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U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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DOMINION ENERGY SOUTH CAROLINA (DESC)
VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) UNIT 1
ALTERNATIVE REQUESTS RR-4-25 FOR ELIMINATION OF REACTOR PRESSURE VESSEL THREADS IN FLANGE EXAMINATION FOR THE REMAINDER OF THE FOURTH 10-YEAR ISI INTERVAL

Pursuant to the provisions of 10 CFR 50.55a(z)(1), Dominion Energy South Carolina (DESC), acting for itself and as an agent for South Carolina Public Service Authority (Santee Cooper) hereby submits the attached request for approval to use an alternative to the volumetric examination In-Service Inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI for the threads in the reactor pressure vessel (RPV) flange.

Specifically, DESC proposes to eliminate the volumetric examination requirements of Section XI of the ASME Code threads in the RPV flange (ASME Code, Section XI, Examination Category B-G-1, "Pressure Retaining Bolting, Greater than 2 inches (50 mm) in Diameter," Item No. B6.40) for the remainder of the VCSNS Unit 1 fourth 10-year ISI interval, which began on January 1, 2014 and is scheduled to end on December 31, 2023.

DESC has determined that the proposed alternative would provide an acceptable level of quality and safety.

A detailed description of the proposed alternative, including basis for use, is enclosed with this letter. DESC requests NRC review and approval of this request by September 30, 2021 to support planning for the fall 2021 refueling outage (RF-26).

Should you have any questions, please contact Mr. Yan Gao at (804) 273-2768.

Respectfully,

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

Commitments contained in this letter: None

Enclosure:

Alternative Request RR-4-25 - Elimination of Reactor Pressure Vessel Threads in Flange Examination for the Remainder of the Fourth 10-Year ISI Interval

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Enclosure

Alternative Request RR-4-25

Elimination of Reactor Pressure Vessel Threads in Flange Examination for the Remainder of the Fourth 10-Year ISI Interval

Virgil C. Summer Nuclear Station (VCSNS) Unit 1
Dominion Energy South Carolina, Inc. (DESC)

Alternative Request RR-4-25

Elimination of Reactor Pressure Vessel Threads in Flange Examination for the Remainder of the Fourth 10-Year ISI Interval

Proposed Alternative Request
Pursuant to 10 CFR 50.55a(z)(1)

Acceptable Level of Quality and Safety

1.0 ASME CODE COMPONENT(S) AFFECTED

Table 1 – ASME Code Components Affected

Element	Description
ASME Code Class	Code Class 1
References	ASME Section XI, Paragraph IWB-2500
Examination Category	B-G-1
Item Number	B6.40
Description	Reactor Pressure Vessel (RPV) – Threads in Flange
Components	Pressure Retaining Bolting, Greater Than 2-Inches in Diameter

2.0 APPLICABLE CODE EDITION AND ADDENDA

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2007 Edition through 2008 Addenda.

3.0 APPLICABLE CODE REQUIREMENT AND PROPOSED ALTERNATIVE

The Reactor Pressure Vessel (RPV) Threads in Flange, Examination Category B-G- 1, Item Number B6.40 are required to be examined using a volumetric examination technique with 100% of the flange ligament areas examined every Inservice Inspection (ISI) interval. The examination area is the one-inch area around each RPV stud hole as shown in ASME Section XI, Figure IWB-2500-12.

Pursuant to 10 CFR 50.55a(z)(1), DESC proposes to eliminate the ASME B&PV Code requirement to volumetrically examine the threads in the RPV flange stud holes during the fourth ISI interval for VCSNS. The fourth ISI interval for VCSNS began on January 1, 2014 and is scheduled to end on December 31, 2023.

4.0 TECHNICAL BASIS FOR THE PROPOSED ALTERNATIVE REQUEST

The technical basis for eliminating the RPV threads in flange volumetric examinations is provided in Electric Power Research Institute (EPRI) Report No. 3002010354, "Reactor Pressure Vessel (RPV) Threads in Flange Examination Requirements" (referred to herein as the EPRI report or Reference 1 [6.1].) The EPRI report [6.1] includes a survey of inspection results from 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 Reactor Coolant System water inventory) of the examination. The technical basis for this alternative is discussed in more detail below.

