SHAWN GIBBY
Vice President

Nuclear Engineering
526 South Church Street, EC-07H
Charlotte, NC 28202
704 519-5138
Shawn.Gibby@duke-energy.com

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U.S. Nuclear Regulatory Commission
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Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325, 50-324
Renewed License Nos. DPR-71 and DPR-62

Catawba Nuclear Station, Unit Nos. 1 and 2
Docket Nos. 50-413, 50-414
Renewed License Nos. NPF-35 and NPF-52

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400
Renewed License No. NPF-63

McGuire Nuclear Station, Unit Nos. 1 and 2
Docket Nos. 50-369, 50-370
Renewed License Nos. NPF-9 and NPF-17

SUBJECT **Response to Requests for Additional Information for Reactor Vessel Closure Stud Exam Extension Alternative**

REFERENCES:

1. Duke Energy Letter RA-19-0352, “Relief Request for Alternative for Reactor Vessel Closure Stud Examinations,” dated December 1, 2020 (Agencywide Document Access and Management System [ADAMS] Accession No. ML20336A033).
2. NRC Letter, “RAIs – Duke Fleet – Alternative Request for Reactor Closure Studs – EPID L-220-LLR-0156(003),” dated December 15, 2021 (ADAMS Accession No. ML21354A861).

Ladies and Gentlemen:

In Reference 1, Duke Energy Carolinas, LLC and Duke Energy Progress, LLC (collectively referred to as Duke Energy) requested U.S. Nuclear Regulatory Commission (NRC) approval for a proposed alternative to American Society of Mechanical Engineers (ASME) Section XI Code requirements for reactor vessel closure stud examinations. Specifically, Duke Energy requested an alternative to ASME Code Section XI, IWB-2500(a), Table IWB-2500-1, Examination Category B-G-1, Item No. B6.20. In Reference 2, the NRC requested additional information to complete its review.

Enclosure 1 to this letter provides Duke Energy's response to the requests for additional information (RAIs). Enclosure 2 provides EPRI report 3002014589, "Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs."

This letter contains no regulatory commitments. Should you have any questions concerning this letter, or require additional information, please contact Lee Grzeck, Fleet Licensing Manager (Acting), at (980) 373-1530.

Sincerely,

Shawn K. Gibby

Shawn Gibby
Vice President – Nuclear Engineering

Enclosures:

1. Response to Requests for Additional Information
2. EPRI Report 3002014589, "Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs"

cc:

L. Dudes, Regional Administrator USNRC Region II
G. Smith, USNRC Senior Resident Inspector – BNP
J. D. Austin, USNRC Senior Resident Inspector – CNS
J. Zeiler, USNRC Senior Resident Inspector – HNP
G. A. Hutto, USNRC Senior Resident Inspector – MNS
L. Haeg, NRR Project Manager – BNP
Z. R. Stone, NRR Project Manager – CNS
A. Hon, NRR Project Manager – HNP
J. Klos, NRR Project Manager – MNS

Enclosure 1

Duke Energy Carolinas, LLC

Catawba Nuclear Station, Units 1 and 2

McGuire Nuclear Station, Units 1 and 2

Duke Energy Progress, LLC

Brunswick Steam Electric Plant, Units 1 and 2

Shearon Harris Nuclear Power Plant, Unit 1

Response to Requests for Additional Information

Request for Additional Information (RAI)-1

Issue

The licensee's proposed alternative request relies heavily on the results of the evaluation in EPRI report 14589, which has not been submitted to the NRC for review. Because of this reliance on the results of a report that has not been submitted for NRC review, the licensee's plant-specific request must include EPRI report 14589 for the NRC staff to make its regulatory findings on the request.

Request

Submit EPRI report 3002014589, "Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs," on the docket.

Duke Energy Response to RAI-1:

EPRI 3002014589 Final Report, November 2018 (Reference 1) is included as Enclosure 2.

RAI-1 References:

1. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

RAI-2

Issue

The NRC staff needs additional information related to the stud preload, as noted below:

a. In Section 3.3.2 of EPRI report 14589, EPRI stated that the stud average preload stress for pressurized water reactors (PWRs) is 46.1 ksi, which is different than the preload membrane stress value of 50.2 ksi given in Table 3-4 of EPRI report 14589.

b. The NRC staff noted that a previous Duke submittal for the RPV threads-in-flange (ADAMS Accession No. ML17221A305) showed actual stud preload stress values for some of the Duke Energy units requested in the current submittal, except for Brunswick, Unit 2 and Catawba, Unit 1. The NRC staff needs to confirm the stud preload stress values for Brunswick, Unit 2 and Catawba, Unit 1.

c. The NRC staff noted that the actual stud preload stress value of 44.328 ksi for Brunswick, Unit 1 (and Brunswick, Unit 2 if information in RAI-2b is confirmed) is higher than the preload stud membrane stress value of 41.6 ksi given in Table 3-3 of EPRI report 14589 for boiling water reactors (BWRs). The lower preload stud membrane stress value of 41.6 ksi in Table 3-3 of EPRI report 14589 could result in nonconservative fatigue crack growth, which could ultimately impact the requested interval extensions for Brunswick.

d. In Section 4.2.1 of EPRI report 14589, EPRI stated that "consistent with Paragraph G-2222(b) [of Section XI of the ASME Code], stresses from bolt preloading are considered primary loads." The NRC staff would expect that the licensee considers stress due to RPV internal pressure a primary stress, but the submittal is not clear whether that is the case.

e. In Attachment 1 of the submittal, the licensee stated that a fracture toughness (symbolized by the parameter K_{IC}) value of 190 ksi√in (discussed in Section 4 of EPRI report 14589) was used for the reactor vessel head closure studs of the subject Duke units. The NRC staff is not clear about the values of the temperatures during stud tensioning and preloading of the studs of the subject Duke units relative to the temperature at the K_{IC} value of 190 ksi√in.

Request

a. Confirm that the higher stud preload membrane stress value of 50.2 ksi given in Table 3-4 of EPRI report 14589 was the value used in the analysis.

b. Confirm that Brunswick, Unit 2 has the same actual stud preload stress value of 44.328 ksi for Brunswick, Unit 1 given in Table 2 of the previous Duke submittal and that Catawba, Unit 1 has the same actual stud preload stress value of 41.144 ksi for Catawba, Unit 2 given in the table.

c. Explain the impact of the higher actual stud preload stress value of 44.328 ksi for Brunswick, as compared to the preload stress value of 41.6 ksi in Table 3-3 of EPRI report 14589 for BWRs on the requested interval extensions for Brunswick, Units 1 and 2.

d. Confirm that in addition to stresses from bolt (i.e., reactor vessel head closure stud) preloading, the other primary stress used in the allowable flaw sizes discussed in Section 4.2 of EPRI report 14589 is RPV internal pressure stress.

e. State the temperature during stud tensioning and preloading of the reactor vessel head closure studs at each of the subject Duke Energy units and compare with the temperature at the K_{IC} value of 190 ksi√in.

Duke Energy Response to RAI-2.a:

The analysis model used for EPRI report 14589 (Reference 1) and output data were reviewed in response to this request. The author of EPRI report 14589 (Dominion Engineering, Inc.) confirms that the 50.2 ksi value given in Table 3-4 of EPRI report 14589 is the value used in the analysis.

RAI-2.a References:

1. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

Duke Energy Response to RAI-2.b:

The actual stud preload stress for Brunswick, Unit 2 is the same stress value as Brunswick, Unit 1 (44.328 ksi). The actual stud preload stress for Catawba, Unit 1 is 41.663 ksi; the slight difference between the two Catawba units is because the RPV designs at the two units are slightly different from each other (Catawba Unit 1 is a Rotterdam RPV and Catawba Unit 2 is a Combustion Engineering RPV).

The RPV threads-in-flange evaluation is based solely on the maximum stud preload stress, and therefore only considers the average axial force carried by the stud. This is an appropriate load to consider for a distributed loading condition like the threads in the closure flange. However, the closure studs are in a more complex loading condition. When an RPV closure stud is preloaded, the head shell and closure flange flex, leading to both axial forces and bending moments. Operating transients further change this stress state when differential thermal expansion between the closure flange and the studs cause changes in the preload and in the closure flexure.

As described in Section 3.4 of EPRI report 14589 (Reference 1), a broad variety of reactor vessel designs are present in the US fleet, including differing values of reactor vessel head radius, reactor vessel head shell thickness, and reactor vessel closure stud diameter. In preliminary studies supporting EPRI report 14589, these parameters were identified as key values leading to conservative prediction of stud bending stress and change in stud bending stress. Specifically, larger values for: 1) the ratio of the head inner radius to the head shell thickness, and 2) the ratio of the head inner radius to the stud diameter led to larger closure flexibility and larger stud bending stresses. EPRI report 14589 defines a bounding model geometry, and the applicability of that model to a given plant geometry is established by comparing the two ratios.

Therefore, for the current submittal, the values of the stud preload stress at the Duke Energy units are not applicable to the actual condition being considered, which is flaw growth caused by change in stud axial plus bending stress and limiting flaw size caused by stud axial plus bending stress.

RAI-2.b References:

1. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

Duke Energy Response to RAI-2.c:

As described in the response to RAI-2.b, the values of the stud preload stress at the Duke Energy units from the current submittal are not applicable to the actual condition being considered, which is flaw growth caused by change in stud axial plus bending stress and limiting flaw size caused by stud axial plus bending stress. The preload condition stud axial plus bending stress at Brunswick has been reviewed and confirmed to be less than the preload condition axial plus bending stress for the bounding model in EPRI report 14589 (Reference 1).

RAI-2.c References:

1. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

Duke Energy Response to RAI-2.d:

In RPV closures (and any other well-designed bolted joint), the stud preload force is greater than the internal pressure force acting on the head. The internal pressure force only acts to unload the compressive force at the closure flange mating surface, and not to add stress to the studs. Therefore, the stress from RPV internal pressure is implicitly considered since the stresses from bolt preload bound those from RPV internal pressure.

Duke Energy Response to RAI-2.e:

The lowest temperature permitted for stud preload by any of the reactor vessel tensioning procedures at the subject Duke Energy units is 60°F. This temperature value compares favorably with the temperatures used to establish the K_{IC} value of 190 ksi$\sqrt{in}$; the 1977 JPVT paper (Reference 1) uses “room temperature” for the temperature value where upper shelf behavior is established and the data show unambiguous upper shelf behavior in both static and dynamic toughness for temperatures greater than about 0°F.

RAI-2.e References:

1. Seeley, R.R. et al., “Fracture Toughness Properties of SA-540 Steels for Nuclear Bolting Applications,” Journal of Pressure Vessel Technology, August 1977.

RAI-3

Issue

The NRC staff needs additional information related to the applied loads, as noted below:

a. Section 3.3.3 of EPRI report 14589 stated that a hydrotest was included in the evaluation. However, in its review, the staff did not receive information with respect to a leakage test. The NRC staff needs this information since the leakage test needs to be accounted for as one of the loading conditions in the flaw tolerance evaluation.

b. In Section 3.3.3 of EPRI report 14589, EPRI stated that for PWRs, the normal operating pressure of 2,185 psi and normal operating temperature of 579°F were applied. The NRC staff noted that some of the Duke Energy PWR units included in the request may have higher normal operating pressure and normal operating temperature than the values used in EPRI report 14589. Catawba, Units 1 and 2, for example, have higher operating pressures and temperatures per the Catawba Updated Final Safety Analysis Report (UFSAR).

c. The NRC staff noted that the flaw tolerance evaluation in EPRI report 14589 did not include seismic loading and loading due to loss-of-coolant accident (LOCA). Section 3.2 of EPRI report 14589, stated that other mechanical loadings, such as those from seismic cases, do not generate significant additional loads for the RPV closure head. The NRC staff noted that seismic and LOCA events could cause the most limiting loads. Seismic loads, for instance, could generate relative motion between the reactor closure head and the reactor vessel closure flange, and thus generate additional loads on the reactor vessel head closure studs. These additional loads combined with operating loads could result in the most limiting flaw size in the studs, when the applied stress intensity factors (SIFs) are compared to fracture toughness in determining the maximum flaw sizes in Section 4.2.3 of EPRI report 14589.

Request

a. Explain how the leakage test performed for each of the Duke Energy units included in the request is bounded, in terms of stress and cycles, by the transients selected in the flaw tolerance evaluation in EPRI report 14589.

b. For each PWR unit included in the request, either confirm that the normal operating pressure of 2,185 psi and temperature of 579°F used in EPRI report 14589 bound the corresponding values for the unit or explain how the 2,185 psi and 579°F are adequate for each PWR unit.

c. Explain how seismic and LOCA events do not generate significant additional loads for the reactor vessel head closure studs of the Duke Energy units in the request.

Duke Energy Response to RAI-3.a:

The leakage test performed each cycle for the Duke Energy units is a pressurization of the RCS to operating pressure. Therefore, the loads associated with the leakage test are bounded by the operating condition stress values.

Duke Energy Response to RAI-3.b:

The operating temperature and pressure values used by the EPRI report model are an adequate representation of the operating conditions for the Duke Energy PWR units included in the request. As described in EPRI report 14589 (Reference 1) Sections 4.2.3 and 4.3.2, the peak stress that defines the limiting flaw size bounds the operating condition stress state, and the transient combination stress ranges that define the fatigue crack growth envelope the operating condition stress state. Therefore, the results of the analysis are insensitive to the precise values of the normal operating temperature and pressure.

RAI-3.b References:

1. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

Duke Energy Response to RAI-3.c:

It is a general characteristic of RPV closures that the stud preload required to resist internal pressure are substantially greater than seismic loadings. As an example, the total preload force applied by the studs for a typical RPV closure is on the order of 50,000,000 pounds; the design preload is generally set to a few percent greater than the blowoff force caused by design pressure acting to the o-ring sealing radius. In contrast, the weight of the reactor closure head is on the order of 500,000 pounds, a factor of 100 less than the preload force. Therefore, a bounding vertical seismic acceleration of 5g would result in a negligible uplift force relative to the stud force holding the head down. Likewise, a bounding horizontal acceleration of 5g would result in a lateral force of 2,500,000 lbs, which is substantially lower than the force required to overcome static friction; assuming a lower bound coefficient of static friction equal to 0.2, a force of 10,000,000 lbs (i.e., 0.2 times the preload force) would be required to overcome it. Other accident loadings, such as LOCA events, tend to reduce the internal pressure and would therefore not change stud loads beyond the transients already considered in the analysis.

RAI-4

Issue

Section 4.2.1 of EPRI report 14589 discusses the methodology for determining the limiting flaw size in the reactor vessel head closure studs. EPRI stated that a safety factor of 2.0 was applied on the primary loads based on the methods in nonmandatory Appendix G, Paragraphs G-2215 and G-2222 of the ASME Code, Section XI. EPRI cited a 2017 NRC safety evaluation (ADAMS Accession No. ML17006A109) that authorized a plant-specific alternative examination request for the reactor vessel threads-in-flange. The reactor vessel threads-in-flange are the components into which the reactor vessel head closure studs are threaded. EPRI stated that the use of the methods of Appendix G of ASME Code, Section XI, is consistent with the NRC position in the 2017 NRC safety evaluation regarding the plant-specific reactor vessel threads-in-flange.

The EPRI report noted that the methods in Paragraphs G-2215 and G-2222 of Appendix G of ASME Code, Section XI, are for vessel components. The report also noted that for bolting materials (i.e., the reactor vessel head closure studs), the recommended methods for evaluating fracture prevention are in Article G-4000 of ASME Code, Section XI, which refers to Welding Research Council Bulletin (WRCB) 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials." EPRI stated that the evaluation methods in WRCB 175 are used primarily to define toughness criteria for bolts with a reference flaw size, and not to evaluate flaws with defined structural (i.e., safety) factors. EPRI stated that WRCB 175 is considerably older than other references cited for fracture mechanics evaluations in bolted joints and none of the solutions discussed in it regarding bolting are specific for bolted joints. Thus, EPRI used the safety factors in Paragraphs G-2215 and G-2222 of Appendix G of ASME Code, Section XI, for vessels, with the SIFs described in Section 4.1 of EPRI report 14589, to define the limiting flaw size for the postulated flaws in the reactor vessel head closure studs.

The NRC staff confirmed the information in Appendix G of ASME Code, Section XI and WRCB 175 that EPRI cited, as discussed above. Even though EPRI acknowledged that the safety factor of 2.0 in Appendix G of ASME Code, Section XI is for vessels, the NRC staff assessed the basis for the safety factor. Chapter 30 of the Companion Guide to the ASME Code (Volume 2) explains that a safety of factor of 2.0 applied on the SIF due to the pressure loading (i.e., primary load) combined with a conservative postulated flaw size (i.e., a depth of one-quarter of the vessel thickness) ensures a safety factor of 3 on primary load that is consistent with the ASME Code, Section III design stress limits on vessels. Given the basis of the safety factor of 2.0 and that the postulated flaw sizes in the reactor vessel head closure studs assumed in EPRI report 14589 are relatively small, the staff is not clear whether applying a safety factor of 2.0 for the reactor vessel head closure studs achieves the same level of margin as the corresponding concept in vessels with regard to the ASME Code, Section III, design stress limits on the reactor vessel head closure studs. The staff accepted the use of a safety factor of 2.0 for the plant-specific reactor vessel threads-in-flange evaluated in the 2017 NRC safety evaluation because the postulated flaw sizes in the reactor vessel threads-in-flange were large (full 360-degree flaw and relatively deep), and therefore provided adequate margin for the reactor vessel threads-in-flange analyses.

Request

Explain how a safety factor of 2.0 applied on the SIF due to primary loads on a postulated semi-circular flaw in the reactor vessel head closure stud ensures an adequate safety factor on primary load that is consistent with the ASME Code, Section III design stress limits on the reactor vessel head closure stud.

Duke Energy Response to RAI-4:

The limiting flaw size for the BWR case will be considered in this response since it is the limiting case for the crack growth calculation. The crack growth for the PWR case is substantially slower, such that the limiting PWR flaw size is not reached even after 80 years of growth.

The limiting flaw size computed in EPRI report 14589 (Reference 1) for the BWR case is 0.789 inch; this value is calculated using the safety factors in Paragraphs G-2215 and G-2222 of Appendix G of ASME Code, Section XI. As described in EPRI report 14589, the flaw case considered is an edge flaw in the bolted connection. A bolt cross section (minor diameter) of 5.84 inches is used in the EPRI model; it is noted that the Brunswick BWR studs have a larger threaded cross section and are therefore bounded by this calculation. Conservatively assuming that the crack front is straight across the stud cross section at the flaw depth, a flaw depth of 0.789 inch has a flawed cross section of 2.16 in^2 and a remaining ligament cross section of 24.63 in^2.

The ASME Code Section III primary design stress requirements for the RPV closure flange studs are provided in Section III Appendix E. For closures that use self-energizing gaskets (like the RPV closure gaskets), the minimum bolt cross section area is effectively defined in Appendix E as the hydrostatic load for the design pressure to the outer o-ring divided by the allowable bolt stress at design temperature.

A minimum bolt cross section area of 21.25 in^2 is calculated using the Brunswick parameters for design pressure, outer o-ring diameter, number of studs in the closure flange, and allowable bolt stress, which is about 15% less than the remaining cross section at maximum flaw size. Thus, the Section III design margins for the bolting material are maintained even with all studs cracked to the limiting flaw size. Assuming that all studs are cracked to the maximum allowable flaw depth permitted is a significant conservatism since: 1) not all studs would be expected to be cracked to the maximum flaw depth, and 2) the maximum flaw depth occurs at a time that is greater than the proposed inspection interval.

RAI-4 References:

1. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

RAI-5

Issue

Section 4 of EPRI report 14589 states a K_{IC} value of 190 ksi√in based on Charpy impact testing and fracture toughness data of SA-540 steels used for reactor vessel closure studs reported in a 1977 paper in the Journal of Pressure Vessel Technology (JPVT). Figures 6 and 7 of the 1977 JPVT paper show the Charpy impact (and lateral expansion) testing and fracture toughness data, respectively. The staff needs clarification on the information in Figure 7 of the 1977 JPVT paper, more information on whether a K_{IC} value of 190 ksi√in is appropriate for the flaw tolerance evaluation, and confirmation of Charpy impact values of the reactor vessel head closure studs of the subject Duke Energy units.

a. The staff noted that Figure 7 of the 1977 JPVT paper has two y-axes: fracture toughness and impact energy. The staff understands that the Charpy impact curves in Figure 7 of the 1977 JPVT paper represent the range of impact energies from Figure 6 of the 1977 JPVT paper. However, because the y-axes of fracture toughness and impact energy are presented side-by-side, it appears that the figure shows the correspondence of fracture toughness and impact energy. For example, the upper-shelf impact energy value at room temperature (70°F) from the lower Charpy impact curve in Figure 7 of the 1977 JPVT paper is about 40 ft-lb, which appears to correspond to a fracture toughness value of 100 ksi√in.

b. The staff verified from Figure 7 of the 1977 JPVT the K_{IC} value of 190 ksi√in selected for analysis. It is the minimum of the range of upper-shelf K_{IC} values at room temperature, as stated in the 1977 JPVT and shown in Figure 7 of the paper. However, since pressure loading occurs at temperatures higher than room temperature and the upper-shelf K_{IC} data in Figure 7 of the 1977 JPVT show that upper-shelf K_{IC} values decrease with increasing temperature, the selection of an upper-shelf K_{IC} value at room temperature may be nonconservative.

c. Figure 7 of the 1977 JPVT paper shows a lower Charpy impact property curve for the SA-540 steel heats used to generate the curve. NB-2333 of Section III of the ASME Code specifies requirements for Charpy impact property values for bolting materials. Because of this requirement, Charpy impact values of the reactor vessel head closure studs of the subject Duke Energy units should be available, and thus may be compared to the lower Charpy impact property curve in Figure 7 of the 1977 JPVT paper.

Request

a. Clarify whether there is correspondence between fracture toughness and impact energy in Figure 7 of the 1977 JPVT paper and if there is correspondence, justify the selection of the high fracture toughness value of 190 ksi√in for the flaw tolerance evaluation.

b. Re-perform the flaw tolerance evaluation using a minimum upper-shelf K_{IC} value at temperatures corresponding to full pressure of PWRs and BWRs or justify the selection of an upper-shelf K_{IC} value of 190 ksi√in at room temperature for the flaw tolerance evaluation, given that pressure loading occurs at temperatures higher than room temperature and upper-shelf K_{IC} values decrease with increasing temperature.

c. Confirm that the available Charpy impact values of the reactor vessel head closure studs of each of the subject Duke Energy units are above the lower Charpy impact property curve shown in Figure 7 of the 1977 JPVT paper.

Duke Energy Response to RAI-5.a:

The first full paragraph on page 422 of the JPVT paper (Reference 1) states the following (emphasis added):

> The static and dynamic fracture toughness properties for these steels (Figs. 7, 8 and 9) exhibit a transitional behavior similar to the Charpy impact properties. ***The scatter band of Charpy properties is shown in these figures for comparison. The transition in fracture toughness and impact properties seem to occur in about the same temperature range.*** The static fracture toughness properties (Fig. 7) begin to develop an upper shelf at temperatures around -25°F (-32°C) to -50°F (-46°C). ***The upper shelf fracture maximum toughness values range from 190 ksi√in (209 MPa√m) to 240 ksi√in (264 MPa√m) at room temperature***.

Based on the "for comparison" statement and on the sentence that follows, it appears that the authors were trying to demonstrate that the transition in fracture behavior for the fracture toughness measurements and the Charpy impact measurements occurred at about the same temperature range. The paper does not establish such a correspondence between the two sets of axes as presented in Figure 7. Furthermore, the authors conclude that the upper shelf behavior results in fracture toughness values ranging from 190 ksi√in to 240 ksi√in at room temperature.

RAI-5.a References:

1. Seeley, R.R. et al., "Fracture Toughness Properties of SA-540 Steels for Nuclear Bolting Applications," Journal of Pressure Vessel Technology, August 1977.

Duke Energy Response to RAI-5.b:

As noted in the response to RAI-5.a, the JPVT paper authors state that an upper shelf toughness range of 190 ksi√in to 240 ksi√in is appropriate for SA-540 bolting steels. The lower bound of this range was selected as a suitably low value for the upper-shelf K_{IC} for comparison. In addition, the Charpy test data did not show any decrease in absorbed energy at temperatures above room temperature.

Duke Energy Response to RAI-5.c:

Duke Energy confirms that all the available Charpy impact values of the reactor vessel head closure studs of each of the subject Duke Energy units are above the lower Charpy impact property curve shown in Figure 7 of the 1977 JPVT paper (Reference 1).

RAI-5.c References:

1. Seeley, R.R. et al., "Fracture Toughness Properties of SA-540 Steels for Nuclear Bolting Applications," Journal of Pressure Vessel Technology, August 1977.

RAI-6

Issue

In Section 4.3.2 of EPRI report 14589, EPRI performed generic fatigue crack growth (FCG) calculations for the reactor vessel head closure studs and stated that the FCG rate was from Nonmandatory Appendix A, Subarticle A-4300 of ASME Code, Section XI. The reactor vessel head closure studs of the subject Duke Energy units are made of high strength bolting materials specified as SA-540 Grade B23 or B24 (yield strength up to 150 ksi or ultimate strength up to 170 ksi), as the licensee stated in Attachments 2 through 8 to the submittal. The NRC noted that, as stated in Subarticle A-1100 of ASME Code, Section XI, the scope of Appendix A applies to ferritic materials 4 inches and greater in thickness with specific minimum yield strengths of 50.0 ksi or less, which implies that the A-4300 FCG rate applies only to vessel materials with specific minimum yield strengths of 50.0 ksi or less.

