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10 CFR 50.55a

2CAN021803

February 5, 2018

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

SUBJECT: Request for Alternative Examination of Reactor Vessel Flange Threads
Request ANO2-ISI-020
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(z)(1), Entergy Operations, Inc. (Entergy) requests approval from the NRC for an alternative to the examination of American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel Code, Section XI, Examination Category B-G-1, Item Number B6.40, Threads in Flange, at Arkansas Nuclear One, Unit 2 (ANO-2). The attachment provides details regarding this request.

In lieu of the inservice requirements for a volumetric ultrasonic examination, Entergy proposes that the alternative provides an acceptable technical basis for eliminating the requirement for this examination because the alternative maintains an acceptable level of quality and safety.

The NRC has approved the use of this alternative for the Southern Company, Exelon Generating Company, and Duke Energy. Details of these approvals are provided in the attachment.

This request is for the Fourth 10-year interval. This interval began March 26, 2010, and will end on March 25, 2020.

There are no new regulatory commitments made in this submittal.

A047
NRR

If you have any questions or require additional information, please contact me.

Sincerely,

SLP/rwc

Attachment: Request for Alternative Examination of Reactor Vessel Flange Threads - ANO2-ISI-020

cc: Mr. Kriss M. Kennedy
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Attachment to

2CAN021803

Request for Alternative Examination of Reactor Vessel Flange Threads

ANO2-ISI-020

REQUEST FOR ALTERNATIVE EXAMINATION OF REACTOR VESSEL FLANGE THREADS

ANO2-ISI-020

1. ASME Code Component(s) Affected

Includes all American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel (BPV) Code, Section XI, Examination Category B-G-1, Item Number B6.40 threads in flange locations at Arkansas Nuclear One, Unit 2 (ANO-2) as follows:

ASME Code Class:	1
Examination Category:	Table IWB-2500-1, Examination Category B-G-1, Pressure Retaining Bolting, Greater Than 2 in. In Diameter
Item Number:	Reactor Vessel B6.40, Threads In Flange (54 Items See Table 1)
Unit / Inspection Interval Applicability:	Arkansas Nuclear One, Unit 2 / Fourth (4th) 10-year interval (March 26, 2010 to March 25, 2020)

2. Applicable Code Edition and Addenda

The ANO-2 Fourth 10-Year Interval Inservice Inspection (ISI) Program is currently written to the 2001 Edition through the 2003 Addenda. However, Nondestructive Examination (NDE), Pressure Testing (PT) and Repair and Replacements (R&R) activities are now being performed to the 2007 Edition through the 2008 Addenda of Section XI per an authorized alternative (Reference 1). Therefore, the current volumetric examination that is required for the Reactor Pressure Vessel (RPV) flange threaded stud holes is performed in accordance with NDE that meets the 2007 Edition through the 2008 Addenda.

3. Applicable ASME Code Requirements

The RPV threads in flange, Examination Category B-G-1, Item Number B6.40, are examined using a volumetric examination technique with 100% of the flange threaded stud holes examined every ISI interval. The examination area is the one-inch area around each RPV stud hole, as shown on Figure IWB-2500-12.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(1), Entergy Operations, Inc. (Entergy) is requesting a proposed alternative from the requirement to perform inservice ultrasonic examinations of Examination Category B-G-1, Item Number B6.40, Threads in Flange. Entergy has worked with the industry to evaluate eliminating the RPV threads in flange examination requirement.

Licensees in the U.S. and internationally have worked with the Electric Power Research Institute (EPRI) to produce Technical Report No. 3002007626, "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements" (Reference 2), which provides the basis for elimination of the requirement. The report includes a survey of inspection results from over 168 units, a review of operating experience related to RPV flange/bolting, and a flaw tolerance evaluation. The conclusion from this evaluation is that the current requirements are not commensurate with the associated burden (worker exposure, personnel safety, radwaste, critical path time, and additional time at reduced water inventory) of the examination.

