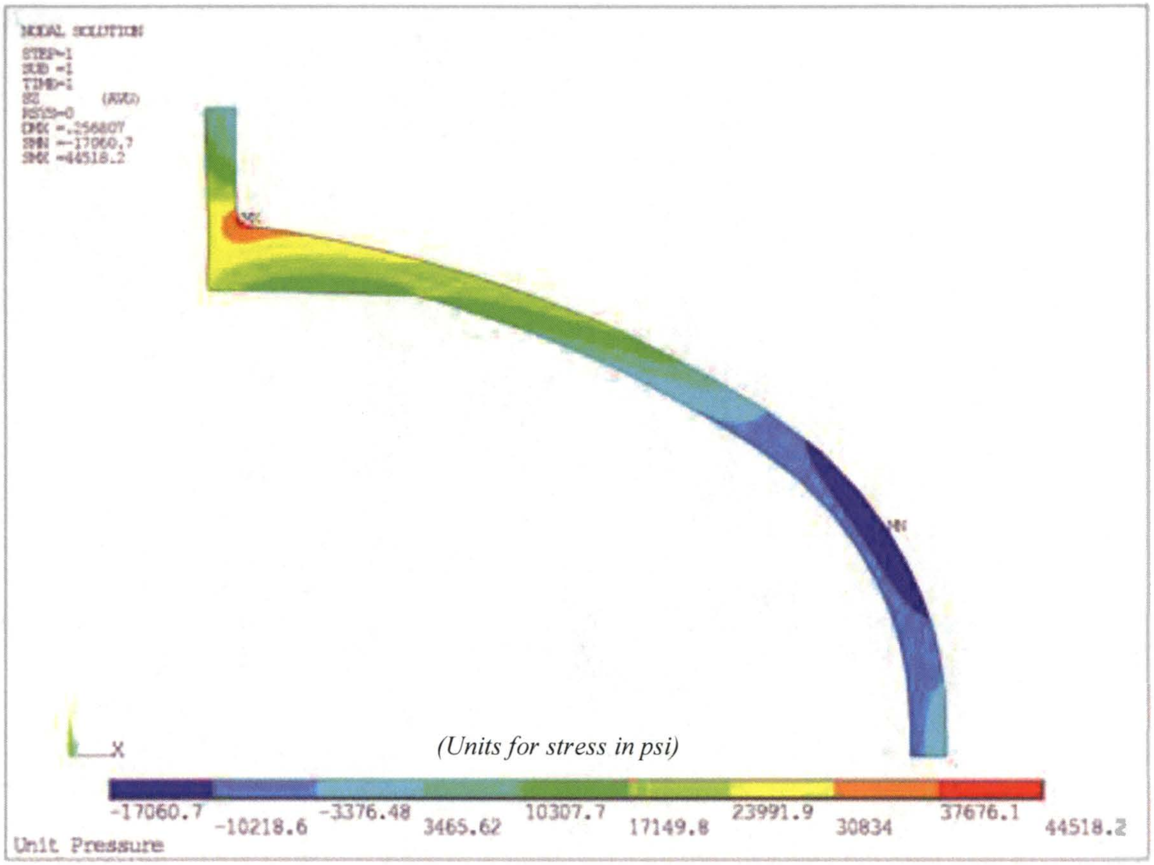


Figure 7-39
Reduced Reinforcement Main Steam Nozzle Pressure Stress Results (Z-direction, nozzle hoop)

For comparison of the stresses, the through-wall stresses were extracted for the two defined paths and plotted up to 80% of the wall thickness (because 80% was assumed as the depth for leakage in Section 8 due to probabilistic fracture mechanics [PFM] software limitations). The nozzle hoop (Z-direction) stresses for the two nozzles evaluated in this section and the original analysis nozzle evaluated in Section 7.1 are shown in Figure 7-40. The hoop direction was selected for the comparison because the nozzle hoop stress was used with the nozzle corner crack stress intensity factor solution (stress normal to the postulated crack).

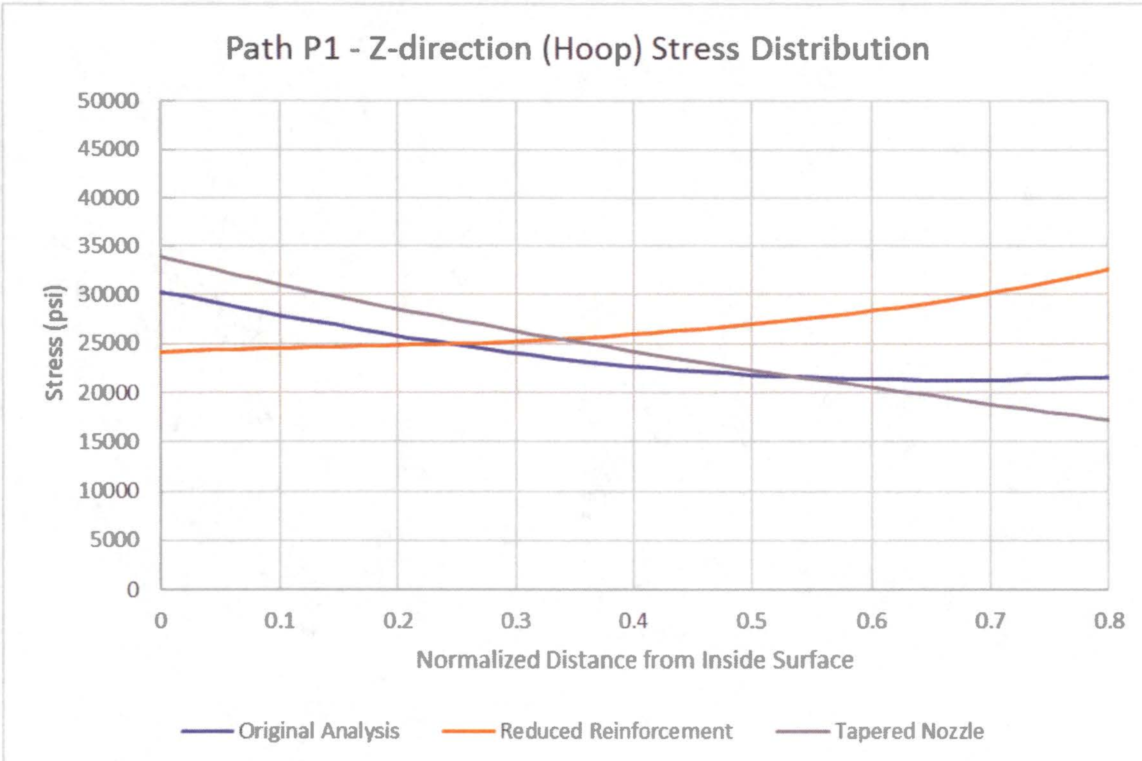


Figure 7-40
Path 1 Through-Wall Stress Distributions

Stress ratios were then plotted up to 80% of the wall thickness to compare the increase or decrease in stress relative to the original analysis nozzle. The Path 1 stress ratios are shown in Figure 7-41. As shown in Figure 7-41, the maximum stress ratio is close to 1.5 at a depth of 80% through-wall for the reduced reinforcement nozzle. When comparing the tapered nozzle to the original analysis nozzle, the differences are no more than 10%.

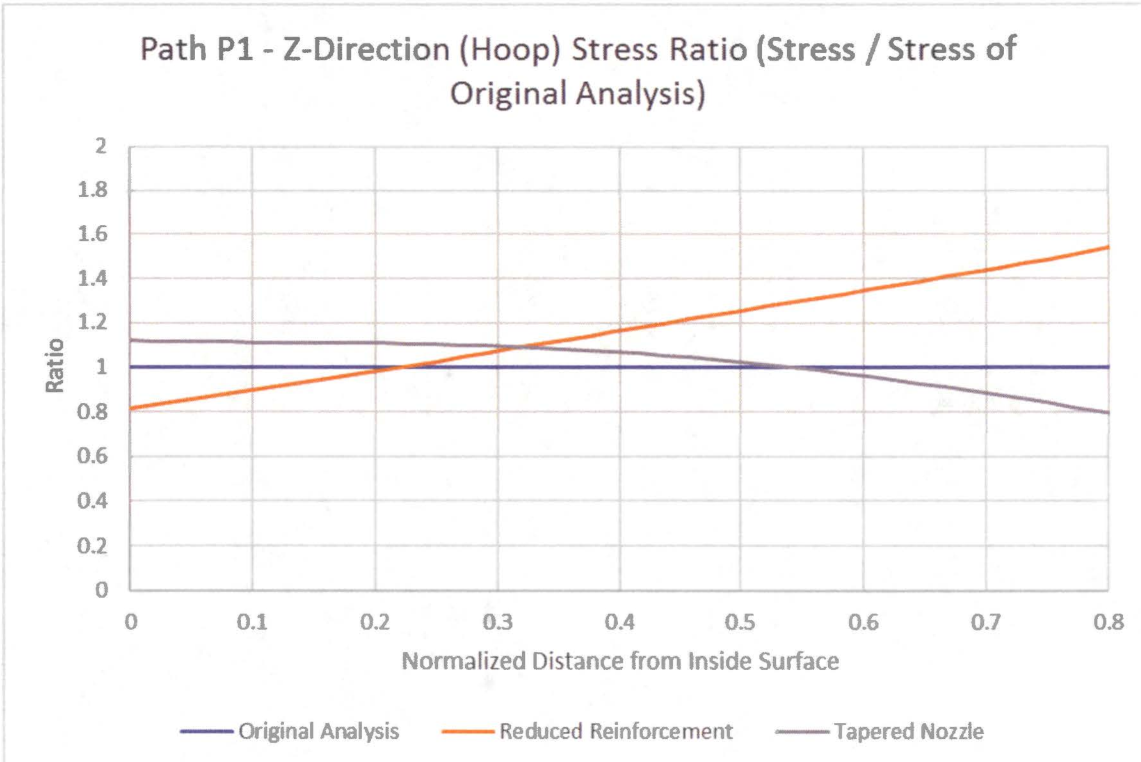


Figure 7-41
Path 1 Through-Wall Stress Distribution Comparisons (ratios)

For the nozzle-to-shell weld (Path 2), the through-thickness stress results are shown in Figure 7-42 for the nozzle-to-head meridional (X) direction (stress component used in the fracture mechanics evaluation). Because the stress values are close to zero near the inside surface, a ratio of the stresses would yield large variations—so stress ratio plots were not used to evaluate this path. Instead, stress differences among the three nozzle designs were plotted, as shown in Figure 7-43. Based on the stress differences shown in this figure, the difference in stress is less than 10 ksi at any point along the stress path for all three nozzle designs.

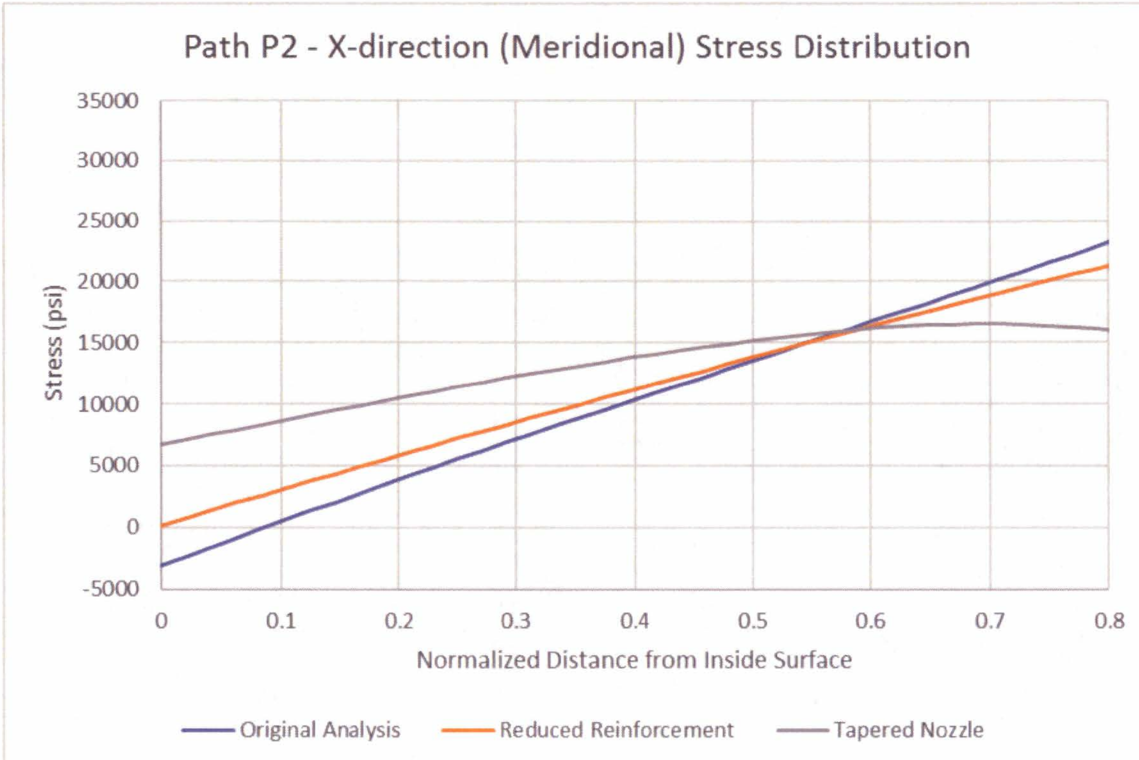


Figure 7-42
Path 2 Through-Wall Stress Distributions

Based on these comparative results from modeling the two extra nozzles, Section 8 describes various sensitivity analyses to investigate increased stresses for both of the main steam nozzle stress paths and the corresponding impacts on the probabilities of rupture or leakage. Evaluations are performed to examine the factor on stress that is needed to reach the acceptability limit. From the results in Section 8, the factor on stress needed for the probabilities to reach the failure limit is 3.4 for the main steam nozzle. This means that the stresses must be more than tripled before the acceptability limits are not satisfied. The results of this section indicate that the stress variations between different nozzle configurations in the fleet are well within the factor of 3.4 used in the sensitivity studies. Therefore, variations in stress caused by different nozzle geometries and reinforcement areas are covered by the fracture mechanics evaluations summarized in this report.

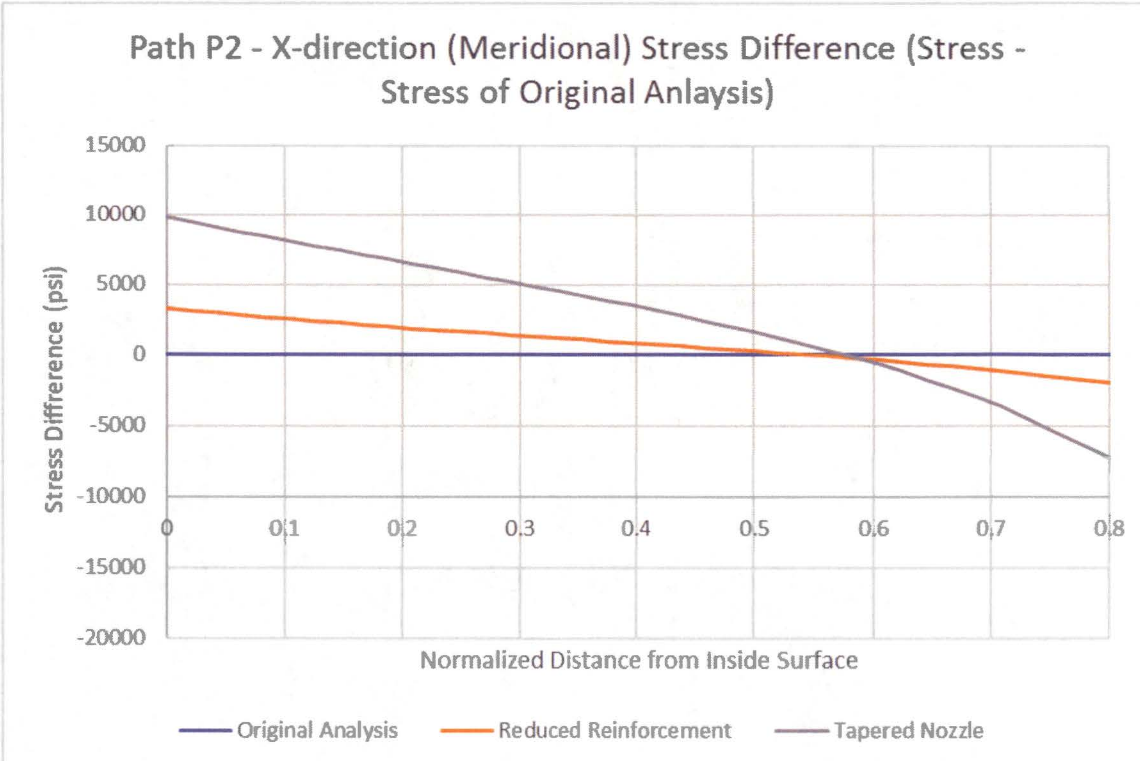


Figure 7-43
Path 2 Through-Wall Stress Distribution Comparisons (differences)

8 PROBABILISTIC AND DETERMINISTIC FRACTURE MECHANICS EVALUATION

This section describes the probabilistic fracture mechanics (PFM) and deterministic fracture mechanics (DFM) analyses performed on the feedwater and main steam nozzles selected in Section 4. Both the nozzle-to-shell welds and the nozzle inside radius sections were evaluated. The primary objective of the PFM and DFM analyses is to assess various inspection frequencies for these nozzles, including the current ASME Section XI requirements discussed in Section 2.

