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Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements

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Technical Update, March 2016

EPRI Project Manager

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All or a portion of the requirements of the EPRI Nuclear Quality Assurance Program apply to this product.

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PRODUCT DESCRIPTION

Anecdotal evidence suggests that examinations required by the ASME Boiler Code for the reactor pressure vessel (RPV) flange have not found service-induced degradation and that the performance of these examinations has negative impacts on worker exposure, personnel safety, and outage critical path time.

Background

Section XI of the ASME Boiler and Pressure Vessel Code provides in-service inspection requirements for nuclear power plant components. It has been shown that some of these examination requirements place an undue burden on plant operators that is not commensurate with the safety benefit provided by the examination. Plants can request relief from these types of examination if an adequate basis can be provided.

Objectives

The objective of this report is to confirm that the anecdotal evidence is correct in suggesting that RPV flange examination requirements are not finding service-induced degradation and that the examinations have negative impacts on worker exposure, personnel safety, and outage critical path time. In addition, the report investigates alternatives to the current requirements and, if appropriate, provides the technical basis for the elimination of the RPV Threads in Flange examination requirement (ASME Section XI Examination Category B-G-1, item number B6.40).

Approach

The approach taken consisted of conducting an industry survey to collect data on the results of the RPV flange examinations as well as to gather insight into the negative aspects of having to conduct these examinations (for example, worker exposure and personnel safety). A literature search was also conducted to identify and assess any related RPV flange operating experience.

Further, a review of several plant-specific and generic industry studies that have been used to assess the structural integrity of the RPV was conducted. These studies have been tailored toward evaluating operating events (for example, inoperable stud), regulatory interactions (for example, the anticipated transient without scram [ATWS] rule), and beyond design basis events (for example, severe accident guidelines).

A flaw tolerance evaluation was also conducted to investigate the robustness of the RPV flange design and, finally, a bounding generic risk impact assessment was conducted. The purpose of the risk impact assessment was to identify the impact of changes to the RPV flange examination requirement on plant risk.

Results

This report provides the basis and recommendation for the elimination of the RPV Threads in Flange examination requirement (ASME Section XI Examination Category B-G-1, item number B6.40). The basis for this recommendation includes an exhaustive survey of inspection practices, a literature review, a flaw tolerance evaluation, and a bounding assessment on risk if these examinations are eliminated. The conclusion from this evaluation is that the risk captured by the current requirement is extremely low and not commensurate with the associated burden (worker exposure, personnel safety, and critical path time). Even with the conservative assumptions included in the analysis, the impact on risk from elimination of these examination requirements is quite low.

Applications, Value, and Use

In the short term, plant operators can use the results of this evaluation to support regulatory interaction requesting plant-specific elimination of these examination requirements. Longer term, it is expected that an ASME Code action will move forward to revise this examination requirement.

Keywords

Examination Category B-G-1
Ligament
RPV flange

EXECUTIVE SUMMARY

The objective of this report is to provide the basis for elimination of the reactor pressure vessel (RPV) Threads in Flange examination requirement (ASME Code Section XI Examination Category B-G-1, item number B6.40). Industry experience indicates that these examinations have not been identifying any service-induced degradation. In addition, these examinations have negative impacts on worker exposure, personnel safety, and outage critical path time. To accomplish this objective, several tasks were performed, including the following:

- Conducting a survey of 168 nuclear plant units to evaluate past inspection results of these components
- Evaluating potential degradation mechanisms
- Performing a flaw tolerance evaluation assuming the presence of an initial ASME Section XI IWB-3500 acceptance flaw
- Considering operating events such as an inoperable stud
- Considering regulatory interactions such as the anticipated transient without scram (ATWS) rule
- Considering beyond design basis events such as severe accidents
- Considering a bounding generic risk impact assessment

Based on the above considerations, it is concluded that the Threads in Flange examinations as mandated in ASME Code Section XI can be eliminated without increasing plant risk or posing any safety concerns for the RPV. Savings gained from the elimination of this inspection can be applied toward more meaningful examinations of other risk-significant plant components.

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1 OBJECTIVE AND SCOPE

Over the past several years, the in-service inspection (ISI) requirements for many components in nuclear reactor pressure vessels (RPVs) have become increasingly rigorous. This significant increase in inspection criteria and associated cost has raised a number of issues concerning the necessity for inspection of some RPV components. One such component is the Threads in Flange (ASME Code Section XI Examination Category B-G-1, item number B6.40 [1]), where it is generally perceived that the inspection does not contribute to the overall safety of the RPV and is therefore not necessary. It is believed that savings from elimination of this inspection can be applied to other more meaningful inspections of other risk-significant plant components.

The objective of this report is to provide the basis for the elimination of the RPV Threads in Flange examination requirement. Industry experience indicates that these examinations have not been identifying service-induced degradation and that they have negative impacts on worker exposure, personnel safety, and outage critical path time.

This report includes the results of an industry survey in which 168 units provided the results of their RPV Threads in Flange examination as well as insight into the impacts of conducting these examinations. A literature search was also conducted to assess any related operating experience (OE) impacting the position that the RPV Threads in Flange examination requirement is providing insufficient value to outweigh the negative impacts associated with performing the examination. In addition, a degradation mechanism evaluation was performed to identify possible mechanisms that could potentially degrade the Threads in Flange component while in service. Potential active degradation mechanisms are then considered in a flaw tolerance evaluation to determine how long it would take a postulated flaw to challenge the integrity of the RPV.

Further, a review was conducted of a number of plant-specific and generic industry studies used to assess the structural integrity of the RPV. These studies were tailored toward evaluating operating events (for example, an inoperable stud), regulatory interactions (for example, the anticipated transient without scram [ATWS] rule), and beyond design basis events (for example, severe accident guidelines). The conclusion drawn from these studies is that the RPV, including the flange, studs, and other connected components (for example, nozzles), has large margins.

Finally, a bounding generic risk impact assessment was conducted. According to this assessment, the risk captured by the current RPV Threads in Flange examination requirement is extremely low and is not considered commensurate with the negative impacts. Even with the conservative assumptions included in the analysis, the impact on risk from the elimination of these examination requirements is low.

2
COMPONENT DESCRIPTION AND EXAMINATION REQUIREMENTS

2.1 Component Description

A typical reactor pressure vessel consists of a cylindrical shell mounted in the vertical direction with a hemispherical bottom head permanently welded to the cylindrical shell. It also has a removable closure head that allows entry into the vessel for maintenance of internal components of the vessel.

The closure head is attached to the vessel by several studs that are threaded into the vessel flange and extend through the closure head flange. The threaded portion of the flange is termed *Threads in Flange*. The number of studs varies among configurations but is generally on the order of 50 to 76. Similarly, the diameter of the studs varies among configurations but is generally on the order of 6 to 7 in. (152 to 178 mm). Table 2-1 presents values for key geometric parameters of the RPV, including the number and diameter of the studs, from seven U.S. pressurized water reactor (PWR) plants (a total of 10 units). Table 2-2 provides the values of these parameters for six boiling water reactor (BWR) units. These values are considered typical.

Table 2-1
Summary of Threads in Flange components at seven U.S. PWR plants (10 units)

Plant	Unit 1	Unit 2	Unit 3	Units 4 and 5	Units 6 and 7	Unit 8	Units 9 and 10
Number of studs	60	54	58	60	54	54	54
Stud diameter (inches)	6.5	6.5	6	6.5	7	7	7
RPV inside diameter (inches)	167	157	155	167	173	173	173
Flange thickness at stud hole (inches)	16	16	15	16	16	16	16
Design pressure (psig)	2500	2500	2500	2500	2500	2500	2500

1 in. = 25.4 mm
1 psig = 6.89 kPa

Table 2-2
Summary of Threads in Flange components at six BWR units

Plant	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6
Number of studs	76	72	64	72	65	56
Stud diameter (inches)	6	6	6	6	6	6
RPV inside diameter (inches)	250	237	217	250	237	217
Flange thickness at stud hole (inches)	13.7	13.25	12.9	14.4	14.5	14.3
Design pressure (psig)	1250	1250	1250	1250	1250	1250

1 in. = 25.4 mm
1 psig = 6.89 kPa

The closure head is held in place by extension nuts that are screwed onto the studs. The closure head is secured to the vessel flange by elongating the stud bolts according to a prescribed tensioning procedure in order to produce the right amount of preload in the studs. A typical Threads in Flange component is shown in Figure 2-1. The flange material is typically low-alloy SA-508 Cl.2 (3/4Ni-1/2Mo-1/3cr-V), and the stud material is typically low-alloy SA-540 (2Ni-3/4Cr-1/4Mo).