4.1 Potential Degradation Mechanisms

An evaluation of potential degradation mechanisms that could impact flange/threads reliability was performed as part of the EPRI report [6.1]. 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, no active degradation mechanisms were identified for the threads in flange component.

The EPRI report [6.1] notes a general conclusion from Reference 2 [6.2], 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 in-service inspections do not provide additional value. As discussed in the operating experience (OE) review summary below, the RPV threads in flange have received the required preservice examinations and over 10,000 in-service inspections with no relevant findings.

To address the potential for mechanical/thermal fatigue, the EPRI report [6.1] provides a stress analysis and flaw tolerance evaluation of the flange thread area to assess mechanical/thermal fatigue potential. The evaluation [6.1] consists of two parts. In the first part, stress analysis is performed considering the applicable loads on the threads in flange component. In the second part, the stresses at the critical locations of the component are used in a fracture mechanics evaluation to determine the allowable flaw size for the component as well as how much time it will take for a postulated initial flaw to grow to the allowable flaw size using guidelines in Subsection IWB-3500 of ASME Section XI [6.3].

4.2 Stress Analysis

A stress analysis was performed and documented in the EPRI report [6.1] to determine the stresses at critical regions of the threads in flange component as input to a flaw tolerance evaluation. 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. The dimensions and geometries of the EPRI analysis model are shown in Figure 1 and Figure 2 at the end of this Enclosure.

The details of the RPV parameters for VCSNS as compared to the bounding values provided in the EPRI analysis are shown in Tables 2 and 3. Table 2 provides a comparison of basic dimensions and loads. The bounding preload stress in the analysis is based on the least number of bolts, the smallest bolt diameter, and the largest RPV inner diameter which results in the highest preload stress. As presented in Table 2, the preload stress used in the analysis bounds that at VCS. The comparison of the basic dimensions provided in Table 3 shows that the dimensions are either the same or within the range of the values considered in the EPRI report. In summary, the comparisons shown in Table 2 and 3 demonstrate that the stress analysis discussed in the EPRI analysis is applicable to and bounding for VCSNS.

Table 2 - Comparison of VCSNS Parameters to Parameters Used in the EPRI Analysis

PLANT	NO. OF STUDS INSTALLED	STUD DIA NOMINAL (INCHES)	RPV ID AT STUD HOLE (INCHES)	RV FLANGE THICKNESS AT STUD HOLE (INCHES)	DESIGN PRESSURE (PSIA)	PRELOAD STRESS (KSI)
VCSNS	58	6.0	155	15	2500	31.6
Bounding Values Used in EPRI Analysis	54	6.0	173	16	2500	42.3

Table 3 - Comparison of VCSNS Flange Thread Parameters to Values Used in the EPRI Analysis

PLANT	RV FLANGE THREAD SPEC	RV FLANGE BOLT HOLE DIA, NOMINAL (INCHES)	FLANGE THREAD PITCH (THREADS/INCH)	FLANGE THREAD DEPTH (INCHES)
VCSNS	6"-8N-2B	6	8	0.0677
Values Used in EPRI Analysis	various	6 - 7	8	0.0635 – 0.0708

The analyzed geometry is shown in Figure 1, and the analytical model is shown in Figure 2. The loads considered in the analysis consisted of:

- A design pressure of 2500 psia at an operating temperature of 600 °F was applied to the internal surfaces exposed to internal pressure.
- The bolt preload on the bounding geometry was calculated as:

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

where

$P_{preload}$ = Preload stress to be applied on modeled bolt (psi)

C = Bolt-up contingencies (10%)

P = Internal pressure (psi)

ID = Largest inside diameter of RPV (inch)

S = Least number of studs

D = Smallest stud diameter (inch)

- The only significant transient (thermal stresses) affecting the flange is heat-up and 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 thread in flange component for the three loads described above.