8.1 Overview of Technical Approach

The PFM evaluations are performed to demonstrate the reliability of the selected components assuming various inspection scenarios, while the DFM evaluations are performed using average and limiting parameters to provide confirmation of the PFM results. Both the DFM and PFM approaches have been used in previous ISI optimization projects involving examination frequency reduction, examination scope reduction, or both, as shown in Table 8-1.

Table 8-1
Previous ISI Projects That Evaluated Inspection Requirements

Document	Description	Approach		Result			Ref.	Revised Inspection Requirement
		D	P	SR	FR	CEM		
BWRVIP-05	BWR vessel welds		X	X			19	Elimination of circumferential weld inspections
BWRVIP-108	BWR inner radius and vessel-to shell welds	X	X	X			2	Sample size reduced from 100% to 25% of each nozzle type
WCAP-16168-NP-A	PWR vessel and nozzle-to-shell welds		X		X		8	Inspection frequency extended from 10 years to 20 years
SIR-94-080	RCP Flywheel. CE and B&W	X			X		52	Inspection frequency extended from 3.5 years to 10 years
WCAP-15666	Westinghouse RCP flywheel	X	X		X		53	Inspection frequency extended from 3.5 years to 10 years
EPRI-3002007626	RPV thread in flange	X			X		30	Inspection frequency extended from 10 years to (at least) 20 years
PVP-2001 (Bamford et al.)	RPV inner radius inspections	X	X	X			17	Elimination of these inspections
PVP2006-ICPVT-11-93892	Regen and residual heat exchangers	X	X			X	12	Inspection method changed from volumetric to visual
PVP2015-45194 supplemented by MRP-82 and NUREG/CR-6934	Appendix L flaw tolerance evaluation to manage fatigue in surge line	X			X		54	Justified maintaining the current 10-year inspection interval
MRP-375	Alloy 690 RPV head nozzle penetration nozzles	X	X	X	X		55	Inspection frequency extended from 10 years to 20 years Scope reduced using sister head concept

D = Deterministic SR = Scope Reduction CEM = Change in Examination Method

P = Probabilistic FR = Frequency Reduction

From Table 8-1, all listed documents involving scope reduction have some probabilistic aspect, while most projects involving frequency reduction have some deterministic aspect. Therefore, to evaluate the potential for possible scope and/or frequency reductions, both probabilistic and deterministic approaches are employed in this report.

8.2 Probabilistic Fracture Mechanics Evaluation

8.2.1 Technical Approach

Monte Carlo probabilistic analysis techniques are used in the PFM analyses to determine the effect of various inspection scenarios on the probability of failure of the components evaluated. The overall technical approach is illustrated in Figure 8-1. An analysis of the sensitivity of the PFM evaluation to various input parameters is also performed to determine the key parameters that influence the results.

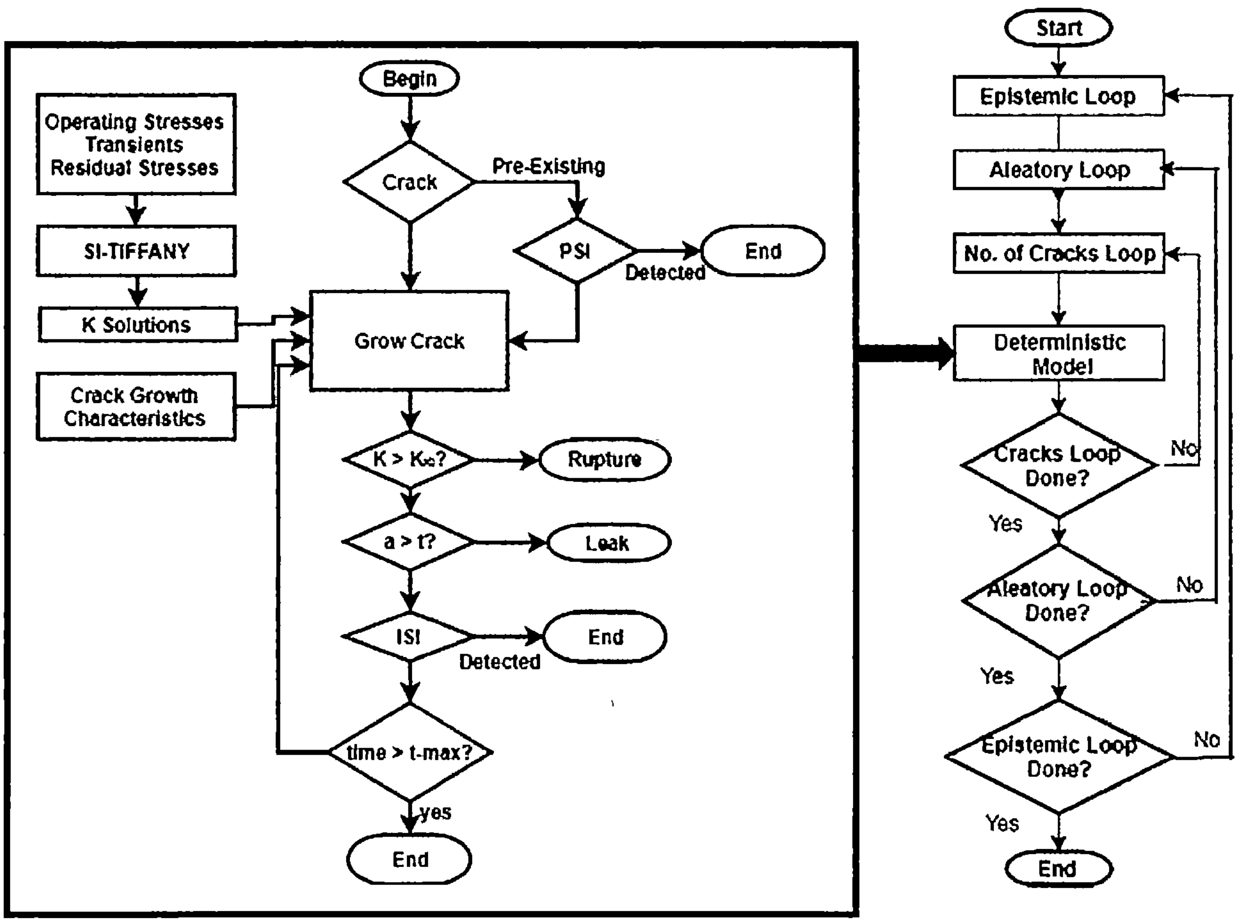


Figure 8-1
Overall Technical Approach

8.2.1.1 Treatment of Uncertainties

Separation of sources of uncertainty is considered in the PFM evaluation, because characterization of the contribution of uncertainty is one of the primary reasons for performing the analyses. There are two contributors to uncertainty in PFM analyses [56].

One contributor is inherent scatter (or random scatter), which is termed *aleatory* uncertainty. In this case, the probability of obtaining each outcome can be measured or estimated, but the precise outcome in any particular instance is not known in advance. Unlike a scenario such as rolling dice, obtaining additional data will not help reduce the variability.

A second contributor to uncertainty is caused by lack of knowledge, which is termed *epistemic* uncertainty. Unlike aleatory uncertainty, gathering more data can be helpful in reducing epistemic uncertainty. Both aleatory and epistemic uncertainties are considered in the PFM evaluation, as shown in Figure 8-1.

8.2.1.2 Sampling Method

Monte Carlo simulation is used for computing failure probabilities. A random number generator with a repeat cycle beyond 10^{15} is employed, with a provision for user input of the random number seed used in the analysis. All random variables are user-selected as being normal, log-normal, Weibull, exponential, uniform, or logistic—or the cumulative distribution can be defined by a tabular input. Provisions are made to perform a deterministic calculation of lifetime, in which case calculations are performed using specified values of the input variables. A user selection for deterministic analysis is provided. Provisions are made to consider variances in inputs due to inherent scatter (aleatory) and due to lack of knowledge (epistemic). This entails two loops in the calculations, as illustrated in Figure 8-1. This means that instead of a single line for probability of failure as a function of time, there will be a band of lines.

8.2.2 Details of Analysis Methodology and Inputs

8.2.2.1 Component Geometry

The geometries of the components considered in the evaluation are presented in Figures 4-8, 4-9, and 4-10 for the Westinghouse 4-loop main steam nozzle, the B&W main steam nozzle, and the Westinghouse 4-loop feedwater nozzle, respectively. Fracture mechanics models used for these configurations are described in Section 8.2.2.5.

8.2.2.2 Initial Flaw Distribution and Number of Flaws per Weld

The initial flaw size (depth) distribution used in the evaluation was previously derived by the NRC during its review of the BWRVIP-05 project in Reference [57], based on flaw data from the Pressure Vessel Research User's Facility (PVRUF) vessel. This distribution was subsequently used in BWRVIP-05 [19] in lieu of the Marshall flaw distribution [58], which was initially used in the ISI optimization of BWR vessel welds. It was also used in the ISI optimization of the nozzle-to-shell welds and nozzle inside radius sections in BWRVIP-108 [2]. The PVRUF flaw distribution was found to be slightly more conservative than the Marshall flaw distribution in References [57] and [58]. Even though the PVRUF data were based on a reactor pressure vessel, they can be applied to an SG vessel because both are large diameter vessels and fabricated from similar plates and forgings processes as well as from the same material (SA-533B and SA-508 Class 2). As such, the fabrication processes should be similar and the PVRUF data are judged to be reasonably representative for this evaluation. The PVRUF flaw distribution derived by the NRC is represented, in the cumulative distribution form, by Equation 8-1:

$$P(a \leq 0.0787) = 0.9054$$

$$\boldsymbol{P(a > 0.0787) = 1 - 0.1616e^{-6.94a} - 0.0139e^{-4.06a}} \qquad \textbf{Eq. 8-1}$$

The entire PVRUF flaw data are provided in Reference [59]. Previous evaluations have used a constant aspect ratio [2, 19] and, as such, once the flaw depth is known, the flaw length can be calculated. In this evaluation, the distribution of crack length is derived from data in NUREG/CR-6817 [60]. This results in a log-normal distribution for the crack length with the following parameters:

$$\mu = 0.74$$

$$\text{median } (\ell - a)/d = 2.08$$

The evaluation procedures for the length distribution are detailed in Reference [61].

In the present work, for the nozzle-to-vessel weld, the number of fabrication flaws was assumed to be 1.0 per nozzle. For the nozzle inside radius section, 0.001 flaws were assumed per nozzle. These values are consistent with those approved by the NRC during the optimization of the BWR nozzle-to-shell welds and nozzle inside radius sections in Reference [3].

8.2.2.3 Probability of Detection (POD) Curve

The POD curve used in this evaluation is based on the Performance Demonstration Initiative (PDI) on full-scale vessel mockups containing realistic defects—the same as used for the optimization of the BWR nozzle-to-shell weld and nozzle inside radius sections in Reference [2]. Three separate POD curves were derived in Reference [2] for automated weld techniques, manual weld techniques, and a combination of the two. The final POD curve used in Reference [2] was the combined curve, which is also used in this evaluation. The combined POD curve is presented in Figure 8-2. This POD curve is more conservative than that used in the development of ASME Code Section XI, Appendix L for carbon steel pipe sections [62], which is presented in Figure 8-3. As shown in this figure, the POD curve used for Appendix L shows a very high probability of detecting a crack (approximately 95%) regardless of flaw depth.

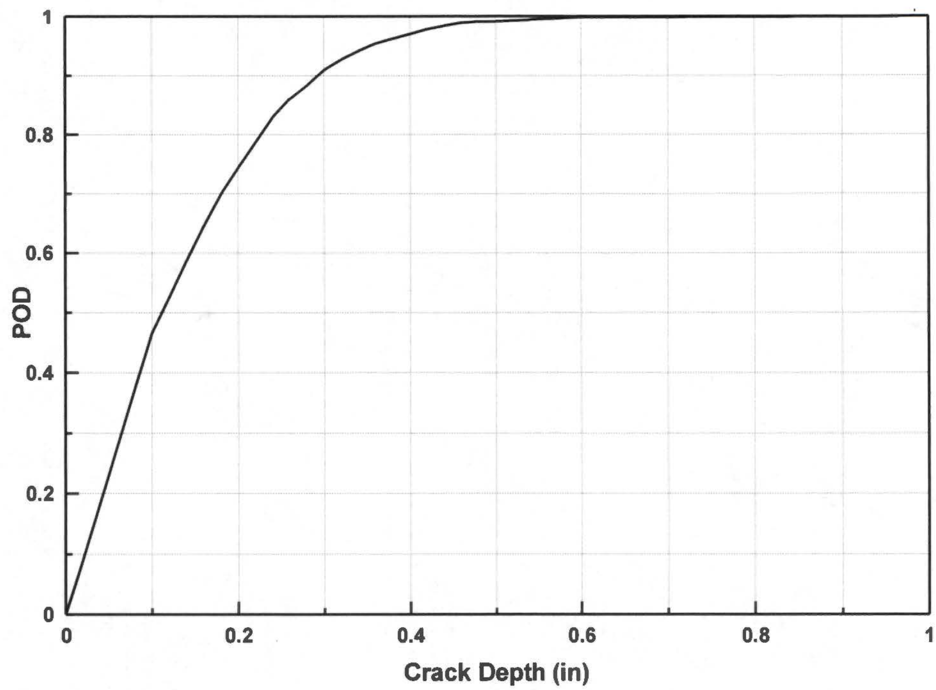


Figure 8-2
POD Curve for Vessels from BWRVIP-108 [2]

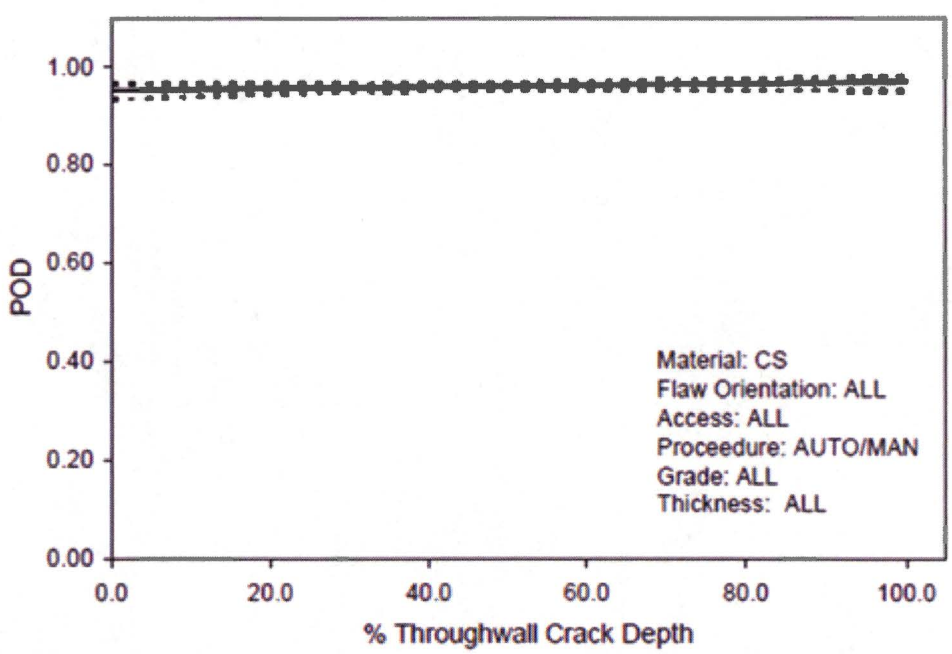


Figure 8-3
Carbon Steel POD Curve for All Pipe Section Thicknesses Used to Develop Section XI Appendix L [62]

8.2.2.4 Applied Stresses

8.2.2.4.1 Operating Transient Stresses

The applied stresses consist of through-wall stresses due to pressure and the thermal transients described in Section 7. For the Westinghouse 4-loop main steam nozzle, the through-wall stress distributions in Figures 7-8 and 7-9 were used as inputs. For the B&W main steam nozzle, the through-wall stress distributions in Figures 7-18 through 7-21 were used. For the feedwater nozzle, the through-wall stress distributions in Figures 7-31 through 7-34 were used. In this evaluation, the transient stresses are assumed to be normally distributed.