Additionally, in support of the EPRI Report Reference 2 above, ASME has developed, approved and published a Section XI, Code Case N-864, (Reference 3) that supports the same conclusion that the EPRI Report does and the code case states that the Section XI Code Requirement to examine the RPV threads in flange is not required. Understanding this code case has not yet been reviewed for acceptance by the NRC staff and is not listed in the latest revision of Regulatory Guide 1.147, Entergy does not plan to adopt this code case as part of this request at this time; however, if NRC approves this code case at a later date, it is planned to be used at ANO-2 in the future subject to any NRC conditions that would need to be addressed at that time.

The technical basis for this alternative is discussed in more detail below.

Potential Degradation Mechanisms

An evaluation of potential degradation mechanisms that could impact flange/threads reliability was performed as part of Reference 2. Potential types of degradation evaluated included pitting, intergranular attack, corrosion fatigue, stress corrosion cracking, crevice corrosion, velocity phenomena, dealloying corrosion and general corrosion, stress relaxation, creep, mechanical wear, and mechanical/thermal fatigue. Other than the potential for mechanical/thermal fatigue, there are no active degradation mechanisms identified for the threads in flange component.

The EPRI report notes a general conclusion from ASME Risk-Based Inspection: Development of Guidelines, Volume 2-Part 1 and Volume 2-Part 2, (Reference 4), which includes work supported by the NRC that when a component item has no active degradation mechanism present, and a preservice inspection has confirmed that the inspection volume is in good condition (i.e., no flaws/indications), then subsequent inservice inspections do not provide additional value going forward. As discussed in the Operating Experience review summary below, the RPV flange ligaments have received the required preservice examinations and over 10,000 inservice inspections have been performed, with no relevant findings.

To address the potential for mechanical/thermal fatigue, Reference 2 documents a stress analysis and flaw tolerance evaluation of the flange thread area to assess mechanical/thermal fatigue potential. The evaluation consists of two parts. In the first part, a stress analysis is performed considering all applicable loads on the threads in flange component. In the second part, the stresses at the critical locations of the component are used in a fracture mechanics evaluation to determine the allowable flaw size for the component as well as how much time it wouldl take for a postulated initial flaw to grow to the allowable flaw size using guidelines in the ASME Code, Section XI, IWB-3500. The Pressurized Water Reactor (PWR) design was selected because of its higher design pressure and temperature. A representative geometry for

the finite element model used the largest PWR RPV diameter along with the largest bolts and the highest number of bolts. The larger and more numerous bolt configuration results in less flange material between bolt holes, whereas the larger RPV diameter results in higher pressure and thermal stresses.

Stress Analysis

A stress analysis was performed in Reference 2 to determine the stresses at critical regions of the threads in flange component as input to a flaw tolerance evaluation. Sixteen nuclear plant units [ten PWRs and six Boiling Water Reactors (BWRs)] were considered in the analysis. The evaluation was performed using a geometric configuration that bounds the sixteen units considered in this effort. The details of the RPV threads in flange parameters for ANO-2 as compared to the bounding values used in the evaluation are shown in Table 1. Additional information further demonstrating the applicability of the generic stress analysis and flaw tolerance evaluation is contained in Tables 2 and 4.

Table 1: Comparison of ANO2 to Bounding Values Used in Analysis

Plant	No. of Studs Currently Installed	Minimum No. of Studs Evaluated (Note 1)	Stud Nominal Diameter (inches)	RPV Inside Diameter at Stud Hole (inches)	Flange Thickness at Stud Hole (inches) (Note 2)	Design Pressure (psia)	Preload Stress (psi) (Note 3)
ANO-2	54	54	6.25	166.658	14.92	2500	36,210
Range for 16 Units Considered	N/A	54 - 60	6.0 - 7.0	157 - 173	15 - 16	2500	42,338
Bounding Values Used in Analysis	**54**	**N/A**	**6.0**	**173**	**16**	**2500**	**42,338**

Notes:

