VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 22, 2020

10 CFR 50.55a(z)(1)

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Serial No.: 20-336
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Docket Nos.: 50-280/281
License Nos.: DPR 32/37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
ASME SECTION XI INSERVICE INSPECTION PROGRAM
REQUEST FOR NRC APPROVAL OF PROPOSED ALTERNATIVE
S1-I5-ISI-05 AND S2-I5-ISI-06

In accordance with 10 CFR 50.55a(z)(1), Virginia Electric and Power Company (Dominion Energy Virginia) requests NRC approval of the proposed alternative request S1-I5-ISI-05 and S2-I5-ISI-06 for Surry Power Station (Surry) Units 1 and 2, respectively. ASME Section XI, Category B-G-1, Item B6.40, requires volumetric examination of the threads in the reactor pressure vessel (RPV) flange stud holes. The proposed alternative would eliminate the volumetric examination requirement in accordance with an industry initiative analyzed in Electric Power Research Institute (EPRI) Report #3002007626, "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements," dated March 2016. The proposed alternative request for Surry Units 1 and 2 is provided in the attachment.

Dominion Energy Virginia requests approval of the proposed alternative by September 30, 2021. If you have any questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Respectfully,

Mark D. Sartain
Vice President – Nuclear Engineering and Fleet Support

Commitments made in this letter: None

Attachment: Proposed Alternative Request S1-I5-ISI-05 and S2-I5-ISI-06

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Attachment

PROPOSED ALTERNATIVE REQUEST S1-I5-ISI-05 AND S2-I5-ISI-06

Virginia Electric and Power Company
(Dominion Energy Virginia)
Surry Power Station Units 1 and 2

Dominion Energy Virginia
Surry Power Station Units 1 and 2
5th 10-Year Interval
Request for Alternative Examination of Reactor Vessel Threads in Flange
Alternative Request S1-I5-ISI-05 and S2-I5-ISI-06

In accordance with 10 CFR 50.55a(z)(1)
Alternative Provides Acceptable Level of Quality and Safety

1. ASME Code Component(s) Affected

The affected components are the Surry Power Station (Surry) Units 1 and 2 Reactor Pressure Vessels (RPVs). Specifically, the American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel (BPV) Code, Section XI, Examination Category B-G-1, Reactor Vessel Item Number B6.40, Threads in Flange, require a volumetric examination.

2. Applicable Code Edition and Addenda

The applicable Code edition for the Surry Units 1 and 2 Fifth 10-Year Interval Inservice Inspection (ISI) Program is the 2004 Edition of ASME Section XI.

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 (Fig 5). The Surry Unit 1 fifth 10-year ISI interval is scheduled to end October 13, 2023, and the Surry Unit 2 fifth 10-year ISI interval is scheduled to end May 9, 2024.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(1), Dominion Energy Virginia 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, for Surry Units 1 and 2.

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 1), 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 1), ASME has developed, approved and published a Section XI, Code Case N-864, (Reference 2) that supports the same conclusion as the EPRI Report, and the code case states that the Section XI Code requirement to examine the RPV threads in flange is not required. Since 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, Surry Power Station does not plan to adopt this code case as part of this request at this time. However, if NRC approves the code case at a later date, Surry plans to use the code case 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 the EPRI report (Reference 1). 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 3), 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, the EPRI report 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 would 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 the EPRI report 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 Surry Units 1 and 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 SPS to Bounding Values Used in Analysis
Minimum RPV

Plant	No. of Studs	Stud Nominal Diameter (inches)	RPV Inside Diameter at Stud Hole (inches)	Flange Thickness at Stud Hole (inches)	Design Pressure (psia)
Surry 1 and 2	58	6.0	155	14.7	2500
Range for 16 Units Considered	54-60	6.0 – 7.0	155 – 173	15 - 16	2500
Bounding Values Used in Analysis	54	6.0	173	16	2500

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:

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

Where:

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

- 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 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.