The staff consulted the Companion Guide to the ASME Code (Volume 2) for additional guidance on the materials under the scope of Appendix A of ASME Code, Section XI. Section 32.1.5 of the guide states that the majority of the A-4300 reference FCG rate were from SA-508 and SA-533 materials, which are common steels used for vessels and have minimum specified yield strengths of 50 ksi or less. The guide also cites Barsom[1] that explained that ferritic steels having a range of yield strengths from 45 ksi to 300 ksi showed similar crack growth behavior. The staff noted that this large range of yield strength includes the yield strength of SA-540 Grade B23 or B24 used for reactor vessel head closure studs. However, the staff compared the FCG rate cited by Barsom with the A-4300 FCG rates in Figure A-4300-1 of the ASME Code, Section XI and noted that the Barsom FCG rate could be higher than the A-4300 FCG rates.

Given the discussion above, the staff noted that the A-4300 FCG rate may not be appropriate for the high strength bolting materials of the reactor vessel head closure studs, specified as SA-540 Grade B23 or B24, of the subject Duke Energy units. The staff is also not clear whether the FCG analysis described in Section 4.3.2 of EPRI report 14589 included safety factors on the applied SIFs due to membrane and bending stresses as the guidance in C-7000 of the ASME Code, Section XI specifies for evaluations using linear elastic fracture mechanics. The staff notes that the evaluation procedures for FCG in Appendix A of the ASME Code, Section XI, cited in Section 4.3.2 of EPRI report 14589 are typically for the reactor vessel and that the evaluation procedures in C-7000 of the ASME Code, Section XI, that specify safety factors for the applied SIFs due to membrane and bending stresses would be more applicable for the reactor vessel head closure studs.

The staff further noted that the licensee is using only deterministic analysis (versus probabilistic analysis) in EPRI report 14589 as basis for eliminating the required ASME Code, Section XI volumetric or surface examination for the reactor vessel head closure studs of the subject Duke units. The A-4300 FCG rates in Figure A-4300-1 of the ASME Code, Section XI used in the deterministic analysis in EPRI report 14589 are based on the median of the data (specifically, 95 percent confidence that the A-4300 FCG rate bounds the median of the data) used to establish those rates. Thus, the A-4300 FCG rates used in EPRI report 14589 are not upper bound rates, and therefore may not be conservative.

[1] Barsom, J. M., “Fatigue Crack Growth Propagation in Steels of Various Yield Strengths,” Trans. of ASME, Journal of Engineering for Industry, Series B, Vol. 93, No. 4, pp. 1190-1196, Nov. 1971.

Request

a. Provide alternate FCG calculations using an FCG rate applicable to high strength bolting materials appropriate for the reactor vessel head closure studs (address using an upper bound curve for the alternate FCG rate, similar to RAI-6c), specified as SA-540 Grade B23 or B24, of the subject Duke Energy units, or justify that the FCG rate in A-4300 of the ASME Code, Section XI, is adequate for the reactor vessel head closure studs specified as SA-540 Grade B23 or B24.

b. Clarify whether appropriate safety factors were applied on the membrane and bending stresses used for calculating the applied SIFs for the reactor vessel head closure studs. If safety factors were not applied, either recalculate the FCG calculations with the appropriate safety factors on the membrane and bending stresses used for the applied SIFs or justify not using safety factors on membrane and bending stresses used for the applied SIFs.

c. Either recalculate the FCG calculations based on upper bound FCG rates or justify how the use of median-based FCG rates (i.e., the FCG rate in A-4300 of the ASME Code, Section XI) provides reasonable assurance of structural integrity of the reactor vessel head closure studs of the subject Duke Energy units without periodic performance monitoring of the studs.

Duke Energy Response to RAI-6.a:

Review of FCG data from other literature sources is summarized as follows:

- Data specific to SA-540 Grade B23/B24 bolting material was not identified in a literature review.
- The Barsom paper (Reference 1) cited in the Companion Guide to the ASME Code, Chapter 32 (Reference 2) was reviewed. This paper compiles measured FCG rate data for two categories of carbon/low-alloy steels: ferritic-pearlitic and martensitic; the SA-540 Grade B23/B24 bolting material is a quenched and tempered martensitic material. Barsom concludes that a conservative estimate of da/dN for martensitic steels is obtained using the relationship $da/dN = 0.66\times10^{-8}\times\Delta K^{2.25}$. This model bounds the data for all martensitic steels tested at a broad range of yield strengths.
- The Barsom model FCG rate for martensitic steel is only modestly different from the A-4300 FCG rate. The Barsom model FCG rate is greater than the A-4300 FCG rate for low values of ΔK, when predicted crack growth is on order of 1×10^{-7} in./cycle or less. At higher predicted FCG rates (i.e., 1×10^{-6} in./cycle and higher), the A-4300 FCG rate and the Barsom model FCG are in much closer agreement.
- A broader review of fatigue crack growth compendia and texts, including the ASM Handbook volume on fatigue and fracture (Reference 3) and Fuchs and Stevens (Reference 4), confirms that yield strength plays a minor role in FCG rate of carbon and low-alloy steels. The ASM Handbook states that at high strengths (130 to 220 ksi), there is no significant variation in FCG rate. Fuchs and Stevens references the Barsom study on FCG rates (Reference 1), and states that the scatter band for growth about a given ΔK varies by a factor of about 2 when the broad range of yield strengths investigated (36 to 191 ksi) are considered.

It is also noted that the analysis in the EPRI report 14589 (Reference 5) applies substantial conservatisms within the crack growth calculations to ensure a conservative result, as follows:

- As noted in EPRI report 14589 Section 4.3.1, the largest fatigue cycle maximum to minimum range was used to bound the full set of operating ranges that occur during plant operation. This maximum range was applied at 1,000 cycles per year to bound all transients. Additional investigation of this conservatism is presented in the response to RAI-6.c.
- As noted in EPRI report 14589 Section 4.3.2, an initial flaw size of 0.3 inch was selected, which is much larger than the minimum detectable flaw size of 0.157 inch.

Therefore, given the FCG rate data observations summarized above and the conservatisms applied in the EPRI report, it is concluded that the A-4300 fatigue crack growth calculations performed are adequate for the reactor vessel closure studs.

<u>RAI-6.a References:</u>

1. Barsom, J. M., “Fatigue Crack Growth Propagation in Steels of Various Yield Strengths,” Trans. ASME, Journal of Engineering for Industry, Series B, Vol. 93, No. 4, pp. 1190–1196, Nov. 1971.
2. Companion Guide to the ASME Boiler & Pressure Vessel Code: Criteria and Commentary on Select Aspects of the Boiler & Pressure Vessel and Piping Codes Volume 2: Fifth Edition, Chapter 32 “Fatigue Crack Growth, Fatigue, and Stress Corrosion Crack Growth: Section XI Evaluation,” ASME, 2018.
3. ASM Handbook, Volume 19 – Fatigue and Fracture, p. 636 “Fracture Mechanics Properties of Carbon and Alloy Steels – Effects of Microstructure and Heat Treatment,” ASM International, 1996.
4. Metal Fatigue in Engineering, Fuchs, H., and Stephens, R., John Wiley and Sons, 1980.
5. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

<u>Duke Energy Response to RAI-6.b:</u>

As described in the response to RAI-4, appropriate safety factors are applied to the calculated crack tip SIF values when establishing the limiting flaw size. In accordance with the standard approach to flaw evaluation of ASME Section XI, safety factors are not applied to the loads when calculating the time for subcritical crack growth.

Duke Energy Response to RAI-6.c:

As described in the response to RAI-6a, substantial conservatisms were applied in the FCG calculations that provide reasonable assurance against uncertainties related to FCG rates. In particular, bounding values for cycle counts were applied to the largest operating transient ranges.

The following additional cases investigate the relative conservatisms for the FCG rate and the transient cycles and applied cycle counts. Applying the existing FCG model as defined in EPRI report 14589 and assuming a FCG rate that is three times the A-4300 calculated value, the following results are obtained for the cases described.

- For PWR studs, applying 50 cycles/year to the Heatup / Loss of Flow combination to bound the lower cycle count transients, then adding 1,000 cycles/year to the maximum range from the normal operating transients (Plant Loading / Unloading, Step Increase / Decrease, and Steady State Fluctuations), a smaller final flaw size at 80 years is calculated than calculated in EPRI report 14589 (Reference 1).
- For BWR studs, applying 50 cycles/year to the Preload / Cooldown combination to bound the lower cycle count transients, then adding 1,000 cycles/year to a bounding normal operating transient range (Operation / Preop Blowdown), the limiting flaw size is reached at 68.7 years, versus the 37.9 years calculated in EPRI report 14589.

It is further noted that the crack growth results for these cases bound the results obtained using the bounding Barsom FCG rate model for martensitic steel in Reference [2]. Therefore, it is concluded that the overall calculations performed in EPRI report 14589 provide reasonable assurance of structural integrity of the reactor vessel head closure studs of the subject Duke Energy units without periodic performance monitoring of the studs.

RAI-6.c References:

1. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.
2. Barsom, J. M., “Fatigue Crack Growth Propagation in Steels of Various Yield Strengths,” Trans. ASME, Journal of Engineering for Industry, Series B, Vol. 93, No. 4, pp. 1190–1196, Nov. 1971.

RAI-7

Issue

In Section 5.0 of Enclosure 1 of the submittal, the licensee stated that a review of Duke Energy's past ISI examination records for reactor vessel head closure studs indicates there have been no occurrences of service-induced degradation. The staff is not clear whether "no occurrences of service-induced degradation" means no relevant indications were detected in the reactor vessel head closure studs during the ASME Code, Section XI ISI examinations.

Request

For each of the Duke Energy units included in the request:

a. Clarify clear whether "no occurrences of service-induced degradation" means no relevant indications were detected in the reactor vessel closure studs during the ASME Code, Section XI ISI examinations

b. Depending on the response in RAI-7a, if relevant indications were detected, explain how the indication was dispositioned and how the size (depth and length) of the indication impacts the postulated semi-circular flaw with an initial flaw depth of 0.3 inch analyzed in the FCG calculation in Section 4.3.2 of EPRI report 14589.

Duke Energy Response to RAI-7.a:

Duke Energy confirms that no relevant indications were detected in reactor vessel closure studs of the subject Duke Energy units during ASME Code, Section XI ISI examinations. The phrase "no occurrences of service-induced degradation" used in Section 5.0, Paragraph 3 of Duke Energy's Alternative submittal (Reference 1) was meant to convey no relevant indications were detected in the reactor vessel closure studs during the ASME Code, Section XI ISI examinations.

RAI-7.a References:

1. Duke Energy Carolinas, LLC & Duke Energy Progress, LLC, "Relief Request for Alternative for Reactor Vessel Closure Stud Examinations", dated December 1, 2020 (ADAMS Accession Number ML20336A033). Relief Request Number RA-19-0352.

Duke Energy Response to RAI-7.b:

No relevant indications were detected in reactor vessel closure studs of the subject Duke Energy units during ASME Code, Section XI ISI examinations. Therefore, no further response is required for RAI-7.b.

RAI-8

Issue

In Attachments 2 to 8 of the submittal, the licensee stated that the Reactor Vessel Closure Stud program of each subject Duke Energy unit includes preventive measures to use stable lubricants, and specifically, to prohibit the use of molybdenum disulfide. The staff noted that this preventive measure is to mitigate the effects of stress corrosion cracking in the reactor vessel closure studs, which is discussed in Section 2.1.1 of EPRI report 14589. For BSEP Units 1 and 2, and SHNPP Unit 1, the staff verified that such a program is included in the updated final safety report (UFSAR) for the units. However, for Catawba, Units 1 and 2, and McGuire, Units 1 and 2, the staff was not able to verify which program listed in Chapter 18, “Aging Management Programs and Activities” of the corresponding UFSAR for the units includes the preventive measure that prohibits the use of molybdenum disulfide for the reactor vessel closure studs of the units.

Request

For Catawba, and McGuire, state which program listed in Chapter 18, “Aging Management Programs and Activities” of the corresponding UFSAR for the units includes the preventive measure that prohibits the use of molybdenum disulfide for the reactor vessel closure studs of the units.

Duke Energy Response to RAI-8:

The Catawba and McGuire application (Reference 1) predate NUREG-1801 (Reference 2) and does not incorporate license renewal aging management programs specific to the Reactor Vessel Closure Studs. Instead, the aging management effects associated with reactor vessel closure studs at Catawba and McGuire are managed by the ISI Program and the Reactor Coolant System (RCS) Operational Leakage Monitoring Program Aging Management Programs (AMPs) as listed in Table 3.1-1 of the license renewal application (Reference 1). This approach is consistent with the licensing basis and found to be acceptable by the NRC in the combined license renewal safety evaluation report for Catawba and McGuire. (Reference 3).

Preventive measures within Catawba and McGuire site maintenance procedures compliant with Regulatory Guide 1.65, Revision 1 (Reference 4) restrict the use of lubricants with deliberately added halogens, sulfur, or lead, including molybdenum disulfide. Catawba and McGuire are committed to Regulatory Guide 1.65, Revision 1 within each site’s UFSAR Chapter 1.

RAI-8 References:

1. Letter from M.S. Tuckman to USNRC, Application to Renew the Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2, dated June 13, 2001 (ADAMS Accession No. ML011660138).
2. U.S. NRC, Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Initial Report, July 2001.
3. NUREG-1772, Safety Evaluation Report Related to the License Renewal of the McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2 (ADAMS Accession No. ML030850251).

4. U.S. NRC, Materials and Inspections for Reactor Vessel Closure Studs, Regulatory Guide 1.65, Rev. 1, April 2010.

RAI-9

Issue

The staff noted that in Section 5.0 of the submittal that for Catawba, Units 1 and 2, McGuire, Units 1 and 2, and Harris, the lengths of the proposed extension of the ISI intervals for the reactor vessel closure studs of these units are more than 20 years (i.e., more than two consecutive 10-year ISI intervals). The staff notes that two consecutive 10-year ISI intervals were determined acceptable for the threads-in-flange in previous requests. Eliminating the volumetric examinations for the reactor vessel closure studs of these units during the proposed extensions eliminates the most effective method for detecting new degradation or changes in degradation *within* the reactor vessel closure studs of the units, and thereby significantly reducing condition monitoring of the reactor vessel closure studs. In Section 5.0 of the submittal, the licensee stated that the detailed procedures used during each refueling outage for the removal, care, and visual inspection of the reactor vessel closure studs and threads-in-flange provide further assurance that degradation is detected.

The staff noted that with the proposed elimination of the volumetric examination of the reactor vessel closure studs, these periodic maintenance procedures would be the only component-specific condition monitoring for the reactor vessel closure studs. It is not clear to the staff how these periodic maintenance procedures would be effective in detecting new degradation or a change in degradation within the reactor vessel closure studs because it does not examine the critical volume around the threads of the reactor vessel closure studs. Additionally, with the volumetric examination eliminated, if there is new degradation or a change in degradation, this would constitute and unanalyzed degradation and thus would not be included in the flaw tolerance analyses in EPRI report 14589. The licensee also stated that the periodic maintenance procedures, coupled with the ASME Code Section XI leak test (Examination Category B-P), provide assurance of pressure boundary integrity. However, the staff noted that the ASME Code Section XI, Examination Category B-P leak test is not component-specific and does not examine the critical volume around the threads of the reactor vessel closure studs.

Request

Given (1) the insufficiency of using only deterministic FCG analyses to justify elimination of volumetric examination longer than 20 years; and (2) the periodic maintenance procedures performed each refueling outage would be the only condition monitoring for the reactor vessel closure studs of Catawba, McGuire, and Harris:

a. Justify how the periodic maintenance procedures (which do not examine the critical volume around the threads of the reactor vessel closure studs) would detect new degradation or changes in degradation within the reactor vessel closure studs of these units (Catawba, Units 1 and 2, McGuire, Units 1 and 2, and Harris) for periods of longer than 20 years.

b. Explain whether these periodic maintenance procedures would be supplemented with other component-specific performance monitoring measures, such as volumetric examination of a sample of the reactor vessel closure studs of the subject Duke units, such that the critical volume around the threads of the reactor vessel closure studs is examined to ensure that new degradation or a change in degradation is detected. If not, justify how not supplementing the periodic procedures would ensure that the critical volume around the threads of the reactor vessel closure studs is examined to ensure that new degradation or a change in degradation is detected.

Duke Energy Response to RAI-9.a:

Controlled maintenance procedures are used each refueling outage to perform reactor pressure vessel (RPV) head tensioning and detensioning using tensioners. Each tensioner is threaded onto the upper part of the RPV stud and interfaces with the castellation on the top of the nut. The tensioner is hydraulically actuated to pull upward to extend the stud while pressing downward on the RPV head flange. The tensioner pressure is controlled and is set to the target value per the relevant plant maintenance procedure. With the RPV stud elongated, the nut is tightened onto the head flange, and the tensioner load is removed. The retained stud load (stud preload) is lower than the applied tensioner load because of compliance of the joint after the tensioner is removed; this effect is accounted for in the stud tensioning procedures. Therefore, the tensioning process results in significantly higher stresses in Pressurized Water Reactor (PWR) studs compared to operating conditions or transients; thus, the tensioning process for Catawba, McGuire, and Harris acts like a proof test for RPV studs, providing an additional measure of confidence in the integrity of the tensioned studs.

Additionally, a RPV stud with limiting flaw size beyond the bounding analysis of the EPRI report 14589 (Reference 1) would likely reveal itself during the proof test by not meeting procedure acceptance criteria for measured elongation. Elongation measurements are taken with the same digital micrometers before and after tensioning to determine the amount of stretch applied to each RPV stud. These measured elongations are verified by Quality Control (QC) examiners and documented in the maintenance procedure to ensure accuracy of these critical measurements. Therefore, an RPV stud not meeting measured elongation acceptance criteria would be identified by QC and/or maintenance during installation and would require further evaluation and/or replacement.

The combination of the controlled maintenance procedures acting as a proof test for the RPV studs and verified measured elongation readings for each RPV stud every refueling outage ensures confidence in the integrity of the tensioned studs for periods longer than 20 years.

RAI-9.a References:

1. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

Duke Energy Response to RAI-9.b:

Site-specific maintenance procedures and practices related to RPV studs are compliant with NRC Regulatory Guide 1.65 (Reference 1) and reflect lessons learned from industry operating experience. These maintenance procedures have been revised to ensure potential degradation mechanisms associated with RPV studs are mitigated. Below is a high-level description of how the use of maintenance procedures at Catawba, McGuire, and Harris mitigate these degradation mechanisms.

Degradation Mechanisms and Maintenance Practices

The design of the reactor vessel closure studs at Catawba, McGuire, and Harris allows them to be completely removed during each refueling outage permitting visual inspection of the RPV stud and the threads in flange to assess protection against degradation. Refueling procedures require that each stud be removed, visually inspected, and placed in a stud rack. After the studs are removed, the stud holes in the vessel flange are sealed with a special plug. The studs are lifted and moved to a storage area prior to the water level being raised in the refueling cavity. Thus, the bolting materials and stud holes are not exposed to the borated refueling cavity water. These procedural steps mitigate exposing the studs to chlorides and potential degradation mechanisms during refueling activities. Additional protection against the possibility of incurring corrosion effects is assured by using a manganese base phosphate surfacing treatment applied to each reactor vessel closure stud for Catawba, McGuire, and Harris.

These activities are performed during each refueling outage and each step is documented per plant procedures. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service.

The generic aging management program for RPV studs in the GALL report (Reference 2) lists four preventative actions that can effectively reduce the potential for Stress Corrosion Cracking (SCC):

- Avoiding studs that are metal plated to reduce the potential for seizing. The metal plating can lead to hydrogen embrittlement or galvanic corrosion at discontinuities when wetted.
- Applying manganese phosphate or other acceptable surface treatments.
- Avoiding the use of molybdenum disulfide as a lubricant, and instead using lubricants that remain stable at operating temperatures.
- Using material with an actual yield strength confirmed by measurement to be less than 150 ksi (newly installed studs) or an ultimate strength of less than or equal to 170 ksi (existing studs)[2].

Duke Energy Nuclear Plants Catawba, McGuire, and Harris satisfy all four bullets above using site-specific periodic maintenance procedures and supply chain controls on procurement of new RPV studs compliant with NRC Regulatory Guide 1.65.

Quantitative Assessment

The EPRI report 14589 (Reference 4) postulated growth of an initial flaw size of 0.05 times the 6.0-inch bolt diameter, or 0.30 inch. This flaw size is consistent with the reference flaw depth recommended in Welding Research Council Bulletin (WRCB) 175 (Reference 5) for bolts greater than 3 inches in diameter, and it is substantially larger than the 0.157-inch minimum detectable flaw size for inspection. The postulated flaw (0.30 inch) in a PWR RPV stud reaches 0.445 inches after 80 years of operation, which is less than the maximum allowable size of 1.06 inches consistent with ASME Code, Section XI, Nonmandatory Appendix G (Reference 6). This demonstrates significant margin for the alternative interval schedule for Catawba, McGuire, and Harris. In addition, the 80-year interval assumes a postulated flaw, however the most recently performed ISI ultrasonic (UT) exams of the Catawba, McGuire, and Harris revealed no relevant indications in any of the RPV studs. Therefore, additional margin beyond 80-years exists, since the RPV studs are free of initiating defects.

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[2] GALL-SLR Section XI.M3 (Reference 3), "Reactor Head Closure Stud Bolting" allows bolting material for closure studs that has an actual measured yield strength of less than 150 ksi for newly installed studs, or 170 ksi ultimate tensile strength for existing studs.

Length of Relief Requested

The maximum length of relief requested as summarized in Table 2 of the Duke Energy Alternative Submittal (Reference 7) is 37-years based on RPV Studs (1-19) being examined in 2009 at Harris. However, the remaining RPV Studs (39-58) were examined more recently in 2016. Therefore, the length of time for those RPV Studs (39-58) would only be 30-years. The length of relief requested for Catawba and McGuire are shown in Table 2 of Reference 7 as 28-years for Catawba Units 1&2 & McGuire Unit 1 and 27-years for McGuire Unit 2. Examination requirements for RPV Studs per ASME Section XI (Reference 6) are deferrable to the end of the interval per IWB-2411(a)(4) and Table IWB-2500-1, Examination Category B-G-1, Item Number B6.20. Therefore, in compliance with current ASME Code requirements of IWA-2430(c)(1) and IWA-2431 the maximum possible time between RPV stud examinations between successive intervals is 21-years. This Duke Energy Alternative Relief Request is only extending the code allowed time between exams a maximum of 16-years for RPV studs (1-19) and a minimum of 9-years for RPV studs (39-58) at Harris. The extension for the remaining Harris RPV Studs (20-38) beyond the code allowed time would be 15-years. Catawba 1&2 and McGuire 1 is a 7-year extension, while McGuire 2 is only 6-years beyond the current code schedule requirements. The EPRI report 14589 calculated a postulated flaw in a PWR RPV stud reaches 0.445 inches after 80-years of operation, which is less than the maximum allowable size of 1.06 inches.

Operating Experience and Corrective Action Program

RPV stud exams in accordance with ASME Section XI requirements continue to be performed at nuclear plants within the Duke Energy Fleet (H.B. Robinson Unit 2 and Oconee Units 1, 2, & 3) and within the Industry. Any relevant indications or new degradation mechanisms identified during those volumetric examinations of the RPV studs would be entered into the Duke Energy Corrective Action Program as required by the applicable administrative procedures. This operating experience would be evaluated and extent of condition examinations (if required per the evaluation) would be performed at Catawba, McGuire, and Harris.

Additionally, industry operating experience associated with RPV studs is discussed extensively within the EPRI report 14589 Section 2.2. The conclusion indicated that preventative measures are addressing relevant degradation mechanisms, which is consistent with the view supported by the Proactive Materials Degradation expert panel (Reference 8).

Conclusion

Existing site-specific periodic maintenance procedures in conjunction with extensive operating experience, bounding quantitative assessments with significant margin against the length of relief, and on-going RPV stud volumetric (UT) exams at other Duke Energy Sites (H.B. Robinson Unit 2 and Oconee Units 1, 2, & 3) provide adequate assurance for detection of new or existing degradation mechanisms associated with RPV studs including the critical volume around the threads of the reactor vessel closure studs for periods longer than 20 years.

RAI-9.b References:

1. U.S. NRC, Materials and Inspections for Reactor Vessel Closure Studs, Regulatory Guide 1.65, Rev. 1, April 2010.
2. U.S. NRC, Generic Aging Lessons Learned (GALL) Report, NUREG-1801, Rev. 2, December 2010.
3. U.S. NRC, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report, NUREG-2191, July 2017.
4. Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.
5. Welding Research Council Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," August 1972.
6. ASME Boiler and Pressure Vessel Code, Section XI, 2007 Edition through 2008 Addenda, American Society of Mechanical Engineers, New York.
7. Duke Energy Carolinas, LLC, Duke Energy Progress, LLC, "Relief Request for Alternative for Reactor Vessel Closure Stud Examinations", dated December 1, 2020. ADAMS Accession No. ML20336A033.
8. Expert Panel Report on Proactive Materials Degradation Assessment, NUREG/CR-6923, February 2007.