4.3 Flaw Tolerance Evaluation

A flaw tolerance evaluation was performed [6.1] using the results of the stress analysis to determine how long it would take an initial postulated flaw to reach the ASME Section XI allowable flaw size. A linear elastic fracture mechanics (LEFM) evaluation consistent with ASME Section XI, IWB-3600 was performed.

At four (4) flaw depths of a 360° inside-surface-connected, partial-through-wall circumferential flaw, stress intensity factors (K) were calculated using finite element analysis techniques with the model described above. The maximum K values around the bolt hole circumference for each flaw depth (a) were extracted and used to perform the crack growth calculations. The circumferential flaw was modeled to start between the 10th and 11th flange threads from the top end of the flange where the largest tensile axial stress occurs. The modeled flaw depth–to–wall thickness ratios (a/t) were 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 3 for the flaw model with a/t = 0.77. The crack tip mesh for the other flaw depths follows the same pattern. When preload was not being applied, the stud, stud threads, and flange threads were not modeled. The model was otherwise unchanged between load cases.

The maximum K results are summarized in Table 4 for four (4) crack depths. Because the crack tip varies in depth around the circumference, the maximum K from all locations at each crack size was conservatively used for the K-versus-a profile.

Table 4 – Maximum Stress Intensity Factor vs. Flaw Depth-to-Wall Thickness Ratio

LOAD		**K at Crack Depth (ksi√in)**			
		0.02 a/t	0.29 a/t	0.55 a/t	0.77 a/t
Case 1	**Preload**	11.2	17.4	15.5	13.9
Case 2	**Preload + Heat-up + Pressure**	13.0	19.8	16.1	16.3

Because a postulated flaw is considered in this evaluation, a conservative LEFM approach consistent with ASME Section XI, Appendix G is used to determine the allowable flaw size. In Appendix G, a structural factor of 2 is applied to the membrane stress, and a structural factor of 1 is applied to the thermal stress. In this evaluation, the conservative structural factor of 2 was applied to all stresses. Hence, the acceptance criterion based on allowable stress intensity factor for the Preload load case, K_I (Preload), is:

$$K_I(Preload) < K_{Ic}/2 = 58.6\ ksi\sqrt{in}$$

Where

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

$= 33.2 + 20.734 exp\ (0.02(T - RT_{NDT}))$

$= 117\ ksi\sqrt{in}$

Table 5 below provides the VCS parameters used to determine the allowable stress intensity factor for the Preload load case, K_I (Preload). These parameters were obtained from plant records [6.6] and the station procedurally-controlled limitation on the minimum required temperature for stud tensioning.

Table 5 - VCSNS Parameters Used to Determine K_I (Preload)

PLANT	RV FLANGE RT_{NDT} (^{0}F)	FLANGE TEMP DURING BOLT PRELOAD (^{0}F)	FLANGE T-RT_{NDT} (^{0}F)
VCSNS	0	70	70

The LEFM approach consistent with ASME Section XI, Appendix G was also applied to the Preload + Heat-up + Pressure load case. The acceptance criterion based on allowable stress intensity factor for the Preload + Heat-up + Pressure load case, K_I(Preload + Heat-up + Pressure), is

$$K_I(Preload + Heat-up + Pressure) < K_{Ic}/2 = 110\ ksi\sqrt{in}$$

Where

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

$= 220\ ksi\sqrt{in}$

The lower bound fracture toughness, K_{Ic}, is obtained from Figure G-2210-1 of Appendix G of ASME Section XI for a material operating in the upper shelf region (normal operating temperature.) In this case, the value of $K_{Ic} = 220\ ksi\sqrt{in}$ was used, which is the maximum value allowed for the applicable conditions in the EPRI study.