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 were 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 (alt) 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 alt = 0.77 *alt* 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 2 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 2: Maximum K vs. a/t				
Load	K at Crack Depth (ksi$\sqrt{in}$)			
	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:

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

Where,

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

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

As can be seen from Table 2, 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 Reference 1 (reproduced in Table 2 above), the maximum K is 19.8 ksi$\sqrt{in}$. The allowable K calculated in Section 6.2.2 of the report is 69.6 ksi$\sqrt{in}$, significantly higher than the calculated value. Assuming an RPV flange with 60 studs originally and one inoperable stud, the increase in K is approximately 1.7% resulting in a maximum K of approximately 20.14 ksi$\sqrt{in}$ which is still significanly less than 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 Surry 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 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 3 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 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 3
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, the EPRI report 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 US NRC 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 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, the EPRI report 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, Dominion Energy Virginia proposes that the EPRI industry report 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, and critical path time for these examinations and additional time at reduced water inventory.

Since there is reasonable assurance the proposed alternative is an acceptable alternate approach to the performance of the ultrasonic examinations, Dominion Energy Virginia requests authorization to use the proposed alternative at Surry Units

1 and 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.

6. Duration of Proposed Alternative

This request for an alternative is applicable to the Surry Unit 1 fifth ISI Interval which began December 14, 2013 and ends on October 13, 2023, and the Surry Unit 2 fifth ISI Interval which began May 10, 2014 and ends on May 9, 2024.

7. Precedents

1. NRC SER, Southern Nuclear Operating Co. Inc., 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, MF8061, MF8062, MF8070, ML17006A109 letter dated January 26, 2017.

2. NRC SER, Exelon Generating Company Braidwood Station, Units 1 and 2; Byron Station, Unit 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 MF8712, MF8729 and MF9548, ML17170A013 letter dated June 26, 2017.

3. NRC SER, Duke Energy, Brunswick Steam Electric Plan, Unit 1; Catawba Nuclear Station, Unit 2; Shearon Harris Nuclear Power Plant, Unit 1; McGuire Nuclear Station, Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3; and H. B. Robinson Steam Electric Plant, Unit 2 – Alternative to Inservice Inspection Regarding Reactor Pressure Vessel Threads in Flange Inspection MF9513: MF9521; EPID L-2017-LLR-0019, ML17331A086 letter dated December 26, 2017.

4. NRC SER, North Anna Units 1 and 2, Reactor Vessel Threads in Flange MF9298 and MF9299, EPIDS L-2016-LLR-0018, ML17332A663 letter dated December 6, 2017.

5. NRC SER, ANO Unit 2, Reactor Pressure Vessel Flange Threads EPID L-2018-LLR-0003, ML18192A104 letter dated August 15, 2018.

6. NRC SER, Millstone Power Station Units 2 and 3, Reactor Pressure Vessel Threads in Flange MF8468 and MF8469, ML17132A187 letter dated May 30, 2017.

8. References

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

2. ASME Code Case N-864, "Reactor Vessel Threads In-Flange Examinations Section XI, Division 1," Approval Date: July 28, 2017.

3. 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 &1998.