Enclosure 2

EPRI report 3002014589, “Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs”

ELECTRIC POWER RESEARCH INSTITUTE

Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs

2018 TECHNICAL REPORT

Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs

3002014589

Final Report, November 2018

EPRI Project Managers
R. Grizzi
D. Kull

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

YES NO

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

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Dominion Engineering, Inc.
12100 Sunrise Valley Dr., Ste. 220
Reston, VA 20191

Principal Investigators
J. Broussard
K. Fuhr
T. Ligon
A. Simon

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Technical Basis for Optimization of the Volumetric Examination Frequency for Reactor Vessel Studs. EPRI, Palo Alto, CA: 2018. 3002014589.

ABSTRACT

Rules for periodic inspections of nuclear power plant components are provided in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Within Section XI, the Class 1 components requiring periodic inspection during each code interval are included in Table IWB-2500-1. Among these, *Examination Category B-G-1, Pressure-Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter, Reactor Vessel Item No. B6.20, "Closure Studs" requires periodic volumetric or surface examination of all reactor pressure vessel (RPV) closure studs every Inspection Interval (nominally 10 calendar years).*

This report develops a technical basis for optimizing the frequency of Item No. B6.20 examinations. The technical basis considers the primary degradation mechanisms applicable to RPV studs, including (1) fatigue, (2) stress corrosion cracking, (3) boric acid corrosion (pressurized water reactors only), and (4) steam cutting. Although the technical basis is oriented toward ASME Code Section XI requirements, the analysis approach and results have merit as a stand-alone technical position. International utilities that use different governing codes and standards for inspections should evaluate how to use the report in conjunction with those standards and regulatory obligations.

Given the operating experience to date for RPV studs, the quantitative assessments in the technical basis report focus on the potential for RPV stud degradation caused by fatigue mechanisms. The technical basis in the report for the optimization of Item No. B6.20 inspections for RPV studs includes (1) assessing typical design basis loads and transients, (2) evaluating the stresses using a finite element analysis of the reactor vessel head closure, (3) identifying and evaluating flaw stability limits, and (4) evaluating fatigue crack growth of a postulated flaw in the RPV studs. The time for the postulated flaw to propagate beyond an acceptable flaw size can be used to optimize an appropriate inspection frequency.

Keywords

ASME Boiler and Pressure Vessel Code (BPVC)
Pressure retaining bolting
Section XI
Surface examination
Volumetric examination

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EXECUTIVE SUMMARY

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

SECONDARY AUDIENCE: Technical staff for nuclear utilities and regulators

KEY RESEARCH QUESTION

This report develops a technical basis for optimizing the frequency of inspections for reactor pressure vessel (RPV) closure studs.

RESEARCH OVERVIEW

The analysis methodology considers the primary degradation mechanisms applicable to RPV studs, including (1) fatigue, (2) stress corrosion cracking, (3) boric acid corrosion (pressurized water reactors [PWRs] only), and (4) steam cutting.

Based on a review of operating experience, the quantitative assessments in this technical basis report focus on the potential for RPV stud degradation caused by fatigue mechanisms. The technical basis in this report for the optimization of Item No. B6.20 inspections for RPV studs includes (1) assessing typical design basis loads and transients, (2) evaluating the stresses using a finite element analysis of the reactor vessel head closure, (3) identifying and evaluating flaw stability limits, and (4) evaluating fatigue crack growth of a postulated flaw in the RPV studs. The time for the postulated flaw to propagate beyond an acceptable flaw size can be used to optimize an appropriate inspection frequency.

KEY FINDINGS

- The fatigue crack growth for PWR RPV studs for a postulated flaw after 80 years of assumed loading cycles remains less than the allowable flaw size using methods consistent with the American Society of Mechanical Engineers (ASME) Code, Section XI, Nonmandatory Appendix G.
- The fatigue crack growth for the boiling water reactor (BWR) RPV studs for a postulated flaw after 37.9 years of assumed loading cycles remains less than the allowable flaw size using methods consistent with ASME Code, Section XI, Nonmandatory Appendix G.
- For PWR RPV studs, the stud tensioning process results in significantly higher applied crack tip stress intensity factor values than normal operating loads. Therefore, the tensioning process for PWR RPV studs is effectively a “proof test” and is therefore the limiting condition for establishing allowable flaw sizes. Because of their higher bending stresses during normal operating conditions, the tensioning process for BWR RPV studs is not the limiting condition and does not act similarly as a proof test.

WHY THIS MATTERS

Optimization of inspection intervals, based on an improved understanding of the potential degradation, provides the benefit of reducing health and safety risk of personnel, promotes as low as reasonably achievable practices, and decreases the overall burden of inspection—all without adversely impacting the safe operations of the nuclear facilities.

ELECTRIC POWER RESEARCH INSTITUTE

EXECUTIVE SUMMARY

HOW TO APPLY RESULTS

This report develops a technical basis using inputs that are designed to evaluate the applicability range of conditions experienced at operating reactors. Section 5.2 provides an approach to define the applicability of this technical basis to a given plant based on key criteria that determine whether the results of this analysis bound actual plant operation.

LEARNING AND ENGAGEMENT OPPORTUNITIES

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

EPRI CONTACTS: Robert Grizzi, Program Manager, rgrizzi@epri.com

PROGRAM: PLANT SUPPORT, NDE

IMPLEMENTATION CATEGORY: Technical Basis

Electric Power Research Institute
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

ACRONYMS

ASME	American Society of Mechanical Engineers
BAC	Boric acid corrosion
BACCP	Boric Acid Corrosion Control Program
BWR	Boiling water reactor
FEA	Finite element analysis
FSAR	Final Safety Analysis Report
GALL	Generic aging lessons learned
GE	General Electric
ID	Inside diameter
LEFM	Linear elastic fracture mechanics
NLR	Netherlands National Aerospace Laboratory
NRC	Nuclear Regulatory Commission
OE	Operating experience
OD	Outside diameter
PWR	Pressurized water reactor
RCP	Reactor coolant pump
RCS	Reactor coolant system
RG	Regulatory guide
RIC SIL	Rapid Information Communication Services Information Letter
RPV	Reactor pressure vessel
SIF	Stress intensity factor
SLR	Subsequent license renewal
SCC	Stress corrosion cracking
UT	Ultrasonic testing
WRCB	Welding Research Council Bulletin

UNIT CONVERSION FACTORS

1 inch = 2.54 cm = 25.4 mm

Temperature (°C) = [Temperature (°F) – 32°F] × 0.5556

1°F Δ = 0.5556°C Δ

1 psi = 6.895 kPa

1 ksi = 6.895 MPa

1 ksi$\sqrt{\text{in}}$ = 1.099 MPa$\sqrt{\text{m}}$

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1
INTRODUCTION

1.1 Background

Rules for periodic inspections of nuclear power plant components are provided by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Within Section XI, the Class 1 components requiring periodic inspection during each Code Interval are included in Table IWB-2500-1. Among these, Examination Category B-G-1, Pressure-Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter, Reactor Vessel Item No. B6.20, "Closure Studs" requires periodic volumetric or surface examination of all reactor pressure vessel (RPV) closure studs every Inspection Interval (nominally 10 calendar years).

EPRI recently developed a technical basis [2] that determined that the volumetric examinations of the threads in the reactor vessel flange (Section XI, Examination Category B-G-1, Item No. B6.40) could be eliminated without increasing plant risk or posing any safety concerns for the RPV. Evaluating the closure studs is a natural progression of that work.

In addition, assessments performed for boiling water reactor (BWR) and pressurized water reactor (PWR) RPV closures (e.g., [4] and [5]) have demonstrated that the vessel closure (i.e., the bolted connection between the RPV head flange and the vessel flange) is tolerant to failure of a single stud, indicating that the assembly has redundancy. These assessments demonstrate that the RPV closure remains leak-tight following failure of any single reactor vessel stud; this result is due to: 1) the self-energized O-rings used, which require a lower seating pressure to fully seal than other gasket types and 2) the stiffness of the flanges. These assessments also demonstrate that the RPV closure components meet all ASME Code, Section III, Division 1 (Class 1) requirements with a single stud out of service.

A description of the RPV closure studs and background on potential aging degradation mechanisms are provided in the following subsections.

1.1.1 Description of RPV Studs

As seen in Figure 1-1, RPV studs are characterized by an upper threaded section that interfaces with the nut, an unthreaded shank in the region of the RPV head flange, and a lower threaded section that is threaded into the vessel flange. Thread specifications are typically 8 pitch (threads per inch) with a nominal diameter about ¼ inch larger than the stud shank diameter. The top and bottom sections of the through-drilled measurement hole are tapped for threaded inserts. The bottom threaded insert is permanently captured and provides the reference surface for measuring stud elongation, and the top threaded insert is removable to protect the measurement hole from damage and debris.

RPV studs are fabricated from low alloy steel, largely AISI Type 4340. An example of efforts to avoid stress corrosion cracking (SCC) issues can be found in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.65 [13]. As originally issued, it specified as acceptable material with an ultimate tensile strength less than 170 ksi (1.17 GPa). This strength limit has since been modified to instead limit the yield strength of the bolting material to 150 ksi (1.03 GPa). RPV studs are typically specified as SA-540 Grade B23 or B24 material, and commonly specified as Class 3 (130 ksi minimum yield strength, 145 ksi minimum ultimate strength). Prior to development and acceptance of the SA-540 material specification into the ASME Code, Code Cases were used to specify and implement the same material requirements to those subsequently found in the SA-540 material specification.

1.1.1.1 Boiling Water Reactors (BWRs)

Operating BWRs have between 64 and 92 RPV studs with outer diameters (at the shank) ranging from 5.625 inches (142.9 mm) to 6.000 inches (152.4 mm); the inner diameter (measurement hole) ranges from 0.625 inch (15.9 mm) to 1.060 inches (26.9 mm). BWR studs are typically about 5.5 ft (1.68 m) long.

BWR RPV studs are typically left in place during refueling outages except for the studs in the "cattle chute" region that are removed to permit fuel movement. Stud covers are installed on the studs that are left in place to prevent corrosion from the aqueous environment while the refueling cavity is flooded.

1.1.1.2 Pressurized Water Reactors (PWRs)

Operating PWRs have between 48 and 60 RPV studs of outer diameter (at the shank) ranging from 5.748 inches (146.0 mm) to 6.979 inches (177.3 mm); the inner diameter (measurement hole) ranges from 0.750 inch (19.1 mm) to 1.125 inches (28.6 mm). PWR studs are typically about 4.5 ft (1.37 m) long.

PWR RPV studs are typically removed from the vessel flange during refueling outages, and stud hole plugs are installed to prevent corrosion of the vessel flange threads.

1.1.2 Potential Causes of RPV Stud Degradation

In the past, lubricants were used that were later determined to be incompatible with the materials and temperatures associated with RPV stud service. These issues were identified in NRC Bulletin 82-02 [1], which reported limited cases of SCC in plant bolting including steam generator manway studs. Based on the information in Bulletin 82-02, lubricants containing molybdenum disulfide were generally prohibited from use on RPV studs in the 1980s. Additionally, since 1990, there have been few volumetric indications and no confirmed cracking detected in these components as part of the Item No. B6.20 examinations.

1.2 Objective

This report develops a technical basis for optimizing the frequency of Item No. B6.20 examinations. The technical basis considers the primary degradation mechanisms applicable to RPV studs, including:

- *Fatigue*. Fatigue and fatigue crack growth formed the original purpose for Section XI inspections, as noted in Chapter 27 of the ASME Code Companion Guide [6]. A technical basis for inspection optimization should therefore consider fatigue crack growth from detectable sizes to allowable sizes.
- *Stress corrosion cracking*. Because studs subject to Examination Category B-G-1 are not normally exposed to a high-temperature aqueous environment (i.e., they remain dry during operation), the main concern for SCC is in the context of lubricants, including those applied historically.
- *Boric acid corrosion (BAC).* (Only applicable for PWRs) In the vicinity of active leakage, boric acid corrosion can occur and cause significant wastage of steel components. Evidence of such leakage is readily observed during direct visual examinations.
- *Steam cutting*. Steam cutting of the RPV flange surface or studs is an erosion corrosion mechanism that is contingent on active leakage from the RPV closure joint. Such leakage is readily detected by leak-off monitoring equipment that alerts the control room operators.

1.3 Scope

This report provides a technical basis for optimizing the frequency of volumetric or surface examinations of the RPV studs per ASME Code, Section XI, Examination Category B-G-1, Item No. B6.20, based on operating experience to-date, and flaw growth and flaw tolerance evaluations. In developing these evaluations, a review of operating experience was performed to assess degradation mechanisms that may contribute to potential flaw propagation, and stress and fracture mechanics analyses were performed to develop stress intensity factors for use in flaw growth and flaw tolerance calculations.

The scope of the analysis is limited to the RPV studs in BWRs and PWRs. Section 5.2 provides criteria for application of the technical basis provided herein for bounding BWR and PWR geometries to any specific plant. The bounding geometries were established based on best available plant information and were developed in the context of finite element analysis (FEA) perturbation studies that were used to assess which geometric parameters have the most significant effects on stud stresses. Consideration was given to bounding load cycle counts that may occur over the operating life of nuclear plants.

The technical basis may be used to revise the inspection requirements and be applied by utilities to obtain relief from the current scope and/or frequency of examinations performed in accordance with ASME Code, Section XI, Examination Category B-G-1, Item No. B6.20. While the technical basis is oriented towards ASME Code Section XI requirements, the analysis approach and results have merit as a standalone technical position. International utilities that use different governing codes and standards for inspections should evaluate how to use this report in conjunction with those standards and regulatory obligations.

1.4 Approach

Given the operating experience to-date for RPV studs, the quantitative assessments in this technical basis document focus on the potential for RPV stud degradation caused by fatigue mechanisms. It is anticipated that the concerns for degradation by SCC are addressed by procedural controls of material strength and lubricant chemistry. The technical basis also credits existing periodic visual examinations of the RPV head for the presence of leakage and, at PWRs, boric acid crystals to address the concern for boric acid wastage and steam cutting. The degradation mechanisms that are a result of leakage (boric acid corrosion and steam cutting) are also of reduced concern due to the aforementioned ability for the RPV to remain leak-tight following the failure of any single RPV stud.

To confirm the appropriateness of focusing on fatigue and fatigue crack growth as the only degradation mechanism requiring quantitative evaluation, a review of relevant operating experience for RPV studs and other large-diameter bolting components was performed. Although it is acknowledged that operating experience alone is not a sufficient basis for inspection relief, operating experience of US plants supplemented by other considerations, such as flaw tolerance evaluations, remains an important input to technical evaluations of components.

The following evaluations were performed as part of developing the technical basis for optimization of Item No. B6.20 inspections for RPV studs:

- *Assess typical design basis loads and transients*. A set of EPRI member utilities were surveyed to characterize the range of loads and operating transients applicable to the RPV studs. These inputs were used to develop bounding design basis loading scenarios and transients, including determination of design basis and representative transient cycle counts. Additionally, survey respondents were queried to provide RPV closure geometries and materials for their plants.

  Based on the results of this assessment, two bounding geometries were utilized for the evaluations contained in this report: one for BWR plants and one for PWR plants.

- *Evaluate stresses using a FEA of the reactor vessel head closure*. The stresses in the RPV studs during tensioning and operating transients were evaluated for both selected geometry cases. The results of these evaluations include the stud membrane and bending stresses, which were used to develop crack tip stress intensity factors for the subsequent evaluations.

- *Evaluate flaw stability limits*. The materials used in RPV studs are high-strength, low alloy steels specified to have substantial ductility even at low temperatures. Using Section XI linear elastic fracture mechanics (LEFM) flaw evaluation methods, flaw stability analyses were performed to demonstrate flaw tolerance for bounding cases. These evaluations utilized conservative material properties and the appropriate bounding loading conditions.

- *Evaluate postulated fatigue crack growth*. Fatigue crack growth analyses were performed to evaluate the expected growth over the component life of a postulated flaw, starting from a size where fatigue cracking for bolted joints changes from nucleation to growth. The time for the postulated flaw to propagate to an allowable flaw size was used to establish and optimize an appropriate inspection frequency.

Criteria are also included to determine the applicability of the evaluations in this report on a plant-specific basis.

1.5 Report Structure

The report is structured along the tasks described in the Scope section, as follows:

Section 1: **Introduction**
The introduction describes the overall approach and provides the scope of the document and its applicability. Background is also provided on RPV studs.

Section 2: **Degradation Mechanisms and Experience**
This section describes the literature review performed to identify degradation mechanisms that may contribute to RPV stud degradation and flaw propagation. A description of current examination requirements is also provided.

Section 3: **Stress Analysis**
This section summarizes the FEA performed to develop the maximum stud bending and membrane stresses encountered for each loading condition or transient.

Section 4: **Flaw Tolerance Assessment**
This section describes the results of the allowable flaw size and fatigue crack growth evaluations. The calculation of crack-tip stress intensity factors is also described.

Section 5: **Conclusions**
This section provides conclusions from the technical evaluations and criteria that are to be met for the technical bases to be considered applicable on a plant-specific basis.

Section 6: **References**
This section lists the works cited.

Appendix A: **Listing of Transients**
This appendix presents the operating transients used to assess stud stresses for the fatigue and structural stability cases.

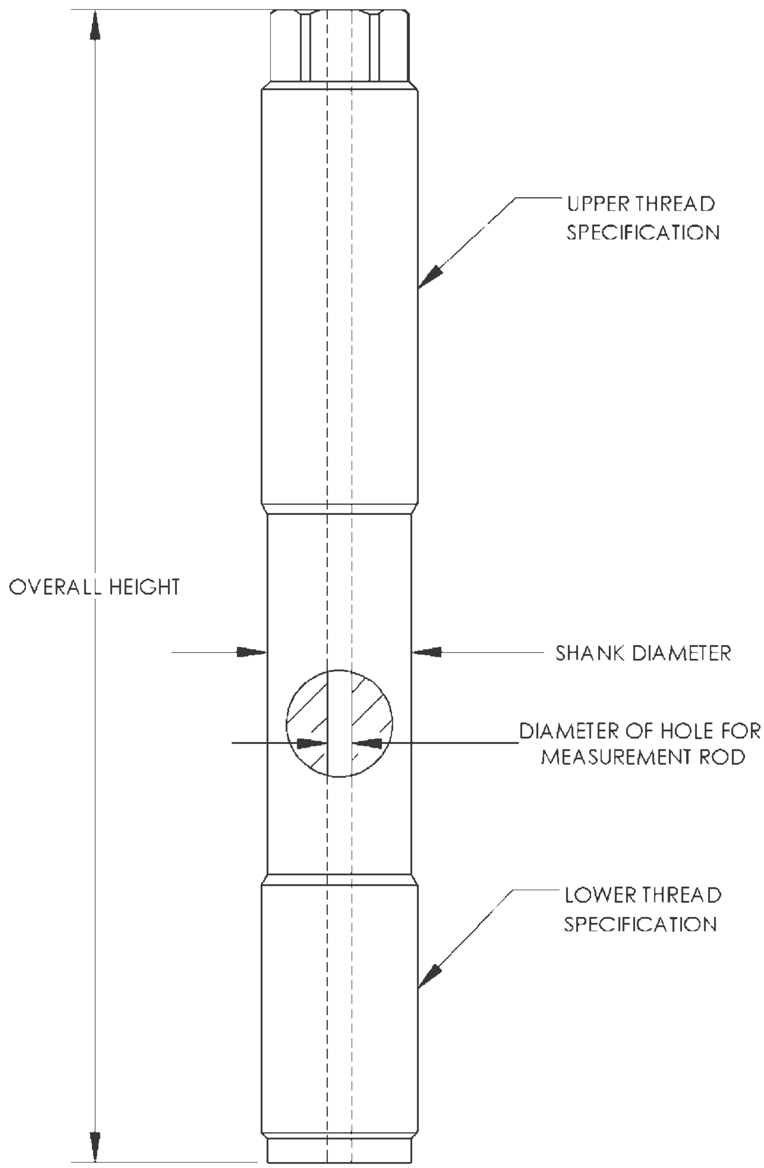

Figure 1-1
Generic RPV stud geometry

2
DEGRADATION MECHANISMS AND EXPERIENCE

This section considers the mechanisms that have the potential to result in significant RPV stud degradation or failure. Prior occurrences of these degradation mechanisms should be considered in an assessment of operating experience. Discussion is also included on inspections currently performed for the RPV studs.

2.1 Potential Degradation Mechanisms

The large number of U.S. licensees pursuing license renewal over the past two decades has resulted in a significant effort to identify active and potential degradation mechanisms for nuclear plant components. Aging management plans have been developed to address these mechanisms, including those associated with the RPV studs. In order to optimize the aging management of RPV studs, degradation mechanisms that have the potential to result in significant degradation or failure of these components are investigated.

In 2007, the expert panel on Proactive Materials Degradation published their assessment [10]. The expert panel considered three degradation mechanisms for RPV studs: erosion corrosion (steam cutting), fatigue, and SCC. The panel considered steam cutting to be a known issue that is possible if there is leakage through the RPV flange joint and assigned it a moderate (yellow) risk categorization. The panel considered fatigue to not be a significant issue (green) based on the load conditions for the studs. In general, the panel considered SCC to be effectively mitigated (dark-green) based on the procedural prohibition of molybdenum disulfide (MoS_2) lubricants, but one panelist noted there was a potential, if wetted, for SCC of high-strength ferritic bolting material due to hydrogen embrittlement.

Significant boric acid corrosion in PWRs would be detected and rectified before leading to significant closure stud degradation because (1) multiple cycles with leakage would be required to cause significant stud degradation and (2) the quantity of leakage would be readily identified during examinations required by PWRs' Boric Acid Corrosion Control Program. BAC at PWRs was considered by the expert panel [10] for other categories of bolting but not for RPV studs.

The Generic Aging Lessons Learned (GALL) report [12] issued by the NRC considers the degradation mechanisms applicable for RPV studs and provides a means for addressing those concerns. In the absence of leakage from the flange (a detectable condition), the possible degradation mechanisms are fatigue and SCC. Similarly, the GALL report for subsequent license renewal (GALL-SLR) [25] contains three sets of aging effects for RPV studs: (1) cumulative fatigue damage or fatigue cracking, (2) cracking due to SCC, and (3) loss of material due to wear, general corrosion, pitting, or crevice corrosion. These three categories of degradation mechanisms are discussed in further detail in the following subsections.

2.1.1 Stress Corrosion Cracking

SCC is typified by a branching crack that propagates in susceptible materials subject to high stress (applied or residual) in the presence of an aggressive environment. Historically, SCC is the degradation mechanism that has led to failures of Class 1 structural bolting [7]. However, the causes of SCC degradation were identified and are now addressed procedurally through controls on procurement, the tensioning process, and lubricant chemical compatibility.

The first major concern for SCC is due to contamination by an aggressive chemical (e.g., halogens, sulfur, and lead). While bolting materials can be contaminated by leaching from insulation, lubricants, or spills/leakage, only contamination by lubricants applies to RPV studs. At operating temperature, the historically applied lubricant MoS_2 degrades and can form H_2S, which promotes SCC in bolting steels. As discussed below, controls are now in place to ensure chemical compatibility and stability of lubricants and surface treatments.

The second major concern is SCC caused by hydrogen embrittlement of the high-strength low alloy steel. This does not appear to be a significant issue for studs with a material yield strength less than 150 ksi (1.03 GPa) and is also mitigated by preventing studs from being wetted during refueling—a source of hydrogen—either by temporary seal covers or by stud removal.

The generic aging management program for RPV studs in the GALL report [12] lists four preventative actions that can effectively reduce the potential for SCC:

- Avoiding studs that are metal plated to reduce the potential for seizing. The metal plating can lead to hydrogen embrittlement or galvanic corrosion at discontinuities when wetted.
- Applying manganese phosphate or other acceptable surface treatments.
- Avoiding the use of molybdenum disulfide as a lubricant, and instead using lubricants that remain stable at operating temperatures.
- Using material with an actual yield strength confirmed by measurement to be less than 150 ksi (1.03 GPa).

Additional details on these recommendations are included in NRC Regulatory Guide (RG) 1.65 on RPV stud material procurement [13]; originally published in 1973, it was revised in 2010 to reflect lessons learned from operating experience.[1]

Since the removal of residual MoS_2 from RPV studs, no additional SCC has been confirmed in RPV studs.[2] Consequently, this mechanism is not considered in the quantitative evaluations of RPV stud degradation in this report.

[1] The major changes between versions include modifying the maximum recommended strength (from a maximum ultimate tensile strength of 170 ksi [1.17 GPa] to a maximum yield strength of 150 ksi [1.03 GPa]), changing the use of MoS_2 from an implicit recommendation to a prohibition (other interim NRC guidance already reflected this prohibition prior to the revision of RG 1.65, as described in [13]), and updated discussion of inspections.

[2] The nature of a recent UT indication at Hatch 2 (see Section 2.2.3.2) has not yet been determined.