8.2.2.4.2 Weld Residual Stresses

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 are characterized in the form of a cosine distribution with a peak stress of 8 ksi [63], as shown in Figure 8-4. Because of the proximity of the nozzle-to-shell weld to the nozzle inside radius section in some configurations, this distribution was applied to both components as a constant value (non-random distribution).

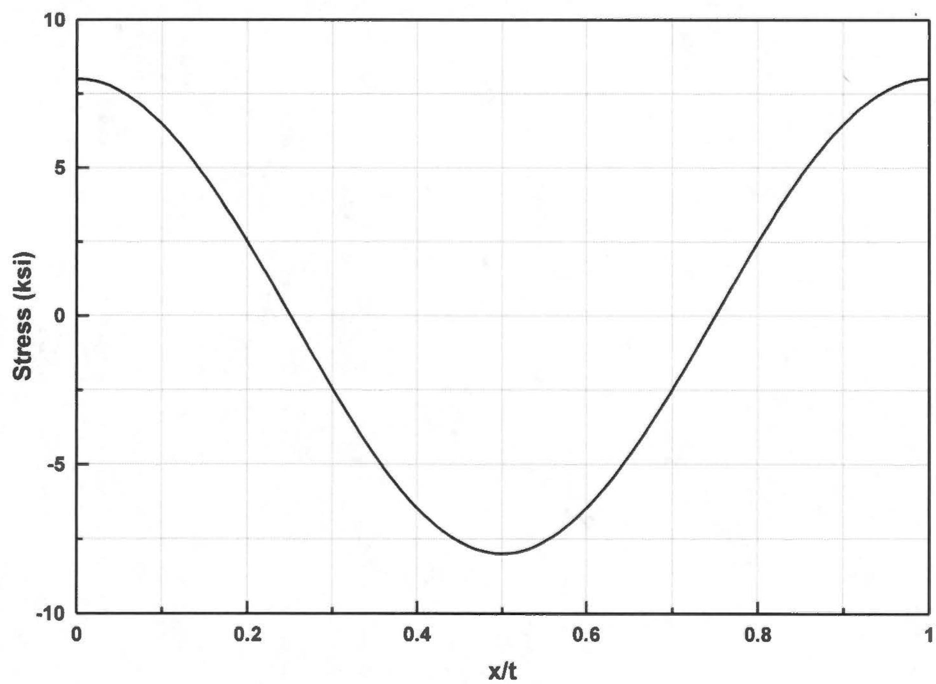


Figure 8-4
Weld Residual Stress Distribution

8.2.2.5 Fracture Mechanics Models

In this evaluation, all pre-existing flaws are conservatively assumed to be surface flaws. Three different fracture mechanics models were used for axial, circumferential, and nozzle corner cracks. For the axial flaw, the stress intensity factor (K) solution for an internal, semi-elliptical crack from API-579/ASME-FFS-1 [64] was used. This model is shown in Figure 8-5. The aspect ratio (a/c) is allowed to vary during crack growth.

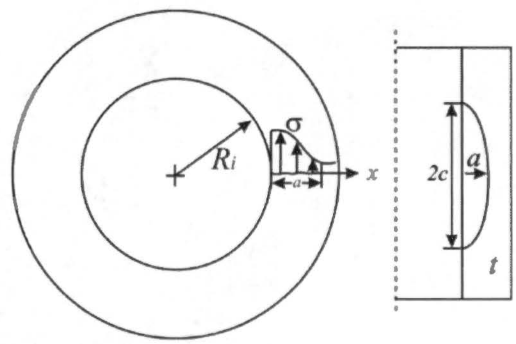


Figure 8-5
Semi-Elliptical Axial Crack in a Cylinder Model

Similarly, for a circumferential flaw, the K solution for an internal, semi-elliptical crack from API-579/ASME-FFS-1 [64] was used. This model is shown in Figure 8-6. The aspect ratio (a/c) is allowed to vary during crack growth.

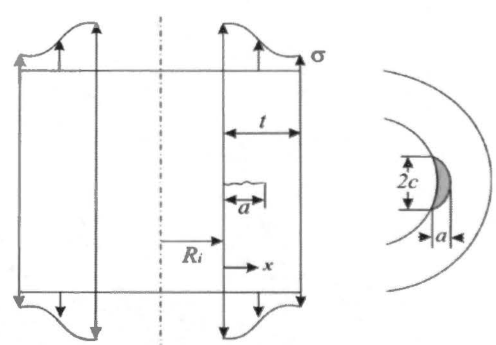


Figure 8-6
Semi-Elliptical Circumferential Crack in a Cylinder Model

For the nozzle corner crack, a weight function-based K solution from Reference [65] was used. The crack is assumed as a circular-arc crack, and the aspect ratio is fixed during crack growth. This model is shown in Figure 8-7.

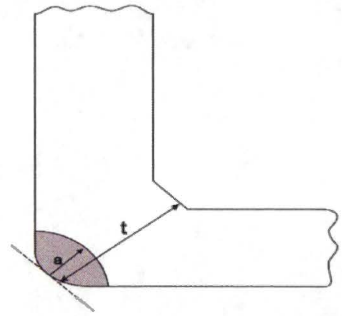


Figure 8-7
Nozzle Corner Crack Model

For the nozzle-to-shell welds, semi-elliptical part-wall cracks are postulated in the axial and circumferential directions, while a circular crack is postulated at the nozzle corner. These fracture mechanics models are incorporated into an SI-developed software called **TIFFANY** [66], which determines the stress intensity factor (K) distribution due to the through-wall stress profiles described in Section 8.2.2.4. The outputs of **TIFFANY** are the maximum and minimum K distributions, as well as the ΔK distribution, for each transient.

8.2.2.6 Fatigue Crack Growth

From the evaluation performed in Section 6, the only potential degradation mechanism identified for the main steam and feedwater nozzle-to-shell welds and nozzle inside radius sections is fatigue (that is, corrosion fatigue, mechanical fatigue, and/or thermal fatigue). The fatigue crack growth (FCG) rate for ferritic steels, as defined in ASME Code Section XI, Appendix A, Paragraph A-4300 [1], is used in the evaluation. The calculated crack growth rate is considered log-normally distributed consistent with the distribution used in the xLPR project [67]. The calculated value of the crack growth based on A-4300 is considered as the median value of the log-normal distribution with a second parameter of 0.467 from Reference [67], which is derived from FCG in ferritic materials in a PWR environment and is considered applicable here. The FCG threshold is also assumed to be log-normally distributed as described in Reference [67].

8.2.2.7 Fracture Toughness

The materials under consideration are all ferritic steels; therefore, the fracture toughness curve provided in ASME Code Section XI, Appendix A (Figure A-4200-1) [1] is used for this evaluation. From Section 5.2, the minimum temperature considering all transients corresponds to the feedwater injection case. Figure 7-30 contains a plot of the temperature during this transient. The temperature near the inner radius region is always above 300°F. The maximum RT_{NDT} for either the SA-533 SG shell material or the SA-508 Class 2 nozzle allowed by Branch Technical Position (BTP) 5-3 [68] is 60°F. Because the temperature is always above 300°F, the temperature to use when entering the ASME Code, Section XI, Figure A-4200-1 is 300°F-60°F=240°F. This temperature is greater than all temperatures in Section XI, Figure A-4200-1, so an upper shelf fracture toughness of at least 200 ksi√in. (220 MPa√m) can be used. Figure 8-8 [69, 70] shows the fracture toughness of vessel steels as a function of temperature and shows that ASME Code Section XI fracture toughness is a lower bound. However, in this evaluation, the ASME Code Section XI fracture toughness will be assumed to be normally distributed with a median of 200 ksi√in. and a standard deviation of 5 ksi√in. consistent with that used in BWRVIP-108 [2].

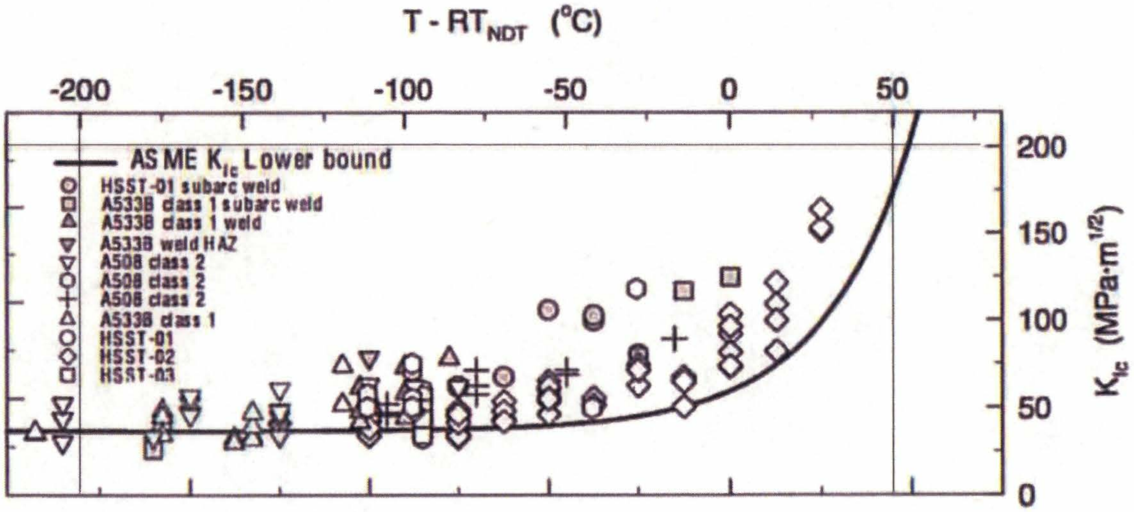


Figure 8-8
ASME Section XI Fracture Toughness Curve for Vessels vs. Experimental Data Points [69, 70]

8.2.2.8 Inspection Schedule Scenarios

The inspection schedule scenarios considered in the evaluation are shown in Table 8-2.

Table 8-2
Inspection Schedule Scenarios

Inspection Schedule	0	10	20	30	40	50	60	70	80
PSI only	X								
PSI + 10	X	X							
PSI + 10 + 20	X	X	X						
PSI + 10 + 20 +30	X	X	X	X					
PSI, 20-year interval	X		X		X		X		
PSI + 10, 20-year interval	X	X		X		X		X	
PSI + 10 + 20, 20-year interval	X	X	X		X		X		
PSI + 10 + 20 + 30, 20-year interval	X	X	X	X		X		X	
PSI, 30-year interval	X			X			X		
PSI + 10, 30-year interval	X	X			X			X	
PSI +10 + 20, 30-year interval	X	X	X			X			
PSI + 10 + 20 + 30, 30-year interval	X	X	X	X			X		

In all scenarios, it is assumed that pre-service inspection (PSI) was performed. ISI is then performed with the current ASME Code Section XI schedule of a 10-year inspection interval up to 30 years followed by 20- or 30-year intervals. This sequence is repeated with a 20- or 30-year inspection interval from the time of the last 10-year inspection. The objective is to determine whether an increased inspection interval can be justified regardless of how long a plant has operated with the current ASME Code Section XI inspection schedule.

8.2.2.9 Acceptance Criteria

The acceptance criterion for the PFM evaluation is that failure frequencies must be less than the NRC safety goal of 10^{-6} failures per year, consistent with that used for the optimization of the BWR vessel shell welds in BWRVIP-05 [19] and the optimization of the BWR nozzle-to-shell weld/nozzle inside radius sections in BWRVIP-108 [2]. This failure criterion was used in the development of alternative fracture toughness requirements for protection against pressurized thermal shock events [71] and has also been adopted for the xLPR project [72].

Failure occurs either by rupture or by leak. Rupture occurs in the probabilistic simulation when, for a given iteration, the applied stress intensity factor (K) exceeds the material fracture toughness (K_{IC}), that is, $K > K_{IC}$. A leak occurs when, for a given iteration, crack depth exceeds the wall thickness ($a > t$). Due to the limitation in the fracture mechanics models used in this evaluation, it is assumed that a leak occurs when the crack depth reaches 80% of the component wall thickness.

8.2.2.10 Limitations and Assumptions

The following assumptions are made in the evaluation:

1. Consistent with the discussion provided in Section 6, SCC is not considered because all materials are ferritic steels in a PWR environment. It is further assumed that no Alloy 82/182 welds are present.
2. Cracks in the nozzle-to-shell welds are characterized as either axial or circumferential cracks in a cylinder. That is, the crack does not grow in a circular direction around the nozzle-to-shell weld because the crack driving force will not drive such growth. The assumed axial and circumferential flaws will ensure that a limiting flaw case is evaluated.

8.2.3 Computer Software Application

8.2.3.1 Software Development

The PFM methodology illustrated in Figure 8-1, and all inputs described in Section 8.2.2, are implemented in an SI-developed software code called **PROMISE** (**PR**obabilistic **O**pti**M**ization of **I**n**S**p**E**ction), Version 1.0. **PROMISE** provides a probabilistic model of initiation and growth of cracks due to SCC and fatigue (although the crack initiation and SCC modules are not used in the present work). The growth and final instability of cracks are treated using linear elastic fracture mechanics (LEFM). Monte Carlo simulation is used to generate numerical results, which include the probability of the crack depth exceeding 80% of the wall thickness ($a/t > 0.8$) and the probability of the applied stress intensity factor exceeding the allowed fracture toughness ($K > K_{IC}$). **PROMISE** consists of the following modules:

- Crack growth
- Stress intensity factor solutions (calculated by **TIFFANY**)
- Loads
- Inspection

Following are the random variables in **PROMISE**:

- Initial crack size (depth and length)
- Loads (transient stresses)
- Weld residual stresses
- Fracture toughness
- Crack detection
- Number of cracks per weld or nozzle
- Inspection frequency
- Crack growth rate

Table 8-3 summarizes the various random variable distributions selected in **PROMISE** for this evaluation.

Table 8-3
Random Variables Distributions

Variable	Distribution
Crack depth	Tabular (PVRUF or NUREG-6817)
Crack length	Log-normal
Transient stresses	Normal
Fracture toughness	Normal
Fatigue crack growth rate	Log-normal
Fatigue crack growth threshold	Log-normal
Crack detection	Tabular (POD curve)
Number of cracks per weld or nozzle	Poisson
Weld residual stresses	Constant
OD, thickness	Constant

8.2.3.2 Software Verification and Validation

The verification and validation (V&V) of the **PROMISE** software involved two phases. In the first phase, a software project plan was developed [73]. The software project plan includes the requirements specification, functional specification, and software V&V plan. The second phase involved comparisons to benchmark problems against existing software codes.

8.2.3.2.1 Software Testing

The **PROMISE** software was tested extensively according to the software V&V plan to ensure that it is producing expected and realistic results. The testing of the software and the results are documented in Reference [74]. A user's manual for **PROMISE** has also been developed [75].

8.2.3.2.2 Benchmarking

A benchmarking exercise was performed to validate the **PROMISE** software against the **VIPERNOZ** Version 1.1 software code [76], which was used for performing the PFM analyses of the BWR vessel nozzle-to-shell welds and the nozzle inner radii in BWRVIP-108 [2]. The **VIPERNOZ** software was chosen for the benchmarking because it was presented to the NRC in BWRVIP-108. The nozzle corner crack model used in **VIPERNOZ** for calculating the stress intensity factors was updated in **PROMISE** using the weight function method [65]. However, in the benchmarking exercise, the stress distribution was chosen such that the two models provided the same stress results. In addition, the PVRUF distribution was used for the initial crack size to be consistent with BWRVIP-05 and BWRVIP-108. The BWRVIP-108 FCG equation was used along with a Weibull distribution for the coefficient of the growth law equation. The FCG threshold was assumed to be zero. A summary of the key inputs to the benchmarking exercise is provided in Table 8-4.

Table 8-4
Benchmarking Inputs

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

Analyses were performed for several combinations of cyclic stress and numbers of load cycles. The cumulative probability of leakage results for the benchmarking exercise are presented in Table 8-5. As can be seen from Table 8-5, reasonably good agreement was obtained between **PROMISE** and **VIPERNOZ**, and the probability of leakage results are similar for the two types of software.