As shown in Table 4, the allowable stress intensity factor for both load cases were not exceeded for crack depths up to the deepest analyzed flaw of a/t = 0.77. Hence, the allowable flaw depth of the 360^0 circumferential flaws was at least 77% of the thickness of the flange. The allowable flaw depth was assumed to be equal to the deepest modeled crack for the purposes of the analysis.

For the crack growth evaluation, an initial postulated flaw size of 0.2 inches (5.08 mm) was chosen to be consistent with ASME Section XI, IWB-3500 flaw acceptance standards. The deepest flaw analyzed was a/t = 0.77 because of the limits of the model. Two load cases were considered for fatigue crack growth: heat-up/cooldown and bolt preload. The heat-up/cooldown load case included the stresses due to thermal and internal pressure loads and was conservatively assumed to occur fifty (50) times per year. The bolt preload load case was assumed to be present and constant during the load cycling of the heat-up/cooldown load case. The bolt preload load case was conservatively assumed to occur five (5) times per year and did 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 was insignificant, the allowable flaw size would 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 stress analysis/flaw tolerance evaluation presented above shows that the thread in flange component at VCSNS is very flaw tolerant and can operate for 80 years without violating ASME Section XI safety margins. This clearly demonstrates that the thread in flange examinations can be eliminated without affecting the safety of the RPV.

4.4 Operating Experience Review Summary

As discussed above, the results of the survey discussed in the EPRI report, which includes inputs from VCSNS, confirmed that the RPV threads in flange examination adversely impacts outage activities (dose, safety, and critical path time with increased time at reduced RCS inventory) while not identifying any service-induced degradations. Specifically, for the US fleet, a total of 94 nuclear units have responded to date and none of these units have identified any type of degradation. As shown in Table 6 below, the 94 nuclear units represent data from 33 BWRs and 61 PWRs. For the BWR units, a total of 3,793 examinations were conducted. For the PWR units, a total of 6,869 examinations were conducted. No service-induced degradation was identified. The response data includes information from the plant designs in operation in the US 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 (Babcock & Wilcox, Combustion Engineering, and Westinghouse.)

Table 6 – Summary of US Fleet Survey Results

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

4.5 Related RPV Assessments

In addition to the examination history and flaw tolerance discussed above, the EPRI report [6.1] discusses studies conducted in response to the issuance of the Anticipated Transient Without Scram (ATWS) Rule [6.4] 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 ATWS events. 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 [6.5] for PWRs, the ASME Service Level C pressure of 3200 psig was assumed be to 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, the EPRI report [6.1] 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, the threads in flange are generally not in contact with primary water at plant operating temperatures/pressures.) The robust design is proven in that plant operation has been allowed at several plants even with a bolt assumed to be out of service. As such, significant degradation of multiple bolts/threads would be needed prior to any RCS leakage.

4.6 Precedents

The NRC has authorized similar requests to eliminate the RPV threads in flange examinations at several operating nuclear plants. Some of the approved requests are listed in [6.7] through [6.12].

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE

In lieu of the requirements for a volumetric ultrasonic examination, DESC proposes eliminating the requirement for the RPV threads in flange examination for the remainder of the fourth 10-year Inservice Inspection interval at VCSNS.

VCSNS has confirmed that VCSNS plant-specific parameters (e.g. vessel diameter, number of studs, ISI findings) are consistent with or bounded by the EPRI report [6.1].

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

To protect against non-service-related degradation, each outage VCSNS uses a detailed procedure for the removal and care of the RPV studs. Care is taken to remove the studs and to inspect the stud threads for damage. If stud thread damage is identified, the corresponding flange stud hole is visually examined for any damage to the threads. Prior to reinstallation, the studs and stud holes are cleaned and lubricated. The studs are then replaced and tensioned into the RPV. This activity is performed each refueling outage with procedure documentation of each step. These controlled maintenance activities provide further assurance that degradation is detected and mitigated prior to returning the reactor to service.

In addition, it is noted that other inspection activities including the system leakage test (Examination Category B-P) are conducted every refueling outage and will continue to be performed every refueling outage.