2.1.2 Fatigue

As noted in Reference [6], Section XI inspections were primarily concerned with the identification of fatigue cracks in nuclear plant components. ASME Code, Section III evaluations of some RPV studs result in fatigue usage factors near 1.0 for the original 40-year design life[3]. However, those calculations included significant amounts of conservatism in terms of load cycle counts, as indicated by the revised fatigue usage calculations for the period of extended operation. Additionally, the ASME Code fatigue life curves include safety factors on stress and cycles to failure. Therefore, it is considered unlikely that fatigue crack initiation will occur during extended plant operation, even in studs with a high design basis usage factor.

Improved tensioning procedures have also contributed to ensuring that loading patterns for RPV studs are optimized during the boltup and detensioning processes. Nevertheless, it is important to consider the potential for fatigue and fatigue crack propagation given the high stresses to which RPV studs are subjected.

2.1.3 Wastage Mechanisms

In addition to cracking by SCC or fatigue, there is the concern for corrosion or erosion of the stud cross-section. Degradation mechanisms that can cause the loss of material include erosion corrosion (steam cutting), general corrosion (such as caused by BAC in PWRs), pitting, or crevice corrosion. Exposure to an air environment and the size of RPV studs prevent pitting from being a relevant failure mode, although small pits on threads or elsewhere can act to promote the initiation of cracking. Despite the crevice-like environment of threads, the region is rarely—if ever—wetted, so crevice corrosion is not relevant as a failure mode. The two relevant mechanisms—BAC (only applicable for PWRs) and steam cutting—are predicated on there being active leakage.

Leakage from the RPV flange is detectable by the equipment that monitors the leak-off space between the two concentric vessel O-rings, and plant technical specifications require shutdown in the event of detected leakage [26]. The absence of leakage at the flange joint precludes the occurrence of steam cutting (erosion corrosion) of the RPV studs, and reduces the extent of possible boric acid corrosion in PWRs. While BAC could occur due to leakage from a source above the head (e.g., conoseal leakage), the extent of wastage is greatly reduced compared to direct impingement because of the remote leakage source. Furthermore, evidence of such degradation would be apparent by a visual examination of the studs and, in the case of BAC, by the large volume of boric acid crystals in the vicinity.

In addition, there have been concerted efforts to address degradation concerns associated with leakage of primary coolant through procedural and training improvements. In the case of BAC, this includes the implementation of Boric Acid Corrosion Control Programs at all US PWRs and training radiation workers to recognize and report the presence of boric acid crystals. Additional discussion on these programmatic controls is provided in 2.3.3.

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[3] It is a common practice to preload bolts/studs to a substantial fraction of their yield strength. Therefore, cycles of unloading/reloading the RPV studs (in order to open the RPV) can result in elevated fatigue usage.

2.2 Operating Experience with RPV Stud Degradation

Most operating experience in the nuclear industry with degradation of primary pressure boundary studs and bolts has been associated with steam generator manway hatch studs and reactor coolant pump (RCP) main flange bolts. Historically, SCC has been the most common cause of fastener failure in the nuclear industry [15], with many instances involving hydrogen embrittlement due to overly hardened studs exposed to a moist, oxygen-rich environment or involving MoS_2. As discussed previously, prior causes of SCC have been addressed for RPV studs through aging management activities. Review of the NRC Public Document Room and plant operating experience indicates that cracking of RPV studs has only been confirmed at one plant to-date. However, there has also been a crack-like indication determined to be corrosion degradation of the threads at one plant, and a recent ultrasonic test (UT) detected an indication that has not yet been subjected to supplemental examination to confirm whether cracking is present.

The second and third most common degradation mechanisms for Class 1 structural bolting in PWRs are BAC and fatigue. Due to the differences in their configuration, and the fact that RPV studs are handled and re-tensioned prior to each operating cycle, the concern for boric acid corrosion of RPV studs over multiple cycles of operation is much lower than for other PWR components.

An additional concern in the industry is counterfeit or fraudulently marked fasteners, but the high-profile procurement of an RPV stud using the guidance such as that found in RG 1.65 [13] ameliorates this concern. Therefore, additional discussion on counterfeit fasteners is not warranted.

Operating experience (OE) for RPV studs indicates that preventative measures are addressing relevant degradation mechanisms, which is consistent with the view supported by the Proactive Materials Degradation expert panel [10].

2.2.1 *Inoperable RPV Studs*

While not actually a degradation mechanism, there have been numerous instances where a RPV stud became stuck and the ability to tension the stud was impacted. These include D.C. Cook (1 stud in 1986), Catawba (1 stud in 1989), Callaway (5 studs in 1987; 4 were repaired prior to restart), Comanche Peak (1 stud in 1992, 3 in 1994), Sequoyah (1 stud in 1996), Seabrook (1 stud in 1997), and Braidwood (1 stud in 1992) [16].[4] In the event that application of a penetrant and vibration are insufficient to remove an RPV stud, an additional corrective action has been to cut off the accessible portion of the stud and bore out the portion remaining in the vessel flange.

As noted in Reference [4], the RPV closure region has been demonstrated to maintain Code structural margins when operating with one stud detensioned. The calculated maximum flange separation during heatup at the design rate is typically no more than a few thousandths of an inch, which is much less than the minimum O-ring springback of around 0.010-0.015 inches (0.25-0.38 mm).

[4] This list is not exhaustive. Callaway has continued to face issues with stuck studs as recently as 2013, but has not had to resort to destructively removing a stud since 1989 [16].

2.2.2 Wastage or Loss of Material

While loss of material due to corrosion has caused significant degradation of steam generator and RCP bolting, RPV studs have experienced only modest degradation from these mechanisms. The larger size of RPV studs tends to increase the time for loss of material to become a significant portion of the cross-sectional area. Wastage experience [5] for RPV studs is limited to boric acid corrosion in PWRs. Typically, this has resulted in limited degradation, but it can become significant if left unaddressed for more than one operating cycle.

The top ends, above the nut, of three RPV studs were corroded at Turkey Point 4 in 1987 because of leakage from a conoseal joint on the RPV head [27]. This leak was discovered in 1986 but was determined to be low-risk and not remediated before operating for another cycle. Similar leakage at Salem 2 [28] resulted in head, but not stud, degradation. Leakage through the RPV closure O-rings occurred at Millstone 2 in 1988 (as identified in a Wolf Creek submittal to the NRC) [5], resulting in minor corrosion of nine RPV studs. Two other instances of RPV closure leakage at Millstone 2 did not result in significant degradation.

While pitting was observed on studs at Dresden 2, the pitting was an initiation site for SCC, as discussed in Section 2.2.3.1, rather than a mechanism that could have caused stud failure.

2.2.3 Cracking

Cracking has been confirmed at only one plant, a BWR. UT indications indicative of cracking have been detected at two other BWRs, one of which was determined to be general corrosion rather than cracking. There has been no reported cracking of RPV studs at PWR plants.

The EPRI Materials Degradation Matrix [11] notes that the performance of high-strength low alloy steel has been good in the absence of the following aggressive conditions relevant to RPV studs: lubricants unstable at operating temperatures or alternating wet/dry conditions, notably for studs with hardness significantly above Rockwell C40. Reference [11] concludes that "*the SCC characteristics of low-alloy steel fastener materials are relatively well understood and industry issues surrounding the use of overly hard bolting material and unstable lubricants are generally resolved.*"

2.2.3.1 Dresden 2 RPV Studs

During the spring 1989 refueling outage, two cracked RPV studs were detected at Dresden Unit 2 [8]. The flaws were identified using UT from the stud end, confirmed using magnetic particle testing, and sized using a custom inner diameter UT probe. The lower half of one stud was destructively examined as part of a root cause evaluation.

The maximum depth of the indications from the UT examination was estimated to be 0.88 inch (22.4 mm) for the stud that was sectioned and 2.09 inches (53.1 mm) for the other stud [8]. Visual examination of the stud revealed pitting and corrosion of the outside surface; magnetic particle examination indicated that the cracking emanated from the roots of threads 14 through 20 below the shank (1.75 to 2.5 inches [44 to 64 mm] below the shank). The metallographic examination of the stud with the smaller maximum flaw size confirmed that the UT probe provided reasonable estimations of the flaw depth (flaw sizes of 0.4 and 0.7 inch [10 and 18 mm] corresponding to UT indications of 0.4 and 0.88 inch [10 and 22 mm]). The crack orientation was about 60° to 70° relative to the stud axis. The metallographic examination indicated that the flaw was a branching SCC crack with a black oxide layer on the crack faces.

Chemical testing and mechanical testing were performed on the material from the stud shank. These tests indicated the chemical composition was nominal, but the material was harder and had 10-20 ksi (69-138 MPa) higher yield strength than indicated by the CMTR (particularly near the outside diameter [OD]). It was concluded that the studs had been subject to tempered martensite embrittlement during operation, which reduces the toughness of the material due to the transformation of austenite retained after heat treatment into string-like carbides and untempered martensite. It was concluded that a further reduction in toughness was not possible because all the austenite had already transformed to martensite [8].

It was thought that the SCC occurred primarily during refueling outages following flood-down and removal of stud covers prior to heatup. During this time, it was postulated that the slight recess of the vessel flange sealing surface trapped water that could flow into the thread region and persist for up to two weeks until it evaporated during heatup [8].

The detection of these cracks resulted in General Electric (GE) issuing the Rapid Information Communication Services Information Letter (RICSIL) No. 055 [18], which recommended augmented examinations of at least five RPV studs at BWRs using a UT method with an improved (at the time) detection threshold of 0.3 inch (8 mm).

2.2.3.2 Hatch 2 RPV Stud (Cracking Not Confirmed)

During the spring 2017 outage, a volumetric indication was detected in one RPV stud at Hatch Unit 2 [4]. This indication was in stud number 33 and was about 1.0 inch (25 mm) long and 0.75 inch (19 mm) deep [9]. The axial position of the indication corresponded to just below or equal to the surface of the vessel flange (where the studs are threaded in), so removal was required to perform a supplemental surface examination. The indication was not confirmed as a surface-breaking flaw because the stud was stuck and unable to be removed using on-site tooling; thus, the supplemental surface examination could not be performed. Note that most BWR RPV studs are not routinely removed during refueling.

FEA was used to demonstrate that it was acceptable for Hatch 2 to return to service with one RPV stud out of service. Hatch 2 received relief to operate for one additional operating cycle (until spring 2019) to provide time for tooling and procedures to be developed to remove and replace the stud, as necessary. It is noted that the stud was fully tensioned before returning to service at the end of the spring 2017 outage.

Such an indication raises the potential for an active degradation mechanism for RPV studs that has not been effectively mitigated to date (e.g., by procedural controls on head tensioning or by control of lubricant materials). Given that surface examination of the stud will not occur until after this report is published, confirmation of the volumetric indication as a surface-breaking flaw remains an open item. While the minimum yield strength was specified as 105 ksi (724 MPa), the actual yield strength of the stud has not been reported [29].

2.3 Examinations

Section XI inspections provide information on the potential for fatigue and other, unknown degradation mechanisms. As degradation mechanisms for RPV studs have been discovered and effectively mitigated, the Section XI examination requirements have continued to evolve.

2.3.1 Current Examination Requirements

RPV stud inspections are performed in accordance with Section XI, Table IWB-2500-1 Examination Category B-G-1, “Pressure Retaining Bolting, Greater Than 2 in. (50 mm) in Diameter,” Reactor Vessel Item No. B6.20, “Closure Studs,” [22]. All RPV studs are required to be examined during each Code Interval and deferral to the end of the Interval is permissible.

A volumetric examination is required, unless a surface examination is performed when the stud is removed. The volumetric examination may be performed in place under tension, when the connection is disassembled, or when the stud is removed.

The examination surface/volume is defined in Figure IWB-2500-12: it extends radially inward from the thread roots by ¼ inch (6.4 mm), and axially from the start of the bottom threads upward to where the threads no longer engage the nut while tensioned. The examination surface/volume is illustrated in Figure 2-1. The examination acceptance criteria are defined in IWB-3515.

For newly procured RPV studs, ASME Code, Section III, NB-2580 requires that the preservice inspection include a visual examination, a magnetic particle or liquid penetrant examination, and an ultrasonic examination. Therefore, newly installed studs are confirmed to be free of surface-breaking defects.

These examination requirements are consistent for the 2010 Edition through the 2017 Edition of Section XI. The examination volume for historic inservice inspections and current preservice inspections includes the entire stud radius inside the thread roots. Consequently, any non-axial subsurface flaws in the volume omitted from contemporary inservice examinations would have previously been detected and dispositioned; therefore, flaws of this nature do not require additional consideration for inspection.

2.3.2 Historic Examination Requirements

Early editions of Section XI of the ASME Code (e.g., the 1971 Edition) required a volumetric and visual or surface examination of the entire RPV stud. Starting in the 1974 Edition, a volumetric examination was required for in-place studs, and a volumetric and surface examination were required when the stud was removed; a visual examination was also required. The 1977 Edition added a figure to define the examination volume (entire volume up to the thread root and the entire stud length) and permitted deferral of surface examinations until the end of the interval. The 1980 Edition limited the axial extent of the examination volume from the start of the lower threads to the upper end of the nut in its as-tensioned location.

Code Case N-307, first published in 1980, limited the radial extent of the ultrasonic examination volume to the outer 0.25 inches (6.4 mm) of the RPV stud when UT examination was performed from the center-drilled hole; the NRC approved this Code Case in Revision 1 of RG 1.147 in 1982. However, UT of the entire stud volume performed from the end of the stud continued to be the standard practice through the early 1990s [19]. Revisions to Code Case N-307 permit UT performed from the end of the stud to use the examination volume adjacent to the thread roots and eliminated the surface examination of the center measurement hole of RPV studs. Operating experience demonstrated that cracking initiates on the OD for RPV studs ([13] and [20]), so examination of the ID was not necessary. The provisions of this Code Case have since been incorporated into Section XI.

In the early 1990s, the operating experience at Dresden 2 prompted the issuance of RICSIL 055. This letter recommended use of "enhanced end-shot" UT, which had a smaller minimum detectable flaw size, on at least five RPV studs at BWRs. Starting in 1999 [22], 10CFR50.55a required the use of Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems" [24], with a maximum notch depth of 0.157 inches (4 mm) and area of 0.059 in^2 (38 mm^2) for blind test qualification specimens for RPV studs (and other bolting greater than 4 inches (100 mm)), ensuring a minimum detectable size smaller than 0.157 inches (4 mm) deep.

According to the Discussion section in RG 1.65 Revision 1 [13], in 2005, the NRC unconditionally accepted Code Case N-652 in RG 1.147, Rev. 15. For RPV studs, this Code Case allowed the surface examination for removed studs to be optional, and permitted the surface examination method to be used in lieu of the volumetric method for removed RPV studs Code Case N-652 also removed the stipulation (added in the 1980s) that deferral until the end of the interval was not permissible if boric acid leakage was detected. The provisions of this Code Case have since been incorporated into Section XI.

2.3.3 Additional Controls on Wastage Mechanisms (Leakage)

The RPV head closure is sealed using two self-energized metallic O-rings. The volume between these O-rings is instrumented to monitor for changes in pressure and any leakage through the inner seal. In combination with low administrative limits for shutdown for leakage through this joint, this monitoring ensures that the potential for any steam cutting (erosion corrosion) to have occurred during any given operating cycle would be known. Consequently, an absence of leakage through the RPV closure is a criterion required for application of this technical basis (see Section 5.2).

While steam cutting is only applicable for leakage through the RPV closure, boric acid corrosion at a PWR may occur due to leakage from components above the RPV studs. The concern for wastage of components, including RPV studs, at PWRs is addressed through a plant's boric acid corrosion control program (BACCP). These programs are designed in-line with PWR Owners Group guidance [30], which is "mandatory" per NEI 03-08 [21]. A BACCP is designed to limit reactor coolant system (RCS) leakage and minimize the potential for boric acid corrosion when leakage does occur. Additional information and best practices are provided by EPRI's Boric Acid Corrosion Guidebook [31], which is not assigned an implementation status per NEI 03-08. In line with reducing leakage, PWR RCS leakage administrative action levels have been revised since the early 2000s in light of operating experience; NEI 03-08 "needed" guidance is provided by Reference [32].

In addition to these controls, the aging management programs at plants with renewed licenses often impose additional requirements.

2.4 Summary of EPRI Survey Results

Two surveys of EPRI members were used as part of developing this report.

All EPRI member utilities were surveyed regarding the number of examinations performed, flaws exceeding Section XI acceptance standards, and the inspection impact on personnel dose and outage impact [33]. The results of this survey, which was not performed within the scope of this report, included data from 70 units and identified the RPV stud examination as one having low-value but high outage impact.

A follow up survey was sent to a selection of EPRI member utilities to obtain detailed geometry and transient cycle information relevant to RPV studs. These EPRI members were selected across a range of RPV designs, with preference for units with higher stud preload and operating stresses. The geometry information requested was sufficient to define the parameters shown in Figure 3-3, and the transient information requested included the Normal, Upset, Emergency, and Faulted transients relevant to the RPV closure region. The survey also included the material specification for the RPV studs and RPV flange, as well as design basis and typical operating conditions for pressure and temperature. Survey responses were obtained from a total of twelve plant configurations (not counting identical units separately), comprising 9 PWR units and 9 BWR units. All respondents provided information on the RPV stud geometry and material, all respondents except one provided operating and design conditions and loads, and seven respondents provided information on reactor coolant system transients. These survey responses were used to form the selection of bounding RPV closure geometries in Section 3 and the fatigue crack growth calculation in Section 4.3.

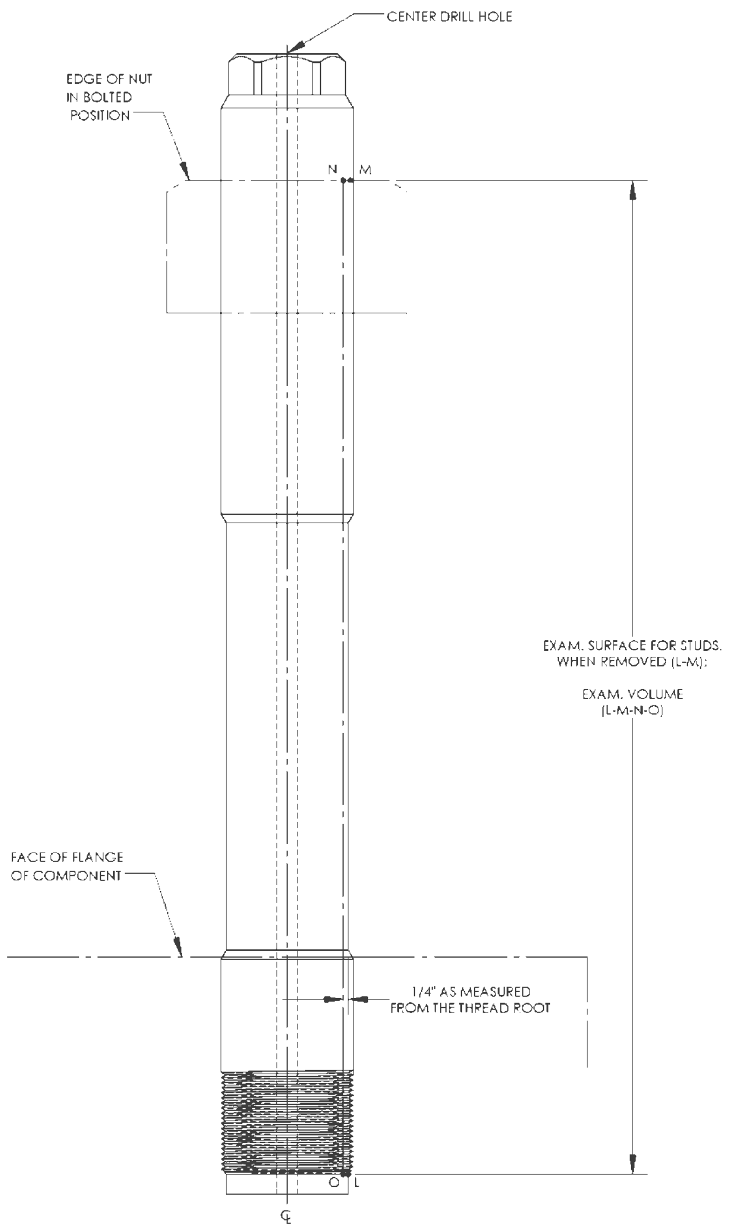

Figure 2-1
RPV stud (Item No. B6.20) examination surface/volume

3
STRESS ANALYSIS

The RPV stud stresses were obtained by FEA of two RPV closure region geometries. FEA model geometries were developed using information provided by EPRI member utilities as part of an industry survey, as discussed in Section 2.4. Two separate FEA models were developed for PWR and BWR geometries because of the differences in the diameters and thicknesses of the RPVs and RPV closure heads. Combinations of model parameters were evaluated to identify two bounding configurations, one for a PWR and one for a BWR.

3.1 Finite Element Model Geometry and Meshing

The FEA models are three-dimensional sections of the RPV top head closure region that span one stud with rotational symmetry. The FEA model geometries conservatively simulate tensioning of all RPV studs at once, which maximizes stud bending stress. The FEA models were developed using ANSYS [34]; the model for the BWR geometry is shown in Figure 3-1 and for the PWR geometry in Figure 3-2. The models include the closure head (purple) and flange (cyan), the upper vessel (purple) and flange (cyan), and the stud and nut/washer stack (red). The O-ring grooves are included in the FEA models, but the O-rings themselves are not included because the O-ring springback force is not significant in this context.

The vessel shell, head, stud, washer/nut, and flange regions were modeled using higher-order three-dimensional structural solid (SOLID95) elements. Given the high contact forces under stud preload, modest amounts of friction ($\mu < 0.1$) are sufficient to prevent slip between the spherical washer and nut of each stud; therefore, these components were modeled as a single component. Contact surfaces (CONTA174 with TARGE170) were used around the stud and between the flanges. Bonded contact was used at the threaded interfaces (stud/nut and stud/vessel flange), as well as between the washer and head flange. The interface between the stud OD and the hole in the RPV head flange is a standard frictionless contact interface. The contact surface between the flange mating surfaces was assigned a rough (no-slip) friction contact.

The nodes at the bottom of the vessel shell were fixed in the vertical direction and allowed to move freely in the radial direction. Symmetry boundary conditions were applied at the circumferential faces of the model: each face was restrained in the circumferential direction but permitted to move freely in the radial and vertical directions. Stud preload was applied using the ANSYS command "PSMESH," which creates a pretension element reflecting the force and displacement nature of preloading. Internal pressure was applied as a surface force on the wetted surfaces of the head and vessel; the internal pressure was applied up to the radius of the inner O-ring of the vessel.

In addition to the structural phase of the FEA, thermal modeling was performed to obtain temperature gradients during transients. In converting the mesh for thermal modeling, SOLID90 elements were used in place of SOLID95 structural elements. Thermal convection coefficients were applied to the head and vessel wetted surfaces to model the heat transfer with the primary coolant, with values based on typical convection boundary conditions for RPV closures. Thermal conduction was modeled through the bulk of material and at the component contact points. ANSYS contact surfaces were used to establish conduction paths where contact occurred in the structural evaluation (at the stud shank / head flange hole interface and between the head flange / vessel flange mating surfaces). Where a gap exists (no contact) along these interfaces, radiative heat transfer (SHELL57) was modeled. The head and vessel outside surfaces were assumed to be adiabatic to represent the insulation surrounding the RPV.

3.2 Finite Element Analysis Approach

The FEA solved for the stud preload condition by applying the preload force (see Section 3.3 for a summary of loads and transients) to the stud at a uniform temperature of 70°F and solving for the stress state. The preload, hydrotest, and steady-state operating cases were assumed to have negligible temperature variation at those conditions, so they were modeled to occur at a uniform temperature without requiring a separate thermal solution step. The hydrotest and steady-state operating cases started from the resulting preload case to model the stud tensioning process.

For each transient case modeled, the analysis was performed by solving the thermal model to establish temperature distributions throughout the FEA mesh, followed by input of the resulting temperatures to the structural model and solving, as follows:

- Reload the RPV closure region stress state following stud tensioning (i.e., the preload case). The heat transfer mechanism between component interface surfaces (conduction vs. radiation) was based on the surfaces that contacted from the application of the stud preload. Surfaces in contact tended to stay in contact during the analysis.
- Solve for the RPV closure region thermal state throughout the transient by varying the bulk temperature at the convection boundary condition along the wetted surfaces. This modeled the changing values of coolant temperature throughout the transient.
- Load the RPV closure region thermal solution into the structural model, apply internal pressure, and perform a structural solution. The stud preload was retained throughout the transient by resuming the solution from the preload stress case. Other mechanical loadings, such as those resulting from seismic cases, do not generate significant additional loads for the RPV closure.

The transients were discretized into a series of time points with the coolant temperature and internal pressure conditions linearly varied between each point. A summary of the RPV stud stresses at an elevation equal to the top of the vessel flange threaded region were extracted from the solution.

3.3 Applicable Loads and Transients

Three types of inputs were applied to the FEA models to simulate the various load states and transients: (1) stud preload force, (2) the coolant temperature along the head and RPV inner surfaces, and (3) the internal pressure. As mentioned in Section 3.1, the stud preload force was effectively applied as an initial strain; the stud preload state was then selected as the starting point for all other evaluated operating states and transients.