Table 8-5
Comparison of Cumulative Probability of Leak Between PROMISE and VIPERNOZ for Benchmarking

Cyclic Stress (ksi)	Cycles/Year	PROMISE	VIPERNOZ
25	500	2.8E-2	3.1E-2
15	500	1.7E-4	3.0E-4

8.2.4 Base Case, Sensitivity Analysis, and Sensitivity Studies

The PFM analysis is performed on the three components selected in Section 4 (that is, Westinghouse main steam nozzle, B&W main steam nozzle, and Westinghouse feedwater nozzle). As discussed in Section 4, multiple stress paths were considered for each nozzle as shown in Figures 7-8, 7-18, and 7-31. For the nozzle inside radius sections, a crack at the inner radius was postulated. For the nozzle-to-shell welds, both axially and circumferentially oriented cracks were considered. The components and the analyzed paths are listed in Table 8-6.

Table 8-6
Components and the Crack Paths Analyzed

Component	Case Identification (1)
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N
	SGW-P2C
	SGW-P2A
B&W Main Steam Nozzle (SGB)	SGB-P1N
	SGB-P2N
	SGB-P3C
	SGB-P3A
	SGB-P4A
	SGB-P4C
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N
	FEW-P2N
	FEW-P3C
	FEW-P3A
	FEW-P4A
	FEW-P4C

Note 1:

P1, P2, P3, P4 – Crack Path (see Figures 7-8, 7-18, and 7-31)

N – Nozzle Corner Crack

C – Circumferential Part-Through-Wall Crack

A – Axial Part-Through-Wall Crack

8.2.4.1 Base Case Evaluation

As detailed in Section 3, the current ASME Code, Section XI inspection requirements for the main steam and feedwater nozzles call for PSI followed by 10-year ISI intervals. To evaluate other inspection possibilities, a scenario of PSI followed by 20-year inspection intervals is considered as the base case. This is similar to the current inspection requirements approved by the NRC for the PWR vessels and nozzles [8]. Other input parameters for the base case evaluation are discussed in Section 8.2.2 and listed in Table 8-7.

The inputs selected for the base case evaluation are considered the most appropriate data for the current report. The selected crack size distribution, POD curve, and weld residual stress profile have been used in similar projects [2, 19] in the past. The crack length distribution is derived from the most recent data on cracks in welds [60, 61]. The FCG curve defined in ASME Code Section XI, Appendix A, A-4300 [1] "falls above the least square best fit to the data" and is based on a 95% global confidence limit of the data. In the present analysis, the FCG curve defined in A-4300 is considered the median curve; however, it is higher than the median curve as discussed in Reference [1] and is therefore conservative. The second parameter of log-normal distribution (analogous to standard deviation) is obtained from the xLPR project [67]. As discussed, the upper shelf ASME Code Section XI fracture toughness (K_{IC}) used in this evaluation is conservative compared to the actual data. All the distributed variables were considered aleatory because they are conservative or based on large sets of data.

Unless mentioned otherwise, 10 million aleatory realizations were performed for all cases investigated in this report.

Table 8-7
PFM Base Case Inputs

No. of realizations	Epistemic = 1, Aleatory = 10 million
No. of cracks per nozzle-to-shell weld	1, constant
No. of cracks per nozzle inside radius section	0.001, 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	20, 40, 60 (20-year interval)
POD curve	BWRVIP-108, Figure 8-2
Fatigue crack growth law and threshold	A-4300, log-normal, Second Par. = 0.467
Uncertainties on transients	None
Weld residual stresses (ksi)	Cosine curve (8, 8), constant (not random)

The results of the base case are listed in Table 8-8 for the probability of rupture and probability of leakage per year. For all components, the probabilities of rupture and leakage are below the acceptance criteria of 10^{-6} per year up to 80 years of operation, with the limiting location being FEW-P3A. This demonstrates that all components are very flaw-tolerant.

Table 8-8
Probability of Rupture (per year) and Probability of Leakage (per year) for Base Case (PSI followed by inspection at 20, 40, and 60 years)

Component	Case Identification	P(Rupture) at 80 Years	P(Leakage) at			
			20 Years	40 Years	60 Years	80 Years
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	5.00E-12	2.50E-12	1.67E-12	1.25E-12
	SGW-P2C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	SGW-P2A	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	5.00E-12	2.50E-12	1.67E-12	1.25E-12
	SGB-P2N	1.25E-12	5.00E-12	2.50E-12	1.67E-12	1.25E-12
	SGB-P3C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	SGB-P3A	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	SGB-P4A	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	SGB-P4C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	4.50E-11	3.75E-11	2.50E-11	1.88E-11
	FEW-P2N	1.25E-12	5.00E-12	2.50E-12	1.67E-12	1.25E-12
	FEW-P3C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	FEW-P3A	1.25E-09	5.00E-09	2.50E-09	3.33E-09	2.50E-09
	FEW-P4A	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09
	FEW-P4C	1.25E-09	5.00E-09	2.50E-09	1.67E-09	1.25E-09

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

8.2.4.1.1 PSI-Only Inspection

Nuclear plants have been in operation for various time periods, and many units have replaced their SGs at various times during their respective plant histories. As such, their inspection intervals and histories are quite different. However, the one inspection that is consistent across all plants is that they all received PSI before entering service. The Section III fabrication examination required for these components was robust, and any Section XI pre-service examinations further contributed to thorough initial examinations. Therefore, in this report, *PSI* refers to these collective initial examinations, and a scenario with PSI only is considered in the PFM studies using the same other input parameters as the base case. The results of this PSI-only case are shown in Table 8-9.

Table 8-9
Probability of Rupture (per year) and Probability of Leakage (per year) for PSI Only

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

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

The following observations are made from the review of the results in Table 8-9:

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

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

8.2.4.1.2 *Effect of Various In-Service Inspection Scenarios*

As shown by the results in Tables 8-8 and 8-9, Case FEW-P3A is the limiting case. This case was chosen to determine the effects of the inspection scenarios listed in Table 8-2 on the probabilities of rupture and leakage. Table 8-10 presents the case in which PSI followed by several 10-year ISI inspections have already been performed followed by 20- or 30-year ISI inspections. The results are also shown graphically in Figure 8-9.

Table 8-10
Probability of Rupture (per year) and Probability of Leakage (per year) for 80 Years for PSI Followed by 10-Year Inspections Followed by 20- or 30-Year Inspections for Case FEW-P3A

ISI Scenario	ISI at	Probability of Rupture (per year)	Probability of Leakage (per year)
PSI only	0	1.25E-09	1.19E-05
PSI followed by 10-year inspections	0,10	1.25E-09	4.93E-06
	0,10,20	1.25E-09	1.36E-06
	0,10,20,30	1.25E-09	1.39E-07
PSI followed by 20-year inspections	0,20,40,60	1.25E-09	2.50E-09
PSI followed by 10-year inspections followed by 20-year inspections	0,10,30,50,70	1.25E-09	2.50E-09
	0,10,20,40,60	1.25E-09	1.25E-09
	0,10,20,30,50,70	1.25E-09	1.25E-09
PSI followed by 30-year inspections	0,30,60	1.25E-09	2.25E-08
PSI followed by 10-year inspections followed by 30-year inspections	0,10,40,70	1.25E-09	3.00E-08
	0,10,20,50	1.25E-09	7.50E-09
	0,10,20,30,60	1.25E-09	2.50E-09

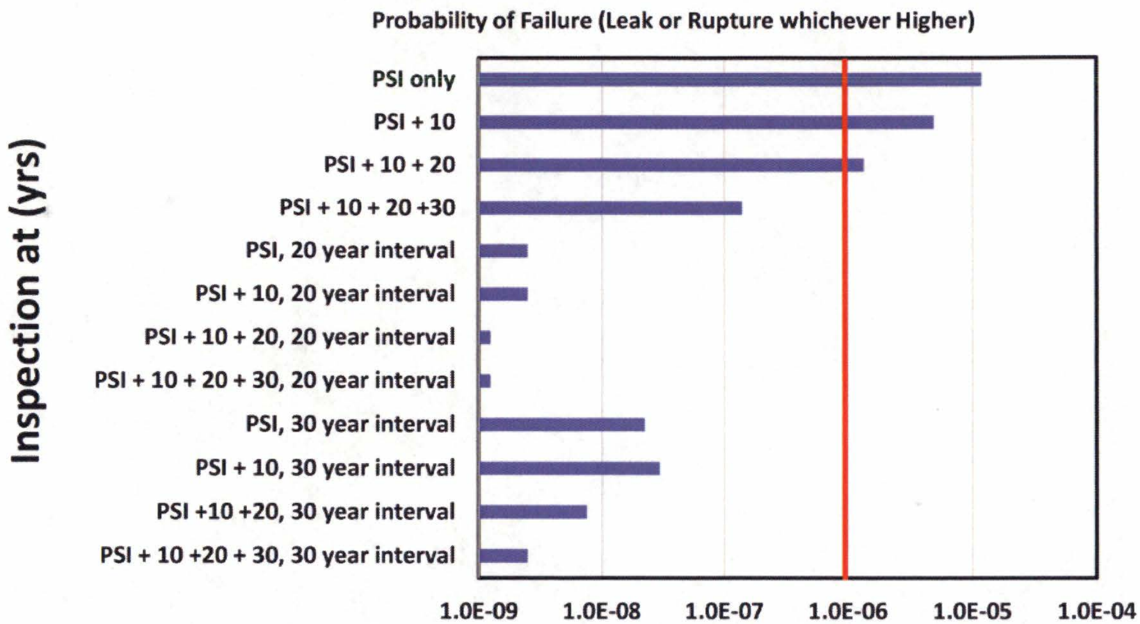


Figure 8-9
Effect of In-Service Inspection on Probability of Failure at Limiting Location (Case FEW-P3A)

As can be seen from Table 8-10 and Figure 8-9, for the PSI only, PSI+10, and PSI+10+20 inspections, the acceptance criteria are not met after 80 years. However, similar to the PSI only case shown in Table 8-9, the acceptance criteria for these cases are met after 60 years (results not shown). The results in Table 8-10 also indicated that, after three 10-year ISI inspections (PSI+10+20+30), the acceptance criteria are met for 80 years. This demonstrates that the current ASME Section XI 10-year inspection schedule meets the acceptance criteria if it is continued through at least 30 years. As shown in Figure 8-9, all other inspection scenarios listed in Table 8-2 involving 20-year ISI inspections, 30-year ISI inspections, or a combination of these with 10-year ISI inspections meet the acceptance criteria. This demonstrates that when one or more 10-year ISI inspections have been performed, a 20- or 30-year ISI interval can be accommodated because the acceptance criteria for probabilities of rupture and of leakage are satisfied for the limiting location (FEW-P3A). Other locations will have more favorable probability of rupture and probability of leakage values.

8.2.4.2 Sensitivity Analysis

The purpose of sensitivity analysis is to identify the random variables that have the most impact on the calculated probabilities (rupture or leakage). The Degree of Separation Method, as developed and documented in Reference [77], is used to rank the effect of the input variables on the output probabilities. Case FEW-P3A was chosen for this evaluation because it is the most limiting in terms of the probability of leakage. The inputs were the same as those used in the base case, except that the mean fracture toughness value was reduced to 150 ksi√in. with a standard deviation of 30 and a stress multiplier of 1.5 (to increase the probability of rupture for the sensitivity analysis). The results of the sensitivity analysis are provided in Table 8-11 in terms of both leakage and rupture probabilities.

Table 8-11
Results of Sensitivity Analysis for Case FEW-P3A

Variable	Importance Factor (%)	
	Probability of Leakage	Probability of Rupture
Fatigue Crack Growth Rate Coefficient	79.16	23.69
Crack Depth	0.02	0.10
Crack Length	8.89	8.43
Fatigue Crack Growth Rate Threshold	0.01	0.00
Fracture Toughness	11.92	67.86

Fracture toughness was the most dominant variable affecting the rupture probabilities (as expected, because the fracture toughness was reduced to 150 ksi$\sqrt{}$in.). On the other hand, the FCG rate coefficient is the most dominant variable affecting the probability of leakage while also having a relatively strong influence on the probability of rupture.

A sensitivity analysis was also performed in which a multiplier was applied to stress and made random. In this example, there were seven random variables in addition to the five variables used in Table 8-11. The 12 random variables and the associated results of this sensitivity analysis are shown in Table 8-12.

Table 8-12
Results of Sensitivity Analysis with Random Stress Multiplier for Case FEW-P3A

Variable	Importance Factor (%) for Probability of Leakage
Fatigue Crack Growth Rate Coefficient	87.82
Crack Length	9.39
Crack Depth	0.00
Fatigue Crack Growth Rate Threshold	0.01
Fracture Toughness	0.01
Crack Face Pressure	0.04
Pressure	1.82
Residual Stress	0.00
Heat-Up/Cooldown	0.01
Loss of Load	0.04
Load Increase (5%)	0.69
Load Decrease (5%)	0.16

In this case, FCG rate coefficient and crack length are the most dominant variables affecting the leakage probabilities.

8.2.4.3 Sensitivity Studies

Even though the sensitivity analysis identified only a few parameters that have significant influence on the probabilities of rupture and leakage, several additional parameters were included in additional sensitivity studies for completeness. Sensitivity to the probability of rupture, probability of leakage, or both are evaluated where appropriate up to 80 years. Sensitivity studies were performed on the following parameters:

1. Fracture toughness (rupture and leakage)
2. Stress (rupture and leakage)
3. Initial crack depth (rupture and leakage)
4. Nozzle flaw density
5. Number of flaws (leakage)
6. Crack size distribution (leakage)
7. FCG (leakage)
8. POD (rupture and leakage)
9. Inspection schedule with two PODs (leakage)
10. Number of realizations (rupture)
11. Combination of key variables

8.2.4.3.1 Sensitivity to Fracture Toughness

Case FEW-P3A was chosen for determining the sensitivity to fracture toughness (K_{IC}). In the base case, K_{IC} was considered normally distributed with a mean of 200 ksi√in. and a standard deviation of 5 ksi√in. The probability of rupture was less than 10^{-8} per year at 80 years. The K_{IC} was then revised to 80 ksi√in. and 100 ksi√in. while keeping the standard deviation constant. The rupture probabilities for all cases are shown in Table 8-13. The rupture probabilities increased with decreasing K_{IC} as expected. However, as discussed previously, even at the minimum temperature of the evaluated transients, the material of the SG vessel is on the upper shelf—therefore, the fracture toughness (K_{IC}) is at least 200 ksi√in. To address NRC concerns regarding the standard deviation on the fracture toughness in the BWRVIP-108 SER [3], a sensitivity study was performed using a standard deviation of 30 ksi√in. As can be seen from Table 8-13, using a standard deviation of 30 ksi√in. and the base case mean toughness of 200 ksi√in. did not change the outcome of the analysis.

The leakage probabilities were calculated for lower values of K_{IC}, and the results are provided in Table 8-14. The change in either K_{IC} or the standard deviation had no impact on the leakage probabilities. This result is consistent with the sensitivity analysis, where K_{IC} was not a significant parameter for leakage probabilities. When rupture probabilities increased, leakage probabilities decreased; this is due to cracks rupturing instead of leaking because of the lower fracture toughness.