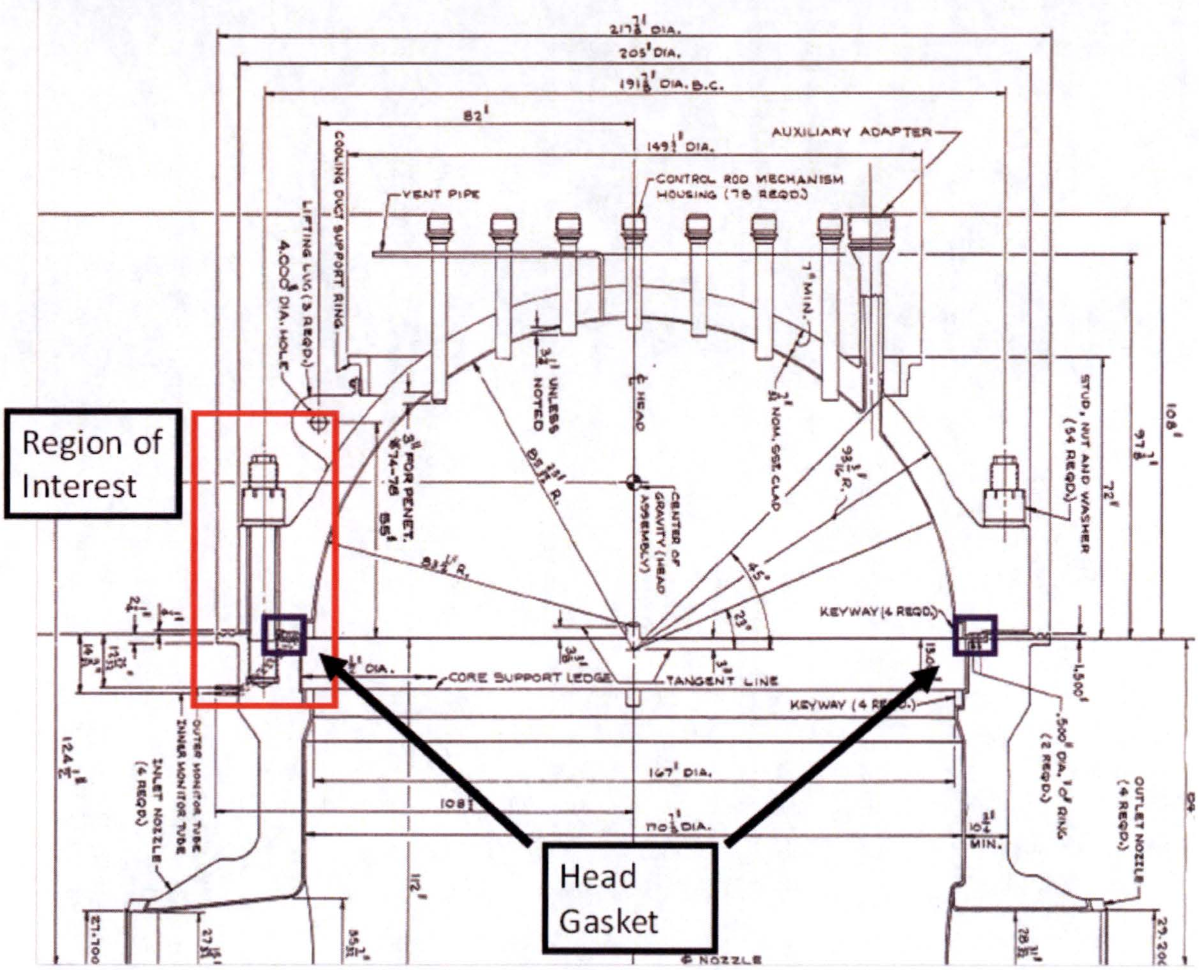
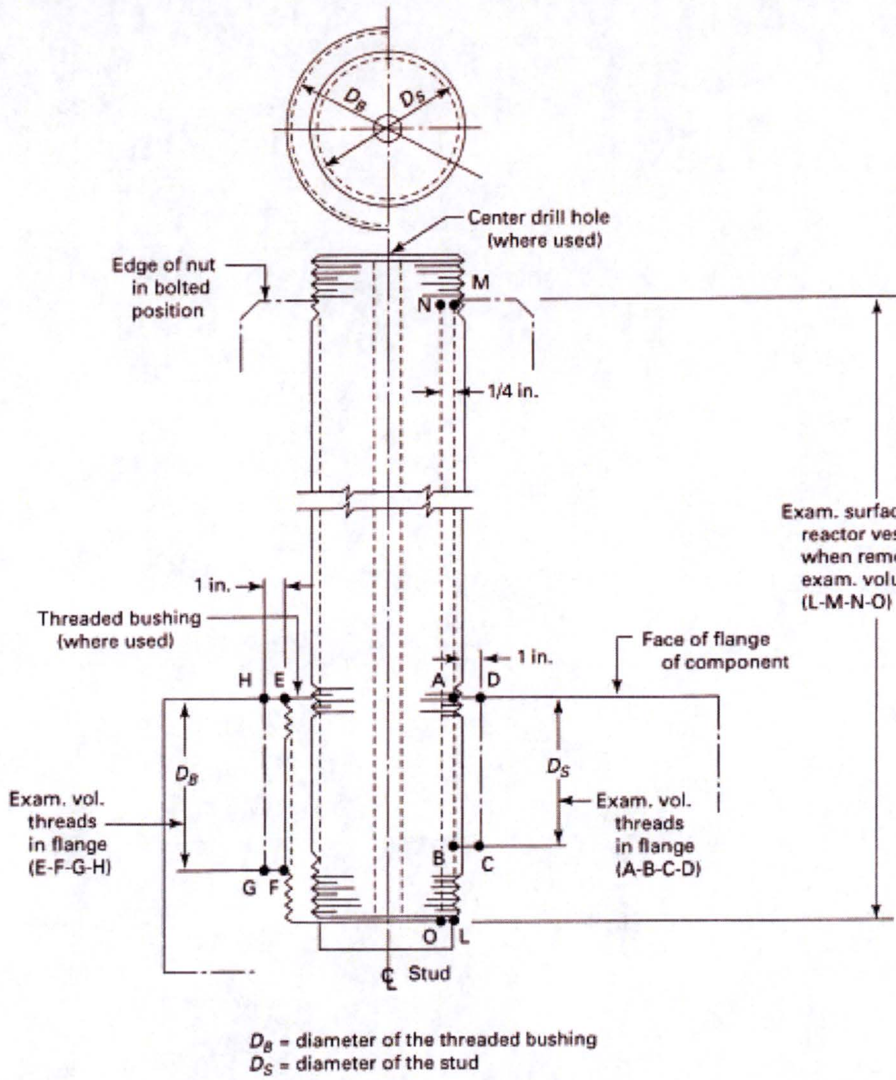


Figure 2-1
Typical Threads in Flange component

Two self-energizing O-ring gaskets are provided on the seating surface of the flange to prevent leakage of the primary coolant, as shown in Figure 2-1. Two gasket grooves are machined on the seating surface of the closure head flange for the placement of the O-ring gaskets, which are composed of polished and silver-plated Ni-Cr-Fe alloy. Two connections for a monitoring device are provided in the vessel flange to detect any leakage past the O-rings. Therefore, the Threads in Flange configuration maintains an air environment and does not contact the reactor coolant fluid.

2.2 Examination Requirements

As part of the ISI requirements, ASME Code Section XI requires that the Threads in Flange component (Category B-G-1, item number B6.40) be inspected during each interval using volumetric examination. The required examination volume is shown in Figure 2-2. The acceptance criteria are such that the size of allowable flaws within the boundary of the examination volume, oriented in a plane normal to the studs, shall not exceed 0.2 in. (5 mm) as measured radially from the root of the thread.



GENERAL NOTES:
(a) 1 in. = 25 mm
(b) ¼ in. = 6 mm

Figure 2-2
Closure stud and Threads in Flange stud hole examination volume (same as ASME Code Section XI, Figure IWB-2500-12)

3 EXAMINATION DATA AND RESULTS

In support of this project, a survey was conducted to confirm the anecdotal evidence that the RPV Threads in Flange examination was not identifying any service-induced degradation while adversely impacting outage activities. This survey was conducted in 2015 and early 2016, and replies were received from 168 units. These units consisted of almost the entire U.S. fleet as well as a number of units operated outside the United States. The data gathered cover all of the plant designs operated within the United States (for example, earlier and later vintage BWRs and PWRs as well as all nuclear steam supply system (NSSS) designs (that is, Babcock & Wilcox, Combustion Engineering, General Electric, and Westinghouse).

The content of the survey and requested data is as follows:

- Plant name
- Unit number
- NSSS vendor
- Type (for example, BWR-4)
- ASME Section XI item number B6.40 inspections (Yes/No)
- Other Code volumetric inspections (for example, Canadian Standards Association [CSA]) of RPV Threads in Flange
- Other owner-defined volumetric inspections of RPV Threads in Flange
- Other inspections (for example, visual and surface) of RPV Threads in Flange
- Total number of RPV Threads in Flange locations
- Total number of examinations
- Number of examinations with no reportable indications
- Number of examinations with reportable indications
- Comment on the impact of these examinations on the site (for example, critical path time)
- Contact information

The results of the survey are provided in the following sections.

3.1 U.S. Fleet

In short, the results of the survey confirmed the anecdotal evidence that the RPV Threads in Flange examination is adversely impacting outage activities (including dose, safety, and critical path time) while not identifying any service-induced degradations. For the U.S. fleet, a total of 94 units have responded to date; not one of these units has identified any type of degradation. As discussed above and as can be seen in Table 3-1, the data are impressive and encompassing. The 94 units represent data from 33 BWRs and 61 PWRs. For the BWR units, a total of 3,793 examinations were conducted; for the PWR units, a total of 6,869 examinations were conducted—again, with no service-induced degradation identified. The response data include information from all of the plant designs in operation in the United States. That is, for the BWR plants, BWR-2, -3, -4, -5, and -6 designs are represented. For the PWR plants, two-loop, three-loop, and four-loop designs are represented, and each of the PWR NSSS designs is also represented (that is, Babcock & Wilcox, Combustion Engineering, and Westinghouse).

Table 3-1
Summary of survey results: U.S. fleet

Plant Type	Number of Units	Number of Examinations	Number of Reportable Indications
BWR	33	3,793	0
PWR	61	6,869	0
Total	94	10,662	0

Appendix A of this report provides additional information on the data gathered through the survey. Some survey information discussed above is not provided in Appendix A because this information is plant unique.

3.2 International Fleet

A number of international plants follow ASME Section XI ISI requirements or follow rules that are very similar to the intent and practice of ASME Section XI. As such, the tasks of collecting and reviewing these data provide additional insight and knowledge into the benefit, or lack of benefit, in conducting the RPV Threads in Flange examinations.

The data provided by the international fleet represented 74 units from Brazil, Canada, France, Japan, Slovenia, and Switzerland. These units comprise eight units of the CANDU design as well as a large number of light water reactors (LWRs), which are of similar design and operation to the U.S. fleet.

Although very consistent with the data provided by the U.S. fleet, there were a few units, in two countries, that did provide some data with regard to reportable indications (see Table 3-2). For the first country, one unit identified that some indications had been found. After further evaluation, the indications were deemed not safety relevant; it was also identified that the terminology in that country with respect to *reportable indication* is somewhat different from that typically used within ASME Section XI. As such, these indications would not be considered "reportable" according to ASME Section XI criteria. In addition, subsequent inspection in a more recent outage revealed no indications at these locations.

For the second country, some indications have been identified at several units. Work is currently underway with the utility to better understand the status and relevance of this information.

Table 3-2
Summary of survey results: International fleet

Plant Type	Number of Units Reporting Data
CANDU	8
BWR	1
PWR	65
Total	74

4
OPERATING EXPERIENCE

A literature search was conducted to assess any related operating experience (OE) that could impact the position that the RPV Threads in Flange examination requirement is providing low value-added safety benefit and is not commensurate with the associated worker exposure, outage complications, and personnel safety concerns.

Table 4-1 provides a summary of this literature review. As can be seen in the table, there has been some OE related to the RPV flange, studs, and related equipment (RPV stud tensioner). However, the RPV Threads in Flange examination requirement would not have prevented any of these occurrences. In addition, there is very little to no relevance of this OE to support the purpose of performing the RPV Threads in Flange examination (that is, flange ligament degradation).

As such, although the lessons learned from the OE identified in this literature search are being implemented by the industry through the normal OE process, this literature search and review further confirm that the RPV Threads in Flange examination requirement is providing low value-added safety benefit and is not commensurate with the associated worker exposure, outage complications, and personnel safety concerns.

Table 4-1
Operating experience review

Plant Type	Date Timeframe	Component	Discussion
PWR	2015	Stuck stud	Abrasion and dust accumulation.
BWR	2009	RPV stud calibration block	
BWR	2005	RPV stud surface examination requirements	Compliance issue between Section III and Section XI requirements.
BWR	2013	RPV stud tension	Tensioner failure.
PWR	2012	RPV stud tensioning out of sequence	RPV studs tensioned out of sequence.
BWR	2011	RPV stud tensioning	Improper stud tensioning resulted in leakage after a mid-cycle outage.
BWR	2009	RPV stud tensioner	Improperly installed tensioner rigging resulting in lost outage time.

Table 4-1 (continued)
Operating experience review

Plant Type	Date Timeframe	Component	Discussion
BWR	2008	Preexisting damage to RPV flange	Steam cutting and mechanical damage on the RPV flange as well as steam cutting and damage on the RPV head flange. The steam cuts most likely occurred as a result of the unrepaired damage to the RPV flange that existed at the time of the Unit 1 restart. The apparent cause of the flange leakage was the existing damage on the RPV flange at the time of the Unit 1 restart. There was mechanical damage to areas on the O-ring groove surface, likely caused by tooling used to remove the O-ring retainer clips.
BWR	2008	Re-performance of reactor vessel head tensioning required	Use of an inappropriate tool resulted in having to completely re-perform all head tensioning with the loss of 11 hours of critical outage time.
BWR	2005	Water on flange on O-ring surfaces prior to RPV head installation	The RPV head and flange areas were inspected, new O-rings were installed into the RPV head, and the RPV head was re-installed at 1513 on April 23, with tensioning completed on April 24 at 2355.
BWR	2004	Indications of reactor inner head seal leak during ASME system leakage test	Upon inspection of the RPV flanges and O-rings, debris was found in several locations, which was determined to be the cause of the leakage.
BWR	2012	RPV head binding occurred during head lift	The RPV head experienced binding, causing the overhead crane to trip on overload. The cause of the event was a lack of reactor head lift experienced personnel assisting the technical director to ensure that the reactor head remained level so that no binding would occur. A subsequent inspection revealed no damage to the reactor flange, the reactor studs, or the stud guides.
BWR	2002	Reactor vessel head became stuck on alignment sleeve during vessel head removal	During the removal of the RPV head, it became stuck on the guide bolts. The consequence of this condition was a 13.5-hour delay to the 1R19 outage vessel disassembly schedule. Equipment damage was limited to thread damage to the upper threads of the #1 vessel head stud, damage to the alignment sleeves, and minor scraping damage to the #1 stud hole in the reactor vessel head. This damage was evaluated and repaired prior to vessel reassembly.