6.0 <u>REFERENCES</u>

6.1 Reactor Pressure Vessel (RPV) Threads in Flange Examination Requirements. EPRI, Palo Alto, CA: 2017. 3002010354, Final Report, December 2017.

6.2 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-Volumes 20-2 and 20-4, ASME Research Task Force on Risk-Based Inspection Guidelines, Washington, D.C., 1992 and 1998.

6.3 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2007 Edition through 2008 Addenda.

6.4 10 CFR 50.62, Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants. Federal Register, Vol. 49, No. 124, June 26, 1984.

6.5 SECY-83-293, "Amendments to 10CFR50 Related to Anticipated Transients Without Scram (ATWS) Events," U.S. Nuclear Regulatory Commission, Washington, D.C., July 19, 1983.

6.6 VCSNS Unit 1 Reactor Vessel Instruction Manual 1MS-94B-036.

6.7 NRC letter to Duke Energy, "Brunswick Steam Electric Plant, Unit No. 1; Catawba Nuclear Station, Unit No. 2; Shearon Harris Nuclear Power Plant, Unit 1; McGuire Nuclear Station, Unit Nos. 1 and 2; Oconee Nuclear Station, Unit Nos. 1, 2 and 3; and H. B. Robinson Steam Electric Plant, Unit No. 2 - Alternative to In-service Inspection Regarding Reactor Pressure Vessel Threads in Flange Inspection (CAC Nos. MF9513 – MF9521; EPID L-2017-LLR-0019), dated December 26, 2017 (ADAMS Accession No. ML 17331A086).

6.8 NRC letter to DTE Electric, "Fermi 2 – Relief from the requirements of the ASME Code (EPID L – 2019-LLR-0023)," dated June 26, 2019. (ADAMS Accession No. ML 191690A315).

6.9 NRC letter to Exelon Generation Company LLC, "Calvert Cliffs Nuclear Power Plant Units 1 and 2 – Issuance of Relief Request I5R-05 RE: Relief from the Requirements of ASME CODE (EPID L-2018-LLR-0392)," dated May 30, 2019. (ADAMS Accession No. ML 19134A373).

6.10 NRC letter to Exelon Generating Company, LLC, "Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Calvert Cliffs Nuclear Power Plant, Units 1 and 2; Clinton Power Station, Unit No. 1; Dresden Nuclear Power Station, Units 2 and 3; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; R. E. Ginna Nuclear Power Plant; and Three Mile Island Nuclear Station, Unit 1 – Proposed Alternative to Eliminate Examination of Threads in Reactor Pressure Vessel Flange (CAC NOS. MF8712 – MF8729 AND MF9548)," dated June 26, 2017. (ADAMS Accession No. ML 17170A013).

6.11 NRC letter to Dominion Nuclear Connecticut, Inc., "Millstone Power Station, Unit Nos. 2 and 3 – Alternative Request RR-04-24 and IR-3-30 for Elimination of the Reactor Pressure Vessel Threads in Flange Examination (CAC Nos. MF8468 and MF8469)," dated May 25, 2017. (ADAMS Accession No. ML17132A187).

6.12 NRC letter to Virginia Electric and Power Company (Dominion, the licensee) North Anna Power Station, Units 1 and 2 – Proposed Inservice Inspection Alternatives N1-14-NDE-009 and N2-14-NDE-004 (CAC Nos. MF9298 AND MF9299; EPIDS L-2016-LLR-0018)," dated December 6, 2017. (ADAMS Accession No. ML 17332A663).