1. Applies only if the plant has been evaluated for operation with one or more non-operational studs.
2. Thickness is in a direction normal to the stud hole (ID to OD), see Figure 1.
3. Preload Stress is calculated using the following equation and inputs Bolt/stud preload – The ANO-2 preload on the bounding geometry is calculated as:

$$P_{preload} = \frac{C \cdot P \cdot ID^2}{s \cdot D^2} = \frac{1.1 \cdot 2500 \cdot 166.658^2}{54 \cdot 6.25^2} = 36{,}210 \text{ psi}$$

where:

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

The analytical model is shown in Figures 2 and 3. The loads considered in the analysis consisted of:

- A design pressure of 2500 psia at an operating temperature of 600 °F was applied to all internal surface exposed to internal pressure.
- Bolt/stud preload – The preload on the bounding geometry is calculated as:

$$P_{preload} = \frac{C \cdot P \cdot ID^2}{s \cdot D^2} = \frac{1.1 \cdot 2500 \cdot 173^2}{54 \cdot 6^2} = 42{,}388 \text{ psi}$$

where:

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

- Thermal stresses – The only significant transient affecting the bolting flange is heat-up/cooldown. This transient typically consists of a steady 100 °F/hour ramp up to the operating temperature, with a corresponding pressure ramp up to the operating pressure.

The ANSYS finite element analysis program was used to determine the stresses in the threads in flange component for the three loads described above as shown in Figures 2 and 3.

Additional design details associated with the RPV threads in flange at ANO2 are shown in Table 2 below:

Table 2: ANO2 RPV Threads in Flange Design Details				
Plant	**Thread Specification**	**Nominal Bolt Hole Diameter in Flange (inches)**	**Pitch**	**Thread Depth (inches)**
ANO-2	6.5"- 8N-2A	6.75	8	0.0759

Flaw Tolerance Evaluation

A flaw tolerance evaluation was performed using the results of the stress analysis in the EPRI Report to determine how long it would take an initial postulated flaw to reach the ASME Code, Section XI, allowable flaw size. A linear elastic fracture mechanics evaluation consistent with ASME Code, Section XI, IWB-3600, was performed.

Stress intensity factors (Ks) at four flaw depths of a 360° inside-surface-connected, partial-through-wall circumferential flaws are calculated using finite element analysis techniques with the model described above. The maximum stress intensity factor (K) values around the bolt hole circumference for each flaw depth (*a*) are extracted and used to perform the crack growth calculations. The circumferential flaw is modeled to start between the 10th and 11th flange threads from the top end of the flange because that is where the largest tensile axial stress occurs. The modeled flaw depth–to–wall thickness ratios (a/t) are 0.02, 0.29, 0.55, and 0.77, as measured in any direction from the stud hole. This creates an ellipsoidal flaw shape around the circumference of the flange, as shown in Figure 4 for the flaw model with a/t = 0.77 a/t crack model. The crack tip mesh for the other flaw depths follows the same pattern. When preload is not being applied, the stud, stud threads, and flange threads are not modeled. The model is otherwise unchanged between load cases.

The maximum K results are summarized in Table 3 for the four crack depths. Because the crack tip varies in depth around the circumference, the maximum K from all locations at each crack size is conservatively used for the *K vs. a* profile.

Table 3: Maximum K vs. a/t				
Load	K at Crack Depth (ksi√in)			
	0.02 a/t	0.29 a/t	0.55 a/t	0.77 a/t
Preload	11.2	17.4	15.5	13.9
Preload + Heatup + Pressure	13.0	19.8	16.1	16.3

The allowable stress intensity factor was determined based on the acceptance criteria in ASME Section XI, IWB-3610/Appendix A, which states that:

$K_I < K_{Ic}/\sqrt{10} = 69.6\ \text{ksi}\sqrt{\text{in}}$

Where,

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

K_{Ic} = Lower bound fracture toughness at operating temperature (220 ksi$\sqrt{\text{in}}$)

Additionally, when determining the preload as it applies to ANO-2 the parameters in Table 4 below apply.

