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Dominion®

October 6, 2016

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

Serial No.	16-375
NRA/WDC	R0
Docket Nos.	50-336/423
License Nos.	DPR-65
	NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNITS 2 AND 3
PROPOSED ALTERNATIVE REQUESTS RR-04-24 and IR-3-30 FOR ELIMINATION OF THE REACTOR PRESSURE VESSEL THREADS IN FLANGE EXAMINATION

Pursuant to 10 CFR 50.55a(z)(1), Dominion Nuclear Connecticut, Inc. (DNC) requests Nuclear Regulatory Commission (NRC) approval of Alternative Request RR-04-24, for Millstone Power Station Unit 2 (MPS2) and Alternative Request IR-3-30 for Millstone Power Station Unit 3 (MPS3). American Society of Mechanical Engineers (ASME) Code, Section XI requires a volumetric examination of Reactor Vessel – Threads in Flange to satisfy nondestructive examination requirements. DNC requests approval of the proposed alternative to the volumetric examination requirements. DNC considers the proposed alternative would provide an acceptable level of quality and safety. The supporting basis for this request is contained in the attachments to this letter.

DNC requests approval of the proposed alternatives to support the next MPS3 refueling outage currently scheduled to occur during fall 2017 (3R18).

The duration of proposed Alternative Request RR-04-24 is for the remainder of the fourth 10-year inservice inspection interval for MPS2 that began on April 1, 2010 and is scheduled to end on March 31, 2020. The duration of proposed Alternative Request IR-3-30 is for the remainder of the third 10-year inservice inspection interval for MPS3 that began on April 23, 2009 and is scheduled to end on April 22, 2019.

If you have any questions regarding this submittal, please contact Wanda Craft at (804) 273-4687.

Sincerely,

Mark D. Sartain
Vice President – Nuclear Engineering

A047
NRR

Attachments:

1. Alternative Request RR-04-24, Proposed Alternative to ASME Section XI for Elimination of Reactor Vessel – Threads in Flange Examination

2. Alternative Request IR-3-30, Proposed Alternative to ASME Section XI for Elimination of Reactor Vessel – Threads in Flange Examination

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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ATTACHMENT 1

ALTERNATIVE REQUEST RR-04-24
PROPOSED ALTERNATIVE TO ASME SECTION XI FOR ELIMINATION OF REACTOR VESSEL – THREADS IN FLANGE EXAMINATION

MILLSTONE POWER STATION UNIT 2
DOMINION NUCLEAR CONNECTICUT, INC.

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(1)

-- Proposed Alternative Provides an Acceptable Level of Quality and Safety --

1. ASME Code Components Affected

ASME Code Class:	Code Class 1
References:	ASME Section XI, Paragraph IWB-2500
Examination Category:	B-G-1
Item Number:	B6.40
Description:	Reactor Vessel – Threads in Flange
Components:	Pressure retaining bolting greater than 2 inches

2. Applicable Code Edition and Addenda

ASME Section XI, 2004 Edition (No Addenda).

3. Applicable Code Requirement

The Reactor Pressure Vessel (RPV) threads in flange 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 Figure IWB-2500-12.

4. Background and Reason for Request

Dominion Nuclear Connecticut, Inc. (DNC) has worked with the industry to evaluate eliminating the RPV threads in flange examination requirement. Licensees in the United States (US) and internationally have worked with the Electric Power Research Institute (EPRI) to produce a technical report (Reference 1) which provides the basis for elimination of the requirement. The report evaluates potential degradation mechanisms and includes a stress analysis / flaw tolerance evaluation. The report also includes a review of operating experience (OE) based on a survey of inspection results associated with RPV flange/bolting and related RPV assessments for over 168 units. The evaluation concludes that the safety benefit of the current examination requirements are not commensurate with the associated impact on worker exposure, personnel safety, radwaste, and increased time at reduced RCS

inventory. The technical basis for the proposed alternative is discussed in more detail below.