Table 4-1 (continued)
Operating experience review

Plant Type	Date Timeframe	Component	Discussion
BWR	2002	Reactor head became stuck on guide pin during removal for reactor disassembly	In preparation for RPV head removal, guide pins were installed at Studs 1 and 39 as specified in the procedure. The lift was stopped at approximately 10 in. to check for vessel water level and dose. When the lift was restarted, the head bound on a guide pin, and the lift was stopped. An attempt was made to lower the head to correct the out-of-level condition, but the head would not move. All work was stopped. A guide pin should have been installed at Stud 34, not Stud 39 as indicated in the procedure.
BWR	1991	Ultrasonic examination indications in RPV head studs	Evaluation suggests that these UT indications are surface discontinuities caused by localized corrosion (pitting) in the threads.
PWR	2015	Imprint in the top of the reactor head flange	An Allen wrench that was being used to troubleshoot a reactor head tensioner was left on the reactor head flange after the tensioner troubleshooting was completed. The tensioner was then set down on top of the Allen wrench, and the stud de-tensioned. When the tensioner was removed, the workers identified an imprint of the Allen wrench left on the reactor flange. The crew verified that the remainder of all flange locations were free of foreign materials. De-tensioning was completed without further issues. The imprint was "blended" outside the flange region area per engineering direction in accordance with operations evaluation.
PWR	2014	Reactor vessel head stud elongation measurement system malfunction	During tensioning of the reactor head studs, the SEMS unit operated properly during the initial no-load and after the first correction pass data collection points. When the second correction pass was completed and stud elongation measurements were ready to be taken, the SEMS unit began displaying negative numbers. The vendor recommended that the utility perform manual calculations to determine proper stretch instead of relying on the SEMS unit. A spreadsheet was also used to perform a peer check of the manual calculations. Multiple stud passes were made using this method, and the head was successfully tensioned.

Table 4-1 (continued)
Operating experience review

Plant Type	Date Timeframe	Component	Discussion
BWR	2013	Reactor head washer installed incorrectly	During reactor head tensioning, the nut at Stud F-6 would not seat properly. The washers are spherical on one side and flat on the other. The flat side is to be installed down so that they are flat against the reactor head flange. The washer at Stud F-6 was discovered to be incorrectly stacked. All of the spherical washers were then checked to determine the extent of the condition. Eight of the washers were found upside down. Six of the eight affected studs had already been tensioned when the error was discovered. An inspection of the reactor head flange, nuts, and washers was performed with no damage found.
PWR	2012	Reactor vessel stud damaged during tensioning	While tensioning the reactor vessel head, a stud tensioner failed to fully engage the stud. This resulted in metal from the tensioner being deposited between the threads of the stud. The stud tensioner had to be repaired, and the stud was removed and machined to remove the deposited metal. There was no damage to the threads on the stud because the gripper is made of a softer material.
PWR	2012	Foreign material discovered on vessel flange during underwater visual examination of the reactor vessel internals (bristles on stud hole cleaner)	The stud hole cleaning machine design was altered, resulting in a differential pressure in the enclosure pipe that caused loose wire strands to be transported into the gap between the reactor head and vessel flanges.
BWR	2012	Misaligned reactor head stud washers prevented seating the stud tensioner onto the vessel flange	The problem was resolved by de-tensioning the stud, re-aligning the washer, and re-tensioning the stud.
PWR	2011	Reactor vessel stud tensioner hoist fell off the seismic hoist rail during installation	During replacement of one of the three reactor vessel stud tensioner hoists, the hoist slipped off the seismic hoist rail and fell approximately 23 feet to the floor of the upper cavity. The hoist had extensive damage, but the reactor studs and cavity shield ring had only superficial damage.

Table 4-1 (continued)
Operating experience review

Plant Type	Date Timeframe	Component	Discussion
PWR	2010	Heavy concentration of boric acid on the RPV O-rings	The boric acid identified on the reactor head O-ring groove was the result of minor O-ring leakage. Detailed inspections of the O-ring seating surface on the reactor head did not identify any indications of an active leakage path. The identified leakage was not pressure boundary leakage; rather, it was leakage past an O-ring seal. The O-ring leakage was very minor in terms of actual leakage rate, based on the observed condition of the O-ring joint. This leakage was well below tech spec limits for unidentified leakage. Results obtained using a calibrated depth gauge indicated a horizontal indication of 0.003 in. across the inner O-ring seating surface. The reactor vessel flange flaw of 0.003 in. was removed by the divers. The reactor vessel head porosity measurements and locations are such that they meet all acceptance criteria guidelines and are fully capable of performing their design function. No further maintenance is required.
BWR	2010	Reactor head not fully tensioned prior to mode change	During reactor head reassembly, supplemental refuel floor supervision used an unapproved spreadsheet to evaluate the stud elongation readings in lieu of the required procedure attachment. Consequently, studs were under-tensioned. During QPA review of the stud elongation data, it was discovered that 20 studs were found to be out of tolerance. Because vessel reassembly had continued, it was necessary to remove the insulation package, reset the tensioning rig, and re-tension the 20 studs.

1 in. = 25.4 mm

5
EVALUATION OF POTENTIAL DEGRADATION MECHANISMS

Potential degradation mechanisms affecting nuclear power plant components (including RPVs) are discussed in References [2] and [3].

Following is a list of mechanisms applicable to the Threads in Flange component.

Corrosion-related mechanisms

- Macro-corrosion mechanisms
  - General corrosion
  - Galvanic corrosion
  - De-alloying corrosion
  - Velocity phenomena
    - Erosion-corrosion
    - Cavitation
    - Flow-accelerated corrosion (FAC)
    - Fretting
    - Impingement
  - Crevice corrosion
- Micro-corrosion mechanisms
  - Pitting
  - Intergranular attack (IGA)
  - Corrosion fatigue
  - Stress corrosion cracking
    - Intergranular stress corrosion cracking (IGSCC)
    - Transgranular stress corrosion cracking (TGSCC)
    - Interdendritic stress corrosion cracking (IDSCC)
  - Microbiologically influenced corrosion (MIC)

Mechanical failure mechanisms

- Fatigue
  - Thermal fatigue
  - Mechanical fatigue
    - Low cycle
    - High cycle
    - Corrosion fatigue

- Mechanical wear
- Creep
- Stress relaxation

5.1 Corrosion-Related Mechanisms

5.1.1 General Corrosion

General corrosion is characterized by an electrochemical reaction that occurs relatively uniformly over the entire surface area that is exposed to a corrosive environment. For carbon and alloy steels, normal reactor water can serve as that corrosive environment. In contrast, austenitic stainless steels are not susceptible to general corrosion in the reactor environment. As discussed in Section 2.1, the Threads in Flange component is not exposed to the reactor water environment; therefore, general corrosion is not a plausible failure mechanism.

5.1.2 Galvanic Corrosion

Galvanic corrosion results when two electrochemically dissimilar materials are in contact with one another in the presence of an electrolyte. The Threads in Flange component has two very similar metals (low-alloy SA-508 Class 2 flange material and low-alloy SA-540 bolting material). In addition, the Threads in Flange component is not exposed to the reactor water environment, so there is no electrolyte to form a galvanic couple. Therefore, galvanic corrosion is not plausible as a failure mechanism.

5.1.3 De-Alloying Corrosion

De-alloying corrosion can occur when one element in an alloy corrodes preferentially as compared to the metal in general. However, for de-alloying corrosion to take place, an electrolytic environment must be present. The absence of reactor water (at the Threads in Flange component) to provide an electrolyte renders de-alloying corrosion an implausible failure mechanism.

5.1.4 Velocity Phenomena

Flow-accelerated corrosion is a form of velocity damage that can occur in low-alloy and carbon steel piping systems (austenitic stainless steel piping systems are not susceptible). Cavitation damage is a form of velocity damage that can occur in austenitic stainless steel piping systems. However, for velocity damage to occur, an electrolyte must be present so that the surface is wetted. The Threads in Flange component is not a piping component and is not exposed to any flow in the reactor water environment. As such, velocity phenomena are not plausible failure mechanisms.

5.1.5 Crevice Corrosion

Crevice corrosion is a form of localized corrosion characterized by a geometrical crevice where a solution is present. In the crevice, both anodic and cathodic reactions can take place, so that a stagnant, potentially aggressive solution is developed within this geometrically restricted area. For crevice corrosion to occur, a solution or electrolyte must be present. The absence of a reactor water environment (at the Threads in Flange component) to provide an electrolyte renders crevice corrosion an implausible failure mechanism.

5.1.6 Pitting

Pitting, like crevice corrosion, occurs in stagnant environments. It is a process that also requires an anode and a cathode to complete the reaction. Pitting is usually characterized by an extended initiation period followed by an autocatalytic propagation. The absence of a reactor water environment (at the Threads in Flange component) to provide an electrolyte renders pitting an implausible failure mechanism.

5.1.7 Intergranular Attack

Intergranular attack is the preferential corrosion of a material's grain boundaries in a corrosive environment due to the presence of segregated alloying elements and/or impurities or the depletion of certain alloying elements. Intergranular attack is often a precursor of IGSCC. The absence of a reactor water environment (at the Threads in Flange component) to provide an electrolyte renders intergranular attack an implausible failure mechanism.

5.1.8 Corrosion Fatigue

Corrosion fatigue is the reduction of the fatigue life of a component due to the synergistic combination of mechanical fatigue and corrosion in a corrosive environment. The reactor water environment has been determined to be sufficiently corrosive to promote corrosion fatigue. However, the absence of a reactor water environment (at the Threads in Flange component) to provide an electrolyte renders corrosion fatigue an implausible failure mechanism.

5.1.9 Stress Corrosion Cracking

Stress corrosion cracking (SCC) is the term given to subcritical crack growth of susceptible alloys under the influence of a tensile stress of sufficient magnitude in a corrosive environment. SCC can be either intergranular or transgranular or, in welds, either interdendritic or transdendritic. The absence of a reactor water environment (at the Threads in Flange component) to provide an electrolyte renders SCC an implausible failure mechanism. Furthermore, the SA-508 Cl. 2 low-alloy steel flange material associated with the RPV Threads in Flange component is not susceptible to SCC.

5.2 Mechanical Failure Mechanisms

5.2.1 Fatigue

Thermal fatigue is a potential failure mechanism in the Threads in Flange component due to the RPV heat-up/cooldown cycles. Mechanical fatigue due to reversible loads (such as fluctuating pressure transients and seismic loadings) is also a plausible mechanism. The effect of these two mechanisms on fatigue crack growth is considered in Section 6 and was found to be negligible. In particular, the crack growth resulting from 40 years of additional operation with an initial flaw equivalent to the ASME Section XI IWB-3500 acceptance standards flaw in the Threads in Flange component was found to be insignificant, indicating that fatigue is not an issue.

5.2.2 Mechanical Wear

Mechanical wear results from the relative motion between two surfaces (adhesive wear), from the influence of hard, abrasive particles (abrasive wear), or from fluid streams (erosion). Once in place, there is no relative motion between the bolt and the threads in the Threads in Flange component. In addition, the component is not exposed to any fluid or abrasive particles. Consequently, mechanical wear is not possible as a failure mechanism.