Table 8-13
Sensitivity of Toughness on Probability of Rupture for 80 Years

Component	Case Identification	Probability of Rupture (per year)			
		Base Case, K_{IC} = 200 ksi√in.	K_{IC} = 100 ksi√in.	K_{IC} = 80 ksi√in.	K_{IC} = 200 SD = 30 ksi√in.
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	1.25E-12	1.25E-12	1.25E-12
	SGW-P2C	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	SGW-P2A	1.25E-09	1.25E-09	1.25E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	1.25E-12	1.25E-12	1.25E-12
	SGB-P2N	1.25E-12	1.25E-12	1.25E-12	1.25E-12
	SGB-P3C	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	SGB-P3A	1.25E-09	1.25E-09	2.50E-09	1.25E-09
	SGB-P4A	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	SGB-P4C	1.25E-09	1.25E-09	1.25E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	4.61E-09	**8.95E-08**	1.75E-11
	FEW-P2N	1.25E-12	1.25E-12	1.25E-12	1.25E-12
	FEW-P3C	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	FEW-P3A	1.25E-09	3.75E-09	3.75E-08	1.25E-09
	FEW-P4A	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	FEW-P4C	1.25E-09	1.25E-09	1.25E-09	1.25E-09

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

Table 8-14
Sensitivity of Toughness on Probability of Leakage for 80 Years

Component	Case Identification	Probability of Leakage (per year)			
		Base Case, K_{IC} = 200 ksi√in.	K_{IC} = 100 ksi√in.	K_{IC} = 80 ksi√in.	K_{IC} = 200 SD = 30 ksi√in.
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	1.25E-12	1.25E-12	1.25E-12
	SGW-P2C	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	SGW-P2A	1.25E-09	1.25E-09	1.25E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	1.25E-12	1.25E-12	1.25E-12
	SGB-P2N	1.25E-12	1.25E-12	1.25E-12	1.25E-12
	SGB-P3C	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	SGB-P3A	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	SGB-P4A	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	SGB-P4C	1.25E-09	1.25E-09	1.25E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.88E-11	1.25E-12	1.25E-12	1.88E-11
	FEW-P2N	1.25E-12	1.25E-12	1.25E-12	1.25E-12
	FEW-P3C	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	FEW-P3A	**2.50E-09**	1.25E-09	1.25E-09	2.50E-09
	FEW-P4A	1.25E-09	1.25E-09	1.25E-09	1.25E-09
	FEW-P4C	1.25E-09	1.25E-09	1.25E-09	1.25E-09

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

8.2.4.3.2 Sensitivity to Stresses

To study the effect of cyclic stresses on failure probability, all cyclic stresses considered for the base case were factored to determine what stress multipliers/factors would result in failure probabilities approaching the acceptance criteria. These stress multipliers account for the effect of geometrical variations (in terms of R/t ratios) discussed in Section 4 of this report. As discussed in Section 4, the R/t ratios for the SG shell can vary by approximately 10%, which suggests that the stresses can vary by 10%. The results presented in Tables 8-15 and 8-16 indicate that a stress multiplier of approximately 1.5 is necessary for the limiting feedwater nozzle location (FEW-P3A) to reach the acceptance criteria (Table 8-16) and a stress multiplier of 2.2 is necessary for the limiting main steam nozzle location (SGB-P3A) to reach the acceptance criteria limit (Table 8-15). These stress multipliers would be even higher for some of the more realistic inspection scenarios listed in Table 8-2 in which one or more 10-year ISI inspections have been performed before switching to a 20-year ISI interval.

Table 8-15
Sensitivity of Stress on Probability of Rupture for 80 Years

Component	Case Identification	Probability of Rupture (per year)		
		Base Case, Stress Multiplier = 1	Stress Multiplier = 1.5	Stress Multiplier = 2.2
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	N/A	1.25E-12
	SGW-P2C	1.25E-09	N/A	1.25E-09
	SGW-P2A	1.25E-09	N/A	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	N/A	1.25E-12
	SGB-P2N	1.25E-12	N/A	1.25E-12
	SGB-P3C	1.25E-09	N/A	1.25E-09
	SGB-P3A	1.25E-09	N/A	**5.23E-07**
	SGB-P4A	1.25E-09	N/A	1.25E-09
	SGB-P4C	1.25E-09	N/A	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	1.25E-12	N/A
	FEW-P2N	1.25E-12	1.25E-12	N/A
	FEW-P3C	1.25E-09	1.25E-09	N/A
	FEW-P3A	1.25E-09	1.25E-09	N/A
	FEW-P4A	1.25E-09	1.25E-09	N/A
	FEW-P4C	1.25E-09	1.25E-09	N/A

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

Table 8-16
Sensitivity of Stress on Probability of Leakage for 80 Years

Component	Case Identification	Probability of Leakage (per year)		
		Base Case, Stress Multiplier = 1	Stress Multiplier = 1.5	Stress Multiplier = 2.2
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	N/A	1.25E-12
	SGW-P2C	1.25E-09	N/A	1.25E-09
	SGW-P2A	1.25E-09	N/A	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	N/A	8.75E-12
	SGB-P2N	1.25E-12	N/A	1.25E-12
	SGB-P3C	1.25E-09	N/A	1.25E-09
	SGB-P3A	1.25E-09	N/A	1.25E-09
	SGB-P4A	1.25E-09	N/A	1.25E-09
	SGB-P4C	1.25E-09	N/A	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.88E-11	1.06E-08	N/A
	FEW-P2N	1.25E-12	1.25E-12	N/A
	FEW-P3C	1.25E-09	1.25E-09	N/A
	FEW-P3A	2.50E-09	**1.04E-06**	N/A
	FEW-P4A	1.25E-09	2.50E-09	N/A
	FEW-P4C	1.25E-09	1.25E-09	N/A

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

8.2.4.3.3 Sensitivity to Initial Crack Depth Distribution

Starting with the base case inputs, a sensitivity study was performed in which the crack depth distribution was replaced by a constant value equal to 5.2% of the wall thickness (which corresponds to the ASME Code Section XI flaw acceptance standard for these components) and the aspect ratio was fixed at 1.0, making the crack semi-circular. The results are shown in Tables 8-17 and 8-18 for the probability of rupture and probability of leakage, respectively. The probability of rupture was not affected. However, the leakage probabilities decreased in 2 out of 15 cases compared to the base case.

Table 8-17
Sensitivity of Initial Crack Size on Probability of Rupture for 80 Years

Component	Case Identification	Probability of Rupture (per year)	
		Base Case, PVRUF	ASME IWB-3500 Acceptance Flaw Size (0.052t, c/a=1)
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	1.25E-12
	SGW-P2C	1.25E-09	1.25E-09
	SGW-P2A	1.25E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	1.25E-12
	SGB-P2N	1.25E-12	1.25E-12
	SGB-P3C	1.25E-09	1.25E-09
	SGB-P3A	1.25E-09	1.25E-09
	SGB-P4A	1.25E-09	1.25E-09
	SGB-P4C	1.25E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	1.25E-12
	FEW-P2N	1.25E-12	1.25E-12
	FEW-P3C	1.25E-09	1.25E-09
	FEW-P3A	1.25E-09	1.25E-09
	FEW-P4A	1.25E-09	1.25E-09
	FEW-P4C	1.25E-09	1.25E-09

Table 8-18
Sensitivity of Initial Crack Size on Probability of Leakage for 80 Years

Component	Case Identification	Probability of Leakage (per year)	
		Base Case, PVRUF	ASME IWB-3500 Acceptance Flaw Size (0.052t, c/a=1)
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	1.25E-12
	SGW-P2C	1.25E-09	1.25E-09
	SGW-P2A	1.25E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	1.25E-12
	SGB-P2N	1.25E-12	1.25E-12
	SGB-P3C	1.25E-09	1.25E-09
	SGB-P3A	1.25E-09	1.25E-09
	SGB-P4A	1.25E-09	1.25E-09
	SGB-P4C	1.25E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.88E-11	6.25E-12
	FEW-P2N	1.25E-12	1.25E-12
	FEW-P3C	1.25E-09	1.25E-09
	FEW-P3A	**2.50E-09**	1.25E-09
	FEW-P4A	1.25E-09	1.25E-09
	FEW-P4C	1.25E-09	1.25E-09

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

8.2.4.3.4 Sensitivity to Nozzle Flaw Density Considering Various Inspection Scenarios

The sensitivity to the flaw density assumed for the nozzle was also performed. In the base case, the flaw density in the nozzle was assumed to be 0.001 consistent with that used in BWRVIP-108 [2]. However, in the safety evaluation of BWRVIP-108 [3], the NRC requested sensitivity studies applying an increase of two orders of magnitude on this flaw density. This sensitivity study is also performed for the base case. The results are presented in Tables 8-19 and 8-20. Only the nozzle locations are affected in the tables. It can be seen that, by increasing the nozzle flaw density by two orders of magnitude, the probabilities of rupture and leakage also increase by two orders of magnitude.

Table 8-19
Sensitivity of Nozzle Flaw Density on Probability of Rupture for 80 Years

Component	Case Identification	Probability of Rupture (per year)	
		Base Case, Flaw Density = 0.001 for Nozzles	Flaw Density = 0.1 for Nozzles
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	**1.25E-10**
	SGW-P2C	N/A	N/A
	SGW-P2A	N/A	N/A
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	**1.25E-10**
	SGB-P2N	1.25E-12	**1.25E-10**
	SGB-P3C	N/A	N/A
	SGB-P3A	N/A	N/A
	SGB-P4A	N/A	N/A
	SGB-P4C	N/A	N/A
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	**1.25E-10**
	FEW-P2N	1.25E-12	**1.25E-10**
	FEW-P3C	N/A	N/A
	FEW-P3A	N/A	N/A
	FEW-P4A	N/A	N/A
	FEW-P4C	N/A	N/A

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

Table 8-20
Sensitivity of Nozzle Flaw Density on Probability of Leakage for 80 Years

Component	Case Identification	Probability of Leakage (per year)	
		Base Case, Flaw Density = 0.001 for Nozzles	Flaw Density = 0.1 for Nozzles
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	1.25E-10
	SGW-P2C	N/A	N/A
	SGW-P2A	N/A	N/A
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	1.25E-10
	SGB-P2N	1.25E-12	1.25E-10
	SGB-P3C	N/A	N/A
	SGB-P3A	N/A	N/A
	SGB-P4A	N/A	N/A
	SGB-P4C	N/A	N/A
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.88E-11	**1.88E-09**
	FEW-P2N	1.25E-12	1.25E-10
	FEW-P3C	N/A	N/A
	FEW-P3A	N/A	N/A
	FEW-P4A	N/A	N/A
	FEW-P4C	N/A	N/A

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

8.2.4.3.5 Sensitivity to the Distribution of Number of Flaws

The sensitivity to the distribution of number of flaws was investigated at a weld location by replacing the constant distribution of flaws in the base case with a Poisson distribution but with a single flaw assumed in each case. The results are shown in Table 8-21. In this case, only the probability of leakage was considered because it is limiting. This table shows that the change in the distribution of the number of flaws only slightly affected the probability of leakage. A multiplier of 1.5 was applied to all stresses.

Table 8-21
Sensitivity to the Distribution of Number of Flaws on Probability of Leakage (Case FEW-P3A)

No. of Flaws	Probability of Leakage (per year) (Stress Multiplier = 1.5)
1 Flaw, Constant	1.04E-06
1 Flaw, Poisson Distributed	7.23E-07

8.2.4.3.6 Sensitivity to Crack Size Distribution

In the base case, the PVRUF crack size distribution was used because it was the one recommended by the NRC in its review of BWRVIP-05 and BWRVIP-108. Two other distributions were investigated for their effect on the probability of leakage per year: the Marshall distribution and a distribution derived by SI based on data from NUREG/CR-6817 [60, 61]. The results of the evaluation shown in Table 8-22 indicate that the PVRUF distribution is similar to the Marshall distribution. This table also indicates that the PVRUF distribution is conservative compared to the distribution derived by SI [60, 61]. A multiplier of 1.5 was applied to all stresses.

Table 8-22
Sensitivity of Crack Size Distribution on Probability of Leakage (Case FEW-P3A) for 80 Years

Flaw Distribution	Probability of Leakage (per year) (Stress Multiplier = 1.5)
PVRUF	1.04E-06
Marshall	1.12E-06
SI	9.20E-07

8.2.4.3.7 Sensitivity to Crack Growth Rate

The sensitivity to crack growth rate was performed by replacing the FCG rate used in the base case with the flaw growth distribution used in BWRVIP-108 [2]. In BWRVIP-108, the crack growth law was derived from data in Reference [78] at a conservative R-ratio of 0.7. This FCG law is compared to the ASME Code Section XI curve shown in Figure 8-10. The results are presented in Table 8-23 and show that the probability of leakage is much lower for the BWRVIP-108 model compared to the ASME Code Section XI, Appendix A, A-4300 model used in the base case. A multiplier of 1.5 was applied to all stresses.

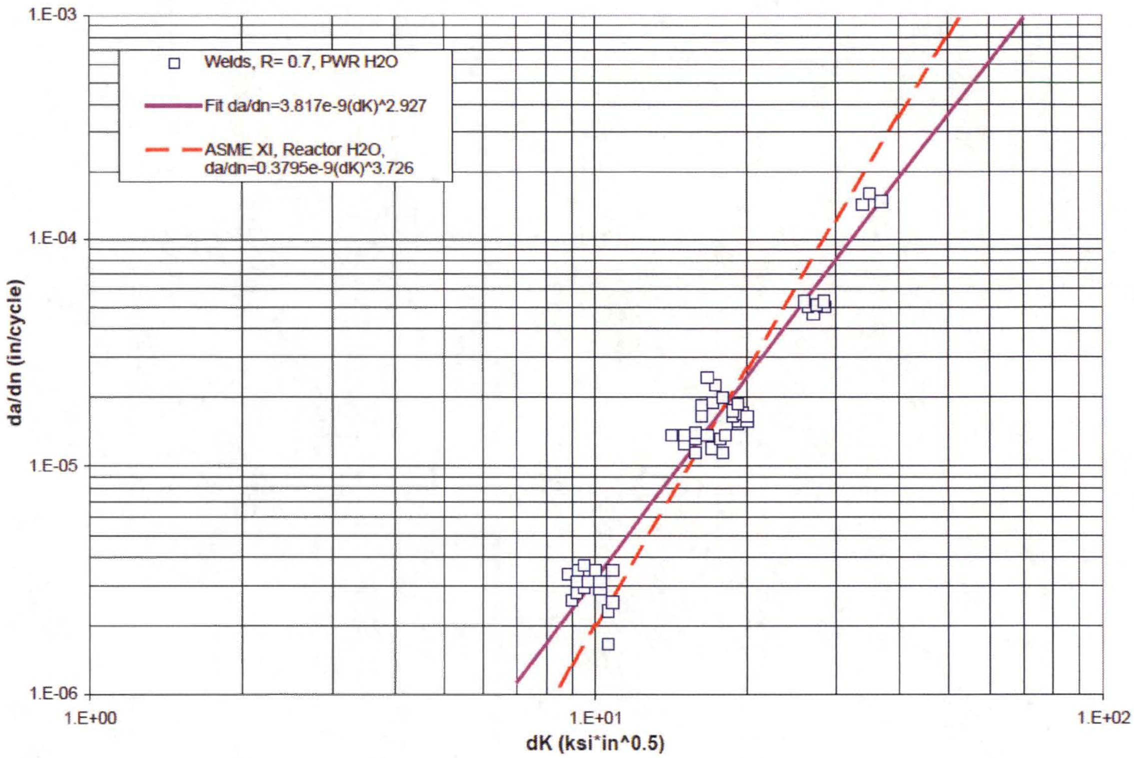


Figure 8-10
Curve Fit for Fatigue Crack Growth at R-Ratio of 0.7 and Comparison to ASME Code Section XI Crack Growth Curve [2]

Table 8-23
Sensitivity to Crack Growth Rate on Probability of Leakage (FEW-P1N) for 80 Years

Distribution	Probability of Leakage per Year
ASME A-4300	3.20E-07
BWRVIP-108	2.50E-12

8.2.4.3.8 Sensitivity to POD Curves

To evaluate the effect of the POD curve on leakage probabilities, the BWRVIP-108 POD curve used in the base case (Figure 8-2) was replaced with the POD curve used in the development of ASME Code Section XI, Appendix L (Figure 8-3). These two POD curves are compared in Figure 8-11. The probabilities of rupture and leakage for the two PODs are shown in Tables 8-24 and 8-25. These tables show only slight variations between the two cases, with the probability of leakage slightly higher in a few of the cases.