Potential Degradation Mechanisms

An evaluation of potential degradation mechanisms that could impact flange/threads reliability was included in the EPRI report. Potential degradation types 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 notes a general conclusion from Reference 2 (which includes work supported by the NRC) that when a component item has no active degradation mechanism present, and a pre-service 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 OE review summary below, the RPV threads in flange have received over 10,000 inservice inspections, 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, 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 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 and documented 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 units (ten PWRs and six Boiling Water Reactors (BWRs)) were considered in the analysis. The evaluation was performed using a geometrical configuration that bounds the sixteen units considered in this effort. The details of the RPV parameters for MPS2, as compared to the bounding values used in the evaluation, are shown in Table 1. As shown in the table, the diameter of the stud used in the analysis is smaller than that at MPS2 and the RPV inside diameter used in the analysis is larger than that at MPS2. The other

parameters are the same. The smaller stud diameter results in higher preload pressure per bolt and the larger RPV inside diameter results in higher pressure and thermal stresses. Hence, the stresses from the analyzed configuration would be conservative in application to MPS2. Dimensions of the analyzed geometry are shown in Figure 1.

Table 1: Comparison of MPS Unit 2 Parameters to Bounding Values Used in Analysis

Plant	No. of Studs	Stud Nominal Diameter (inches)	RPV Inside Diameter at Stud Hole (inches)	Flange Thickness at Stud Hole (inches)	Design Pressure (psia)
MPS2	54	7	172	16	2500
Range for 16 Units Considered	54 - 60	6.0 – 7.0	157 - 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 the internal surfaces 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 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.

Flaw Tolerance Evaluation

A flaw tolerance evaluation was performed using the results of the stress analysis 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.

At four flaw depths of a 360° inside-surface-connected, partial-through-wall circumferential flaw, stress intensity factors (Ks) 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 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 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√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:

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

Where,

K_I = Allowable stress intensity factor (ksi√in)
K_{Ic} = Lower bound fracture toughness at operating temperature (220 ksi√in)

As shown in Table 2, the allowable K is not exceeded for 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.

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 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 load case 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 does 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 thread in flange component at MPS2 is very flaw tolerant and can operate for 80 years without violating ASME Code, Section XI safety margins. This demonstrates that the thread in flange examinations can be eliminated without affecting the safety of the RPV.

<u>Operating Experience Review Summary</u>

As discussed above, the results of the survey, which includes results from MPS2, 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, data from a total of 94 nuclear units has been obtained to date, with no degradation identified. As shown in Table 3 below, the 94 units represent data from 33 BWRs and 61 PWRs. For the BWR units, a total 3,793 examinations were conducted and for the PWR units, a total of 6,869 examinations were conducted. No service-induced degradation was identified. The response data includes information from BWR and PWR plant designs in operation in the US. 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 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 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 a concern that steam generator tubes might fail before other RCS components, with a resultant bypass of containment. With respect to this alternative request, in these studies 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 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 and the threads in flange are generally not in contact with primary water at plant operating temperatures/pressures). The robustness of the design has been affirmed by analysis at several plants that supported continued plant operation even with a bolt/stud assumed to be out of service.

5. Proposed Alternative and Basis for Use

In lieu of the requirements for a volumetric ultrasonic examination, DNC proposes eliminating the requirement for the RPV threads in flange examination for MPS2.

DNC has confirmed that MPS2 plant-specific parameters (e.g. vessel diameter, number of studs, ISI findings) are consistent with or bounded by the EPRI report.

Since there is reasonable assurance that the proposed alternative is an acceptable alternate approach to the performance of the ultrasonic examinations, DNC 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, DNC uses a detailed procedure for the removal and care of the RPV studs. After the studs are removed, the stud threads and stud hole threads are inspected for damage. RPV stud hole plugs are installed to protect the flange threads from damage. Prior to reinstallation, the studs and stud holes are cleaned as necessary and lubricated. The studs are then replaced and tensioned into the RPV. This activity is performed each refueling outage 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.