Table 4: Additional Information					
Plant Name	Flange RT_{NDT} (°F)		Preload Temp (°F)	Minimum T-RT_{NDT} (°F) (Note 1)	Flange K_{1c} (ksi$\sqrt{\text{in}}$) (Note 2)
	From Plant Records	From NRC RVID2 Database (Note 3)			
ANO-2	10	N/A	$\geq$70	60	102.03

Notes:

1. T represents the minimum permissible flange temperature required for preload to the RPV head studs.
2. K_{1c} is determined in accordance with ASME Section XI, 2007 Edition with 2008 Addenda, Nonmandatory Appendix A, A-4200. K_{1c} is to be determined for the preload temperature listed in Table 3.
3. NRC RVID2 Database has no information for the subject RPV Flanges, which are not in the Vessel Beltline.

As can be seen from Table 3, the allowable stress intensity factor is not exceeded for all crack depths up to the deepest analyzed flaw of a/t = 0.77. Hence the allowable flaw depth of the 360° circumferential flaw is at least 77% of the thickness of the flange. The allowable flaw depth is assumed to be equal to the deepest modeled crack for the purposes of this analysis.

As seen from the stress intensity factor (K) calculation documented in Table 6-1 of Reference 2 (reproduced in Table 3 above), the maximum K is 19.8 ksi$\sqrt{\text{in}}$. The allowable K calculated in Section 6.2.2 of the report is 69.6 ksi$\sqrt{\text{in}}$, significantly higher than the calculated value. Assuming an RPV flange with 60 studs originally and one inoperable stud, the increase in K is about 1.7% resulting in a maximum K of about 20.14 ksi$\sqrt{\text{in}}$ which is still significantly less than the allowable value.

For the crack growth evaluation, an initial postulated flaw size of 0.2 in. (5.08 mm) is chosen consistent with the ASME Code, Section XI IWB-3500, flaw acceptance standards. The deepest flaw analyzed is a/t = 0.77 because of the inherent limits of the model. Two load cases are considered for fatigue crack growth: heat-up/cooldown and bolt preload. The heat-up/cooldown load case includes the stresses due to thermal and internal pressure loads and is conservatively assumed to occur 50 times per year. The bolt preload is assumed to be present and constant during the load cycling of the heat-up/cooldown load case. The bolt preload load case is conservatively assumed to occur five times per year, and these cycles do not include thermal or internal pressure. The resulting crack growth was determined to be negligible due to the small delta K and the relatively low number of cycles associated with the transients evaluated. Because the crack growth is insignificant, the allowable flaw size will not be reached and the integrity of the component is not challenged for at least 80 years (original 40-year design life plus additional 40 years of plant life extension).

The bounding stress analysis/flaw tolerance evaluation presented above shows that the threads in flange component at ANO-2 in this alternative request is very flaw tolerant and can operate for 80 years without violating ASME Code, Section XI, safety margins. This clearly demonstrates that the thread in flange examinations can be eliminated without affecting the safety of the RPV.

Operating Experience Review Summary

As discussed above, the results of the survey, which includes results from the Entergy plants in this request for alternative, confirmed that the RPV threads in flange examination are adversely impacting outage activities (worker exposure, personnel safety, radwaste, critical path time, and additional time at reduced water inventory) while not identifying any service induced degradations. Specifically, for the U.S. fleet, a total of 94 units have responded to date and none of these units have identified any type of degradation. As can be seen in Table 5 below, the data is encompassing. The 94 units represent data from 33 BWRs and 61 PWRs. For the BWR units, a total of 3,793 examinations were conducted and for the PWR units a total of 6,869 examinations were conducted, with no service-induced degradation identified. The response data includes information from all of the plant designs in operation in the U.S. and includes BWR-2, -3, -4, -5 and -6 designs. The PWR plants include the 2- loop, 3-loop and 4-loop designs and each of the PWR NSSS designs (i.e., Babcock & Wilcox, Combustion Engineering and Westinghouse).