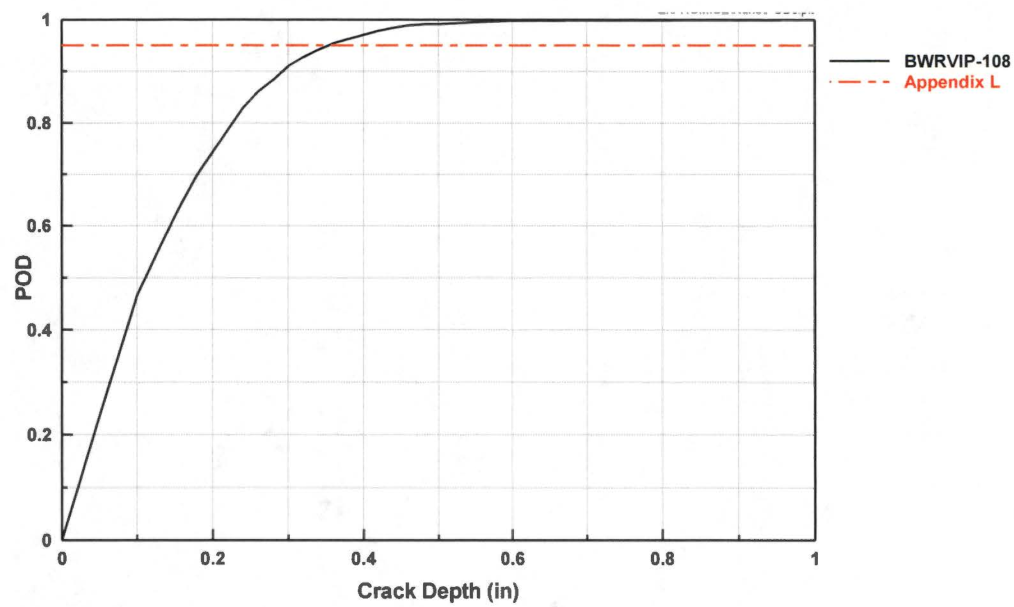


Figure 8-11
POD Curves for Sensitivity Study

Table 8-24
Sensitivity of POD on Probability of Rupture for 80 Years

Component	Case Identification	Probability of Rupture (per year)	
		Base Case, BWRVIP-108 POD Curve	Appendix L POD Curve
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	1.25E-12
	SGW-P2C	1.25E-09	1.25E-09
	SGW-P2A	1.25E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	1.25E-12
	SGB-P2N	1.25E-12	1.25E-12
	SGB-P3C	1.25E-09	1.25E-09
	SGB-P3A	1.25E-09	1.25E-09
	SGB-P4A	1.25E-09	1.25E-09
	SGB-P4C	1.25E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	1.25E-12
	FEW-P2N	1.25E-12	1.25E-12
	FEW-P3C	1.25E-09	1.25E-09
	FEW-P3A	1.25E-09	1.25E-09
	FEW-P4A	1.25E-09	1.25E-09
	FEW-P4C	1.25E-09	1.25E-09

Table 8-25
Sensitivity of POD on Probability of Leakage for 80 Years

Component	Case Identification	Probability of Leakage (per year)	
		Base Case, BWRVIP-108 POD Curve	Appendix L POD Curve
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	1.25E-12
	SGW-P2C	1.25E-09	1.25E-09
	SGW-P2A	1.25E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	1.25E-12
	SGB-P2N	1.25E-12	1.25E-12
	SGB-P3C	1.25E-09	1.25E-09
	SGB-P3A	1.25E-09	1.25E-09
	SGB-P4A	1.25E-09	1.25E-09
	SGB-P4C	1.25E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.88E-11	5.00E-11
	FEW-P2N	1.25E-12	1.25E-12
	FEW-P3C	1.25E-09	3.75E-09
	FEW-P3A	2.50E-09	**2.25E-08**
	FEW-P4A	1.25E-09	3.75E-09
	FEW-P4C	1.25E-09	3.75E-09

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

8.2.4.3.9 Sensitivity to Inspection Schedules Using Two Different PODs

Sensitivity studies were performed to determine the effect of several of the inspection schedules listed in Table 8-2. The evaluation was performed using the limiting component in terms of leakage (FEW-P3A), considering both the base case POD curve and the Appendix L POD curve. Because the base case FEW-P3A leak probabilities are very low, the stress multiplier was changed to 1.5 with a standard deviation of 0.2 to generate the results. These scenarios, along with the results for both the POD curves, are shown in Table 8-26. The results show that the base case POD curve (BWRVIP-108) yields conservative results when compared to Appendix L POD curve for most inspection scenarios. The results for both POD curves are plotted as a function of time in Figure 8-12 for the case of inspections at 0, 10, and 20 years, showing that the leakage probabilities are similar.

Table 8-26
Sensitivity of Inspection Schedule on Probability of Leakage (Case FEW-P3A) for 80 Years

	Probability of Leakage (per year)	
Inspection at (years)	**BWRVIP-108 POD (Stress Multiplier = 1.5, SD = 0.2)**	**Appendix L POD (Stress Multiplier = 1.5, SD = 0.2)**
0	1.19E-03	1.08E-04
0,10	3.58E-04	5.45E-06
0, 10, 20	3.94E-05	2.95E-07
0, 10, 20, 30	2.15E-06	5.08E-08
0, 20, 40, 60	1.44E-06	9.14E-07
0, 10, 30, 50, 70	4.20E-07	2.07E-07
0, 10, 20, 40, 60	3.38E-08	5.95E-08
0, 10, 20, 30, 50, 70	1.00E-08	4.11E-08

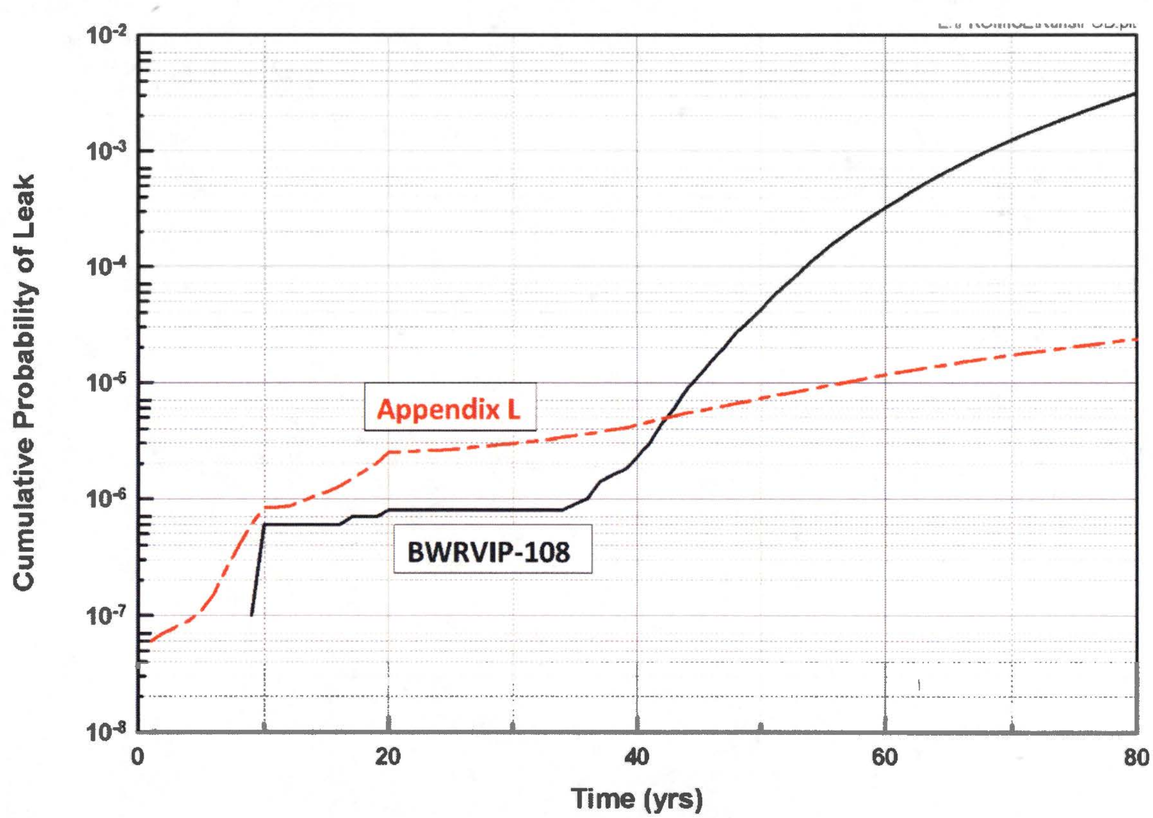


Figure 8-12
Sensitivity of Inspection Schedule and POD Curves on Probability of Leakage

8.2.4.3.10 Sensitivity to Number of Realizations

A sensitivity study was performed to determine the effect of the number of realizations used in the evaluations. Case FEW-P3A with a stress multiplier of 1.5 was selected for this study. As shown in Table 8-27, with 10^7 and 10^8 aleatory realizations, both cases resulted in approximately the same leakage probability.

Table 8-27
Convergence Test for Case FEW-P3A

No. of Aleatory Realizations	Probability of Leakage (per year)
10^7	1.04E-06
10^8	1.09E-06

8.2.4.3.11 Sensitivity Study to Determine Combined Effect of Important Parameters

Based on the sensitivity studies performed, the following are the most important input parameters:

- Fracture toughness
- Stress
- Flaw density (for the nozzle inner radius location)

A sensitivity study was performed to determine the combined effect of these three variables. A mean fracture toughness of 200 ksi$\sqrt{}$in. with a conservative standard deviation of 30 ksi$\sqrt{}$in., a stress multiplier of 1.5, and a nozzle inner radius flaw density of 0.1 (instead of the base case of 0.001) were used in the sensitivity study. For the nozzle-to-shell welds, the flaw density was kept at 1.0 per nozzle. Two cases were considered. In the first case, the analysis was performed using the base case inspection scenario, which consists of PSI followed by 20-year ISI intervals. The results of this case are shown in Table 8-28. The results indicate that, even with the conservative inputs, the acceptance criteria are met at all locations except one (which only slightly exceeds the acceptance criteria). In the second case, the analysis was performed assuming a realistic inspection scenario of PSI, followed by two 10-year ISI inspections followed by two 20-year ISI inspections. The results of this case are shown in Table 8-29. As this table indicates, once a 10-year ISI inspection has been performed before changing to 20-year ISI intervals, the acceptance criteria are met at all locations. This second case is very representative of operating plants because most have performed at least one 10-year ISI inspection. It should be noted that, because of the increase in the nozzle inner radius flaw density by two orders of magnitude from the base case, the limiting case for this sensitivity study is a nozzle inner radius location (FEW-P1N).

Table 8-28
Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years: Case 1

Component	Case Identification	Probability of Rupture (per year)	
		Base Case	Combined Case K_{IC} = 200 ksi√in. SD = 30 ksi√in. Stress Multiplier = 1.5 Nozzle Flaw Density = 0.1 PSI+20+40+60
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	1.25E-10
	SGW-P2C	1.25E-09	1.25E-09
	SGW-P2A	1.25E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	2.50E-10
	SGB-P2N	1.25E-12	1.25E-10
	SGB-P3C	1.25E-09	1.25E-09
	SGB-P3A	1.25E-09	1.25E-09
	SGB-P4A	1.25E-09	1.25E-09
	SGB-P4C	1.25E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	**5.30E-06**
	FEW-P2N	1.25E-12	1.25E-10
	FEW-P3C	1.25E-09	1.25E-09
	FEW-P3A	1.25E-09	3.13E-07
	FEW-P4A	1.25E-09	1.25E-09
	FEW-P4C	1.25E-09	1.25E-09

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

Table 8-29
Sensitivity to Combined Effects of Fracture Toughness, Stress, and Nozzle Flaw Density for 80 Years: Case 2

Component	Case Identification	Probability of Rupture (per year)	
		Base Case	Combined Case K_{IC} = 200 ksi√in. SD = 30 ksi√in. Stress Multiplier = 1.5 Nozzle Flaw Density = 0.1 PSI+10+20+40+60
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	1.25E-12	2.50E-10
	SGW-P2C	1.25E-09	1.25E-09
	SGW-P2A	1.25E-09	1.25E-09
B&W Main Steam Nozzle (SGB)	SGB-P1N	1.25E-12	3.75E-10
	SGB-P2N	1.25E-12	1.25E-10
	SGB-P3C	1.25E-09	1.25E-09
	SGB-P3A	1.25E-09	1.25E-09
	SGB-P4A	1.25E-09	1.25E-09
	SGB-P4C	1.25E-09	1.25E-09
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	1.25E-12	**2.02E-07**
	FEW-P2N	1.25E-12	1.25E-10
	FEW-P3C	1.25E-09	1.25E-09
	FEW-P3A	1.25E-09	2.50E-08
	FEW-P4A	1.25E-09	1.25E-09
	FEW-P4C	1.25E-09	1.25E-09

Note: Changed values from the base case are shaded in yellow, and the limiting value is displayed in **red bold** text.

8.2.4.3.12 Summary of Sensitivity Studies

The sensitivity studies performed in the previous sections show that probabilities of failure are not significantly affected by using alternative distributions for the input variables. The key variables that affect the outcome of the analysis (fracture toughness, stress, and nozzle flaw density) were combined with realistic inspection scenarios, and it was concluded that the acceptance criteria for probability of rupture and leakage are still met. The results of some of the key sensitivity studies are summarized in Figure 8-13.

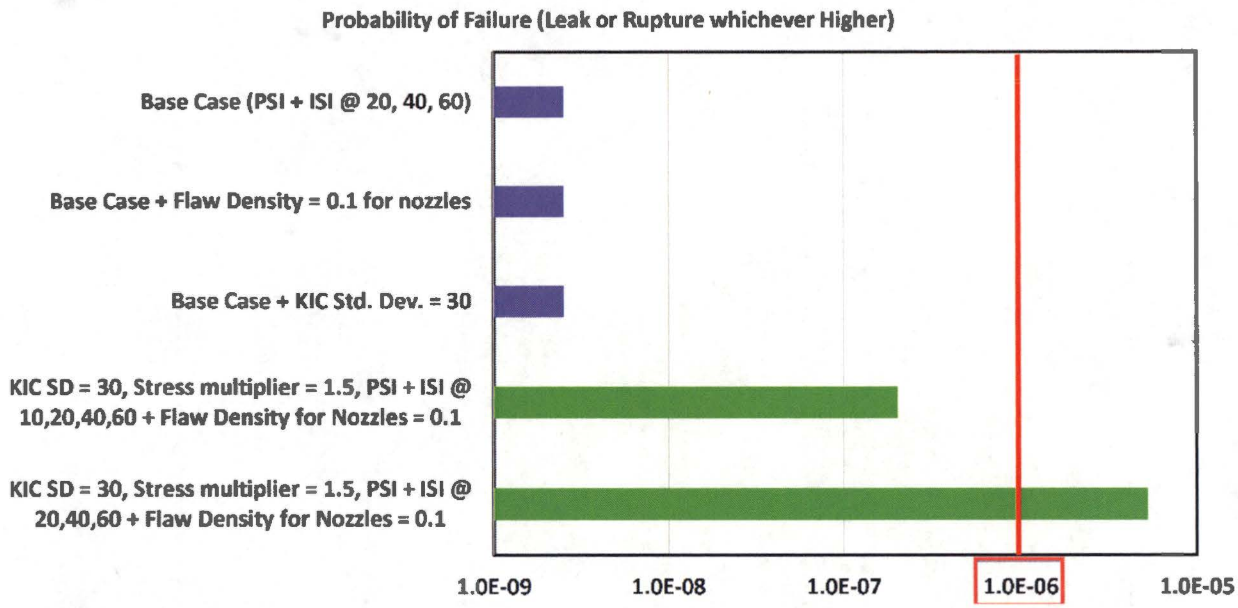


Figure 8-13
Summary of Key Sensitivity Analyses

8.2.5 Inspection Coverage

The evaluations presented so far assume that 100% of the inspection volume specified in ASME Code Section XI for the main steam and feedwater nozzles has been achieved during the examinations assumed in each of the inspection scenarios. However, as discussed in Section 4, there have been instances in which 100% coverage of the components has not been achieved during ISI examinations for some plants because of access restrictions and limitations. These components did receive 100% coverage during PSI through a combination of the required Section III fabrication and Section XI pre-service examinations. Therefore, the main steam and feedwater nozzles for all plants, at a minimum, had complete 100% coverage PSI examinations followed by at least one partial coverage ISI examination, depending on the length of service of replacement SGs at each plant.

As discussed in Section 8.2.4.1.1, by performing complete 100% coverage PSI examinations, no other inspections are required for the components for 80 years to maintain safe plant operation due to the relatively low failure probabilities for rupture and leakage. Therefore, any additional ISI examinations (including limited inspections resulting from partial coverage) after the PSI examination reduces the already low probability of rupture and leakage for these components. As such, the results of the evaluations presented herein may also be applied to components with partial coverage.

8.3 Deterministic Fracture Mechanics Evaluation

A DFM evaluation was performed using average parameters to provide verification of the PFM evaluation results. The very low probability of failure obtained from the PFM evaluation indicates that a long operating period should be obtained from a DFM evaluation before failure is predicted. All configurations considered in the PFM evaluation are also considered in the DFM evaluation.

8.3.1 Technical Approach

The technical approach used in the DFM evaluation is to assume an initial flaw size equivalent to the ASME Code Section XI acceptance standard and use the ASME Code Section XI FCG law with the through-wall stress distributions from Section 7 and the fracture mechanics models described in Section 8.1.2.3 to determine the length of time for the postulated 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.