The information in this section was developed from public sources as well as EPRI member responses to the survey discussed in Section 2.4. Additional information was obtained from design basis documents available on the NRC ADAMS database, including final safety analysis reports (FSARs).

3.3.1 Tensioning Load

The maximum stress during tensioning was determined directly, without the use of the FEA models. The method by which RPV studs are tensioned results in the maximum stress (maximum elongation) occurring as pure membrane stress based on the maximum applied tensioner force. It is noted that many stud tensioning sequences are performed at pressures below the maximum tensioner force.

RPV head tensioning and detensioning are performed using a set of two to eight tensioners suspended from a carousel over the RPV head. Each tensioner is threaded onto the upper part of the RPV stud and interfaces with the castellation on the top of the nut. The tensioner is hydraulically actuated to pull upward to extend the stud while pressing downward on the RPV head flange. The tensioner pressure is controlled and is set to the target value per the relevant plant procedure. With the RPV stud elongated, the nut is tightened onto the head flange, and the tensioner load is removed. The retained stud load is lower than the applied tensioner load because of compliance of the joint after the tensioner is removed; this effect is accounted for in the stud tensioning procedures.

The tensioning process described above is not simulated in the FEA modeling. Rather, the stud is directly preloaded to a desired stud preload, as described in Section 3.3.2.

3.3.2 Stud Preload

The RPV stud preload was based on the maximum stud average stress values from upper bound elongation tolerances multiplied by the stud cross-sectional area. A preload force of 1014.5 kips (4513 kN), equivalent to a stud average stress of 41.6 ksi (287 MPa), was applied to the stud for the BWR cases. Similarly, a preload force of 1375.0 kips (6116 kN), equivalent to a stud average stress of 46.1 ksi (318 MPa), was used for the PWR cases. These preload values are consistent with the preload force for plants with the geometry used to create the limiting RPV closure geometry. As noted in Section 3.1, the stud pretension was applied using the ANSYS "PSMESH" command to reflect the displacement nature of this load. A uniform temperature of 70°F was applied during these load steps to reflect the fact that stud tensioning occurs at ambient conditions.

3.3.3 Hydrotest and Steady-State Operation

Starting from the stud preload conditions, internal pressure was applied as a surface force on the wetted surfaces of the FEA models, which included all elements on the inner diameter of the vessel and head out to the inner O-ring diameter. For the BWR cases, the hydrotest was applied at 1,250 psi, and the operating pressure was applied at 1,060 psi. For the PWR cases, the hydrotest was applied at 3,125 psi, and the operating pressure was applied at 2,185 psi.

For the steady-state, normal operation cases, representative operating temperatures (553°F for BWRs, 579°F for PWRs) were applied uniformly to all elements in the FEA models; the hydrotests were performed at a uniform temperature of 100°F. After application of the internal pressures and uniform temperatures, a structural solution was performed to obtain the stress state.

The "Inner Seal Leakage" case for the BWR geometry considers leakage past the inner O-ring of the RPV closure. The case was also run as a steady-state evaluation using the same temperature and pressure as the steady state, normal operation case, except that internal pressure was applied out to the outer O-ring instead of to the inner O-ring.

3.3.4 Transients

Four BWR transients and eleven PWR transients were evaluated. The transients were selected based on the information obtained from the EPRI plant survey, and are considered representative of typical plant design basis transients. Appendix A contains a detailed description of the transients evaluated in this work.

For the BWR model, the following operating states and transients were considered:

- *Steady State*: preload, hydrotest, normal operation, and inner seal leakage.
- *Transient*: startup, shutdown, loss of feedwater pumps, and pre-op blowdown.

For the PWR model, the following operating states and transients were considered:

- *Steady State*: preload, hydrotest, and normal operation.
- *Transient*: heatup, cooldown, plant loading/unloading, small step increase/decrease, large step decrease, steady state fluctuations, loss of load, loss of flow, and reactor trip.

Because the stud preload is applied as a displacement load (see Section 3.3.2), changes in the stud load due to conditions such as differential thermal expansion are accurately considered.

3.4 Development of Limiting RPV Closure Geometries

A range of vessel geometries from the respondents of the EPRI survey, as well as geometries available for other U.S. BWRs and PWRs [3], were considered in this work. Given the significant differences in closure geometries, separate models were developed for BWR and PWR configurations. BWR RPV heads are much thinner and have a larger inner radius because of the lower operating pressure compared to PWRs. The ratio of the RPV head radius to its thickness plays a prominent role in determining RPV stud bending stresses. Results for different model parameter values (parameters identified in Figure 3-3) were compared to develop a single bounding configuration for each of the PWR and BWR geometries.

To identify bounding geometries, the preload stud stresses were compared for the available geometries. Non-dimensional ratios were developed related to vessel and stud properties that were expected to impact stud stresses:

- Head inner radius to thickness (R/t)
- Head inner radius to stud outer diameter (R/D_S)
- Head area to stud area (A_H/A_S)
- Head inner radius to stud thickness (R/t_s)

Based on these ratios and stud preload stresses, a subset of cases was evaluated using the ANSYS model approach discussed in the preceding subsections. The stresses for preload, hydrotest, and operating conditions were evaluated, and it was determined that the highest maximum stud stresses were obtained for model geometries that had the highest R/t ratio (which predominantly affects stud bending stress) as well as the highest R/D_s ratio (which predominantly affects stud membrane stress). Based on the results from these initial analyses, bounding geometries were developed that maximized these ratios. Analyses of the bounding geometries confirmed that they produced bounding stud stresses at preload, hydrotest, and operating conditions compared to the initial subset of analysis cases. Specifically, the BWR and PWR bounding geometries were developed as follows:

- The selected BWR bounding geometry corresponds to the plant configuration with the highest ratio of the RPV head inner radius to head thickness; however, the stud geometry for this configuration was modified to use the smallest currently operating BWR stud geometry of the cases considered.
- The selected PWR bounding geometry corresponds to the plant configuration with the highest ratio of the RPV head inner radius to stud outer diameter; however, the head thickness from this configuration was reduced to result in the largest ratio of RPV head inner radius to head thickness for the PWR cases considered.

The values of the model parameters for the bounding BWR and PWR cases are listed in Table 3-1, and the values of these ratios for the bounding cases are listed in Table 3-2.

3.5 Results

RPV stud membrane and bending stresses were evaluated throughout each transient considered. For both the BWR and PWR FEA models, the largest bending stresses in the stud were located at the top of the lower threads, where the stud interfaces with the vessel flange. As shown in Figure 3-1 and Figure 3-2, the studs are loaded in tension and bending. The preload in the stud leads to bending in the RPV head and head flange; flange rotation then leads to bending stress in the studs. There are no transients or conditions that lead to reversals in the bending moment orientation. Therefore, the maximum stud stress, as shown in Figure 3-1 and Figure 3-2, is always located at the radially inboard edge of the stud relative to the RPV centerline.

For each of the load conditions listed in Section 3.3, the stress results are presented in Table 3-3 for the BWR model and in Table 3-4 for the PWR model. These tables provide the FEA results for the maximum and minimum stud membrane stress for each model during each transient, as well as the maximum and minimum values for stud total stress during each transient. The difference between the stud maximum stress and stud membrane stress at a given time point may be taken as the stud bending stress, since the stud is modeled as a smooth cylinder with no stress concentrations. While not evaluated using the FEA model, the maximum stud membrane stress during tensioning is also shown in the tables. The tensioning values were calculated using the typical maximum tensioner force and typical stud cross-section and are purely membrane stresses (no bending).

For both the BWR and PWR cases, the operating membrane stresses were slightly less than the preload value due to the effects of operating temperature and pressure on the bolted closure joint. The mating surface of the preloaded closure joint remains in compression after application of internal pressure, and little additional load is transferred to the studs. The BWR stud stress results are characterized by significantly more bending stresses compared to the PWR stud stresses. Most transients have only a modest effect on the RPV stud stresses, but the BWR startup transient results in significantly higher maximum stud stresses. On the other hand, the tensioning process generally results in higher stresses in PWR RPV studs compared to operating conditions or transients; thus, the tensioning process for PWRs acts like a proof test for the RPV studs, providing an additional measure of confidence in the integrity of the tensioned studs. The sole exception to this case is the ASME Code hydrotest at 3,125 psi internal pressure; however, it is noted that this condition is not experienced following the plant entering service.

These stress results are used in Section 4.3 to model propagation of a postulated flaw by fatigue; the maximum stress cases are used to develop limiting flaw sizes in Section 4.2. The inputs for these calculations are the net tension and bending stresses for the bolt cross section; peak stresses (which are used to perform design basis fatigue calculations) are not an input used for the flaw models.

While not evaluated here, it is noted that analyses have been performed for many units to demonstrate that the RPV head remains in compliance with ASME Code, Section III requirements, as well as maintaining leak-tight conditions, following failure of any single RPV stud and select combinations of multiple studs. An example of these calculations is described in Reference [4].

Table 3-1
Bounding RPV closure geometries (see Figure 3-3)

Dimension	BWR	PWR
HIR – RPV closure head inside radius (in)	126.25	87.25
HTK – RPV closure head thickness (in)	3.625	6.00
VIR – Vessel inside radius (in)	127.0	84.188
VTK – Vessel thickness (in)	6.125	12.00
FLTK – RPV closure head flange thickness (in)	28.75	30.00
BCR – Bolt circle radius (in)	133.875	93.375
nS – Number of studs	68	60
SDIA – Stud OD (in) [Taken as the diameter of the stud shank, the minimum cross-section]	5.625	6.250
SID – Stud ID (in) [Diameter of internal hole for measurement rod]	0.625	1.000
WDIA – Diameter of washer (in)	8.855	9.75

Table 3-2
Values of bounding criteria ratios

Ratio	BWR	PWR
HIR / HTK	34.8	14.5
HIR / SDIA	22.4	14.0

Table 3-3
BWR model stud stress results

Transient	Max Stud Stress Location (ksi)		Stud Membrane Stress (ksi)	
	Max in Transient	Min in Transient	at Max Stress Time	at Min Stress Time
Tensioning	65.0		65.0	
Preload	91.9		41.6	
Hydrotest	96.3		44.5	
Operation	89.7		38.6	
Inner Seal Leakage	89.5		38.8	
Startup	111.0	91.5	48.7	41.2
Shutdown	89.7	65.2	38.6	28.0
Loss of Feedwater Pumps	92.9	79.6	39.9	33.7
Pre-Op Blowdown	92.2	79.7	39.5	33.7

Table 3-4
PWR model stud stress results

Transient	Max Stud Stress Location (ksi)		Stud Membrane Stress (ksi)	
	Max in Transient	Min in Transient	at Max Stress Time	at Min Stress Time
Tensioning	70.0		70.0	
Preload	66.7		50.2	
Hydrotest	76.6		54.0	
Operation	68.9		47.5	
Heatup	69.8	60.1	55.5	42.5
Cooldown	67.2	60.1	50.4	42.5
Plant Loading	60.9	59.4	43.2	42.7
Plant Unloading	61.6	59.5	42.5	41.6
Small Step Increase	61.2	60.2	42.9	42.2
Small Step Decrease	61.0	60.3	42.9	42.3
Large Step Decrease	61.5	59.6	42.6	41.6
Steady State Fluctuations	61.1	60.2	42.9	42.2
Loss of Load	62.5	60.3	43.4	42.4
Loss of Flow	61.7	59.1	42.7	41.4
Reactor Trip	62.0	59.8	42.8	41.9

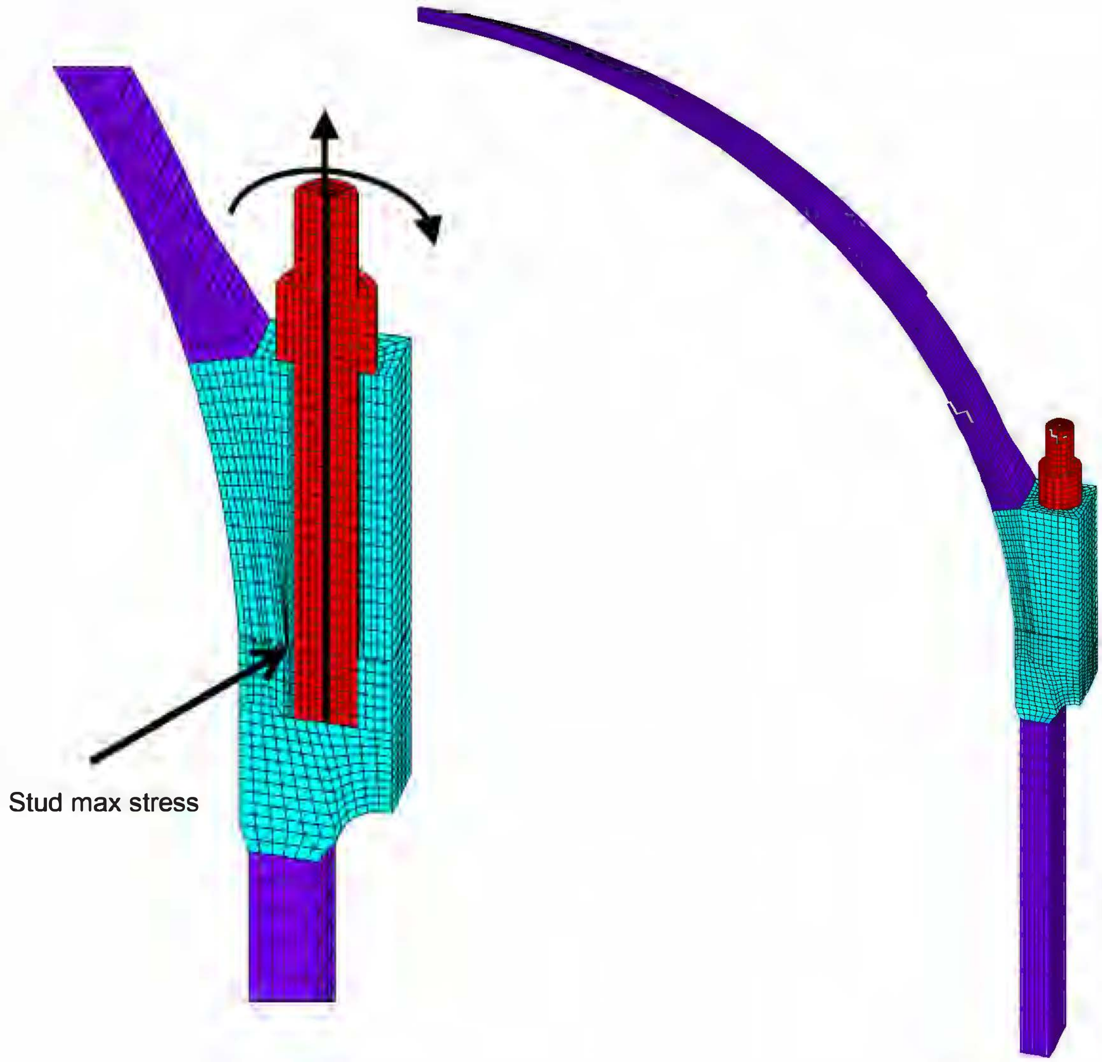

Figure 3-1
BWR RPV closure region FEA

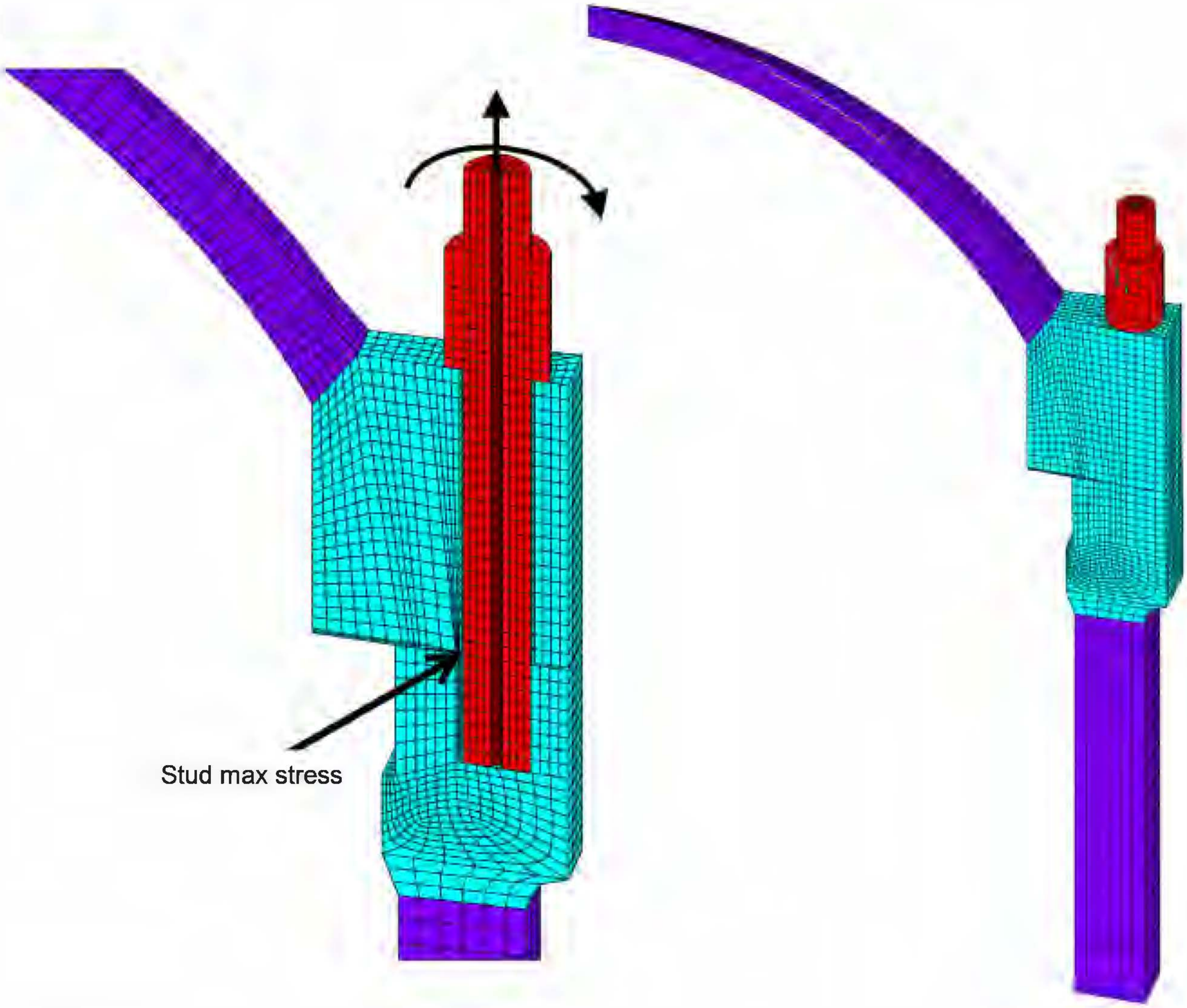

Figure 3-2
PWR RPV closure region FEA

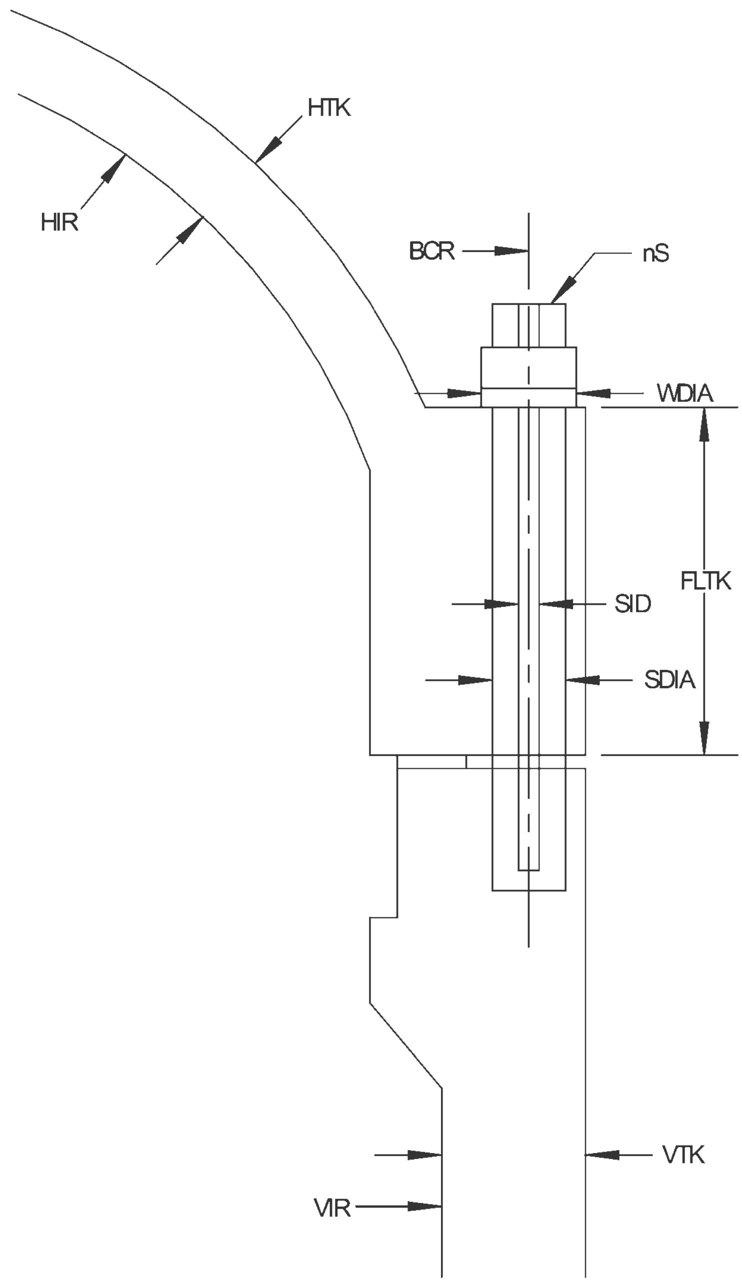

Figure 3-3
RPV head closure geometry (see Table 3-1)

4
FLAW TOLERANCE ASSESSMENT

The flaw tolerance of the RPV studs in the presence of preload plus design basis transient loadings was evaluated using an approach consistent with ASME Code, Section XI methodology. LEFM was used to calculate the crack tip stress intensity factor (SIF), K_I, for a range of flaw sizes postulated in the stud under different loads from Section 3. These K_I values are compared to K_{IC} at fracture for bolting steel materials with a structural factor to obtain a maximum allowable flaw size. Fatigue crack growth calculations were performed using the applied ranges of K_I values, and the limiting flaw size was compared to the allowable flaw size.

4.1 Part-Depth Flaw Stress Intensity Factor Calculation

As described in Section 3.5, the RPV studs are subjected to tension and bending stresses. For both the BWR and PWR configurations, the maximum stud stress, as shown in Figure 3-1 and Figure 3-2, is located at the radially inboard edge of the stud relative to the RPV centerline. Given these stress conditions, flaw evaluations were performed using a postulated surface flaw originating at the location of maximum stress and propagating horizontally across the stud cross section; a 360° flaw was not considered. This approach is supported by Welding Research Council Bulletin 175 (WRCB 175) Paragraph 7 [36], which considered but did not use the more conservative 360° flaw case when developing minimum toughness requirements for bolting.

The crack tip SIF solution for a surface flaw in a threaded bolt under tension and bending stresses is available from several sources, including the following:

- NASGRO v8.2 software crack case SC08 [38], which is based on results described in a paper from the Netherlands National Aerospace Laboratory [39]
- British Standard BS 7910 Cases M10.4.1 and M10.4.2 [40]

Additional discussion for the calculation methodology used from these sources is as follows.

4.1.1 NASGRO v8.2 Case SC08

NASGRO [38] was developed and distributed by the NASA Johnson Space Center and Southwest Research Institute; it is a suite of software programs that evaluate fracture mechanics behavior. Of specific interest for this evaluation is the calculation of crack tip SIF values using the solutions library built into the software. The solution for a specific combination of crack shape and component geometry is referred to as a "crack case" in NASGRO. The software has more than 100 crack cases, and they include solutions for through-wall cracks, corner cracks, embedded cracks and surface cracks in plates, cylinders, and other specialized geometries. Among these geometries, different crack cases are used for different types of loading inputs and boundary conditions.

The NASGRO SC08 crack case considers a semi-elliptical surface crack in a threaded bolt under tension and bending stresses. As shown in Figure 4-1, the case considers a flaw with a depth, *a*, measured from the surface of the bolt across the bolt diameter. The flaw is characterized by a ratio of the flaw depth to bolt minor diameter *a/d*, and the flaw depth to length aspect ratio (*a/c*). The SC08 case considers a single aspect ratio, *a/c* = 1.0 (flaw total length equals twice the flaw depth). The notes for this case state that it applies to rolled threads only, and that machined thread cases need to be accounted for using additional methods.