5.2.3 Creep

Creep is primarily an intergranular fracture phenomenon (occurring at relatively high temperatures) and is defined as the progressive deformation of a material at constant stress. Creep failure is also known as *stress rupture*. Operation below 700°F (371°C) in ferritic steel and below 800°F (427°C) in austenitic steel can alleviate concerns about creep [4]. Creep is not generally a major consideration in LWRs due to their operation at approximately 650°F (343°C) or less. Consequently, creep is considered a very unlikely failure mechanism for the Threads in Flange component.

5.2.4 Stress Relaxation

Stress relaxation is a phenomenon related to creep, in which the stress in a member decreases when a constant amount of deformation is applied to the component. Stress relaxation is typically observed in bolted joints and in shrink-fit or press-fit components. The Threads in Flange is a bolted joint. The tensioning procedures developed by the plants ensure that the preload is maintained and minimal stress relaxation occurs. Consequently, stress relaxation is considered a very unlikely failure mechanism for the Threads in Flange component. However, the applied preload was considered in the flaw tolerance evaluation described in Section 6.

5.3 Summary of Degradation Mechanism Evaluation

In summary, other than the potential for mechanical/thermal fatigue, there are no active degradation mechanisms identified for the Threads in Flange component. The effect of fatigue is evaluated in Section 6 and was found to have an insignificant effect on the component over the postulated 80 years (original 40-year design life plus additional 40 years of plant life extension). It is therefore concluded that there are no active degradation mechanisms that could potentially prevent the Threads in Flange component from performing its intended function.

6
STRESS ANALYSIS AND FLAW TOLERANCE EVALUATION

In the previous section, mechanical/thermal fatigue was identified as the only plausible degradation mechanism. In this section, a flaw tolerance evaluation is performed considering this degradation mechanism. The evaluation consists of two parts. In the first part, stress analysis is performed considering all applicable loads on the Threads in Flange component. In the second part, the stresses at the critical locations of the component are used in a fracture mechanics evaluation to determine the allowable flaw size for the component as well as how much time it will take for a postulated initial flaw to grow to the allowable flaw size using guidelines in the ASME Code, Section XI IWB-3500.

6.1 Stress Analysis

Due to the relatively complicated geometry of the component, finite element analysis (FEA) methods are used to determine the stresses due to all applicable loads.

6.1.1 Finite Element Model

To perform the flaw tolerance calculation, a bounding three-dimensional (3-D) finite element model is developed using the ANSYS FEA software package [5] to represent a typical Threads in Flange component in the domestic US fleet. As discussed in Section 2.1 and as shown in Tables 2-1 and 2-2, there are slight variations in configuration across the U.S. fleet. In general, the shape of the bolting flange is similar, but there are variances in bolt size, number of bolts, and RPV diameter.

To create a representative geometry for the finite element model, the PWR design was selected because of its higher design pressure and temperature. The largest RPV diameter of the PWRs was used along with the largest bolts and the highest number of bolts. The larger and more numerous bolt configuration results in less flange material between bolt holes, whereas the larger RPV diameter results in higher pressure and thermal stresses (see Figure 6-1). The dimensions of the finite element model based on this configuration are provided in Figure 6-2.

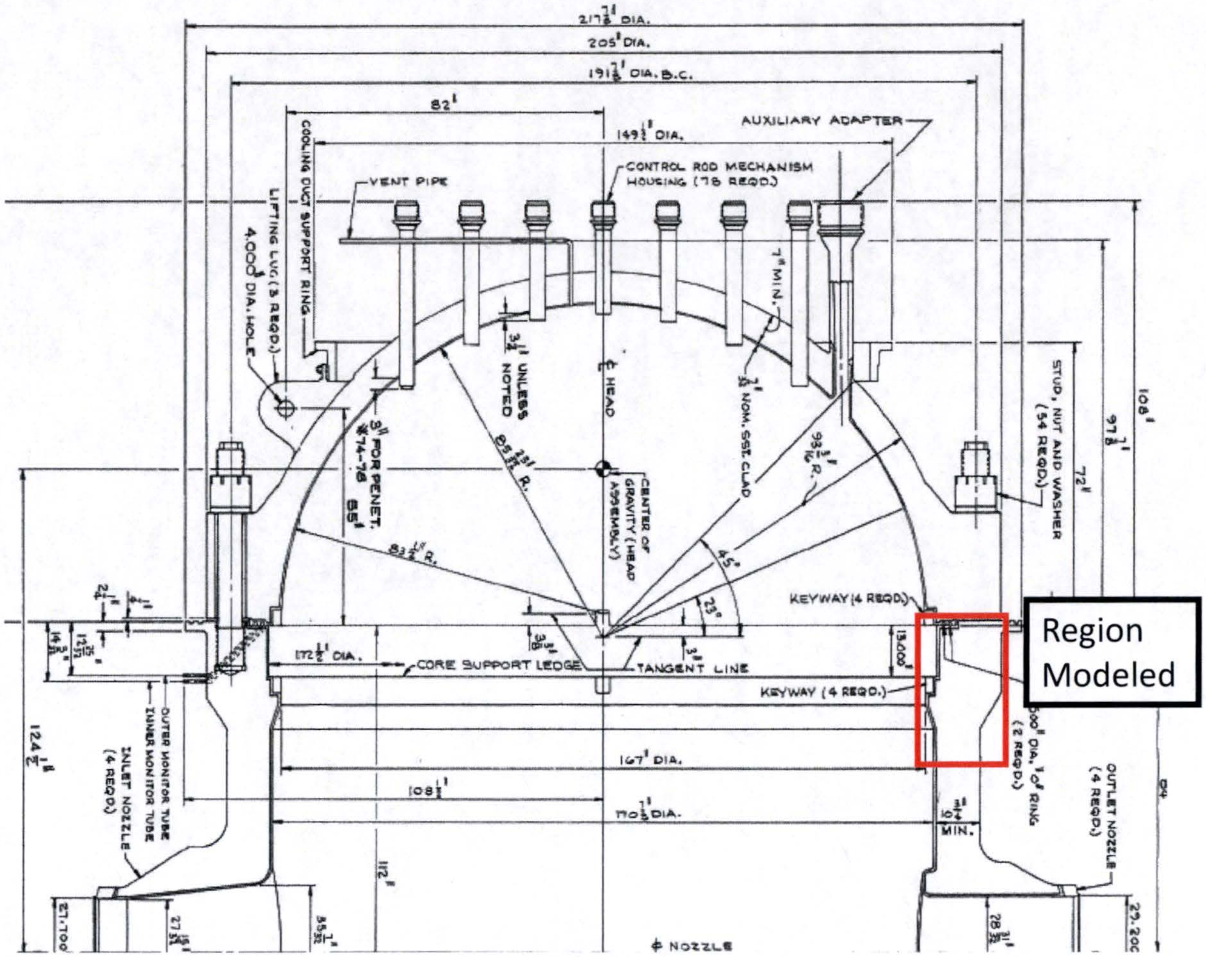


Figure 6-1
Region of interest on RPV

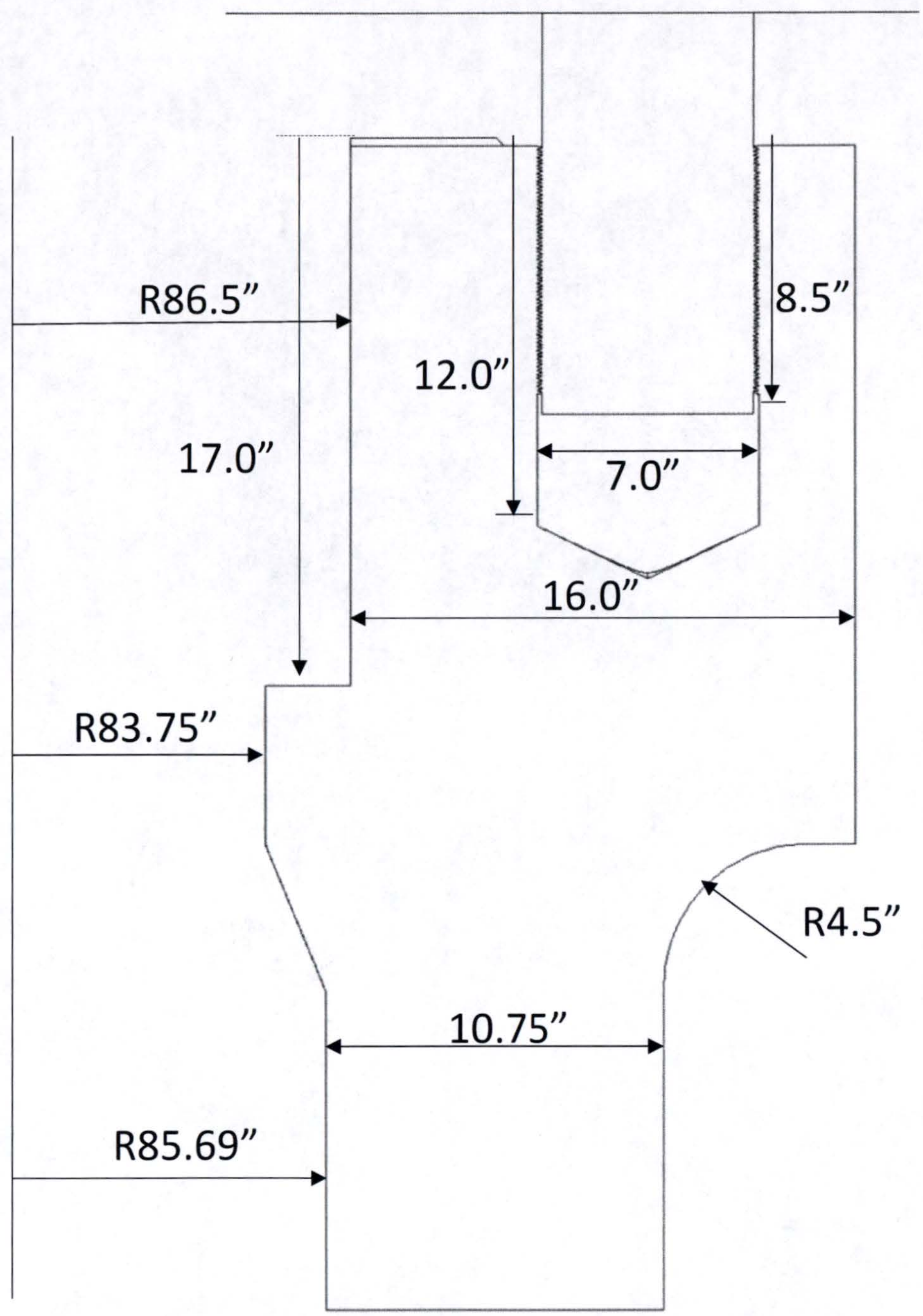


Figure 6-2
Modeled dimensions

The model is a symmetric model along two radial planes, containing a local portion of an RPV bolting flange and a threaded bolt and consists of a symmetric one-half section of the bolt and a symmetric mid-section between two bolts, as shown in Figure 6-3. The mesh of the finite element model is shown in Figure 6-4. Contact surfaces are used in the thread interface between the bolt and the flange during preload application. To simulate the RPV head, the top of the cladding surface is fixed in the axial direction, and the bottom of the flange is coupled axially to simulate the rest of the vessel. Symmetric boundary conditions are applied on both symmetry faces of the model.