Figure 1

Modeled Dimensions

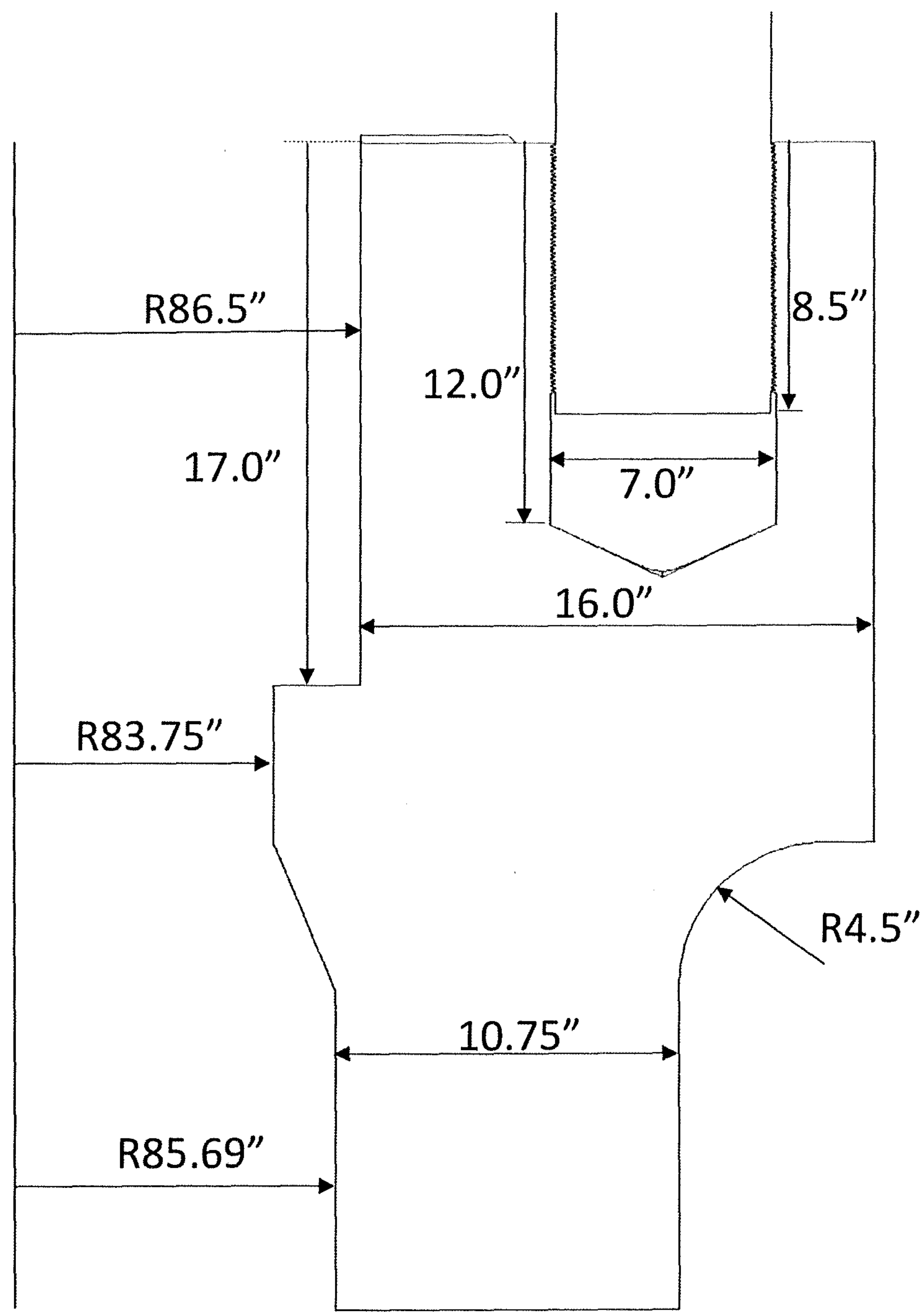


Figure 2

Finite Element Model Showing Bolt and Flange Connection