Table 5: Summary of Survey Results – US Fleet

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

Related RPV Assessments

In addition to the examination history and flaw tolerance discussed above, Reference 2 discusses studies conducted in response to the issuance of the Anticipated Transient Without Scram (ATWS) Rule by the NRC. This rule was issued to require design changes to reduce expected ATWS frequency and consequences. Many studies have been conducted to understand the ATWS phenomena and key contributors to successful response to an ATWS event. In particular, the reactor coolant system (RCS) and its individual components were reviewed to determine weak links. As an example, even though significant structural margin was identified in USNRC SECY-83-293 for PWRs, the ASME Service Level C pressure of 3200 psig was assumed to be an unacceptable plant condition. While a higher ASME service level might be defensible for major RCS components, other portions of the RCS could deform to the point of inoperability. Additionally, there was the concern that steam generator tubes might fail before other RCS components, with a resultant bypass of containment. The key take-away from these studies is that the RPV flange ligament was not identified as a weak link and other RCS components were significantly more limiting. Thus, there is substantial structural margin associated with the RPV flange.

In summary, Reference 2 identifies that the RPV threads in flange are performing with very high reliability based on operating and examination experience. This is due to the robust design and a relatively benign operating environment (e.g., the number and magnitude of transients is small, generally not in contact with primary water at plant operating temperatures/pressures, etc.). The robust design is manifested in that plant operation has been allowed at several plants even with a bolt/stud assumed to be out of service. As such, significant degradation of multiple bolts/threads would be needed prior to any RCS leakage.

5. Proposed Alternative and Basis for Use

In lieu of the inservice requirements for a volumetric ultrasonic examination, Entergy proposes that the industry report (Reference 2) provides an acceptable technical basis for eliminating the requirement for this examination because the alternative maintains an acceptable level of quality and safety.

This report provides the basis for the elimination of the RPV threads in flange examination requirement (ASME Section XI Examination Category B-G-1, Item Number B6.40). This report was developed because evidence had suggested that there have been no occurrences of service-induced degradation and there are negative impacts on worker dose, personnel safety, radwaste, critical path time for these examinations and additional time at reduced water inventory.

Since there is reasonable assurance that the proposed alternative is an acceptable alternate approach to the performance of the ultrasonic examinations, Entergy requests authorization to use the proposed alternative at ANO-2 pursuant to 10 CFR 50.55a(z)(1) on the basis that use of the alternative provides an acceptable level of quality and safety.

To protect against non-service related degradation, ANO-2 uses detailed procedures for the care and visual inspection of the RPV studs and the threads in flange each time the RPV closure head is removed. Care is taken to inspect the RPV threads for damage and to protect threads from damage when the studs are removed. Prior to reinstallation, the studs and stud holes are cleaned and lubricated. The studs are then replaced and tensioned into the RPV flange. This activity is performed each time the closure head is removed, and the procedure documents each step. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service.

6. Duration of Proposed Alternative

This request for an alternative is applicable to the ANO-2 Fourth 10-Year ASME Section XI ISI Interval which began on March 26, 2010, and is scheduled to end on March 25, 2020.

7. Precedents

At the time of this submittal for ANO-2, multiple Licensee submittals have been made to the NRC to use the same alternative as described in this request; however, the precedent listed below for Southern Company, Exelon Generating Company, and Duke Energy are the only known to have received authorization/approval to use this alternative per the Safety Evaluation Reports (SER) listed below.