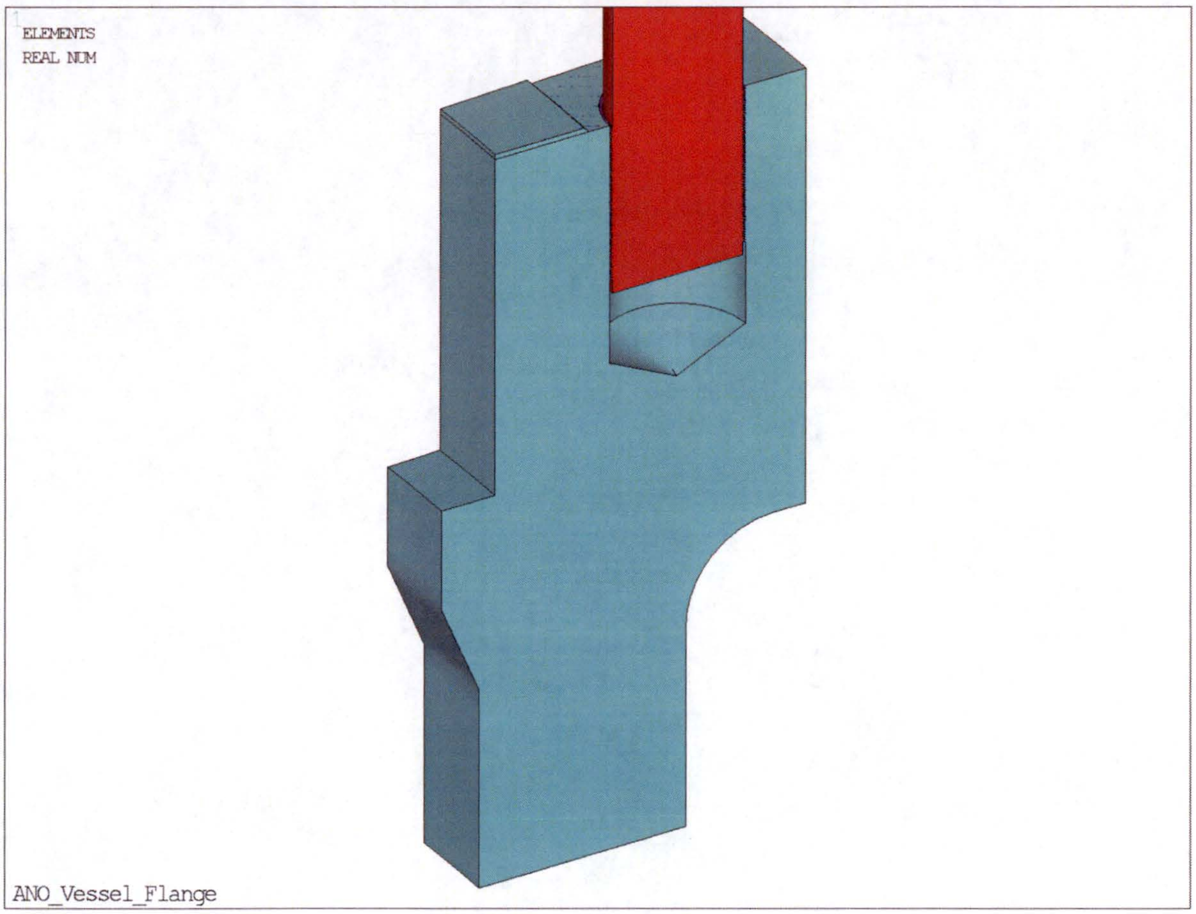


Figure 6-3
Finite element model showing bolt and flange connection

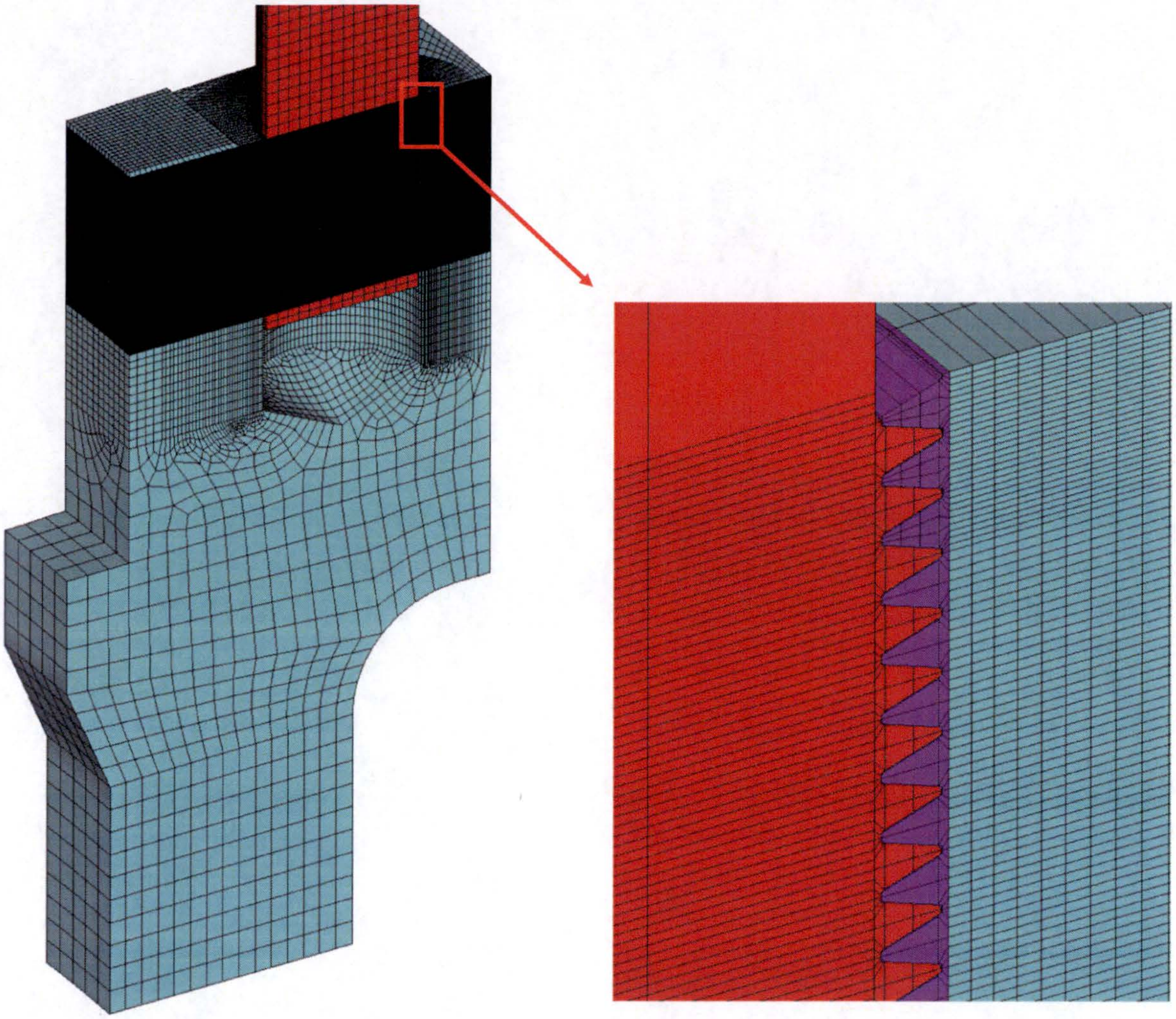


Figure 6-4
Finite element model mesh with detail at thread location

6.1.2 Applied Loads

The loading considered for the Threads in Flange component consists of the following:

- Design/operating pressure
- Bolt preload
- Thermal transients

The design pressure and temperature for the U.S. PWR fleet (typically 2500 psig [17.2 MPa] and 600°F [316°C]) are much higher than those for BWRs (typically 1250 psig [8.6 MPa] and 550°F [288°C]); therefore, PWR loading conditions are used to bound the U.S. fleet.

The bounding preload is calculated using the method outlined in several RPV manuals. For this analysis, the preload will be based on the largest RPV diameter, smallest bolts, and least number of bolts to obtain the largest preload and bound the fleet:

$$P_{preload} = \frac{C \cdot P \cdot ID^2}{S \cdot D^2} = \frac{1.1 \cdot 2500 \cdot 173^2}{54 \cdot 6^2} = 42{,}338 \text{ psi}$$

Where:

$P_{preload}$	=	Preload pressure to be applied on modeled bolt (psi)
P	=	Internal pressure (psi)
ID	=	Largest inside diameter of RPV (in.)
C	=	Bolt-up contingencies (+10%)
S	=	Least number of studs
D	=	Smallest stud diameter (in.)

Figure 6-5 shows the resulting axial stress due to bolt preload, with bolt removed from the model for clarity.

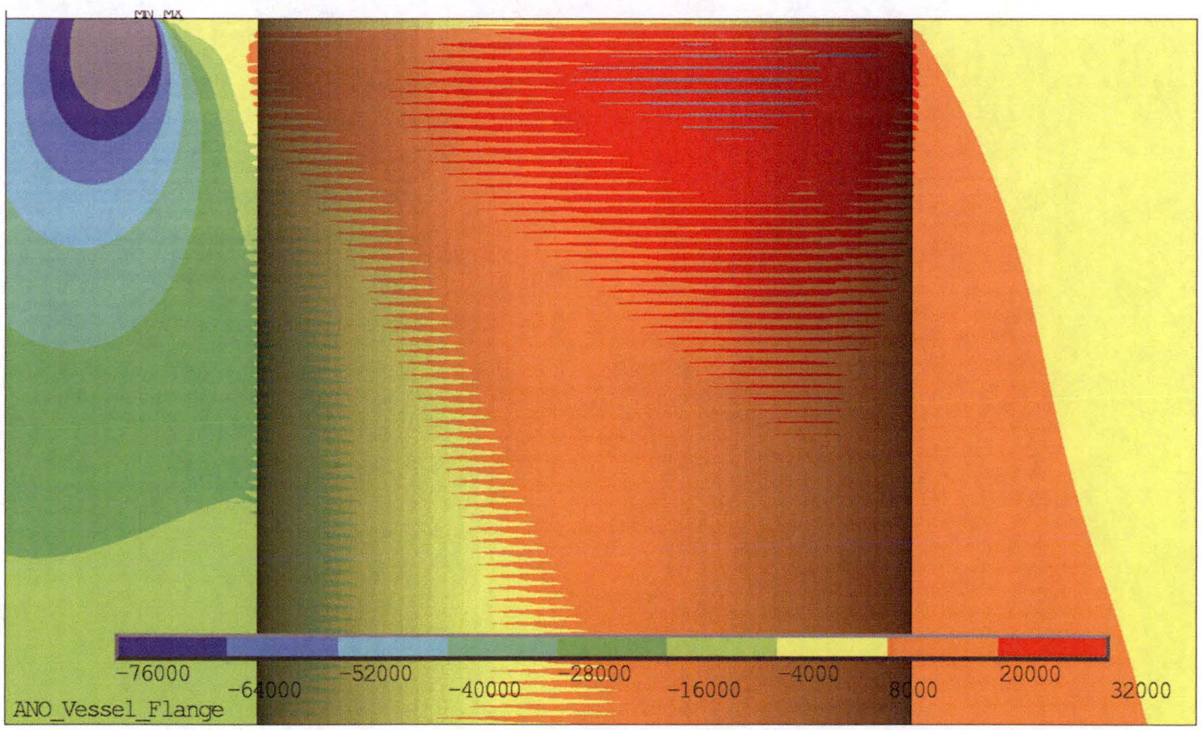


Figure 6-5
Axial stress contour for preload

The only significant transient affecting the bolting flange is heat-up/cooldown. This transient typically consists of a steady 100°F/hour ramp up to the operating temperature, with a corresponding pressure ramp up to the operating pressure. Typical values of design pressure and temperature are considered for conservatism. Figure 6-6 shows the corresponding axial stress at time = 19,080 seconds, which is the time of the greatest through-wall temperature difference.