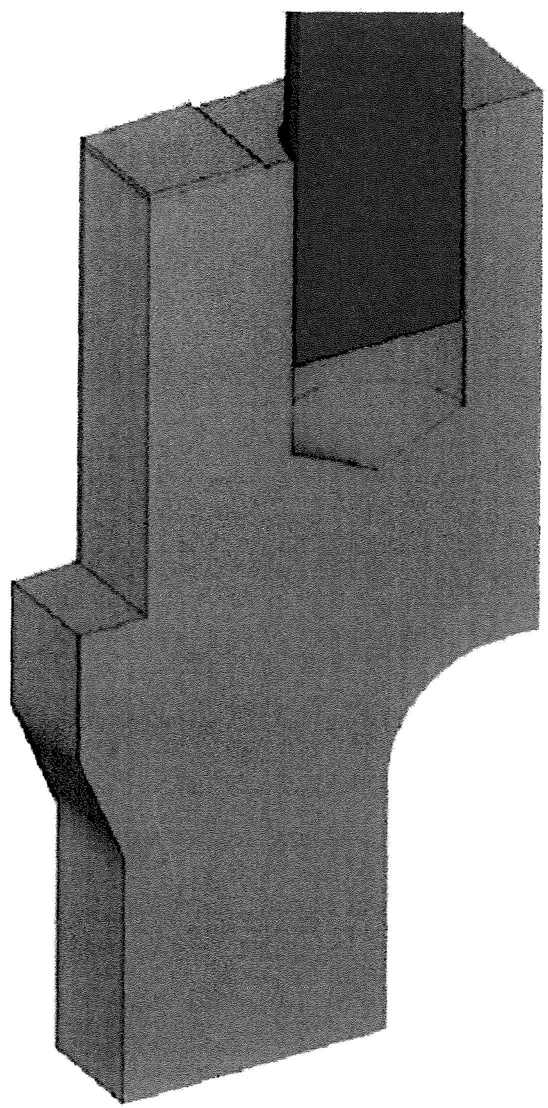


Figure 3

Finite Element Model Mesh with Detail at Threaded Location

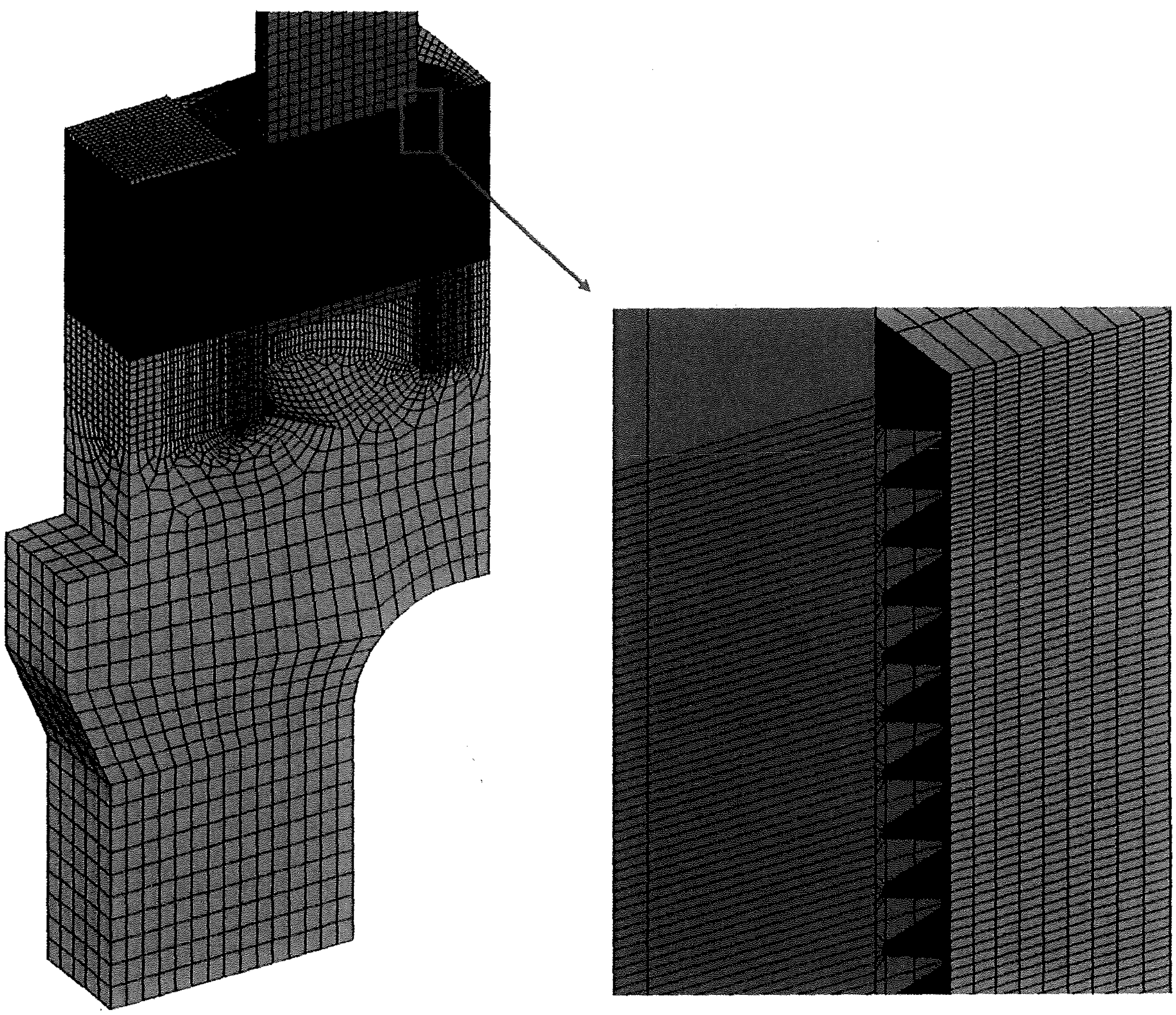


Figure 4