1. NRC SER from M. T. Markley to Southern Nuclear Operating Co. Inc., C. R. Pierce, regarding "Vogtle Electric Generating Plant, Units 1 and 2 and Joseph M. Farley Nuclear Plant, Unit 1" - Alternative to Inservice Inspection Regarding Reactor Pressure Vessel Threads in Flange Inspection, (CAC Nos. MF8061, MF8062, MF8070), dated, January 26, 2017, (ML17006A109)

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

3. NRC SER from Undine Shoop to Duke Energy, S. Capps "Brunswick Steam Electric Plan, Unit No. 1; Catawba Nuclear Station, Unit No. 2; Shearon Harris Nuclear Power Plant, Unit 1; McGuire Nuclear Station, Units Nos. 1 and 2; Oconee Nuclear Station, Unit Nos. 1, 2, and 3; and H. B. Robinson Steam Electric Plant, Unit No. 2 – Alternative to Inservice Inspection Regarding Reactor Pressure Vessel Threads in Flange Inspection (CAC Nos. MF9513 – MF9521; EPID L-2017-LLR-0019)", dated December 26, 2017 (ML17331A086)

8. References

1. NRC SER from Robert J. Pascarelli to Entergy Operations Inc., Bryan S. Ford, regarding Arkansas Nuclear One, Unit 2 Relief Request EN-ISI-16-1 and Indian Point Nuclear Generating Station Unit No. 3, Regarding Use of Later Edition and Addenda of the ASME Code (CAC NOS. MF8905 and MF8906)), dated, July 12, 2017, (ADAMS Accession No.: ML17174B144)

2. Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements. EPRI, Palo Alto, CA: 2016. 3002007626. (ADAMS Accession No. ML16221A068)[1]

3. ASME Code Case N-864, "Reactor Vessel Threads In Flange Examinations Section XI, Division 1," Approval Date: July 28, 2017, Published in ASME Nuclear Code Cases Book 2017 Edition, Supplement 2

4. American Society of Mechanical Engineers, Risk-Based Inspection: Development of Guidelines, Volume 2-Part 1 and Volume 2-Part 2, Light Water Reactor (LWR) Nuclear Power Plant Components. CRTD-Vols. 20-2 and 20-4, ASME Research Task Force on Risk-Based Inspection Guidelines, Washington, D.C., 1992 and 1998.

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[1] As stated in the second precedent, the Staff's evaluation contained in the first precedent concluded that the generic stress analysis and flaw tolerance evaluation provided in Reference 2 are acceptable and can be used to support eliminating examination of threads in the RPV flange.