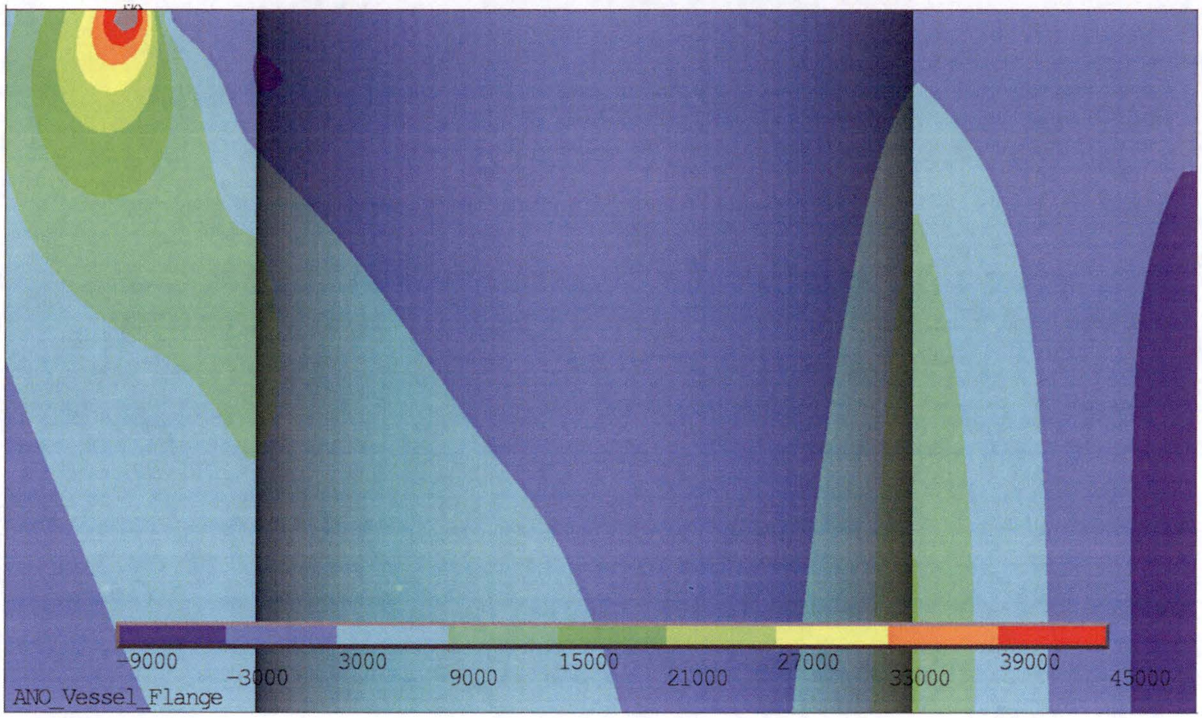


Figure 6-6
Axial stress contour for heat-up transient at time = 19,080 seconds

Since only a local portion of the RPV bolting flange in the axial direction is modeled, a pressure is applied at the bottom far end of the model to include the effect of an end cap pressure. The pressure end cap load is determined as follows:

$$P_{\text{end cap}} = \frac{P \cdot ID^2}{\left(OD^2 - ID^2\right)} = \frac{2500 \cdot 171.375^2}{\left(192.875^2 - 171.375^2\right)} = 9{,}376 \text{ psi}$$

Where:

$P_{\text{end cap}}$	=	End cap pressure on RPV wall (ksi)
P	=	Internal pressure (psi)
ID	=	Inside diameter of RPV (in.)
OD	=	Outside diameter of RPV (in.)

Figure 6-7 shows the axial stress due to the internal pressure load case.

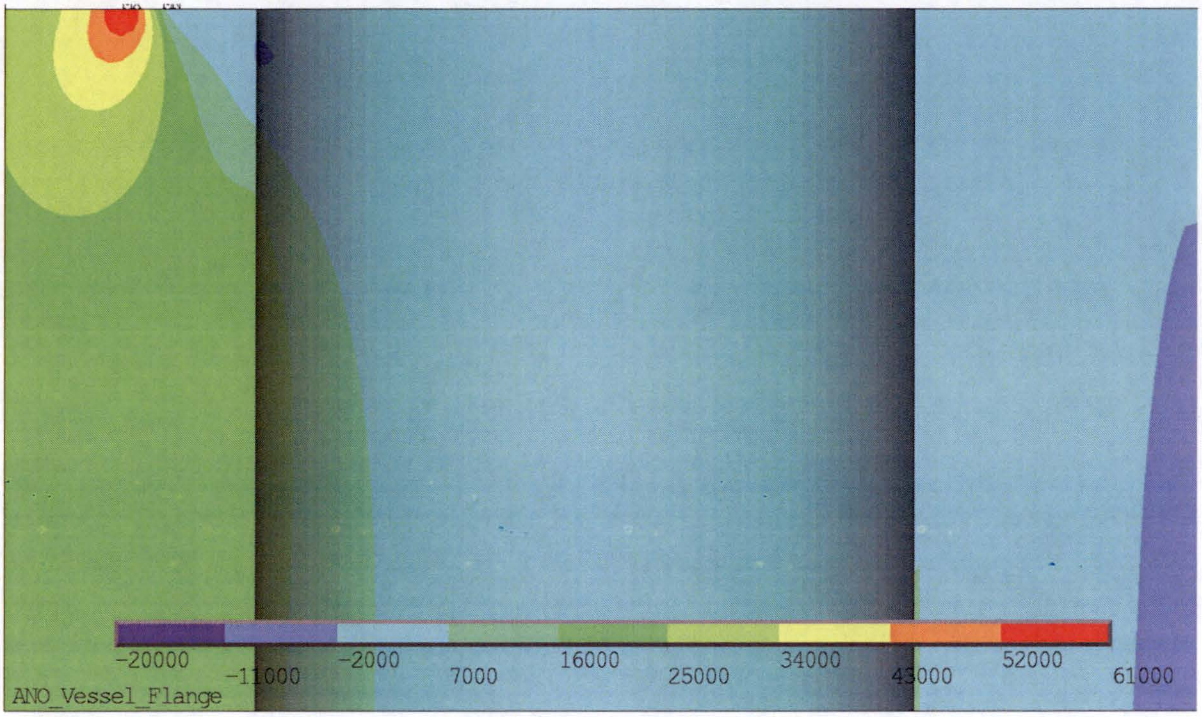


Figure 6-7
Axial stress contour for internal pressure

6.2 Flaw Tolerance Evaluation

A flaw tolerance evaluation is performed using the stresses to determine the allowable flaw size and how long it will take an initial postulated flaw size to reach the allowable flaw size. Even though the flange of the vessel operates in the upper shelf region, which would justify the use of elastic-plastic fracture mechanics (EPFM) principles for determining the allowable flaw size, a conservative linear elastic fracture mechanics (LEFM) approach is used for calculating the allowable flaw size since higher structural factors are used per ASME Section XI for determination of allowable flaw sizes using LEFM. LEFM is also employed for performing the crack growth evaluation. The use of LEFM requires the determination of the stress intensity factor (K), which depends on the geometry of the configuration, the applied stresses, and the flaw size.

6.2.1 Stress Intensity Factor Determination

Stress intensity factors (Ks) at four depths for 360° inside-surface-connected, part-through-wall circumferential flaws are calculated using FEA with the model described in Section 6.1. The maximum K values around the bolt hole circumference for each flaw depth (a) are extracted and used as input into pc-CRACK software [6] to perform the crack growth calculations. Because the *K vs. a* profile is used as input, the shape of the component is not relevant.

The circumferential flaw is modeled to start between the 10th and 11th flange threads from the top end of the flange because that is where the largest tensile axial stress occurs. The modeled flaw depth–to–wall thickness ratios (a/t) are 0.02, 0.29, 0.55, and 0.77, as measured in any direction from the bolt hole. This creates an ellipsoidal flaw shape around the circumference of the flange, as shown in Figure 6-8 for the flaw model with a/t = 0.77 a/t crack model. The crack tip mesh for the other flaw depths follows the same pattern. When preload is not being applied, the bolt, bolt threads, and flange threads are not modeled. The model is otherwise unchanged between load cases.

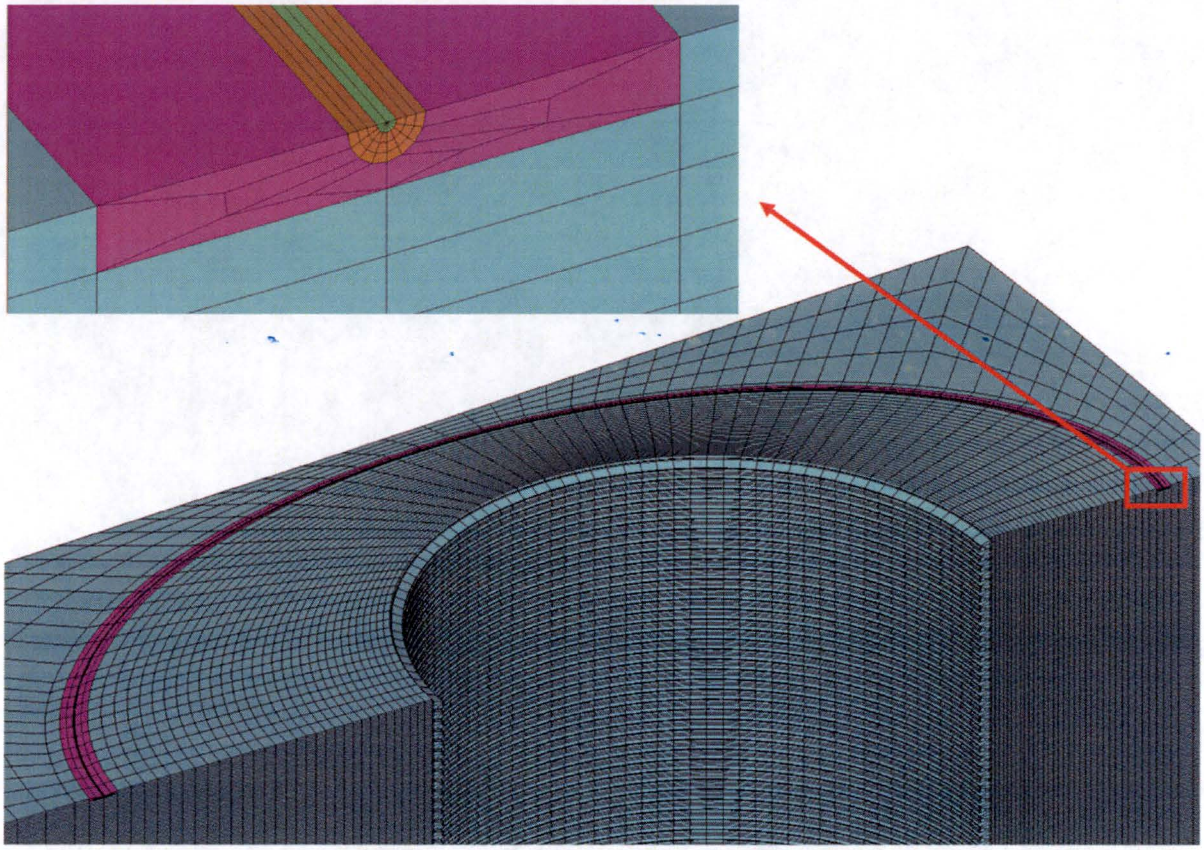


Figure 6-8
Cross section of circumferential flaw with crack tip elements inserted after 10th thread from top of flange

The maximum K results are summarized in Table 6-1 for the four crack depths. Because the crack tip varies in depth around the circumference, the maximum K from all locations at each crack size is conservatively used for the *K vs. a* profile.