Additional investigation of the SC08 case background identified report NLR-TP-99313 [39], a report from the Netherlands National Aerospace Laboratory (NLR). This report summarizes the technical basis for the SC08 influence coefficients, which includes fracture mechanics calculations and fatigue flaw propagation testing. According to the report, surface cracks in bolts with rolled threads benefitted from the residual compressive surface stress caused by the rolling process, leading to a flaw depth to length aspect ratio (*a/c*) of 1.0, whereas flaws for machined threads propagated as a longer flaw with a smaller aspect ratio (*a/c*) of 0.645. This result explains the note used in NASGRO for the SC08 case, which includes influence coefficients only for the case of an aspect ratio *(a/c*) of 1.0; the NASGRO case is the less conservative of the two aspect ratio cases.

The NLR paper reports influence coefficients for both machined threads and rolled threads. The influence coefficients are used to calculate a single value for K_I, and not individual results at the maximum crack depth and surface locations. The calculated value is considered the bounding value for the entire crack front. The calculation method uses as inputs: (1) the nominal bolt stress due to axial tension and (2) the nominal bolt stress due to bending. Separate influence coefficients are used for the bolt tension and bolt bending stresses.

The influence coefficients for the more conservative machined thread case were used to perform the crack tip SIF calculations. However, the influence coefficients for machined threads start at a value of *a/d* equal to 0.1; therefore, additional steps were taken to define smaller SIF compounding factors at values of *a/d* less than 0.1 as described in Section 4.1.2.

4.1.2 BS 7910 Case M10.4.2

British Standard BS 7910 [40] includes two solutions for semi-elliptical surface flaws in bolts under tension and bending loads. Case M10.4.1 (Solution 1) was developed specifically for ISO M8 × 1.0 bolt geometry, while Case M10.4.2 (Solution 2) was developed for UNF fasteners and has a wider range of applicability. Case M10.4.2 (Solution 2) provides a solution for tension loads that is based on a combination of solutions for semi-circular surface flaws and straight-fronted flaws in round bars, together with thread effects for $a/d < 0.1$. The range of applicability for this solution is as low as $a/d = 0.004$. Case M10.4.2 also includes an expression for bending loading, but the solution does not include thread effects. Similar to NASGRO Case SC08, a single value for K_I is calculated; this value is a bounding value for the entire crack front.

Crack tip SIF values were calculated using Case M10.4.2 for tensile loads only (i.e., no bending load) and for *a/d* values of 0.004, 0.025, and 0.05. The ratios of these Case M10.4.2 values to the corresponding values calculated using the NLR-TP-99313 [39] influence coefficients for the SC08 rolled threads case were used to define the influence coefficients for the machined thread case at these values of *a/d*. It is noted that Case M10.4.2 does not include a solution with thread effects for bending loads. However, it was considered reasonable to apply the ratios calculated for tension loads to the bending load influence coefficients.

4.1.3 Thread Dimensions

The minor thread dimensions for a typical 6-inch diameter stud with UN thread as specified for RPV studs were used in the calculations. A 6-inch stud is the smallest size specified for RPV studs, at either BWRs or PWRs. WRCB 175 Paragraph 7 [36] notes that the thread root radial depth of these studs is typically 0.08 inches; therefore, a 6-inch thread will have a minor diameter equal to 5.84 inches. It is noted that the stud dimensions cited in Section 3 are for the stud shank, which typically has a slightly smaller diameter than the stud thread nominal diameter. The stud threads are more likely to initiate flaws given the small radii of the thread roots. Therefore, the flaw calculations are slightly conservative, since tension and bending stresses are calculated for the smaller stud shank, and then applied to the larger diameter stud threads.

4.1.4 Calculation Results

The influence coefficients used to perform the K_I calculations are summarized in Table 4-1. As noted previously, the models are used to calculate a single value for K_I, and not individual results at the maximum crack depth and surface locations. The calculated value is the bounding value for the entire crack front.

Crack tip SIF values for flaw depths ranging from 0.25 inches to 1.5 inches were calculated for selected RPV stud tension and bending stresses reported in Section 3.5. The resulting K_I vs. flaw depth values are presented in Figure 4-2 for PWR studs and in Figure 4-3 for BWR studs.

4.2 Allowable Flaw Size

The allowable flaw sizes for the RPV studs are developed in this section based on ASME Code, Section XI approaches and from material properties for bolting steels.

4.2.1 Limiting Flaw Size Evaluation Method

The limiting flaw size evaluation is performed using the methodology specified in Appendix G of ASME Code, Section XI [22]. This appendix is appropriate because a postulated flaw is considered for this evaluation, rather than a method for accepting a flaw indication identified during inspection. This is consistent with previous NRC position [41].

Article G-2000, which provides methodology for vessels, is used for the evaluation. While Appendix G also includes Article G-4000 "Bolting," this Article references WRCB 175 [36] Paragraph 7 and notes that Paragraph 7 provides procedures "for evaluating various defect sizes and associated toughness levels in bolting materials." A review of WRCB 175 Paragraph 7 reveals that the evaluation methods in this paragraph are used primarily to define the minimum toughness criteria for bolts with a 0.3-inch deep reference flaw (for bolts 3 in. to 6 in. in diameter), and not to evaluate flaws using defined structural factors. It is further noted that WRCB 175 is considerably older than other references cited for fracture mechanics evaluations in bolted joints; none of the solutions discussed in Paragraph 7 are specific for bolted joints. Therefore, the structural factors provided in Article G-2000 for vessels, in combination with the crack tip SIF calculation methods previously described, are used to define the limiting flaw size for the postulated flaws in RPV studs.

Consistent with Paragraphs G-2215 and G-2222, the method of calculation used to define the limiting flaw size is to apply a structural factor of 2 to K_I due to primary loads and a structural factor of 1 to K_I due to secondary loads; then the sum of these two values must be less than K_{IC}. Consistent with Paragraph G-2222(b), stresses from bolt preloading are considered primary loads. It is noted that the stud stresses reported in Table 3-3 and Table 3-4 include a combination of primary and secondary stresses. Therefore, the K_I value due to primary loads is calculated first, and then the remainder of the K_I value for the combined stud load state that is not primary is considered secondary. The primary load for this evaluation is taken as the stud membrane (without bending) stress at preload conditions. This approach is consistent with the calculation of the RPV stud preload required to meet ASME Code, Section III, Appendix E [23], which calculates the minimum stud bolting area under primary loads. It is further noted that, due to the bounding nature of the model, the model-calculated stud membrane stress at preload conditions is substantially greater than the RPV stud preload required to meet Section III Appendix E.

4.2.2 K_{IC} for RPV Stud Material

ASME Code, Section XI, Nonmandatory Appendix A provides values for K_{IC} for ferritic pressure vessel steels, but similar values for bolting steels are not provided. RPV studs are well above the reactor beltline and subject to much lower fluence than portions of the vessel shell; therefore, no reduction in fracture toughness due to irradiation effects needs to be considered.

A significant amount of data on the fracture toughness of the SA-540 steels used for RPV studs was identified in a 1977 paper in the Journal of Pressure Vessel Technology [35]. The paper summarized the fracture toughness, yield strength, and Charpy V-notch impact properties that were measured for five commercial heats of SA-540 steels. The paper states: (1) static fracture toughness properties begin to develop an upper shelf around -25°F (-32°C) to -50°F (-46°C), and (2) the upper shelf maximum fracture toughness values range from 190 ksi√in (209 MPa√m) to 240 ksi√in (264 MPa√m). Based on these stated results from the paper, a K_{IC} value of 190 ksi√in (209 MPa√m) is selected as the value for comparison.

4.2.3 Maximum Flaw Size

Reviewing the crack tip SIF results presented in Figure 4-2 for PWR RPV studs and Figure 4-3 for BWR RPV studs, the following conclusions are made:

- The limiting load case for PWR studs is the maximum tensioner load, which is not an operating condition. This result demonstrates that, for PWR studs, tensioning is a “proof test” for subsequent loads, providing an additional measure of confidence in the integrity of the tensioned studs. The limiting operating condition for PWR studs is the Heatup condition.
- The limiting load case for BWR studs is the Startup condition. This result is due to the larger bending stresses present in BWR studs compared to PWR studs.

Using the Appendix G methodology described in Section 4.2.1, the sum of $2K_I$ due to primary stresses and K_I due to secondary stresses was calculated for these limiting load cases, and compared to the fracture toughness of 190 ksi√in (209 MPa√m). Because the upper shelf conditions for RPV stud material are reached well below room temperature, it is not necessary to separately compare preload conditions.

The calculation methodology is shown graphically in Figure 4-4 for PWR studs and in Figure 4-5 for BWR studs. The following results were obtained:

- The maximum allowable flaw size for PWR studs occurs during Heatup conditions and is equal to 1.063 inches (27.0 mm).
- The maximum allowable flaw size for BWR studs occurs during Startup conditions and is equal to 0.789 inches (20.1 mm).

4.3 Fatigue Crack Growth

Fatigue crack growth calculations were performed for the PWR and BWR RPV stud cases using the ranges of crack tip SIF values identified in Figure 4-2 for PWR studs and Figure 4-3 for BWR studs.

4.3.1 Load Transient Frequency

As noted in Section 3.5, RPV stud stresses are always tensile at the peak stress location. Therefore, the largest fatigue cycle occurs between the unloaded condition (zero stress) and the peak stress condition. Other fatigue cycles will be a smaller change in SIF than this cycle. Therefore, the fatigue crack growth evaluation used the largest fatigue cycle loading to bound the full set of operating ranges that occur during plant operation. The transient frequency is based on the results of the EPRI survey of plants for this project. ASME Code hydrotest at 3,125 psia is not included in the transient frequency, since this condition is not experienced once the plant enters operation.

The PWR stud stress cases and the associated fatigue cycles selected for analysis, are described as follows:

Cycle Peak Condition	Cycle Minimum Condition	Assumed Cycle Frequency
Tensioner Load	Unloaded	1 cycle/year
Tensioner Load	Preload	4 cycles/year
Heatup (Max)	Loss of Flow (Min)	1000 cycles/year

As previously noted, the tensioner/unloaded cycle represents boltup. The tensioner/preload cycle represents potential re-seating of the nuts to achieve the required stud elongation during tensioning. The final cycle, using stresses from the heatup and loss of flow transients, is used to bound the magnitude and frequency of the remaining transients that occur during plant operation. For transients, the maximum condition is the maximum stud stress during the transient; the minimum condition is the minimum stud stress during the transient that occurs at the same location as maximum stud stress.

The BWR stud stress cases and the associated fatigue cycles selected for analysis, are described as follows:

Cycle Peak Condition	Cycle Minimum Condition	Assumed Cycle Frequency
Startup (Max)	Unloaded	1 cycles/year
Startup (Max)	Shutdown (Min)	12.5 cycles/year
Preload	Shutdown (Min)	1000 cycles/year

The first cycle provides the greatest K_I range, which is between the startup transient and unloaded for BWRs. Because the maximum tensioner load does not result in the highest stud stress for BWRs, it is not used. The next major cycle to consider is between the maximum of the startup transient and the minimum of the shutdown transient.

To bound the remaining transients, the cycle between the preload stress state and the minimum of the shutdown transient is used. This cycle results in greater crack growth than the cycle between startup (max) and operation.

4.3.2 Fatigue Crack Growth

Using the load cycles described in Section 4.3.1, the change in crack tip SIF (ΔK_I) for each transient was calculated using an initial flaw size, a, of 0.05 times the 6.0-inch bolt diameter, or 0.30 inch (7.6 mm). This flaw size is consistent with the reference flaw depth recommended in WRCB 175 [36] for bolts greater than 3 inches in diameter, and it is substantially larger than the 0.157-inch (4 mm) minimum detectable flaw size for inspection. Additionally, an initial postulated flaw depth of $0.05d$ is consistent with the flaw size where fatigue cracking for bolted joints changes from nucleation to growth, as identified in testing performed in Reference [37].

The fatigue crack growth rate was taken from ASME Code, Section XI, Nonmandatory Appendix A, A-4300 for carbon and low alloy ferritic steels exposed to air environments. Using the relationship between K_I and flaw depth (a) developed in Section 4.1 for the different load cases considered, the $K_{I,max}$ and $K_{I,min}$ values were calculated for use with the A-4300 equations to calculate the fatigue crack growth per cycle defined in Section 4.3. The crack growth per cycle was multiplied by the applied number of cycles per year identified in Section 4.3.1 to compute the incremental flaw depth. This process was iterated to calculate the flaw depth as a function of time.

The results of the crack growth calculations for the PWR and BWR load cases are presented graphically in Figure 4-6 and Figure 4-7 respectively, which show the initial flaw size, the maximum allowable PWR flaw size and the maximum allowable BWR flaw size from Section 4.2. The results of the calculations indicate the following:

- The postulated flaw in a PWR RPV stud reaches 0.445 inches (11.3 mm) after 80 years of operation, which is less than the maximum allowable size of 1.06 inches (26.9 mm)
- The postulated flaw in a BWR RPV stud reaches the maximum allowable size of 0.789 inches (20.0 mm) after 37.9 years

Table 4-1
Influence coefficients for flaws in stud threads

a / d Flaw Depth / Stud Diameter	F_t Axial Stress Coefficient	F_b Bending Stress Coefficient
0.025	1.6008	0.9918
0.05	1.1088	0.7128
0.1	0.95	0.61
0.2	0.9	0.54
0.3	0.98	0.55
0.4	1.29	0.64
0.5	2.05	0.84

$F_t = K_{It} / (\sigma_m\sqrt{\pi a})$ where σ_m is the stud membrane stress and a is the flaw depth

$F_b = K_{Ib} / (\sigma_b\sqrt{\pi a})$ where σ_b is the stud bending stress and a is the flaw depth

$K_I = K_{It} + K_{Ib}$

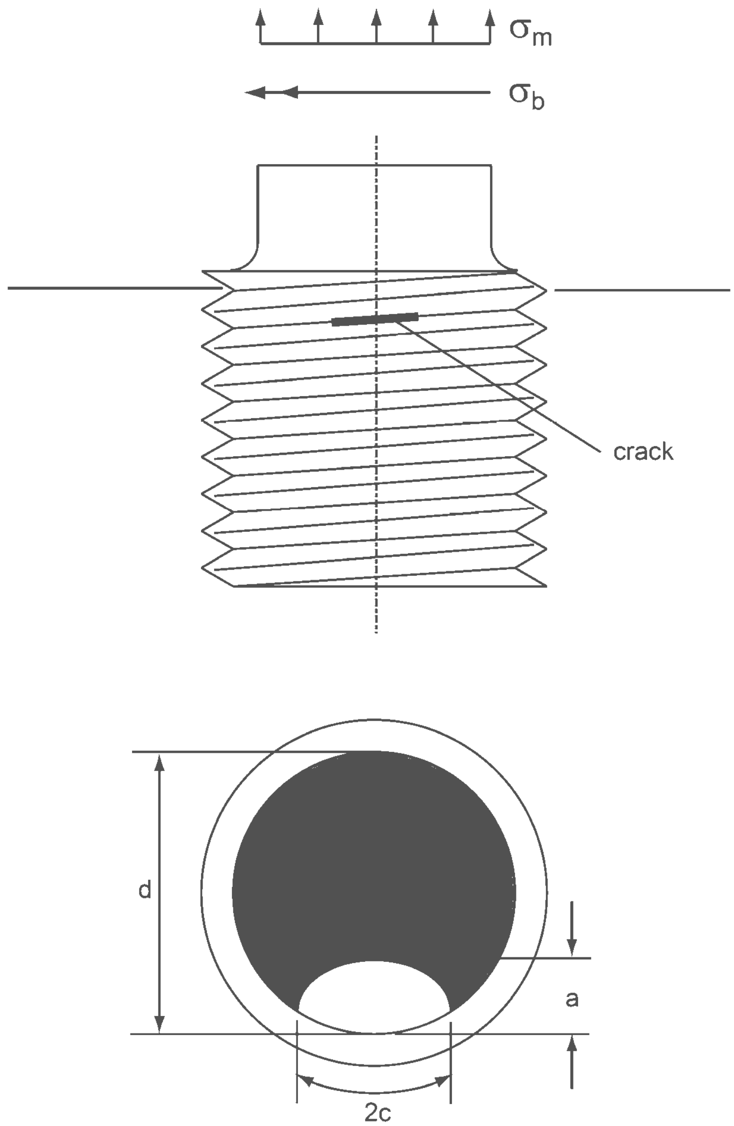

Figure 4-1
Stress intensity factor solution for fasteners flaw geometry

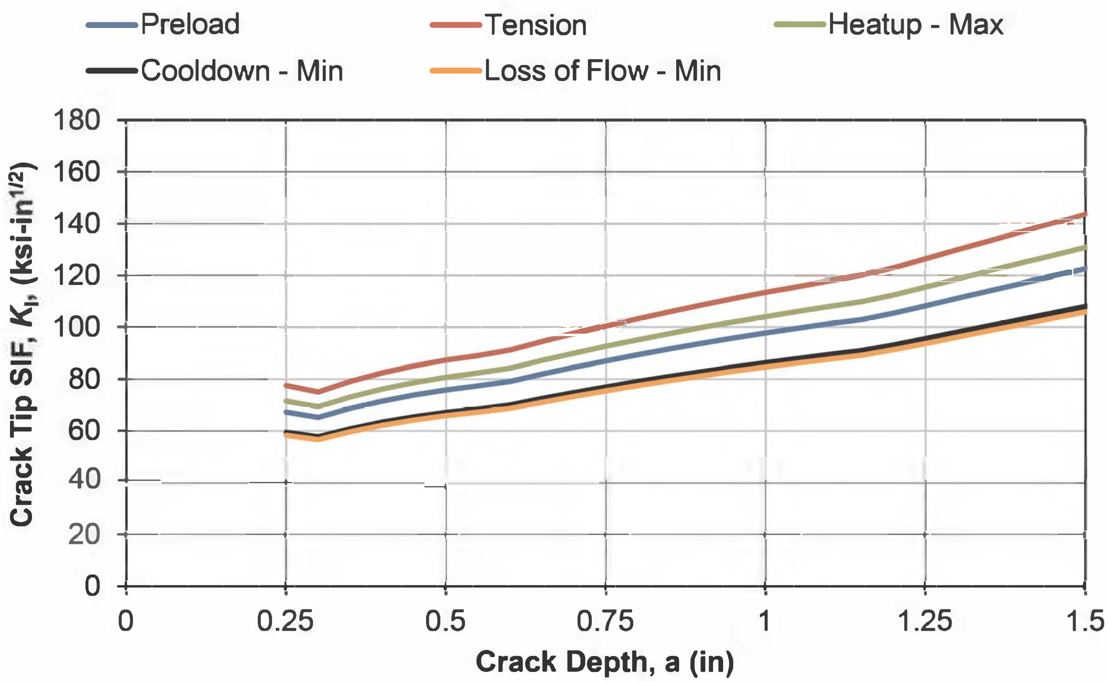

Figure 4-2
Stress intensity factor (K_I) as function of crack depth, for a postulated crack at the thread root in a PWR RPV stud

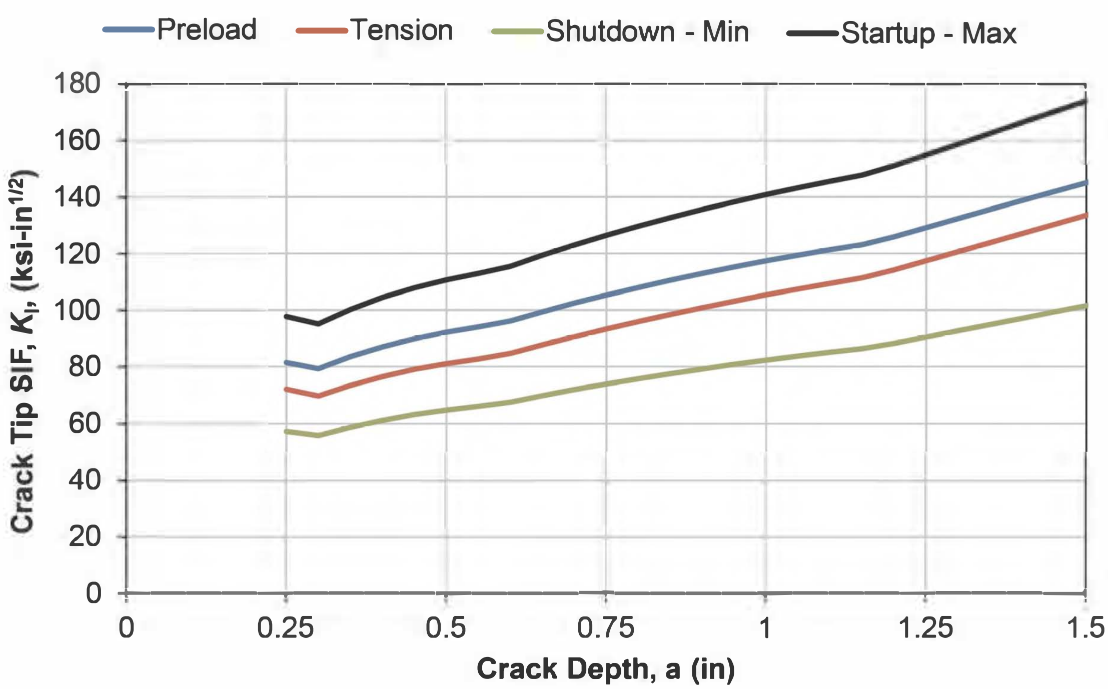

Figure 4-3
Stress intensity factor (K_I) as function of crack depth, for a postulated crack at the thread root in a BWR RPV stud