Figure 1

Modeled Dimensions

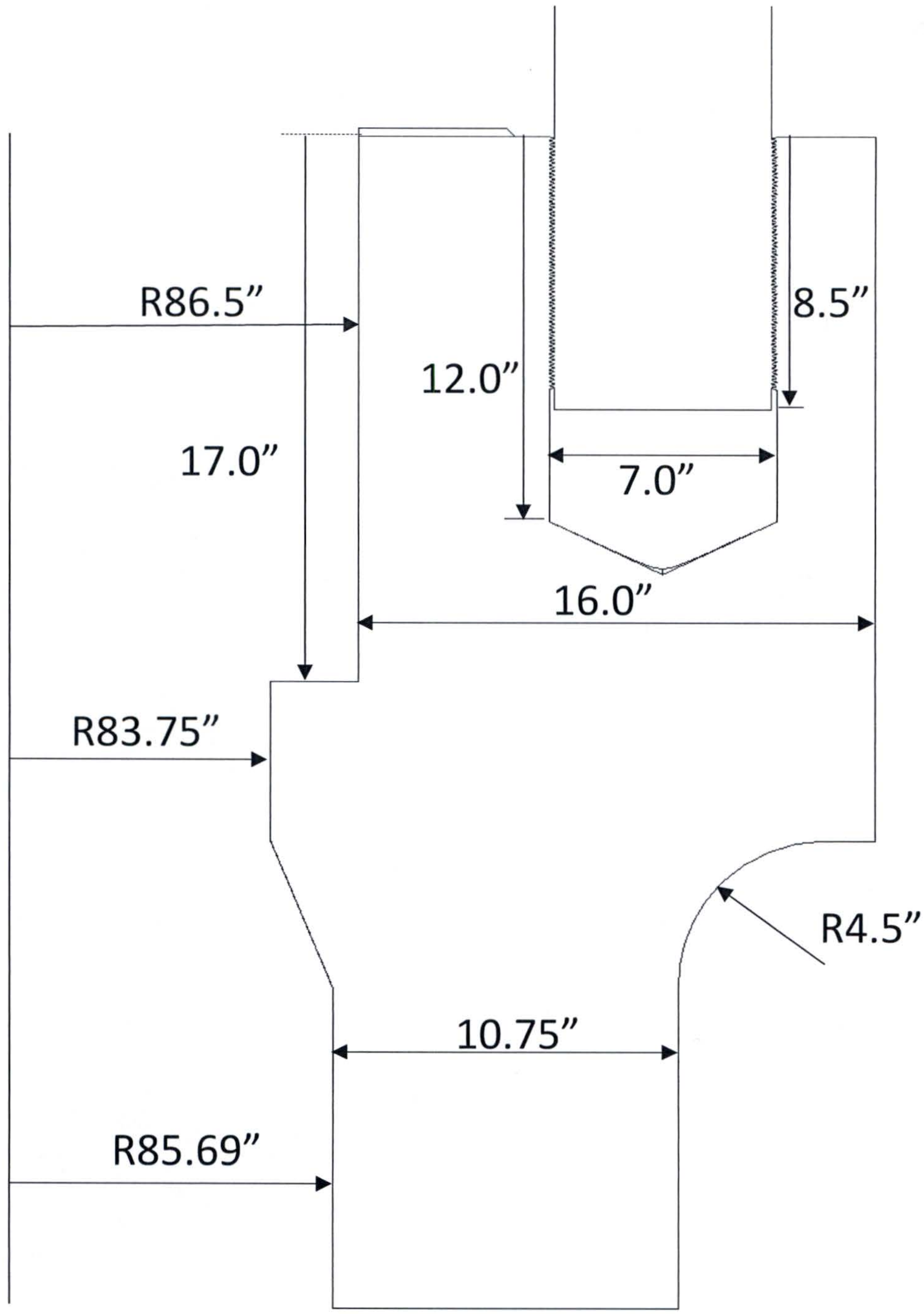

Figure 2

Finite Element Model Showing Bolt and Flange Connection

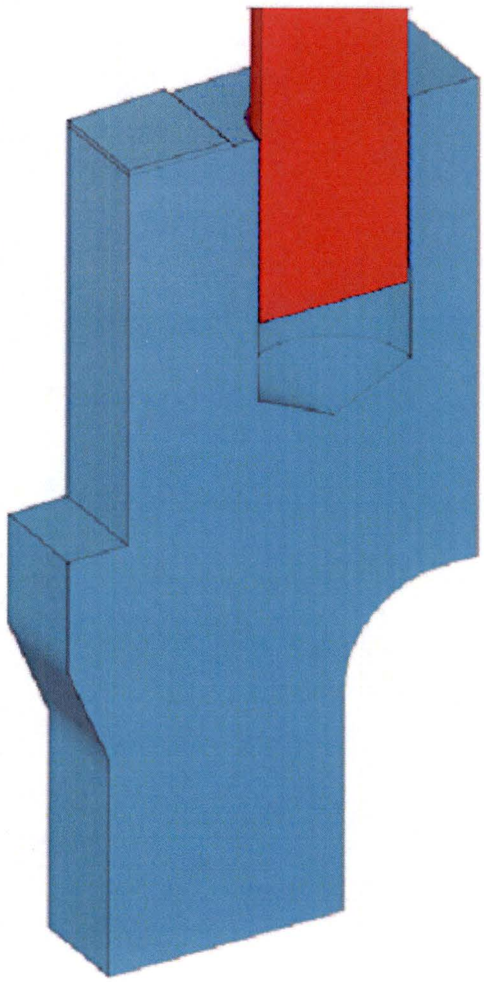

Figure 3