8.3.2 Design Inputs

The design inputs used in the DFM evaluation are shown in Table 8-30, which are essentially the same as those used in the PFM evaluation, but they are all deterministic.

Table 8-30
DFM Base Case Inputs

Geometry	Same as used in the PFM
Initial crack size	5.2% of the thickness, c/a = 1
Fracture toughness	200 ksi$\sqrt{\text{in.}}$
Fatigue crack growth law	ASME Code, Section XI Appendix A, A-4300
Operating transient stresses	Same as used in the PFM
Residual stresses	Cosine curve (8, 8)

8.3.3 Results of Deterministic Fracture Mechanics Evaluation

The results of the DFM evaluation are summarized in Table 8-31. As expected, the period required for a flaw to leak is considerably long, which provides verification of the PFM evaluation. Because the DFM evaluation is considering a hypothetical postulated flaw, a structural factor of 2 on primary loads and 1 on secondary loads consistent with ASME Code Section XI, Appendix G can be applied. Because the most dominant load is pressure, which results in a primary stress, the structural factor of 2 is conservatively applied to the fracture toughness of 200 ksi$\sqrt{\text{in}}$. This results in an allowable fracture toughness of 100 ksi$\sqrt{\text{in}}$. Reviewing the results in Table 8-31, the maximum K value is either below or (in one case) only slightly exceeds this allowable fracture toughness after 80 years.

Table 8-31
Results of Deterministic Fracture Mechanics Analyses

Component	Case Identification	Years to Leak	Max. K at 80 Years (ksi√in.)
Westinghouse Main Steam Nozzle (SGW)	SGW-P1N	795	38
	SGW-P2C	950,000	8
	SGW-P2A	8221	18
B&W Main Steam Nozzle (SGB)	SGB-P1N	716	40
	SGB-P2N	4660	16
	SGB-P3C	32,640	14
	SGB-P3A	1086	36
	SGB-P4A	36,190	12
	SGB-P4C	5529	20
Westinghouse Feedwater Nozzle (FEW)	FEW-P1N	**147**	**101**
	FEW-P2N	1890	13
	FEW-P3C	4612	15
	FEW-P3A	396	27
	FEW-P4A	3468	15
	FEW-P4C	7998	13

Note: Limiting value is displayed in **red bold** text.

8.4 Concluding Remarks on PFM and DFM Evaluations

From the PFM and DFM evaluations performed in this section, the following observations are made:

- All components considered in this study are very flaw-tolerant. According to the PFM evaluation, once PSI has been performed, the failure probabilities (in terms of both rupture and leakage) are very low and are significantly below the acceptance criteria of 10^{-6} failures per year after 60 years of operation.
- For the base case consisting of PSI followed by 20-year ISI inspections, the failure probabilities (in terms of both rupture and leakage) are significantly below the acceptance criteria of 10^{-6} failures per year after 80 years of operation.

- Sensitivity analyses were performed on most input parameters to identify the key variables that significantly affect the results of the evaluation. The input parameters that significantly impact the results are the fracture toughness of the SG vessel for the probability of rupture and the crack growth rate for the probability of leakage. The following items were also considered in the sensitivity studies:
  - A conservative upper shelf toughness value of 200 ksi√in. was used as the mean toughness in the PFM analysis. The sensitivity study indicated that, even if the fracture toughness was as low as 80 ksi√in., the acceptance criteria would still be met.
  - An alternative crack growth rate (used in BWRVIP-108) was also used to determine the effect on the probability of leakage and resulted in lower values compared to the ASME Section XI Appendix A FCG law.
  - When PSI is followed by 20-year ISI intervals, stress multipliers of 1.5 for the feedwater nozzle and 2.2 for the main steam nozzle can be applied and still meet the acceptance criteria. If the 20-year ISI inspections are preceded by the 10-year ISI inspections already performed by most plants to-date, these stress multipliers would increase further.
- The DFM evaluation verified the results of the PFM evaluation by demonstrating that it would take a very long operating period (approximately 100 years) for a postulated initial flaw (with a depth equal to the ASME Code Section XI acceptance standards) to propagate through-wall and cause leakage. After 80 years, the maximum K obtained from the analysis is below or close to the ASME Code Section XI allowable fracture toughness, including the Section XI Appendix G structural factor for primary stress, indicating that ASME Code Section XI structural margins have been met.
- The PFM and DFM evaluations presented in this section incorporate many conservative inputs and, as such, the results are considered very conservative. The following summarizes the primary inputs and their conservatisms:
  - **Stresses:** The temperature gradients and number of cycles used for this analysis are conservative. For example, typical design cycles for heat-up and cooldown are 200 cycles for 40 years, which linearly projects to 300 cycles for 60 years. Typically, the actual number of operating cycles is less than the design values, making the number of cycles used conservative.
  - **Fracture toughness:** The mean fracture toughness value (200 ksi√in.) used in this analysis is a lower bound, as described in Section 8.2.2.6. In addition, a standard deviation of 5 ksi√in. is used, which will produce a K_{IC} below the lower bound in 50% of the realizations.
  - **Fatigue crack growth:** The ASME Code A-4300 curve, with a 95% confidence limit on the data, is used as the median curve in this analysis, making it conservative.

Based on the evaluations performed in this section and the above observations regarding conservatism of the inputs, it is concluded that the current ASME Code Section XI inspection schedules for the main steam and feedwater nozzle-to-shell welds and nozzle inside radius sections listed under Item Nos. C2.21, C.2.22, and C2.32 in Table IWC-2500-1 (essentially every 10 years) are conservative. The evaluations have shown that, after PSI, no other inspections are necessary up to 60 years of plant operation to meet the NRC safety goal of 10^{-6} failures per year. When an inspection scenario consisting of PSI followed by 20-year ISI inspections is considered, the NRC safety goal is met beyond 80 years of plant operation.

9 PLANT-SPECIFIC APPLICABILITY

The evaluations performed in this report used representative geometries, materials, and loading conditions associated with PWR SG feedwater and the main steam nozzles. However, there are variations in nozzle configurations among PWR plants, especially with the main steam nozzle. Sensitivities studies were therefore performed to determine the effects of key input variables such as geometry, material properties, and stress on the ability to meet acceptance criteria. This section describes the parameters that must be evaluated by plants to determine the plant-specific applicability of this work.

9.1 Geometric Configurations

As discussed in Section 4, the feedwater nozzle geometry is relatively uniform across the various PWR designs. Typical diameters of the feedwater piping range from 14-in. to 18-in. NPS with a corresponding nominal thickness. The stress sensitivity study performed in Section 8 showed that for the feedwater nozzle, a stress multiplier of at least 1.5 can be applied to account for geometric differences and the component would still meet the failure acceptance criteria. A feedwater nozzle design with a thermal sleeve (the standard design) was used in this evaluation; therefore, the analysis presented in this report applies to all feedwater nozzles with thermal sleeves. The analysis is also limited to main feedwater nozzles and does not apply to auxiliary feedwater nozzles directly connected to the steam generator.

In addition, as discussed in Section 4, the main steam nozzle geometry differs across the various NSSS vendors; therefore, Westinghouse/CE and B&W designs were analyzed separately. For the Westinghouse and CE designs, typical diameters of the main steam piping range from 28-in. to 36-in. NPS. The stress sensitivity study performed in Section 8 showed that for the main steam nozzle, a stress multiplier of at least 2.2 can be applied to account for geometric differences and the component would still meet the failure acceptance criteria. This multiplier is much larger than the variation in stress of 1.5 developed in Section 7.4 for some of the most significantly varying main steam nozzle designs; therefore, all Westinghouse/CE main steam nozzles are addressed by the evaluations in this report. This report did not evaluate SGs with multiple main steam nozzles or ones that significantly protrude into the steam generator (that is, the CE System 80 design). Otherwise, the analysis presented in this report applies to all Westinghouse and CE (non-System 80 design) main steam nozzles.

For the B&W design, the typical diameter of the main steam piping is 24-inch NPS, with no variations noted across the plants surveyed. The sensitivity study performed in Section 8 showed that for the main steam nozzle, a stress multiplier as high as 2.2 could potentially be applied to account for any geometric differences and the component would still meet the failure acceptance criteria. As such, small deviations from this standard geometry can be accommodated. Therefore, the analysis presented in this report applies to all B&W main steam nozzles.

9.2 Material Properties

The material properties used in performing the stress analyses are typical of low-alloy ferritic steel pressure vessels. Most (if not all) SG nozzles and shell material are fabricated from low-alloy ferritic steel and possibly some with carbon steel (B&W design); therefore, the material properties used should be applicable to most SGs. The fracture toughness used in the evaluation is also based on the materials being low-alloy ferritic steels and conforms to the requirements of ASME Code Section XI, Appendix G, Paragraph G-2110. Therefore, the analysis presented in this report applies to all plants with low-alloy (and carbon) ferritic steel SG nozzle and shell materials.

9.3 Operating Transients

The operating transients and associated cycles evaluated in this report are summarized in Table 5-5 and are typical of historical PWR operations, though variations may exist among plant designs and among individual plants. In the fracture mechanics evaluations performed for this report, the transients that contribute the most to crack growth are heat-up and cooldown events. A total of 300 such events were considered during a 60-year plant life. The sensitivity study performed in Section 8 showed that stress multipliers of 1.5 for the feedwater nozzle and 2.2 for the main steam nozzle can accommodate small variations in the pressures and temperatures of the evaluated transients. Design transient severities were used in this evaluation, which have been shown through many years of industry fatigue monitoring to be conservative compared to actual transient severities experienced from plant operations. Therefore, the stress analysis presented in this report applies to all plants with transient numbers of cycles less than or equal to those shown in Table 5-5.

9.4 Criteria for Technical Basis Applicability

Based on the previous three sections, the following summarizes the criteria that must be met for the results of this report to be applied to a specific PWR plant.

9.4.1 General

- The nozzle-to-shell weld must be one of the configurations shown in Figure 1-1 or Figure 1-2.
- The materials of the SG shell, feedwater nozzle, and main steam nozzle must be low-alloy ferritic steels that conform to the requirements of ASME Code, Section XI, Appendix G, Paragraph G-2110.
- The SG must not experience more than the number of all transients shown in Table 5-5 over a 60-year operating life.

9.4.2 SG Feedwater Nozzle

- The piping attached to the feedwater nozzle must be 14-in. to 18-in. NPS.
- The feedwater nozzle design must have an integrally attached thermal sleeve.
- Auxiliary feedwater nozzles connected directly to the SG are not covered in this evaluation.

9.4.3 SG Main Steam Nozzle

- For Westinghouse and CE SGs, the piping attached to the main steam nozzle must be 28-in. to 36-in. NPS.
- For B&W SGs, the piping attached to the main steam nozzle must be 22-in. to 26-in. NPS.
- The SG must have one main steam nozzle that exits the top dome of the SG. For B&W plants, there may be more than one main steam nozzle; it will exit the side of the SG.
- The main steam nozzle shall not significantly protrude into the SG (for example, see Figure 4-7) or have a unique nozzle weld configuration (for example, see Figure 4-6).

10
SUMMARY AND CONCLUSIONS

This report has developed the technical bases for inspection requirements of the nozzle-to-shell welds and the nozzle inside radius sections of PWR steam generator feedwater and main steam nozzles, which are listed in ASME Code, Section XI, Table IWC-2500-1 under the following item numbers:

- Item No. C2.21: Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds
- Item No. C2.32: Nozzle-to-shell (nozzle-to-head or nozzle-to-nozzle) welds when inside of vessel is accessible
- Item No. C2.22: Nozzle inside radius section

To establish the technical bases for these inspections, several topics were addressed—the findings are summarized as follows.

- A detailed study was performed on previous related projects that have involved ISI technical bases in the industry; it was concluded that there are several precedents in the industry for such a technical basis. Inspection requirements for many similar components such as Class 1 BWR feedwater nozzles and Class 1 BWR and PWR RPV welds have been addressed in the past.
- A comprehensive survey was conducted for the U.S. and international PWR fleet to collect the number of examinations performed and associated examination results for Item Nos. C2.21, C2.32, and C2.22, including the PWR SG feedwater and main steam nozzle components evaluated in this report. The survey results showed that of a total of 727 examinations for plants responding to the survey that were performed on Item Nos. C2.21, C2.32, and C.22 BWR and PWR components, only one (PWR) examination (for Item No. C2.21) identified indications that exceeded the acceptance criteria of ASME Code Section XI (they were dispositioned by light grinding). The vast majority of the 563 of these examinations performed for PWRs were for the SG feedwater and main steam nozzle components evaluated in this report.
- PWR SG designs and operating experience were reviewed. Information was also reviewed regarding variability among SG designs in terms of dimensions, design pressures and temperatures, and ASME Code design considerations as well as information on configurations with known limitations. The main conclusion was that, although variations exist among SG designs, the configurations are similar enough from a stress standpoint that any differences could be addressed in the PFM and DFM evaluations through sensitivity analyses using stress multipliers. In this manner, and because all plants worldwide did not respond to the survey, representative components were selected for evaluation based on the plants that did respond to the survey, the factors discussed in Section 4.3, and a set of related criteria.

- SG operating transients were reviewed, and a set of representative transients with design basis severities and conservative numbers of occurrences for 60 years was established for the feedwater and the main steam nozzles. The operating transients were then used in stress analyses to determine through-wall stresses at critical locations of the feedwater and main steam nozzle-to-shell welds and nozzle inside radius sections. The stress analyses included evaluation of one representative feedwater nozzle and two representative main steam nozzle configurations. Additional sensitivity analyses were performed to address variations in nozzle geometries across the fleet.
- A degradation mechanism evaluation was performed for all PWR SG Item No. C2.22 nozzle inside radius section components as well as all PWR SG Item Nos. C2.21 and C2.32 nozzle-to-shell welds. It was concluded that the only potential degradation mechanism for these components is fatigue (that is, corrosion fatigue, mechanical fatigue, and/or thermal fatigue). Therefore, this mechanism was considered in the PFM and DFM evaluations for these components.
- Comprehensive PFM evaluations were performed using the NRC safety goal of 10^{-6} failures per year as the acceptance limit for various inspection scenarios. These evaluations were supplemented by DFM evaluations. The results of the PFM and DFM evaluations showed the following:
  - All components considered in this study are very flaw-tolerant. From the PFM studies, once PSI has been performed, the probability of rupture is below the NRC safety goal of 10^{-6} failures per year after 80 years of plant operation, while the probability of leakage is about an order of magnitude below the safety goal for 80 years. Therefore, from a safety viewpoint, no other inspections are required through 80 years of plant operation. The evaluation also considered limited coverage during subsequent ISIs to address components for which full examination coverage cannot be obtained due to physical obstructions present for some components in some plants.
  - For an inspection scenario consisting of PSI followed by 20-year ISI inspections, the failure probabilities (both in term of rupture and leakage) are significantly below the acceptance criteria of 10^{-6} failures per year after 80 years of operation. The results are even more favorable if previous 10-year inspections are combined with the 20-year inspection interval.
  - A sensitivity analysis was performed to identify key input parameters that have a significant effect on the results of the evaluation. Sensitivity studies were then performed on all key parameters. The most significant input parameters that affect the probabilities of failure and leakage were determined to be the fracture toughness of the SG vessel (for the probability of rupture) and the crack growth rate (for the probability of leakage). Other key results included the following:
    - A conservative upper shelf toughness value of 200 ksi$\sqrt{}$in. was used as a mean value with a standard deviation of 5 ksi$\sqrt{}$in. in the PFM analysis. The sensitivity study indicated that, even if the fracture toughness was as low as 150 ksi$\sqrt{}$in. or the standard deviation was increased to 30 ksi$\sqrt{}$in., the acceptance criteria would still be met.

  - An alternative crack growth rate that was investigated in BWRVIP-108 was used to evaluate the effect of crack growth rate on the probability of leakage. The alternative crack growth rate resulted in lower probabilities of leakage compared to the Section XI Appendix A crack growth rate relationship.
  - When PSI is followed by 20-year ISI intervals, stress multipliers of 1.5 for the feedwater nozzle and 2.2 for the main steam nozzle can be applied and still meet the acceptance criteria. If the 20-year ISI inspections are preceded by the 10-year ISI inspections already performed by most plants to-date, these stress multipliers would increase further.