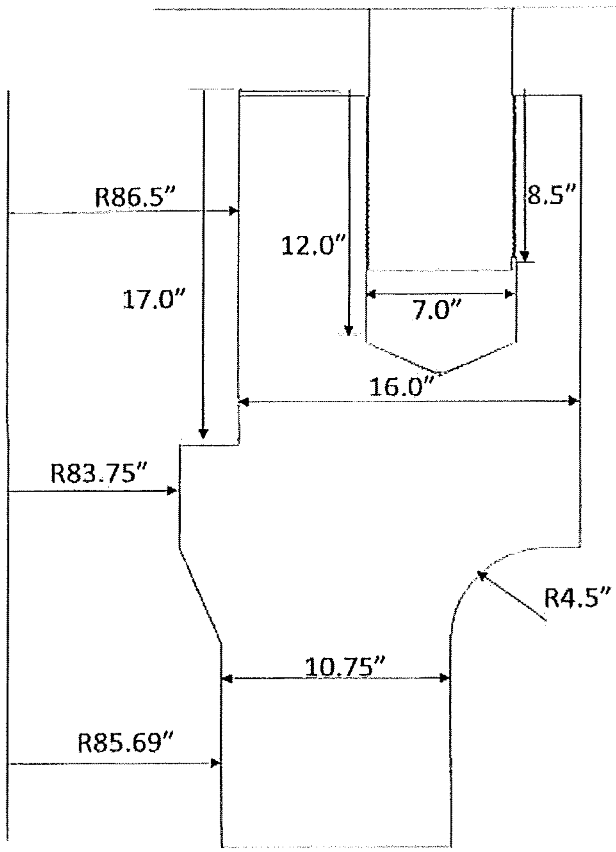

Figure 1: Modeled Dimensions

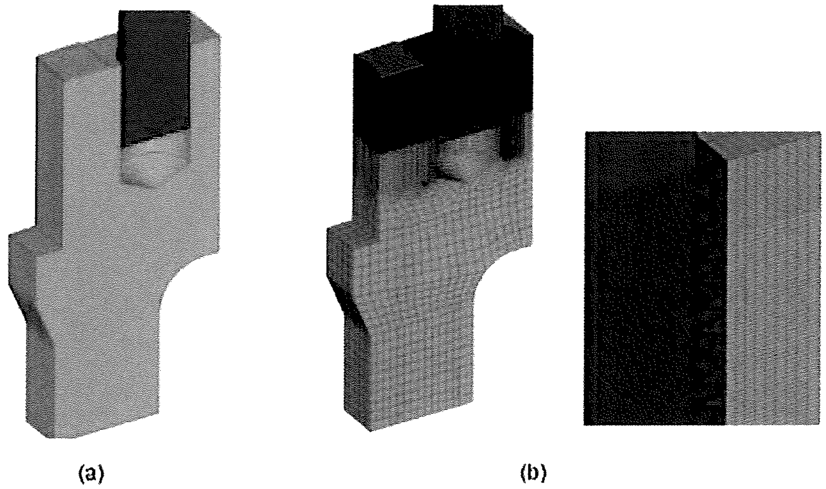

Figure 2: (a) Finite Element Model Showing Bolt and Flange Connection and,
(b) Finite Element Model Mesh with Detail at Thread Location

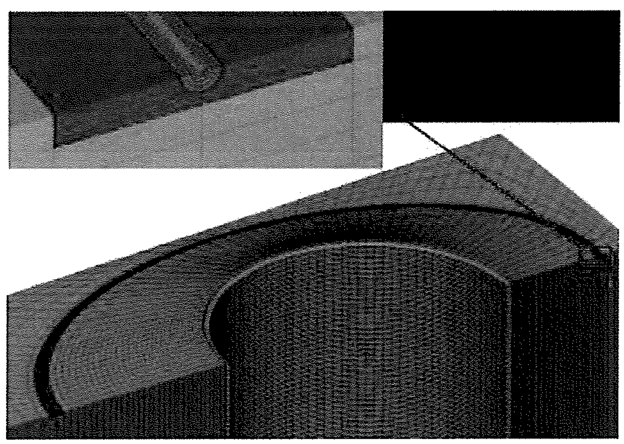

Figure 3: Cross Section of Circumferential Flaw with Crack Tip Elements Inserted after 10th Thread from Top of Flange