Table 6-1
Maximum K vs. a/t

Load	K at Crack Depth (ksi√in)			
	0.02 a/t	**0.29 a/t**	**0.55 a/t**	**0.77 a/t**
Preload	11.2	17.4	15.5	13.9
Preload + Heatup + Pressure	13.0	19.8	16.1	16.3

6.2.2 Allowable Flaw Size Determination

As discussed above, a conservative LEFM approach consistent with ASME Code Section XI IWB-3600 is used to determine the allowable flaw size. Based on IWB-3612, the acceptance criterion based on allowable stress intensity factor is:

$K_I < K_{Ic}/\sqrt{10}$ for normal operating condition

Where:

K_I = Applied stress intensity factor (ksi$\sqrt{\text{in.}}$)

K_{Ic} = Lower bound fracture toughness at operating temperature

The fracture toughness K_{Ic} is obtained from Figure A-4200-1 of Appendix A of ASME Code Section XI for a material operating in the upper shelf region. In this case, the value of K_{Ic} = 220 ksi$\sqrt{\text{in.}}$ is used, which is the maximum value allowed for the applicable conditions in this study. Hence $K_{Ic}/\sqrt{10}$ results in an allowable K value of 69.6 ksi$\sqrt{\text{in.}}$ As can be seen from Table 6-1, the allowable K is not exceeded for all crack depths up to the deepest analyzed flaw of a/t = 0.77. Hence the allowable flaw depth of the 360° circumferential flaw is at least 77% of the thickness of the flange.

6.2.3 Crack Growth Evaluation

For the crack growth evaluation, an initial postulated flaw size of 0.2 in. (5.08 mm) is chosen consistent with the ASME Code Section XI IWB-3500 acceptable flaw size (see Section 2.2). The deepest flaw analyzed is a/t = 0.77 because of the inherent limits of the model.

Two load cases are considered for fatigue crack growth: heat-up/cooldown and bolt preload. The heat-up/cooldown load case includes the stresses due to thermal and internal pressure loads and is conservatively assumed to occur 50 times per year. The bolt preload is assumed to be present and constant during the load cycling of the heat-up/cooldown load case. The bolt preload load case is conservatively assumed to occur five times per year, and these cycles do not include thermal or internal pressure.

The resulting crack growth as calculated by pc-CRACK is negligible due to the small delta K and the relatively low number of cycles associated with the transients evaluated. Because the crack growth is insignificant, the allowable flaw size will not be reached and the integrity of the component is not challenged for at least 80 years (original 40-year design life plus additional 40 years of plant life extension), indicating that the component is very flaw tolerant. This clearly demonstrates that the Threads in Flange examinations can be eliminated without affecting the safety of the RPV.

7
RELATED RPV ASSESSMENTS

A number of plant-specific and generic industry studies have been conducted assessing the integrity of the RPV. These studies have been tailored toward evaluating operating events (for example, inoperable stud), regulatory interactions (for example, the ATWS rule), and beyond design basis events (for example, severe accident guidelines). The conclusion drawn from these studies is that the RPV, including the flange, studs, and other connected equipment (for example, nozzles), has large design margins. Several of these efforts are summarized next.

7.1 Inoperable Stud

A PWR plant was evaluated [7] for a hole at the main vessel flange, which contained a stud. During an outage, the hole was machined to accept a threaded sleeve that would allow installation of a closure stud. In the process of machining the stud hole so that the threaded sleeve could be installed, the vendor equipment malfunctioned. It was decided to leave the stud hole in its present condition with partial machining done.

A calculation was performed to evaluate the redistribution of forces and stresses in the RPV flange closure studs when one closure stud is out of service. The calculation accounted for the geometries of the closure studs and stud locations in accordance with the design drawings. The results reaffirmed the structural integrity of the RPV and that all of the stress intensity and cumulative fatigue usage factor limits of the applicable ASME Boiler and Pressure Vessel (B&PV) Code were still satisfied with one closure stud out of service. The calculation concluded that both the maximum stud service stress and the maximum average service stress in the closure studs adjacent to the out-of-service closure stud would still be less than the ASME B&PV Code limit due to the increased loading. The cumulative fatigue usage factor of the closure studs remained below the ASME B&PV Code allowable limit for the rest of the operational life of the reactor vessel. RPV flange separation at the O-ring gaskets was also evaluated with the finding that the O-rings will remain sealed during reactor operation, given the increased load in the closure studs adjacent to the out-of-service closure stud.

In addition, the conclusions of the plant-specific analyses for the affected plant were expanded to be applicable to three other units within the utility's fleet. Also as part of this effort, reference was made to another plant that had received U.S. Nuclear Regulatory Commission (NRC) approval to operate with an inoperable stud [8].

7.2 ATWS Rule

According to Reference [9], NRC issued what is known as the *Anticipated Transient Without Scram* (ATWS) rule. This rule was issued to require design changes to reduce expected ATWS frequency and consequences. The basis for the ATWS rule is provided in References [10, 11, 12]. Many studies have been conducted to understand the ATWS phenomenon and key contributors to successful response to an ATWS event. In particular, the reactor coolant system (RCS) and its individual components were reviewed to determine weak links. As an example, even though significant structural margin was identified in SECY-83-293 [10], for PWRs the ASME Service Level C pressure of 3200 psig (22 MPa) was assumed be to an unacceptable plant condition. Although a higher ASME service level might be defensible for major RCS components, other portions of the RCS could deform to the point of inoperability. In addition, there was the concern that steam generator tubes might fail before other RCS components, with a resultant bypass of containment. The key takeaway for these studies is that the RPV flange ligament was not identified as a weak link, and other RCS components were significantly more limiting. Therefore, there is substantial structural margin associated with the RPV flange.

8 RISK IMPACT

As discussed in previous sections, operating experience and inspection experience identify that the RPV Threads in Flange are performing with very high reliability. This is due to the robust design and a relatively benign operating environment (for example, the number and magnitude of transients are small, generally not in contact with primary water at plant operating temperatures and pressures). The robust design is manifested in the fact that plant operation has been allowed at several plants even with a bolt/stud assumed to be out of service.

Conducting a plant-specific quantitative risk impact analysis for the elimination of these examination requirements is not warranted based on the bounding analysis provided next. Conducting a plant-specific quantitative risk impact analysis would require additional unnecessary resources and would not add any new risk insights. To have an impact on risk, some measurable failure potential must exist (that is, some type of operative degradation) along with conditions that could create a credible consequence (for example, cause a plant initiating event, impact the plant's mitigative ability, or some combination of both); the end result needs to have a measurable impact on plant risk.

With this in mind, the following assessment is provided. This would bound any impact of eliminating the RPV flange ligament examination.

To determine the impact on risk, a determination of initiating event (IE) and conditional core damage probability (CCDP) given the event is needed. If leakage were to occur, it would be very limited and less than events modeled as a small loss-of-coolant accident (LOCA) in plant probabilistic risk assessments (PRAs). As such, small-small LOCA (very small LOCA [VSLOCA]), normal plant trip (NPT), and manual plant shutdown (MSHTDN) were used to bound the CCDP value. (Note: it is believed that manual plant shutdown is the more representative case.) These assumptions are based on the significant operating and inspection experience as documented in the previous sections and on robust design margin for the RPV flange, studs, and ligaments. A survey was conducted to determine representative values for use as a CCDP. From this survey, an upper bound value for CCDP is 7E-5 for VSLOCA, a value of 2E-06 for NPT, and a lower bound value of <1E-06 for MSHTDN will be used as representative values.

With respect to IE, there have been no occurrences of pressure boundary leakage related to the RPV flange ligament inspection. If we were to assume one event in the 16,000 reactors years of operation, we could conservatively assess the benefit of continuing the RPV flange ligament examinations. And while it is anticipated that with the elimination of the examinations there will continue to be no leakage, an upper bound of 1 event per 10 years with 100 plants in the U.S. industry will be used for the case with the ligament examination requirements no longer in effect. The basis for this assumption is that if the event were to occur, according to industry OE review requirements, other applicable plants would need to assess the applicability of the event to their operating regime/practices prior to any significant new occurrences. In addition, due to the size of the U.S. fleet and its associated fuel cycle (12 month, 18 month, 24 month), a large number of plants undergoes refueling outages each year. As such, there is ample opportunity—irrespective of volumetric examination of the RPV flange ligament—to visually assess the overall condition of the RPV flange connection during these refueling outages. Therefore, the IE frequency for the base case (that is, continue with the existing ligament examination requirements) will be conservatively assumed to be 6.3E-05 (that is, 1/16,000), and the frequency for the proposed action (elimination of the examination requirement) will be 1.0E-03 (1/(10*100)) (see Tables 8-1 and 8-2).

Because the mitigative ability of the plant is not impacted by the proposed action, the change in risk associated with the proposed action is simply a result of the postulated change in leakage frequency. Based on the above, the change in leakage IE frequency would be 9.4E-04 (1E-3 – 6.3E-05 = 9.4E-04) or conservatively set at 1E-03. As can be seen next, the risk captured by the current requirements is extremely low and not commensurate with the associated burden (worker exposure, personnel safety, and critical path time). In addition, even with the conservative assumptions discussed previously, the impact on risk from industrywide adoption of the proposed action is quite low.

In addition to the above, it is important to note that all other inspection activities—including the system leakage test (examination category B-P), which is conducted each refueling outage—will continue going forward.

Finally, as discussed in Reference [13] (which includes work supported by NRC), without an active degradation mechanism present, it was concluded that if pre-service inspection has confirmed that the inspection volume contains no flaws or indications and subsequent ISIs do not provide any additional value. As discussed in earlier sections, the RPV flange ligaments have not only received the required pre-service examinations, more than 10,000 ISIs have been carried out—with no relevant findings.

Table 8-1
Bounding risk captured by current requirements

Current	$IE_{frequency}$	CCDP	CDF
MSHTDN	6.3E-05	<1.0E-06	<6.3E-11
NPT	6.3E-05	2.0E-06	1.3E-10
VSLOCA	6.3E-05	7.0E-05	4.4E-09

Table 8-2
Bounding risk increase due to the proposed action

Proposed Action	$IE_{frequency}$	CCDP	CDF
MSHTDN	1.0E-03	<1.0E-06	<1.0E-09
NPT	1.0E-03	2.0E-06	2.0E-09
VSLOCA	1.0E-03	7.0E-05	7.0E-08

9
SUMMARY

This report provides the basis for the elimination of the RPV Threads in Flange examination requirement (ASME Section XI Examination Category B-G-1, item number B6.40). This report was developed because industry experience suggested that these examinations have not been finding service-induced degradation and that there are negative impacts on worker dose, personnel safety, and critical path time due to these examinations.

This report includes the results of an industry survey in which 168 units provided the results of their RPV Threads in Flange examinations as well as insight into the negative aspects of having to conduct these examinations (for example, worker exposure and personnel safety). A literature search was also conducted to assess any related OE that could impact the position that the RPV Threads in Flange examination requirement is providing low value added and is not commensurate with the associated worker exposure, outage complications, and personnel safety concerns.

A degradation mechanism evaluation was performed to identify possible mechanisms that could potentially degrade the Threads in Flange component while in service. It was concluded that the only potential active degradation mechanism is mechanical/thermal fatigue. This degradation mechanism was then addressed in a flaw tolerance evaluation using a representative configuration of a typical PWR plant to determine how long it would take a postulated flaw with this mechanism to challenge the integrity of the RPV. The allowable flaw size was determined to be at least 77% of the component thickness. A fatigue crack growth analysis was performed with an initial postulated flaw corresponding to the ASME Section XI acceptance standards flaw. Crack growth was determined to be insignificant over 80 years of plant life (original 40-year design life plus additional 40 years of plant life extension). This indicates that the integrity of the RPV would not be challenged by any potential degradation mechanism.

Further, a review of a number of plant-specific and generic industry studies that have been used to assess the structural integrity of the RPV was conducted. The conclusion drawn from these studies is that the RPV, including the flange, studs, and other connected equipment (for example, nozzles), has large margins.