- The DFM evaluations provided verification of the PFM results by demonstrating that it would take in excess of 100 years for an initial postulated flaw (with a depth equal to the ASME Code Section XI acceptance standards) to grow to 80% of the wall thickness (conservatively assumed as a leak for the purposes of this evaluation). The maximum K obtained from the DFM analyses remained below or very close to the ASME Code Section XI allowable fracture toughness (K_{IC} divided by a structural factor of 2), indicating that the ASME Code Section XI structural margins are maintained.

- The evaluations presented in this report incorporate conservative inputs and, as such, the results are considered conservative. The following two items are the primary conservatisms:
  - Fracture toughness: The mean fracture toughness, K_{IC}, value of 200 ksi√in. was used in this analysis. This value is a lower bound value, as described in Section 8.2.2.6. In addition, a standard deviation of 5 ksi√in. was used, which produced a K_{IC} of less than the lower bound in 50% of the realizations. Sensitivity studies were performed for K_{IC} values as low as 80 ksi√in. and a standard deviation of 30 ksi√in. that still showed acceptable results.
  - Fatigue crack growth: The ASME Code, Section XI A-4300 fatigue crack growth curve was used. This relationship, which represents a 95% confidence limit on the data, was assumed to be a median curve in this analysis.

Based on the evaluations performed in this report, technical bases were developed for various ASME Code, Section XI inspection schedules for the main steam and feedwater nozzle-to-shell welds and nozzle inside radius sections listed under Item Nos. C2.21, C.2.22, and C2.32 in Table IWC-2500-1 (essentially every 10 years). These results can be used by PWR plants, using the plant-specific applicability requirements in Section 9, to establish optimized inspection schedules for PWR SG main steam and feedwater nozzles.

11
REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, 2017 Edition.
2. *BWRVIP-108: BWR Vessels and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii*. EPRI, Palo Alto, CA: 2002. 1003557.
3. U.S. NRC, Safety Evaluation of Proprietary EPRI Report, *BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP-108)*, December 19, 2007, ADAMS Accession Number ML073600374.
4. *BWRVIP-241: BWR Vessels and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii*. EPRI, Palo Alto, CA: 2010. 1021005.
5. U.S. NRC, Safety Evaluation of Proprietary EPRI Report, *BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Shell Welds and Nozzle Blend Radii (BWRVIP-241)*, April 19, 2013, ADAMS Accession Numbers ML13071A240 and ML13071A233.
6. Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds," *ASME Code Section XI, Division 1*, Approval Date: February 20, 2004.
7. U.S. NRC Regulatory Guide 1.147, Revision 17, "In-Service Inspection Code Case Acceptability," *ASME Code Section XI, Division 1*, August 2014.
8. B. A. Bishop, C. Boggess, and N. Palm, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval," *WCAP-16168-NP-A, Rev. 3*, October 2011.
9. Code Case N-691, "Application of Risk-Informed Insights to Increase the Inspection Interval for Pressurized Water Reactor Vessels," *ASME Code Section XI, Division 1*, Approval Date: November 18, 2003.
10. U.S. NRC, "Revised Safety Evaluation by the Office of Nuclear Reactor Regulation; Topical Report WCAP-16168-NP-A, Revision 2, *Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval*, Pressurized Water Reactor Owners Group, Project No. 694," July 26, 2011, ADAMS Accession Number ML111600303.
11. Code Case N-706-1, "Alternative Examination Requirements of Table IWB-2500-1 and IWC-2500-1 for PWR Stainless Steel Residual and Regenerative Heat Exchangers," *ASME Code Section XI, Division 1*, Approval Date: January 10, 2007.

12. W. Bamford, B. Bishop, R. Haessler, and M. Bowler, "Technical Basis for Revision of Inspection Requirements for Regenerative and Residual Heat Exchangers," Paper No. PVP2006-ICPVT-11-93892, *Proceedings of ASME 2006 Pressure Vessels and Piping Division Conference*, Vancouver, British Columbia, Canada, July 23–27, 2006.

13. Letter from R. Necci (Northeast Nuclear Energy) to U.S. NRC, Millstone Nuclear Power Station Unit No. 2, Request to Use an Alternative to ASME Code Section XI Code Case N-619, October 22, 1999, Docket No. 50-336, B17868, ADAMS Accession Number ML993070099.

14. Letter from J. B. Beasley (Southern Company) to U.S. NRC, Vogtle Electric Generating Plant, Request for Relief – Steam Generator Nozzle Inner Radius Examination, April 26, 2000, Docket No. 50-424, ADAMS Accession Number ML003710824.

15. Letter from B. S. Allen (PG&E) to U.S. NRC, ASME Section XI Inspection Program Request for Relief NDE-SG-MS-IR, Main Steam Nozzle Inner Radius Examination Impracticality, August 24, 2015, Docket Nos. 50-275, OL-DPR-80, and 50-323, OL-DPR-82, Diablo Canyon Power Plant (DCPP) Units 1 and 2, ADAMS Accession Number ML15236A376.

16. Code Case N-619, "Alternative Requirements for Nozzle Inner Radius Inspection for Class 1 Pressurizer and Steam Generator Nozzles," *ASME Code Section XI, Division 1*, Approval Date: February 15, 1999.

17. W. H. Bamford, B. A. Bishop, D. Kurek, N. Closky, G. Stevens, L. Becker, and S. Ranganath, "Technical Basis for Elimination of Reactor Vessel Nozzle Inner Radius Inspections," *Proceedings of ASME 2001 Pressure Vessels and Piping Conference*, Atlanta, Georgia.

18. Code Case N-648-1 "Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles," *ASME Code Section XI, Division 1*, Approval Date: September 7, 2001.

19. *BWR Vessels and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05).* EPRI, Palo Alto, CA: 1995. TR-105697.

20. Code Case N-560-2, "Alternative Inspection Requirements for Class 1, B-J Piping Welds," *ASME Code Section XI, Division 1*, Approval Date: March 28, 2000.

21. *Revised Risk-Informed In-Service Inspection Evaluation Procedure, Revision B-A*. EPRI, Palo Alto, CA: 1999. TR-112657.

22. Code Case N-577-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping. Method A," *ASME Code Section XI, Division 1*, Approval Date: March 28, 2000.

23. Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping. Method B," *ASME Code Section XI, Division 1*, Approval Date: March 28, 2000.

24. WCAP-14572, *Westinghouse Owners Group, Application of Risk-Informed Methods to Piping In-Service Inspection Topical Report*, Revision 1-NP-A, February 1999.

25. Code Case N-716-1, "Alternative Piping Classification and Examination Requirements," *ASME Code Section XI, Division 1*, Approval Date: January 27, 2013.

26. Report No. 2002-02A-01, *Risk-Informed/Safety Based (RIS_B) Technology, A Blended Approach to Defining Examination Requirements, ASME Section XI, Working Group on the Implementation of Risk-Based Examinations, White Paper in Support of Code Case N716 (Item Nos. RI-02-02A, SGWCS 0301, BC04-318),* Revision 0, October 2005.

27. Report No. 2002-02A-01, *Risk-Informed/Safety Based (RIS_B) Technology, A Blended Approach to Defining Examination Requirements, ASME Section XI, Working Group on the Implementation of Risk-Based Examinations, White Paper in Support of Revision 1 to Code Case N716 (Item Nos. RI-02-02A, SGWCS 0301, BC04-318),* Revision 1, October 2010.

28. Code Case N-613-2, "Ultrasonic Examination of Full Penetration Nozzles in Vessels, Examination Category B-D, Reactor Nozzle-to-Vessel Welds and Nozzle Inside Radius Section, Figs. IWB-2500-7(a), (b), (c), and (d)," *ASME Code Section XI, Division 1*, Approval Date: December 20, 2010.

29. Code Case N-552-1, "Alternative Methods – Qualification of Nozzle Inside Radius Section from the Outside Surface," *ASME Code Section XI, Division 1*, Approval Date: June 2012.

30. *Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements.* EPRI, Palo Alto, CA: 2016. 3002007626.

31. Letter from M. T. Markley (NRC) to C. R. Pierce (Southern Nuclear Operating Co. Inc.), Vogtle Electric Generating Plant, Units 1 And 2, and Joseph M. Farley Nuclear Plant, Unit 1 - Alternative to In-Service Inspection Regarding Reactor Pressure Vessel Threads in Flange Inspection (CAC Nos. MF8061, MF8062, MF8070), January 26, 2017, ADAMS Accession Number ML17006A109.

32. Letter from J. G. Danna (NRC) to D. G. Stoddard (Dominion Nuclear Connecticut Inc.), Millstone Power Station, Unit Nos. 2 and 3 - Alternative Requests RR-04-24 And IR-3-30 for Elimination of the Reactor Pressure Vessel Threads in Flange Examination (CAC Nos. MF8468 and MF8469)," May 25, 2017, ADAMS Accession Number ML17132A187.

33. Letter from D. J. Wrona (NRC) to B. C. Hanson (Exelon Generating Company, LLC), Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; R. E. Ginna Nuclear Power Plant; and Three Mile Island Nuclear Station, Unit 1 – Proposed Alternative to Eliminate Examination of Threads in Reactor Pressure Vessel Flange (CAC Nos. MF8712-MF8729 and MF9548)," June 6, 2017, ADAMS Accession Number ML17170A013.

34. Letter from U. Shoop (NRC) to S. Capps (Duke Energy), Brunswick Steam Electric Plant, Unit No. 1; Catawba Nuclear Station, Unit No. 2; Shearon Harris Nuclear Power Plant, Unit 1; McGuire Nuclear Station, Unit Nos. 1 and 2; Oconee Nuclear Station, Unit Nos. 1, 2, and 3; and H. B. Robinson Steam Electric Plant, Unit No. 2 - Alternative to In-Service Inspection Regarding Reactor Pressure Vessel Threads in Flange Inspection (CAC Nos. MF9513-MF9521; EPID L-2017-LLR-0019)," December 26, 2017, ADAMS Accession Number ML17331A086.

35. *Evaluation of Basis for Periodic Visual Examination of Accessible Areas of Reactor Vessel Interior per Examination Category B-N-1 of ASME Section XI, Division 1.* EPRI, Palo Alto, CA: 2018. 3002012966.

36. *Identification and Assessment of Low-Value Nondestructive Evaluation Examinations with High Outage Impacts.* EPRI, Palo Alto, CA: 2018. 3002012965.

37. NUREG-1841, *U.S. Operating Experience with Thermally Treated Alloy 690 Steam Generator Tubes,* U.S. NRC, 2007.

38. Cofie, N. G. et al., "Management of Steam Generator Feedwater Nozzle Cracking in PWRs," *ASME PVP-Vol. 286*, pp. 31–43, 1994.

39. ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition.

40. South Texas Project Amended Request for Relief from ASME Section XI Code Requirements for Steam Generator Nozzle Nondestructive Examination (RR-ENG-3-01) (TAC ME4766/ME4768)" December 2, 2010, ADAMS Accession Number ML103470281.

41. U.S. NRC Regulatory Guide 1.193, Revision 5, *ASME Code Cases Not Approved for Use,* August 2017.

42. Indian Point Unit No. 2 Section 10-Year Interval In-Service Inspection Program Relief Request No. 25, October 16, 1987, ADAMS Accession Number ML100330890.

43. Indian Point Unit No. 3 Attachment II to IPN-89-062, Pre-Service Examination Report for Replacement Steam Generators, ADAMS Accession Number ML100280148.

44. Point Beach Nuclear Plant Unit 2 10 CFR 50.55a Request, Relief Request RR-10 Examination of the Unit 2 Steam Generator Feedwater Nozzle Extension to Nozzle Weld Fifth 10-Year Inservice Inspection Program Interval, October 13, 2014, ADAMS Accession Number ML14286A094.

45. Palo Verde Nuclear Generating Station Unit 2 Relief Request 58 – Unit 2 Impractical Examinations for the Third 10-Year In-Service Inspection Interval, March 17, 2018, ADAMS Accession Number ML18079A725.

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

47. Farley Nuclear Plant FSAR Section 5.0, "Reactor Coolant System and Connected System," ADMAS Accession Number ML17117A369.

48. *Materials Reliability Program: Characterization of U.S. Pressurized Water Reactor (PWR) Fleet Operational Transients (MRP-393).* EPRI, Palo Alto, CA: 2014. 3002003085.

49. *Component Life Estimation: LWR Structural Materials Degradation Mechanisms.* EPRI, Palo Alto, CA: 1987. NP-5461.

50. Not used.

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

52. Letter from B. W. Sheron (U.S. NRC) to D. C. Mims, Acceptance for Referencing of Report SIR-94-080 - Relaxation of Reactor Coolant Pump Flywheel Inspection Requirements, May 21, 1997.
53. WCAP-15666, *Extension of Reactor Coolant Pump Motor Flywheel Examination,* July 2001.
54. T. D. Gilman, A. Alleshwaram, and T. Satyan-Sharma, "Industry First NRC Approved Appendix L Flaw Tolerance Evaluation to Manage Fatigue in a Surge Line," PVP2015-45194, *Proceedings of the ASME 2015 Pressure Vessels and Piping Conference*, July 19–23, Boston, MA, USA.
55. *Materials Reliability Program: Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375).* EPRI, Palo Alto, CA: 2014. 3002002441.
56. A. Der Kiureghian and O. Ditlevsen, "Aleatory or Epistemic? Does it Matter?" *Structural Safety,* Vol. 31, pp. 105–112, 2009.
57. Letter from B. W. Sheron (U.S. NRC) to C. Terry (BWRVIP), Transmittal of NRC Staff's Independent Assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for Additional Information (TAC NO. M93925), August 14, 1997, ADAMS Legacy Library Accession Number 9709090456.
58. Marshall, W., *An Assessment of the Integrity of PWR Pressure Vessels,* Study Group Report, Service Branch, UKAEA, London, June 1982.
59. G. J. Schuster et al. *Characterization of Flaws in U.S. Reactor Pressure Vessels,* NUREG/CR-6471, Volumes 1, 2, and 3.
60. F. A. Simonen, S. R. Doctor, G. J. Shuster, and P. G. Heasler, *A Generalized Procedure for Generating Flaw-Related Inputs to the FAVOR Code*, U.S. Nuclear Regulatory Commission Report, NUREG/CR-6817, August 2013.
61. Structural Integrity Associates Calculation Package 1700313.301, *Fabrication Flaw Size Distribution, Revision 0.*
62. G. L. Stevens, S. R. Gosselin, and J. M. Davis, "Technical Basis Document for Section XI Nonmandatory Appendix L Revisions," *Appendix L Technical Basis Document*, 9/21/2006.
63. 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.
64. API Standard 579-1/ASME FFS-1, *Fitness-for-Service, Second Edition*, June 2016.
65. Structural Integrity Associates Calculation Package, *Stress Intensity Factors (SIFs) for Nozzle Corner Cracks,* DEV1801.302, 3/12/2018.
66. SI-TIFFANY 3.1, Structural Integrity Associates, September 2018.
67. xLPR Version 2.0: Technical Basis Document PWSCC and Fatigue Crack Growth and Coalescence Module Development, June 26, 2016.
68. U.S. NRC Branch Technical Position MTEB 5-2, *Fracture Toughness Requirements,* Rev. 1, July 1981.

69. 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.

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

71. M. T. Erickson and T. L. Dickson, *Recommended Screening Limits for Pressurized Thermal Shock*, NUREG/CR-1874, March 2010.

72. Letter from B. E. Thomas (U.S. NRC) to J. W. Lubinski (U.S. NRC), Acceptance Criteria to Use with xLPR Code, Version 2.0, November 7, 2016, ADAMS Accession Number ML16271A436.

73. Structural Integrity Associates Report DEV1806.401, *Software Project Plan, Software Requirements, Functional Specification, and Software Verification and Validation Plan for PROMISE Version 1.0*, Revision 0.

74. Structural Integrity Associates Report DEV1806.403, *PROMISE 1.0 Verification and Validation Report*, Revision 0.

75. Structural Integrity Associates Report DEV1806.402, *PROMISE 1.0 Theory and User's Manual*, Revision 0.

76. SI Calculation 1600481.302, *Verification of Software VIPERNOZ Version 1.1,* Revision 1.

77. *Sensitivity Analysis – Comparison Studies*, DEI, EMC2, SIA, Version 5, September 2018.

78. W. H. Bamford, "Application of Corrosion Fatigue Crack Growth Rate Data to Integrity Analysis of Nuclear Reactor Vessels," *Journal of Engineering Material and Technology*, Volume 101, 1979.

79. Letter from F. A. Kearney (Exelon) to U.S. NRC, Byron Station Unit 2 90-Day In-Service Inspection Report for Interval 3, Period 3, (B2R17), July 29, 2013, Docket No. 50-455, ADAMS Accession Number ML13217A093.

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