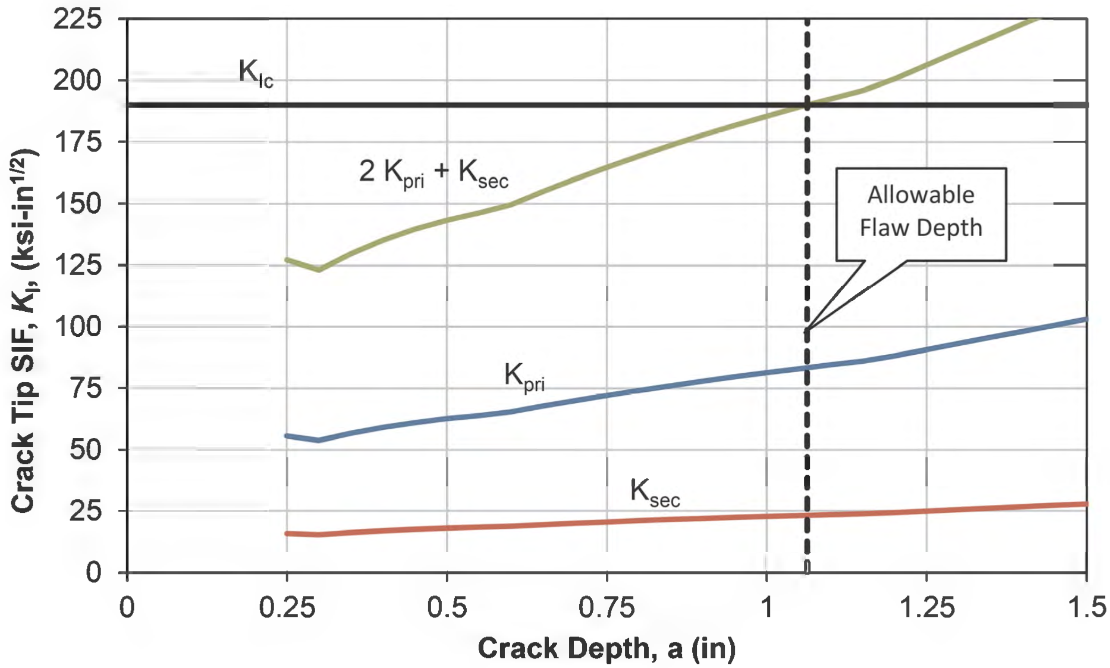

Figure 4-4
Limiting flaw size calculation – PWR RPV stud

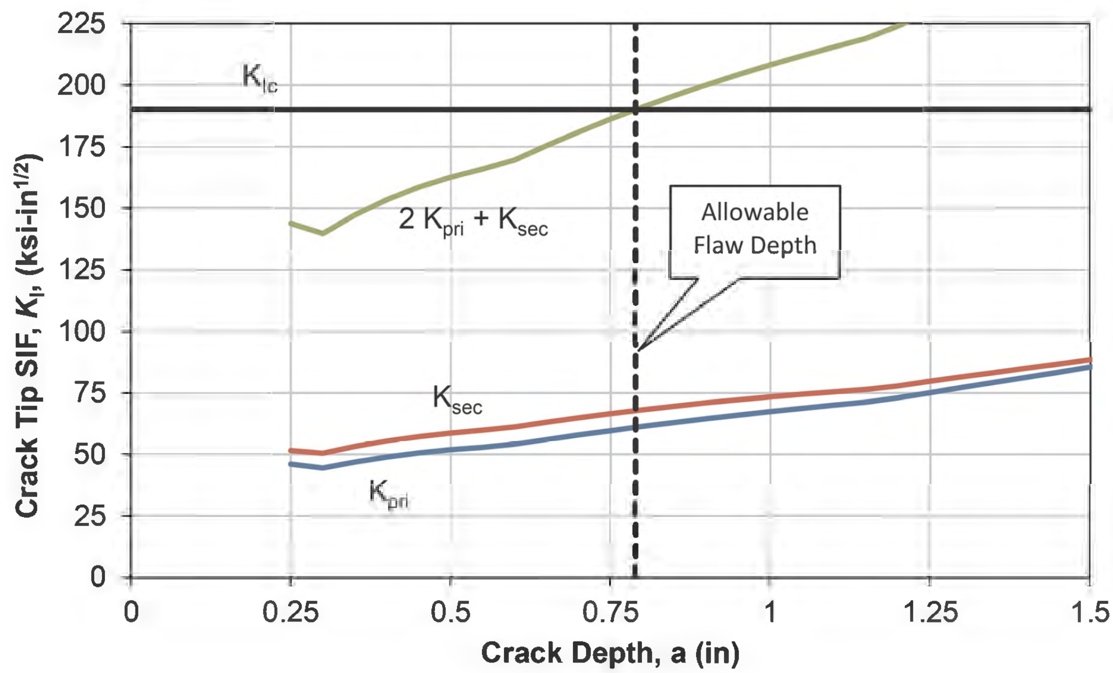

Figure 4-5
Limiting flaw size calculation – BWR RPV stud

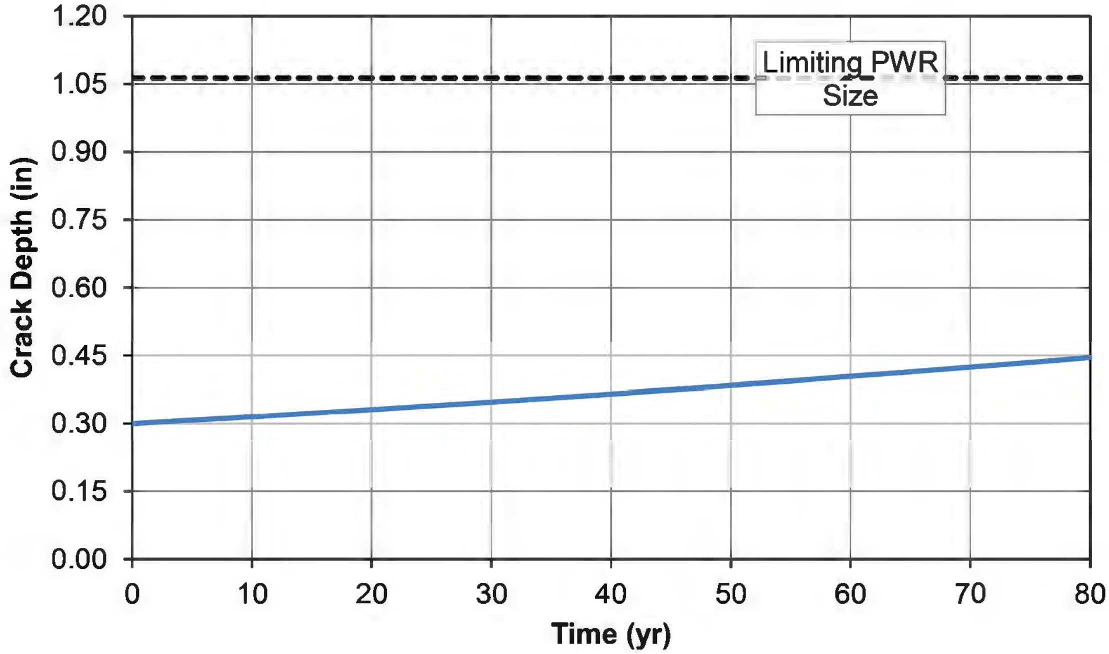

Figure 4-6
Fatigue crack growth evaluation results – PWR RPV studs

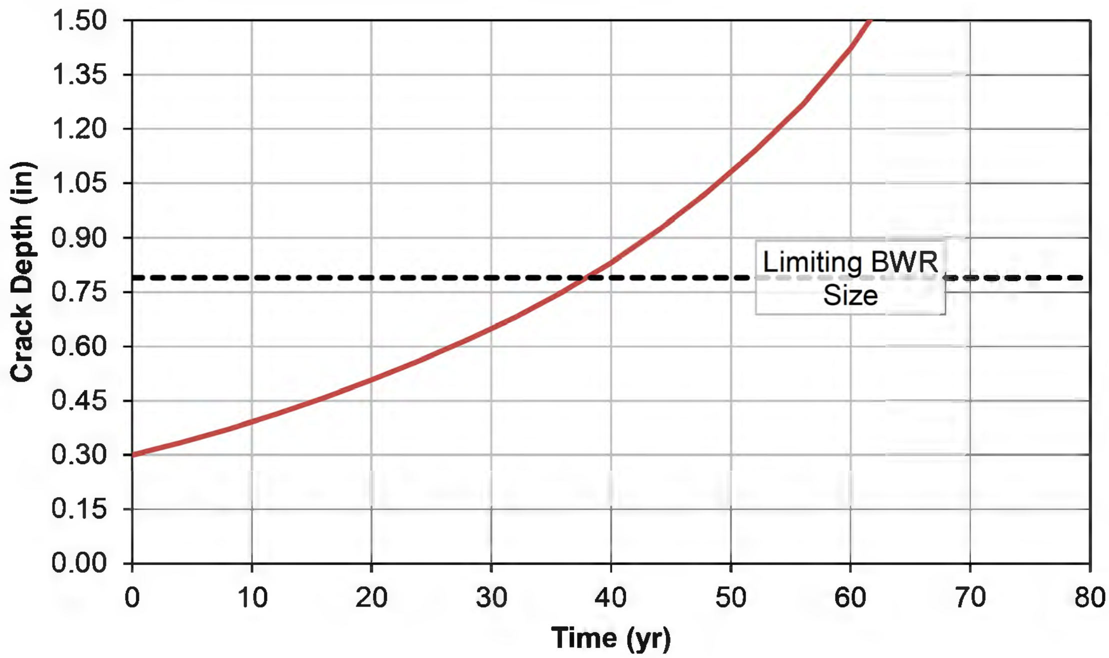

Figure 4-7
Fatigue crack growth evaluation results – BWR RPV studs

5 CONCLUSIONS

This report develops a basis for optimizing the inspection requirements of ASME Code, Section XI, Examination Category B-G-1, Item No. B6.20, "Closure Studs" of reactor pressure vessels. The report evaluates potential degradation mechanisms, and provides a detailed technical evaluation for fatigue crack growth in RPV studs. While the technical basis is oriented towards ASME Code Section XI requirements, the analysis approach and results have merit as a standalone technical position. International utilities that use different governing codes and standards for inspections should evaluate how to use this report in conjunction with those standards and regulatory obligations.

5.1 Summary of Technical Evaluation

A review of all relevant degradation mechanisms for RPV studs was performed. The review identified fatigue and SCC as possible aging mechanisms applicable to RPV studs in non-leaking RPV closures. Issues related to SCC have been addressed by preventative measures that limit the yield strength of stud material and by prohibiting use of lubricants that can promote SCC. These issues have been implemented by all U.S. licensees and are addressed in aging management programs documented in U.S. plant license renewal applications. Issues related to leaking connections, which could lead to boric acid corrosion and steam cutting, are readily identified by visual observation including operator observation and maintenance activities.

Accordingly, fatigue is the degradation mechanism requiring continued aging management in the absence of leakage for studs. Flaw tolerance evaluations were therefore performed to estimate the growth of a postulated fatigue crack in PWR and BWR RPV studs to provide a technical basis for modifying the required inspection frequency for these items. Stress analyses of bounding PWR and BWR RPV closures were performed that included the effects of boltup and typical operating transients. These analyses provided RPV stud membrane and bending stresses for the different transients considered. The stud stresses were used in fracture mechanics evaluations to calculate crack tip SIF values for postulated surface cracks in the RPV studs originating at the location of highest stress. The change in crack tip SIF values were used to perform fatigue crack growth calculations using appropriately bounding or representative load combinations and cycles.

The flaw tolerance calculations summarized in this report identified the following conclusions:

- The fatigue crack growth for PWR RPV studs for a postulated flaw after 80 years of assumed loading cycles remains less than the allowable flaw size using methods consistent with ASME Code, Section XI, Nonmandatory Appendix G.
- The fatigue crack growth for BWR RPV studs for a postulated flaw after 37.9 years of assumed loading cycles remains less than the allowable flaw size using methods consistent with ASME Code, Section XI, Nonmandatory Appendix G.

- For PWR RPV studs, the stud tensioning process results in significantly higher applied crack tip SIF values than normal operating loads. Therefore, the tensioning process for PWR RPV studs is effectively a "proof test," providing an additional measure of confidence in the integrity of the tensioned studs. Due to their higher bending stresses during normal operating conditions, the tensioning process for BWR RPV studs is not the limiting condition and does not act similarly as a proof test.

Furthermore, while not evaluated here, it is noted that analyses have been performed for many units to demonstrate that the RPV head remains in compliance with ASME Code, Section III requirements, as well as maintaining leak-tight conditions, following failure of any single RPV stud and select combinations of multiple studs. An example of these calculations is described in Reference [4].

5.2 Criteria for Technical Basis Applicability

This report develops a technical basis using inputs that are designed to evaluate the applicability range of conditions experienced at operating reactors. Consequently, the applicability of this technical basis is contingent on the conditions at a given plant being consistent with key criteria that determine whether the results of this analysis bounds actual plant operation. When assessing plant-specific applicability using this technical basis, owners should establish that the RPV closure region meets the following criteria:

- The RPV head inner radius-to-head thickness ratio is smaller than 34.8 for BWRs and 14.5 for PWRs, which is the bounding ratio used in the underlying evaluation (see Table 3-2).
- The RPV head shell inner radius-to-stud diameter ratio is smaller than 22.4 for BWRs and 14.0 for PWRs, which is the bounding ratio used in the underlying evaluation (see Table 3-2).
- The applicable transients are bounded by the transients shown in Appendix A.
- The number of transients projected through the end of the applicable operating period is less than the number of transients identified in Section 4.3.1.
- All RPV studs remain in service and are successfully tensioned.
- All RPV studs are fabricated from material with a yield strength of less than or equal to 150 ksi (165 MPa).
- RPV studs are specified as SA-540, Grades B23 or B24 material, or the RPV stud material specification is consistent with all SA-540 Grade B23/B24 requirements.
- No leakage from the RPV flange has been observed since the most recent volumetric/surface examination.

A plant not meeting any of the above criteria should perform a plant specific evaluation to define the deviation from the above criteria and to evaluate the impact of the deviation.

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A
LISTING OF TRANSIENTS

This appendix presents pressure and temperature profiles for all transients applied to the limiting geometry FEA models.

A summary listing of the transients considered is provided in Table A-1. The BWR transient profiles are shown in Figure A-1 through Figure A-5, and the PWR transient profiles are shown in Figure A-6 through Figure A-17.

Table A-1
Summary of transients

Transient Case #	BWR Case	PWR Case
1	Preload (*Steady State*)	Preload (*Steady State*)
2	Hydrotest (*Steady State*)	Hydrotest (*Steady State*)
3	Operation (*Steady State*)	Operation (*Steady State*)
4	Inner Seal Leakage (*Steady State*)	Heatup
5	Startup	Cooldown
6	Shutdown	Plant Loading
7	Loss of Feedwater Pumps	Plant Unloading
8	Pre-Op Blowdown	Small Step Increase
9	-	Small Step Decrease
10	-	Large Step Decrease
11	-	Steady State Fluctuations
12	-	Loss of Load
13	-	Loss of Flow
14	-	Reactor Trip