Finally, a bounding generic risk impact assessment was conducted. According to this assessment, the risk captured by the current requirements is extremely low and not commensurate with the associated burden (worker exposure, personnel safety, and critical path time). Even with the conservative assumptions included in the analysis, the impact on risk from elimination of these examination requirements is low.

Based on these considerations, it is concluded that the Threads in Flange examination as mandated in ASME Code Section XI can be eliminated without increasing plant risk or posing any safety concerns for the RPV. Savings gained from the elimination of this inspection can be applied toward more meaningful examinations of other risk-significant plant components.

10
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A
SURVEY RESULTS: U.S. FLEET

Table A-1 provides additional U.S. fleet details. Additional plant-unique information is not provided in this report.

Table A-1
U.S. fleet details

NSSS Vendor	Type	Total Number of RPV Threads in Flange Locations	Number of Examinations with No Reportable Indications[1]	Number of Examinations with Reportable Indications	Comment on the Impact of These Examinations on the Site (for example, Critical Path Time)
B&W	PWR	60	60	0	No impact – scheduled off critical path.
CE	PWR	54	54	0	No impact – scheduled off critical path.
West.	PWR-3	58	174	0	Critical path time.
West.	PWR-3	58	174	0	Critical path time.
West.	4 Loop	54	108	0	Performed with head suspended over flange for dose reduction, safety concern.
West.	4 Loop	54	108	0	Performed with head suspended over flange for dose reduction, safety concern.
GE	BWR-4	92	92	0	No impact on critical path.
GE	BWR-4	92	92	0	No impact on critical path.
GE	BWR-4	92	92	0	No impact on critical path.
GE	BWR-4	64	192	0	Critical path time.
GE	BWR-4	64	192	0	Critical path time.
West.	PWR 4 Loop	54	162	0	Critical path/dose
West.	PWR 4 Loop	54	162	0	Critical path/dose.
West.	PWR 4 Loop	54	54	0	8 hours.
CE	PWR	54	54	0	Critical path exam.
CE	PWR	54	54	0	Potential impact to critical path.
West.	PWR 4 Loop	54	162	0	Critical path time.
West.	PWR 4 Loop	54	162	0	Critical path time.
GE	BWR-6	64	64	0	Perform 1/3 over each period.

Table A-1 (continued)
U.S. fleet details

NSSS Vendor	Type	Total Number of RPV Threads in Flange Locations	Number of Examinations with No Reportable Indications[1]	Number of Examinations with Reportable Indications	Comment on the Impact of These Examinations on the Site (for example, Critical Path Time)
GE	BWR-5	76	304	0	~4 work-hours critical path per exam. Last exam dose estimate was approximately 0.1 rem.
GE	BWR-4	52	52	0	Exam time (critical path) around 2 hours.
B&W	PWR	60	240	0	These exams consistently impact critical path time during the outage. In addition, due to the proximity to the vessel (and plant conditions), these exams require UT personnel to receive a substantial amount of dose.
West.	PWR 4 Loop	54	54	0	
West.	PWR 4 Loop	54	54	0	
West.	PWR 4 Loop	54	54	0	Rad dose.
West.	PWR 4 Loop	54	54	0	Rad dose.
GE	BWR-3	92	184	0	Not a big deal.
GE	BWR-3	92	368	0	Not a big deal.
GE	BWR-4	240	240	0	Approximately 1 to 2 hours - 60 bolt/stud holes examined over 4 intervals.
West.	PWR 3 Loop	58	226	0	
West.	PWR 3 Loop	58	174	0	
GE	BWR-4	68	68	0	Typically, this takes 3–4 hours prior to cavity flood up.
GE	BWR-3	52	0	0	

Table A-1 (continued)
U.S. fleet details

NSSS Vendor	Type	Total Number of RPV Threads in Flange Locations	Number of Examinations with No Reportable Indications[1]	Number of Examinations with Reportable Indications	Comment on the Impact of These Examinations on the Site (for example, Critical Path Time)
GE	BWR-5	76	25	0	
GE	BWR	52	255	0	
GE	BWR	56	254	0	
GE	BWR-4	92	92	0	These exams require 1 hour of critical path time.
West.	PWR	54	54	0	
West.	PWR	54	54	0	
GE	BWR-5	68	9	0	1 hr critical path and 100 mrem.
GE	BWR-5	76	9	0	1 hr critical path and 100 mrem.
GE	BWR-4	76	202	0	1/3 of threaded holes are inspected each period. Inspection is performed after the vessel head is removed prior to flood up. Biggest impact is dose; impact to critical path is no more than 4 hours.
GE	BWR-4	76	202	0	1/3 of threaded holes are inspected each period. Inspection is performed after the vessel head is removed prior to flood up. Biggest impact is dose; impact to critical path is no more than 4 hours.
West.	PWR 4 Loop	54	162	0	Critical path time.
West.	PWR 4 Loop	54	162	0	Critical path time.
CE	PWR 2 Loop	54	54	0	Impacts critical path/ALARA.

Table A-1 (continued)
U.S. fleet details

NSSS Vendor	Type	Total Number of RPV Threads in Flange Locations	Number of Examinations with No Reportable Indications[1]	Number of Examinations with Reportable Indications	Comment on the Impact of These Examinations on the Site (for example, Critical Path Time)
West.	PWR 4 Loop	54	54	0	Critical path/ALARA.
GE	BWR-3	67	67	0	Dose and critical path time.
GE	BWR-5	79	79	0	Dose and critical path time.
West.	PWR 3 Loop	58	174	0	This exam definitely impacts critical path because it is time on vessel.
West.	PWR 3 Loop	58	174	0	Critical path because it is time on vessel.
B&W	PWR 2 Loop	60	180	0	Critical path time.
B&W	PWR 2 Loop	60	180	0	Critical path time.
B&W	PWR 2 Loop	60	180	0	Critical path time.
GE	BWR-2	64	128	0	
CE	PWR 2 Loop	54	8	0	No comment.
CE	PWR	54	54	0	
CE	PWR	54	54	0	
CE	PWR	54	54	0	
GE	BWR-4	92	92	0	Remaining threads to be examined in next refueling outage.
GE	BWR-4	92	92	0	Remaining threads to be examined in next refueling outage.

Table A-1 (continued)
U.S. fleet details

NSSS Vendor	Type	Total Number of RPV Threads in Flange Locations	Number of Examinations with No Reportable Indications[1]	Number of Examinations with Reportable Indications	Comment on the Impact of These Examinations on the Site (for example, Critical Path Time)
GE	BWR-6	72	240	0	Very high dose field and contaminated zone; must be done with the vessel head suspended, which is a safety risk. Exams are done 1/3 every period, so about 4 hours of critical path time a period.
GE	BWR 3	56	56	0	
West.	PWR 2 Loop	192	192	0	2 to 4 hours/exam; approximately 1 to 2 hours; 48 bolt/stud holes examined over 4 intervals.
West.	PWR 2 Loop	192	192	0	2 to 4 hours/exam; approximately 1 to 2 hours; 48 bolt/stud holes examined over 4 intervals.
West.	PWR 2 Loop	48	48	0	Critical path, dose, safety.
West.	PWR 2 Loop	48	48	0	Critical path, high dose, and safety concern.
GE	BWR-3	92	13	0	Slows down IVVI work.
GE	BWR-3	92	14	0	Slows down IVVI.
West.	PWR 2 Loop	48	96	0	Suspended load safety concern and dose.
GE	BWR-6	64	0	0	
West.	PWR 3 Loop	50	200	0	Critical path time.
West.	PWR 4 Loop	54	54	0	These exams require 1 hour of critical path time each period and an estimated dose of approximately 40 to 60 mrem.

Table A-1 (continued)
U.S. fleet details

NSSS Vendor	Type	Total Number of RPV Threads in Flange Locations	Number of Examinations with No Reportable Indications[1]	Number of Examinations with Reportable Indications	Comment on the Impact of These Examinations on the Site (for example, Critical Path Time)
West.	PWR 4 Loop	54	54	0	These exams require 1 hour of critical path time each period and an estimated dose of approximately 40 to 60 mrem.
West.	PWR 4 Loop	108	108	0	Approximately 1 to 2 hours; 54 stud/bolt holes examined over 2 intervals.
West.	PWR 4 Loop	54	54	0	Remote inspections performed during spring 2015 Unit 1 outage. Requires use of fuel bridge after core offload, while at full cavity. Other option for exam is to put examiners in the cavity scanning manually while RPV head is suspended and used for shielding, or the suspended load can be removed and then you have open access to the reactor vessel with no shielding.
West.	PWR 3 Loop	58	116	0	Critical path time.
West.	PWR 4 Loop	36	36	0	Critical path.
West.	PWR 4 Loop	36	36	0	Critical path.
CE	PWR 4 Loop	162	162	0	54 bolt/stud holes examined over 3 intervals.
CE	PWR 4 Loop	162	162	0	54 bolt/stud holes examined over 3 intervals.
West.	PWR 3 Loop	58	58	0	We received approximately 150 mrem of dose related to this examination. It did occur on critical path. Another big concern is that of FME into the vessel from the UT transducer or tool.

Table A-1 (continued)
U.S. fleet details

NSSS Vendor	Type	Total Number of RPV Threads in Flange Locations	Number of Examinations with No Reportable Indications[1]	Number of Examinations with Reportable Indications	Comment on the Impact of These Examinations on the Site (for example, Critical Path Time)
West.	PWR 3 Loop	58	58	0	
GE	BWR-4	76	12	0	Only those threads that are revealed when studs are removed for installation of the cattle chute are inspected as allowed by Section XI.
GE	BWR-4	76	12	0	Only those threads that are revealed when studs are removed for installation of the cattle chute are inspected as allowed by Section XI.
B&W	PWR 4 Loop	60	200	0	Dose issues with regard to plenum move.
West.	PWR 3 Loop	232	232	0	58 bolt/stud holes examined over 4 intervals.
West.	PWR 3 Loop	232	232	0	58 bolt/stud holes examined over 4 intervals.
West.	PWR 3 Loop	58	58	0	3rd interval inspection, performed 11/6/2012; critical path time 14 hours.
West.	PWR 4 Loop	54	144	0	
West.	PWR 4 Loop	54	144	0	
CE	PWR	54	27	0	Scheduled off critical path. Exams are scheduled in two outages.

Table A-1 (continued)
U.S. fleet details

NSSS Vendor	Type	Total Number of RPV Threads in Flange Locations	Number of Examinations with No Reportable Indications[1]	Number of Examinations with Reportable Indications	Comment on the Impact of These Examinations on the Site (for example, Critical Path Time)
West.	PWR 4 Loop	54	162	0	Approximately 6 hours' impact to critical path schedule with an approximate average of 1.2 person-rem per interval to complete flange ligament inspections from flange face and 0.050 person-rem per interval with long-handled tool and cavity flooded. Relief request was submitted to, and approved by, the NRC to extend Interval 3 by three months in order to complete the exams in the next refueling outage if the NRC determined that the examination performed utilizing the long-handled tool did not provide adequate inspection coverage as compared to the direct flange technique.
West.	PWR 4 Loop	54	54	0	The methods used to perform these exams in U1R2, U1R4, and U1R7 in the first 10-year interval generated a large amount of dose for the examiners. Also, this previous method requires pausing the removal or installation of the reactor vessel head, a short distance off of the RV flange (for shielding) with cribbing. This is a critical path activity delay along with being an impact to a critical lift. This method is also considered a safety concern due to having people within or in close proximity to the load drop zone.

[1] Most plants inspect 100% of the locations in a single outage while a small number of plants inspects only a fraction of the locations in an outage. Several of these plants provided data only for the most recent outage.

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