Finite Element Model Mesh with Detail at Thread Location

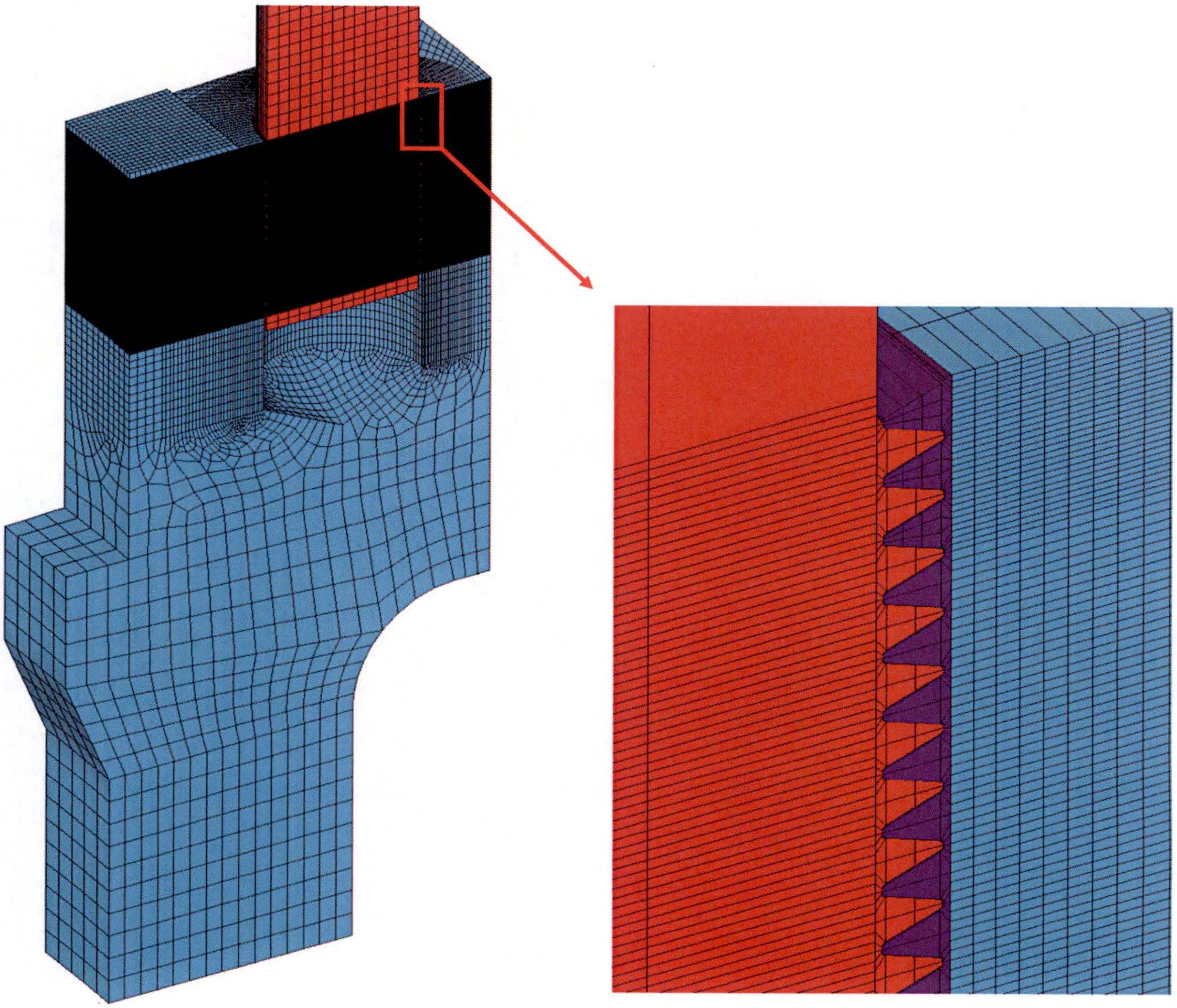

Figure 4

Cross Section of Circumferential Flaw with Crack Tip Elements Inserted After 10th Thread from Top of Flange