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

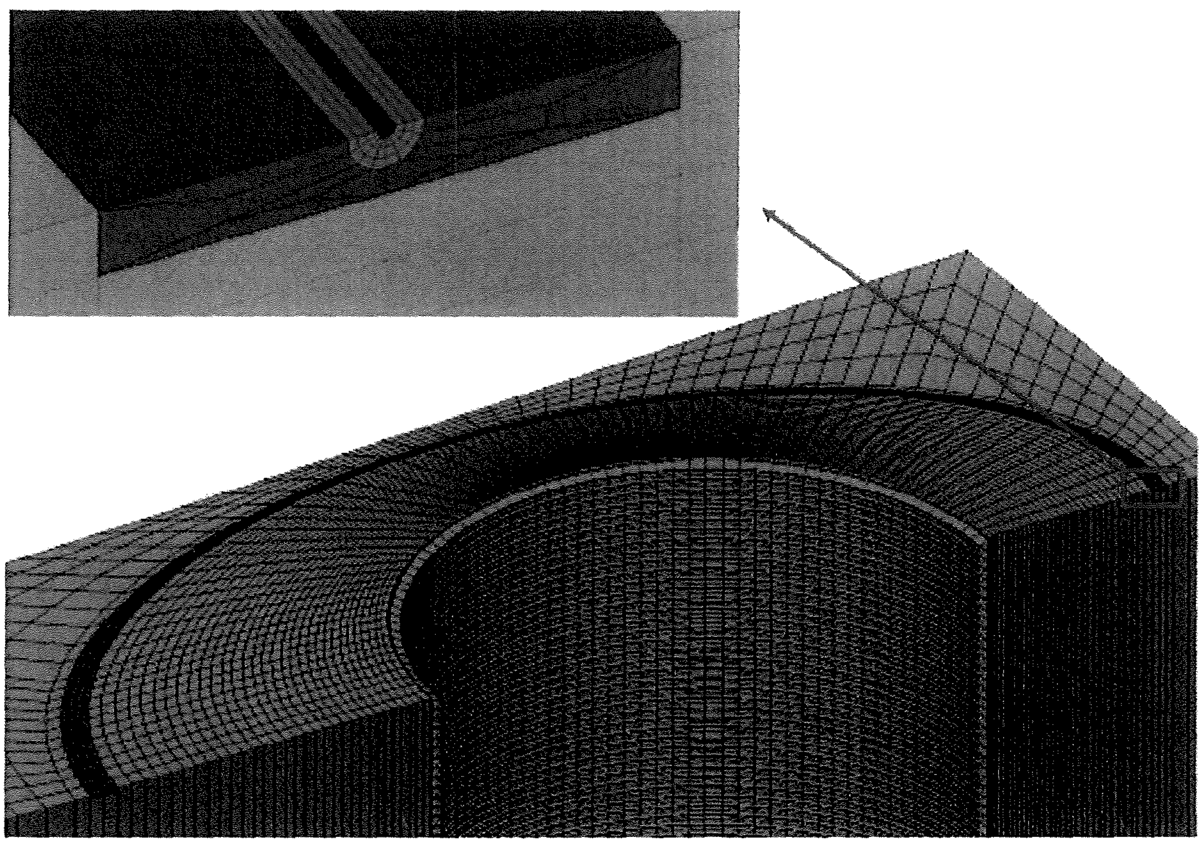
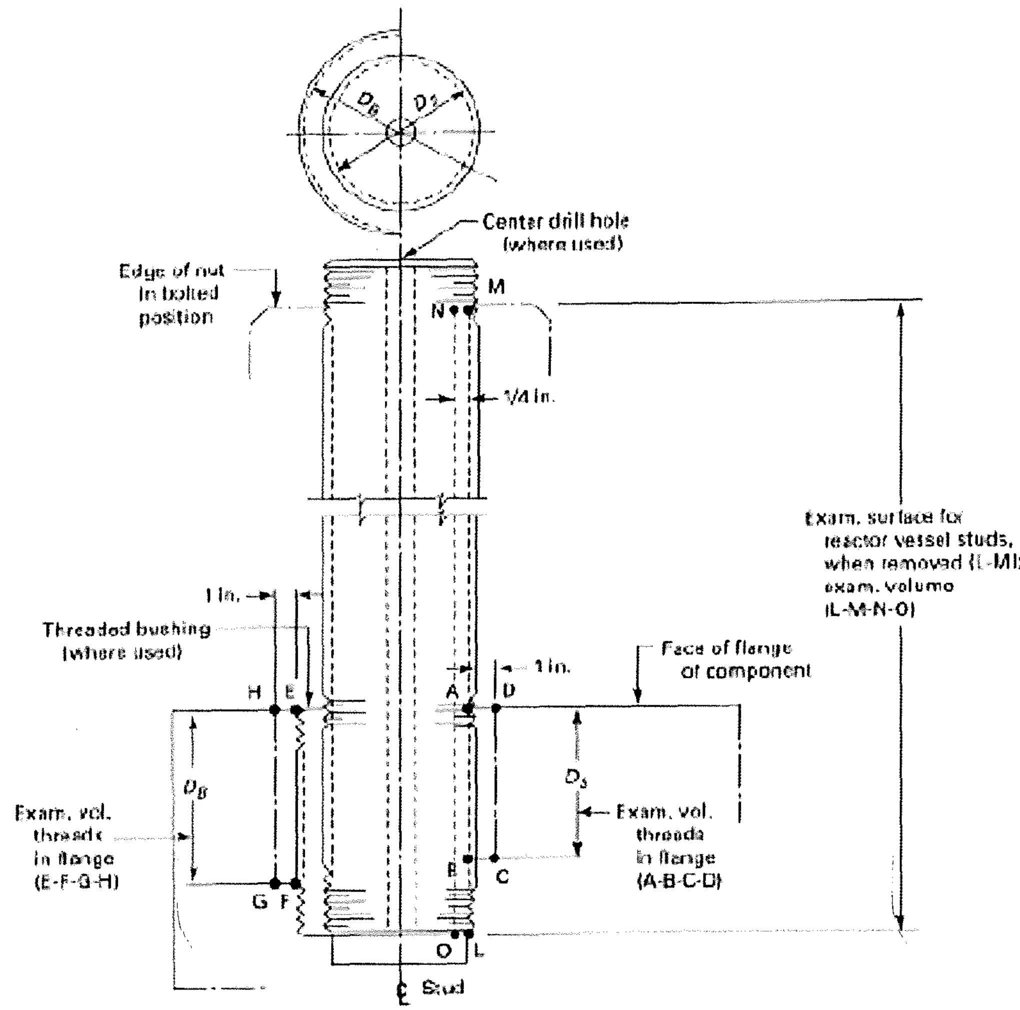


Figure 5

Closure Stud and Threads in Flange Stud Hole



D_B = diameter of the threaded bushing
D_S = diameter of the stud

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