A.1 BWR Transients

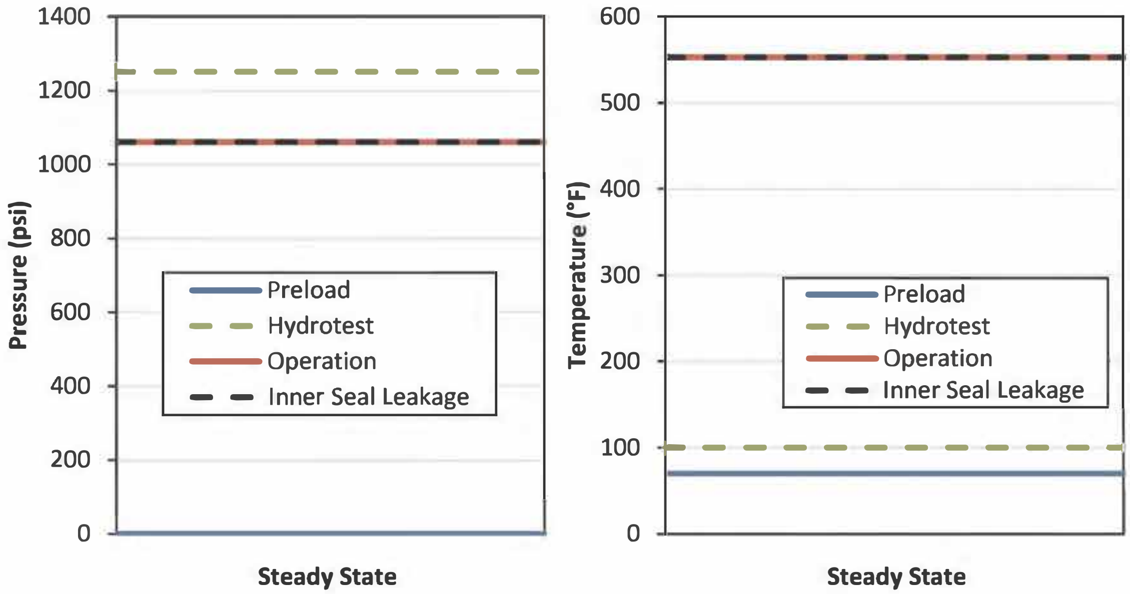

Figure A-1
Steady state cases

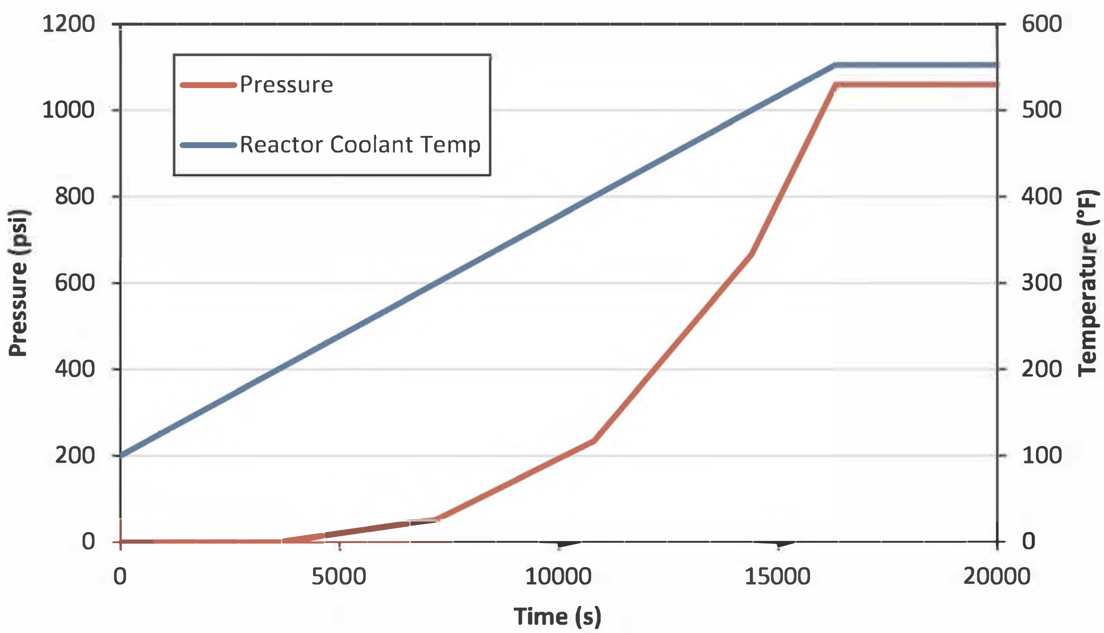

Figure A-2
Startup transient

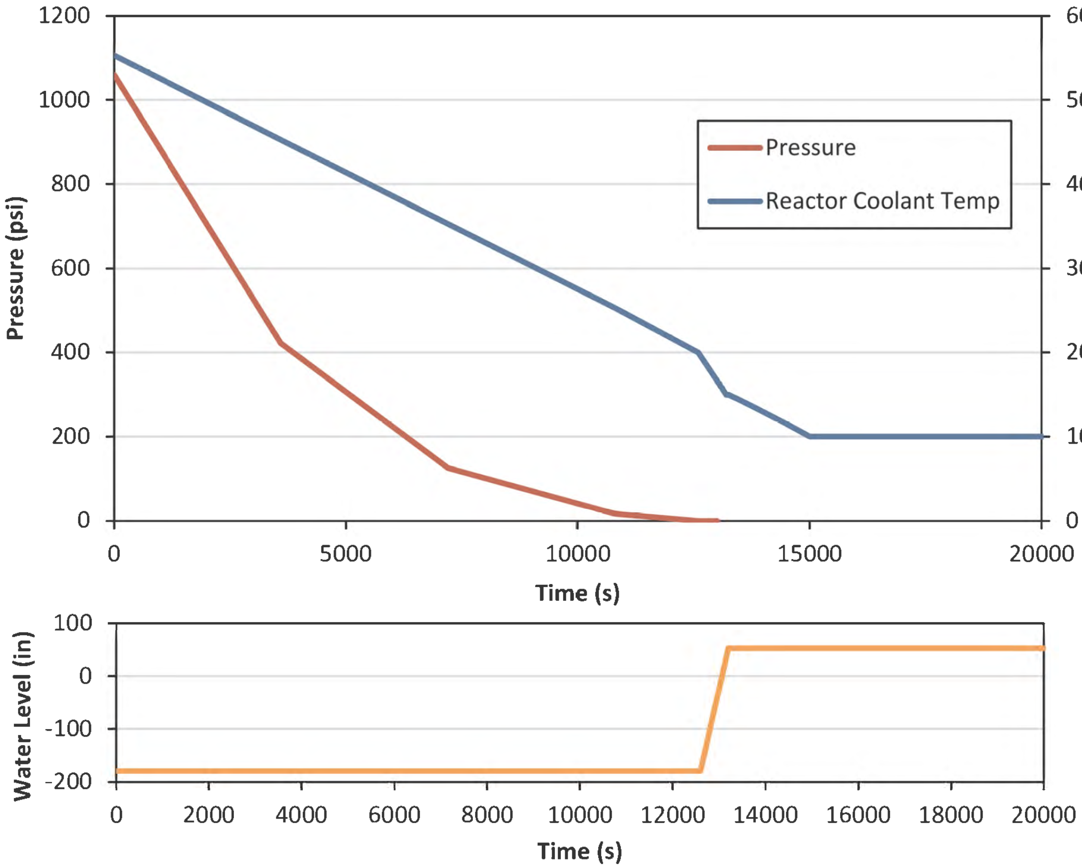

Figure A-3
Shutdown transient

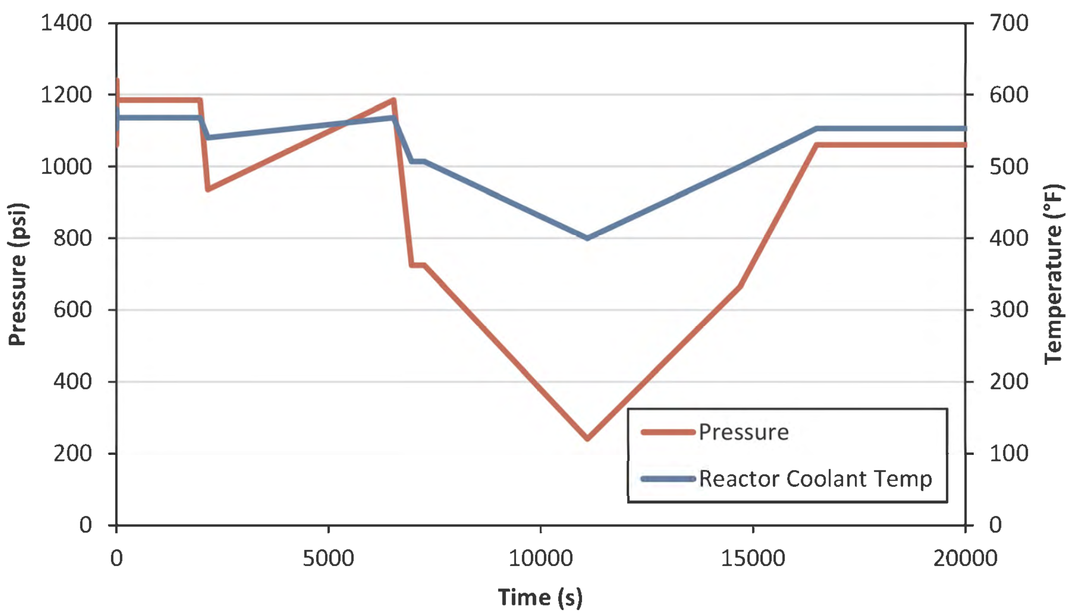

Figure A-4
Loss of feedwater pumps transient

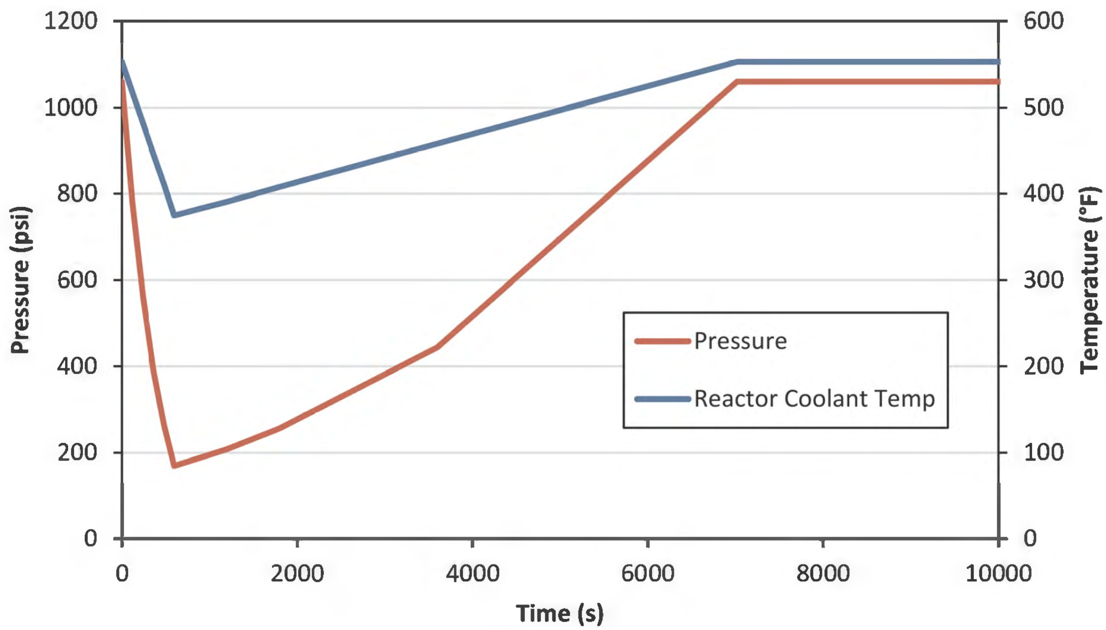

Figure A-5
Pre-operation blowdown transient

A.2 PWR Transients

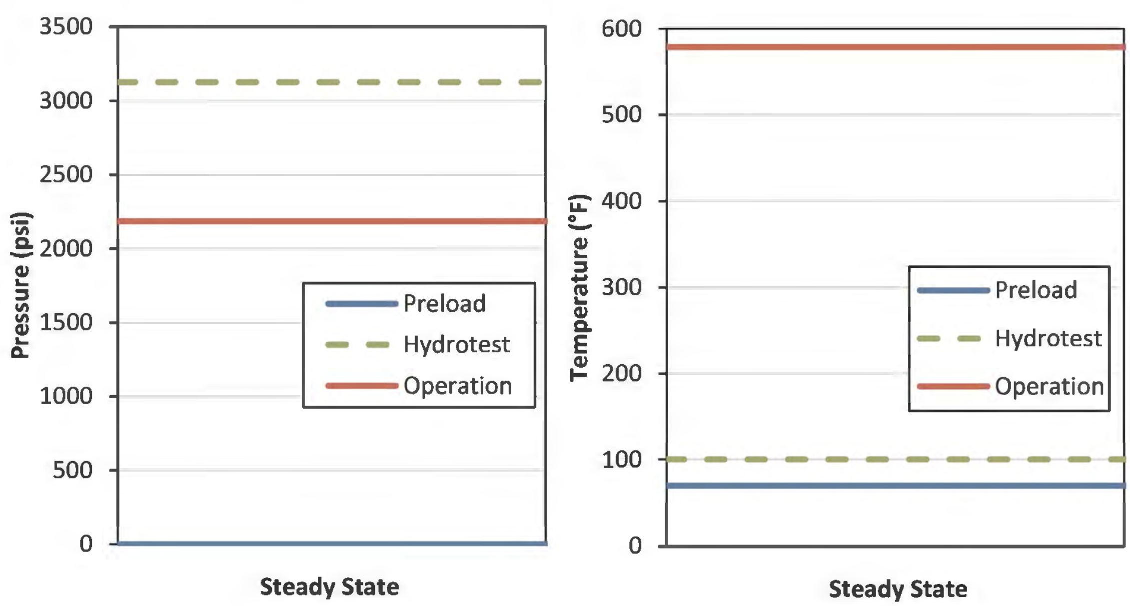

Figure A-6
Steady state cases

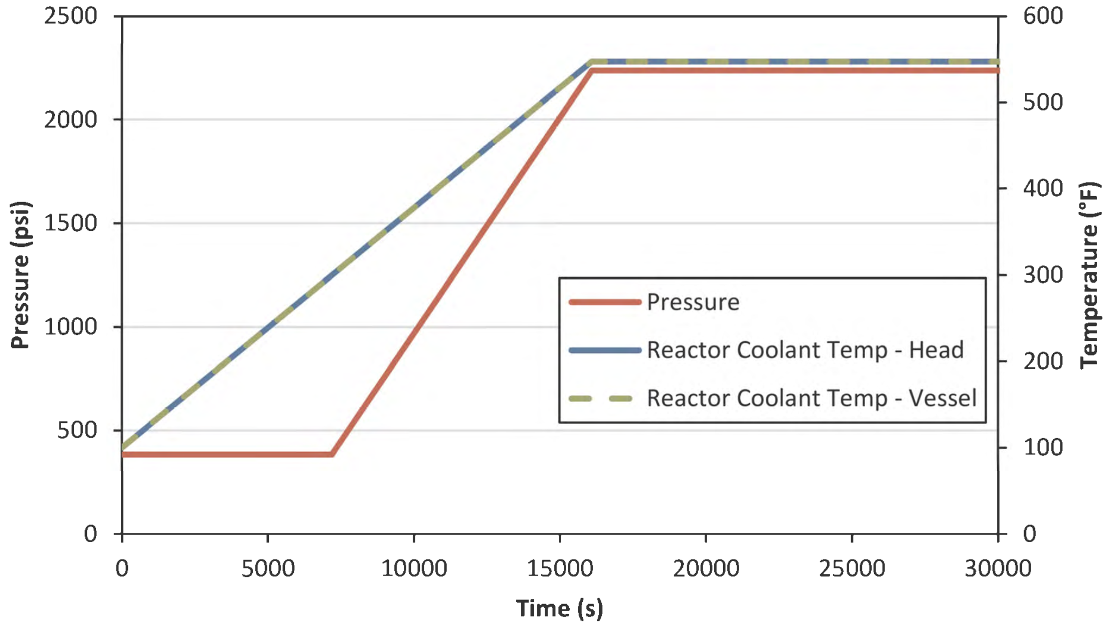

Figure A-7
Heatup transient

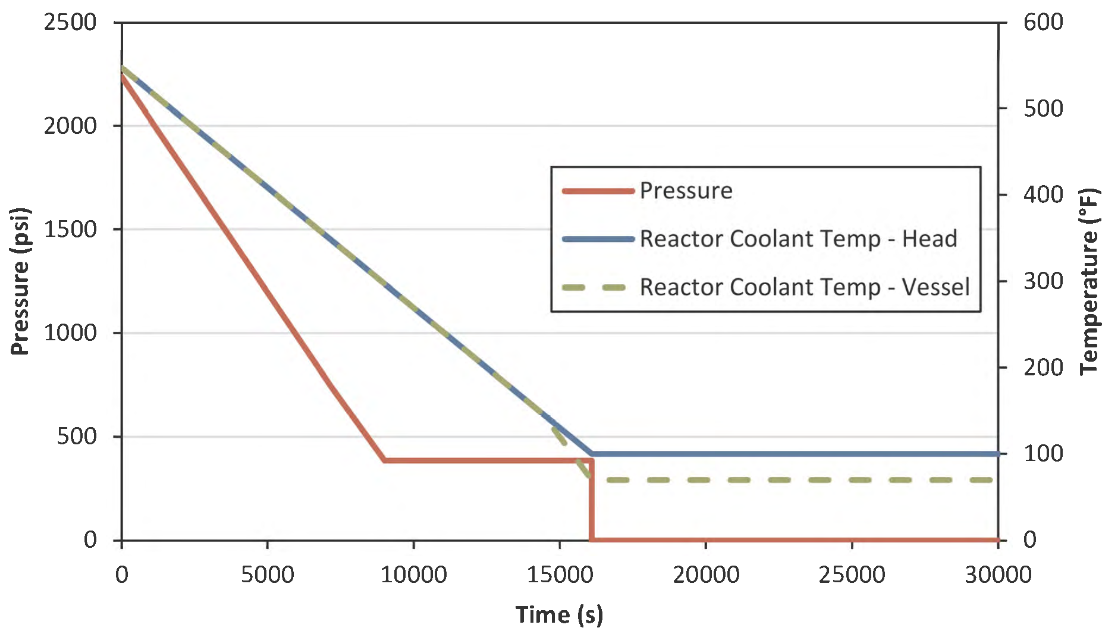

Figure A-8
Cooldown transient

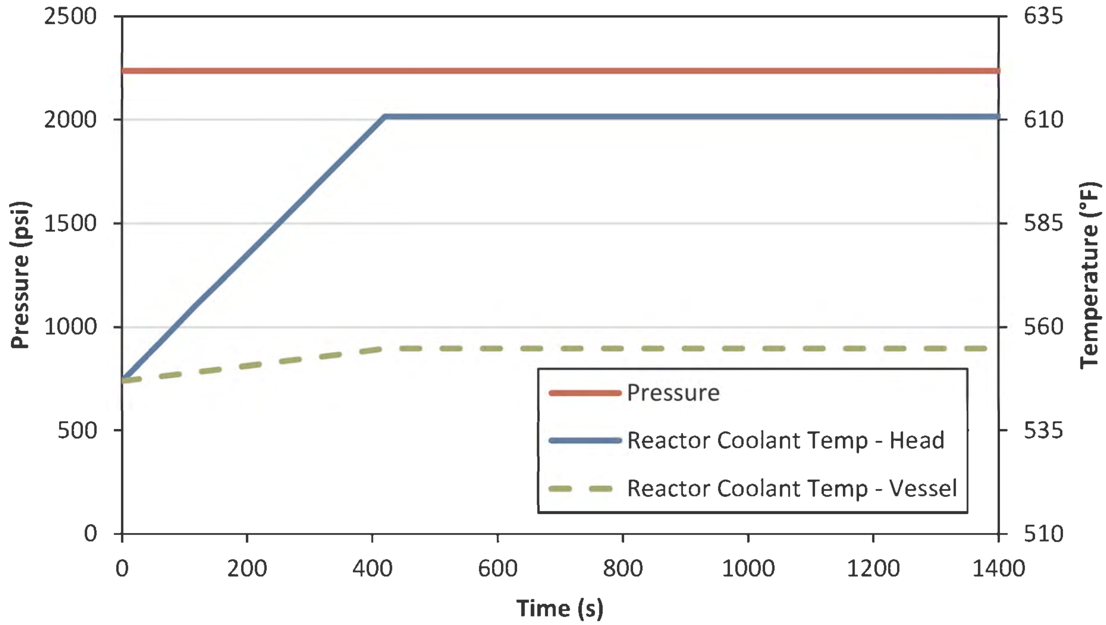

Figure A-9
Plant loading transient

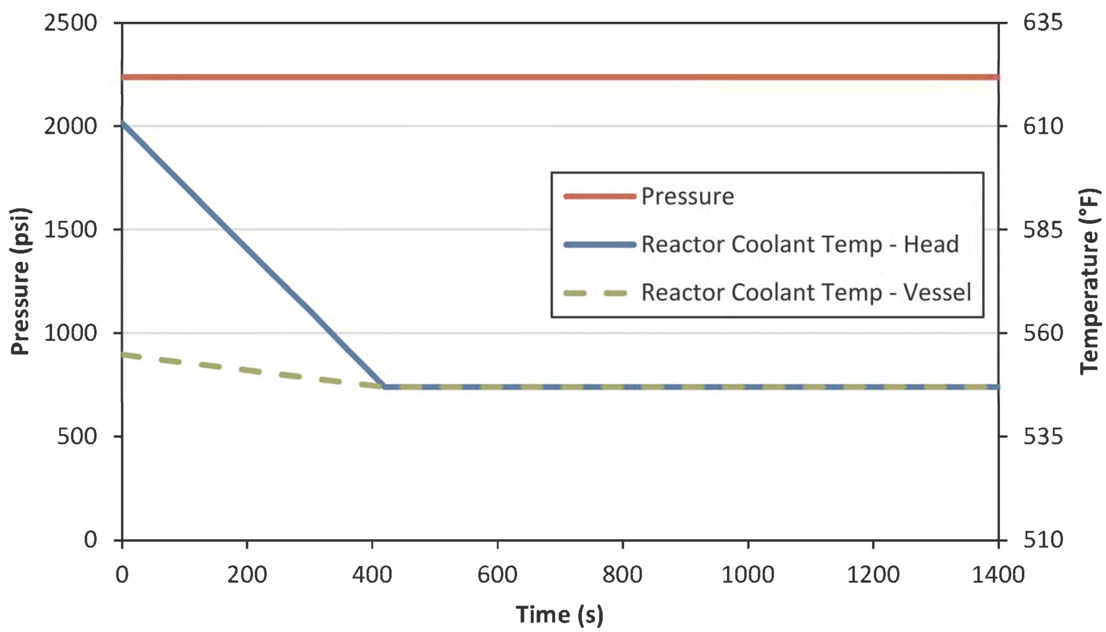

Figure A-10
Plant unloading transient

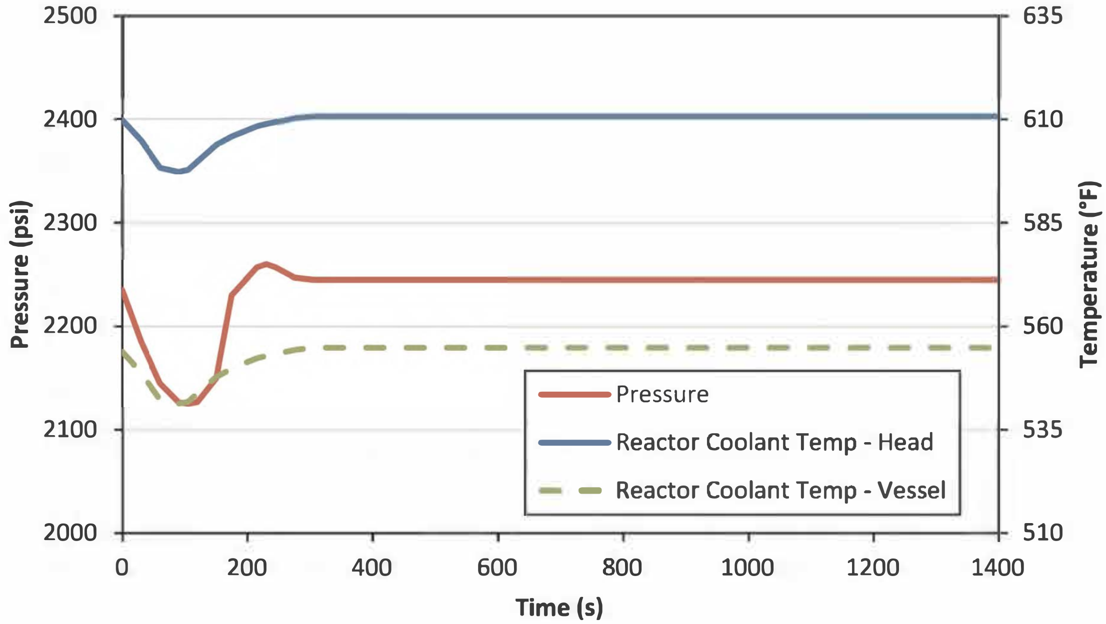

Figure A-11
Small step increase transient

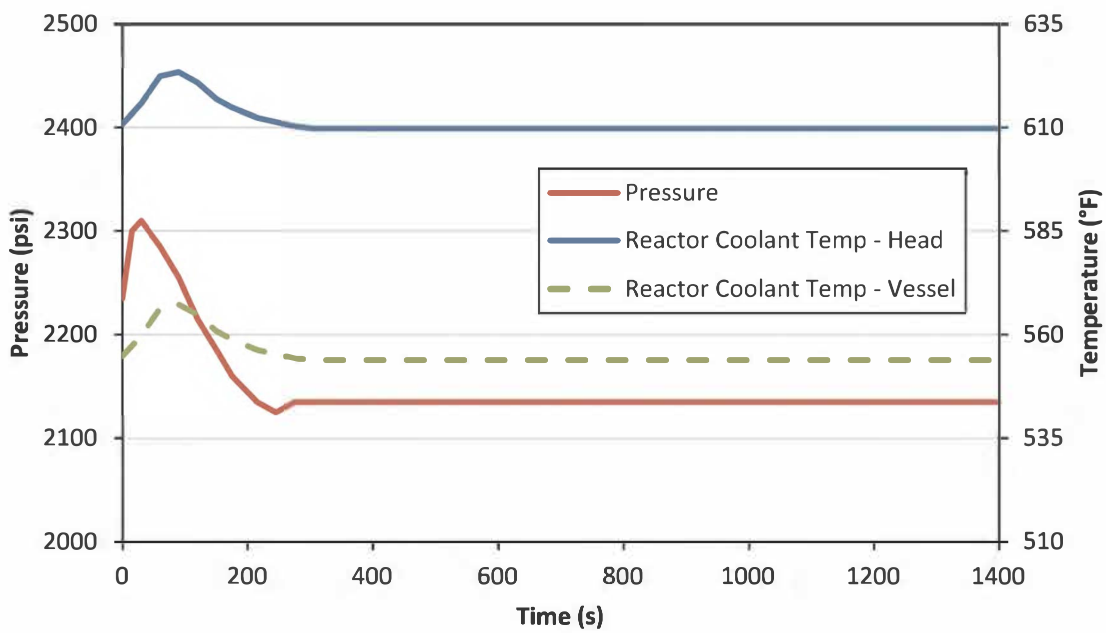

Figure A-12
Small step decrease transient

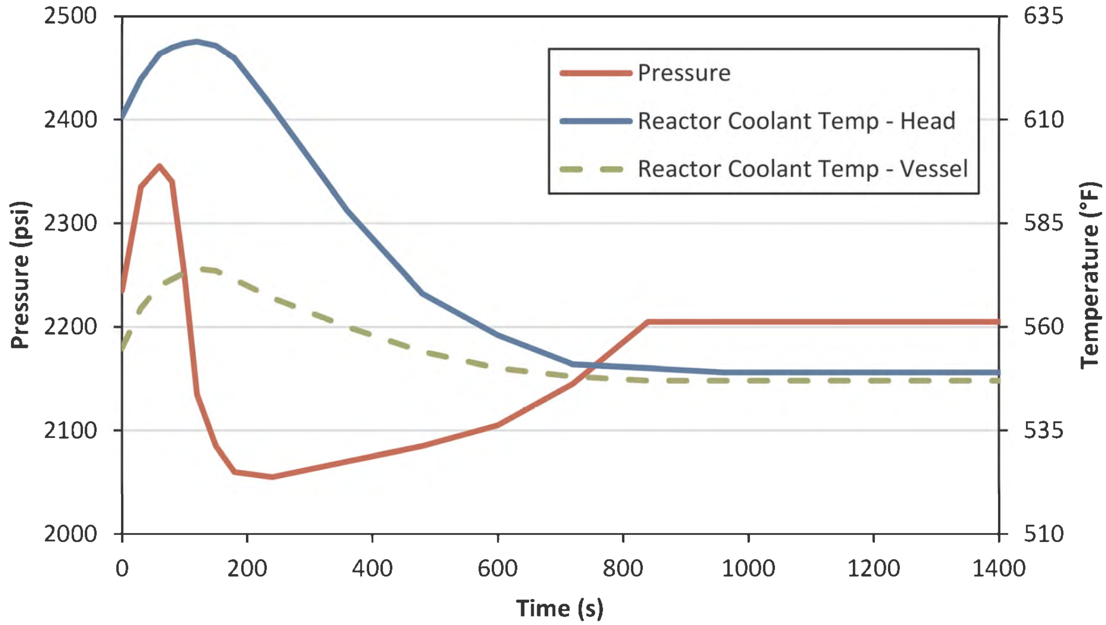

Figure A-13
Large step decrease transient

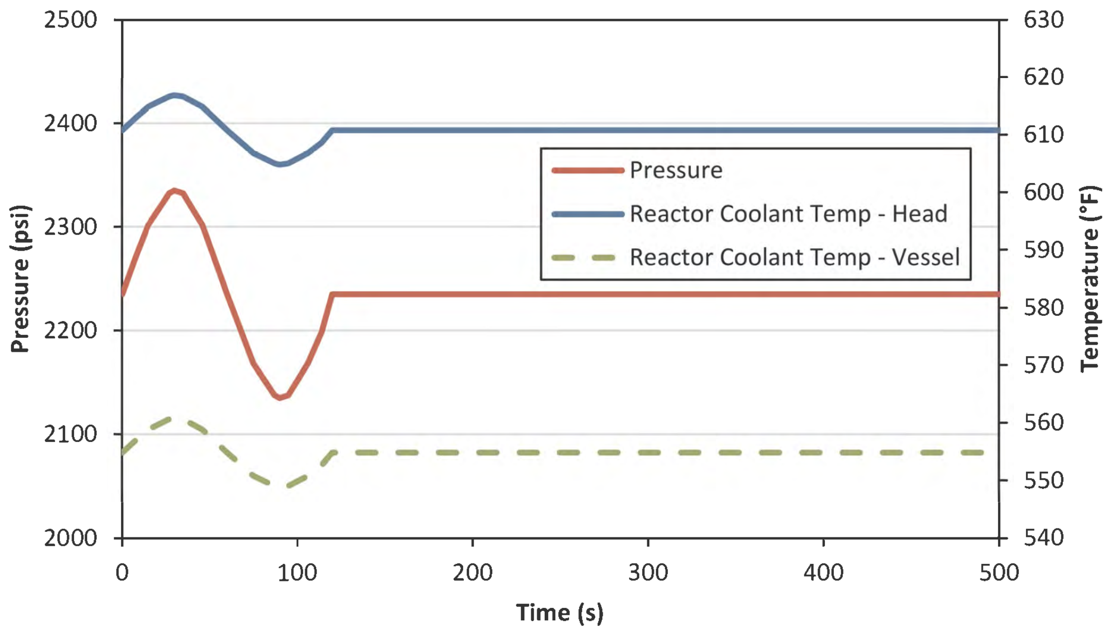

Figure A-14
Steady state fluctuation transient

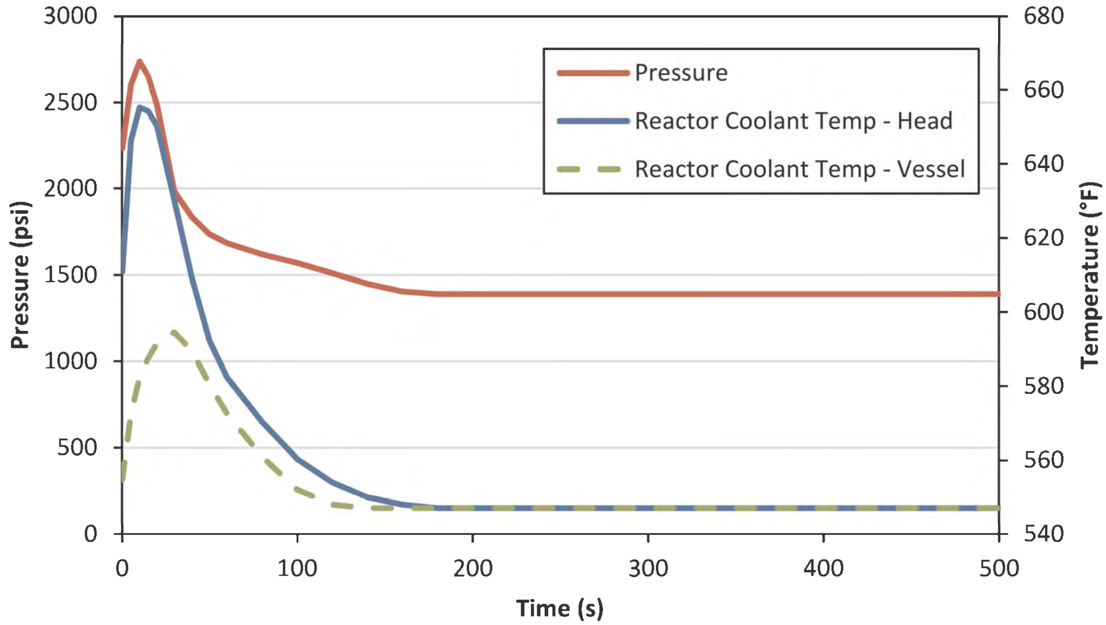

Figure A-15
Loss of load transient

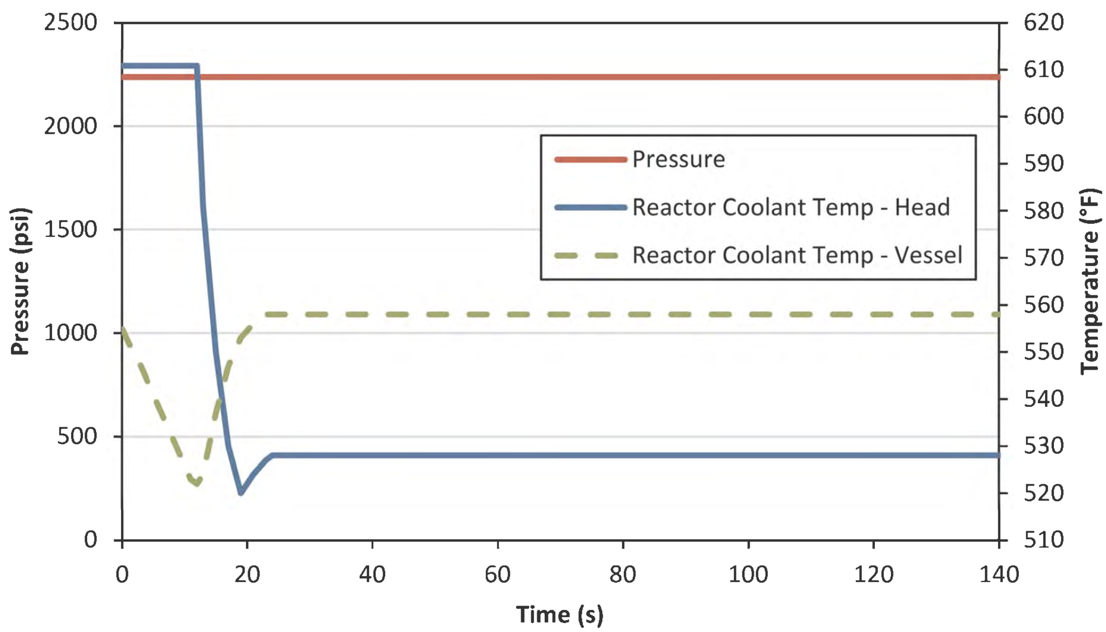

Figure A-16
Loss of flow transient

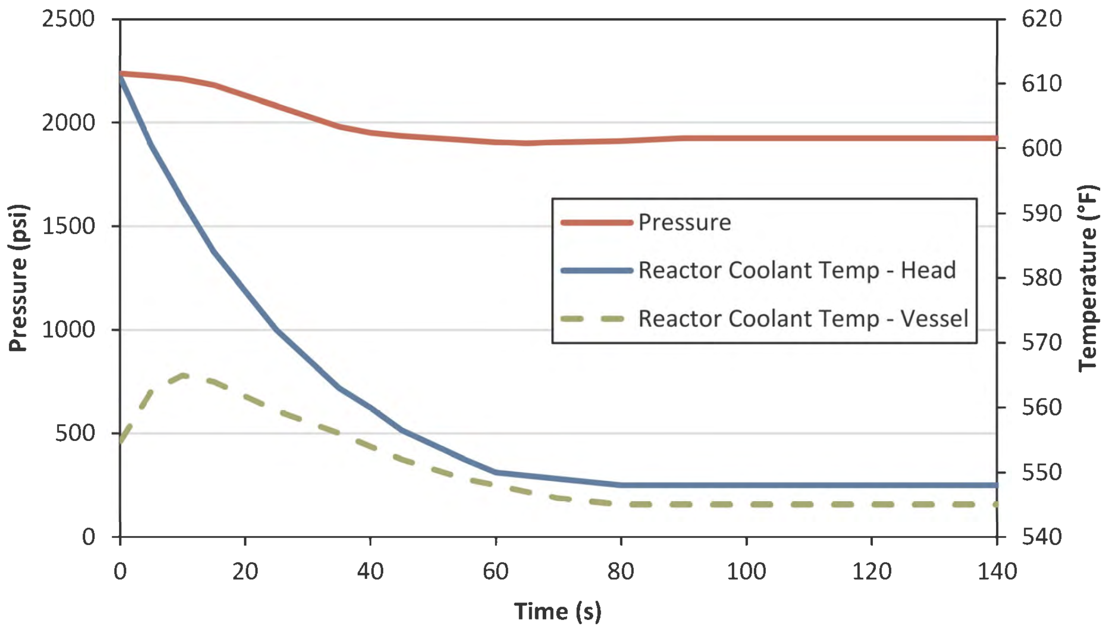

Figure A-17
Reactor trip transient