In addition, other inspection activities, including the system leakage test (examination category B-P), which is conducted each refueling outage, will continue to be performed.

6. Duration of Proposed Alternative

DNC requests approval of this alternative for the remainder of the fourth 10-year inservice inspection interval for MPS2 that began on April 1, 2010 and is scheduled to end on March 31, 2020.

7. References

1. EPRI Nondestructive Evaluation Report- Reactor Pressure Vessel Threads in Flange Examination Requirements. 3002007626; dated March 2016, Electric Power Research Institute (EPRI). ADAMS Accession Number ML16221A068

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

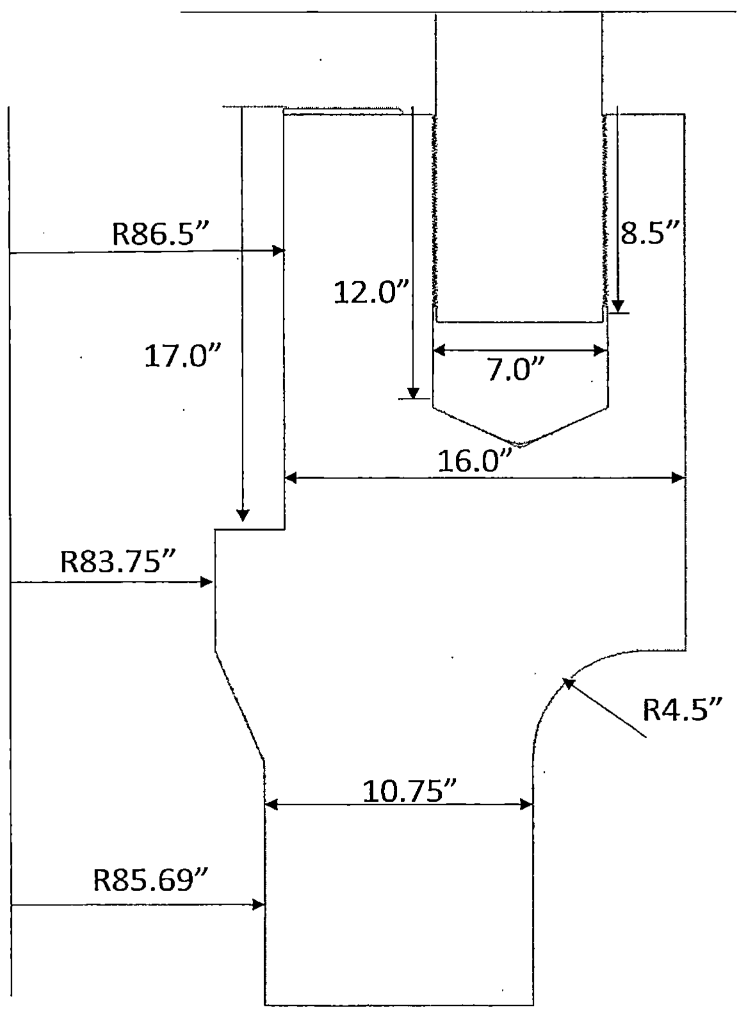


Figure 1
Modeled Dimensions

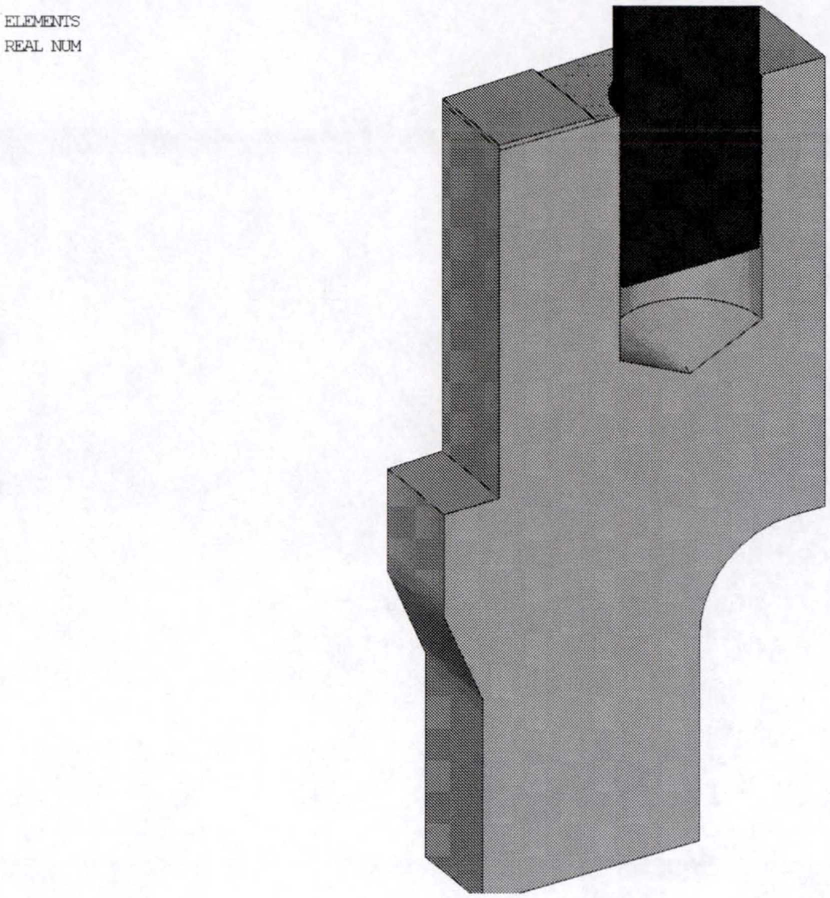
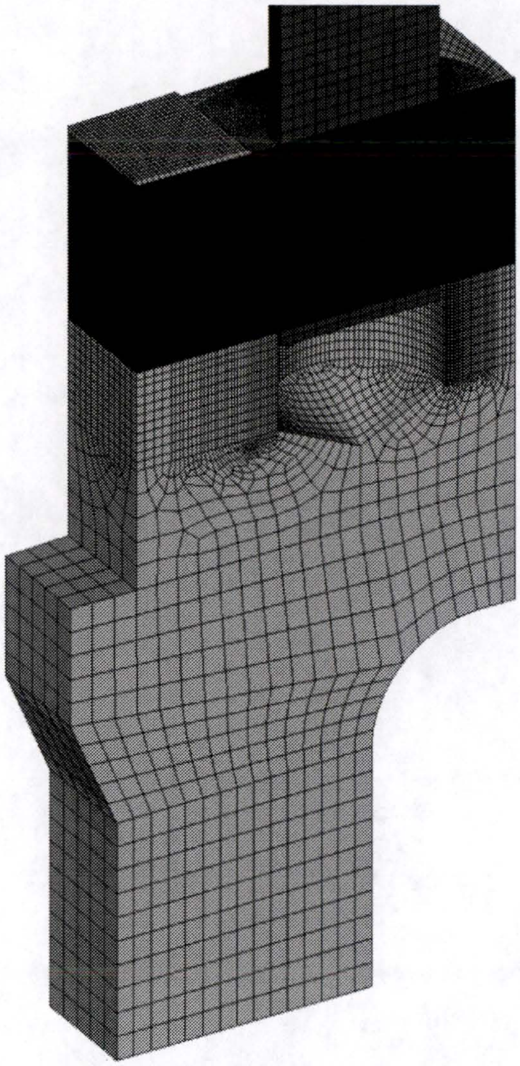


Figure 2
Finite Element Model Showing Bolt and Flange Connection



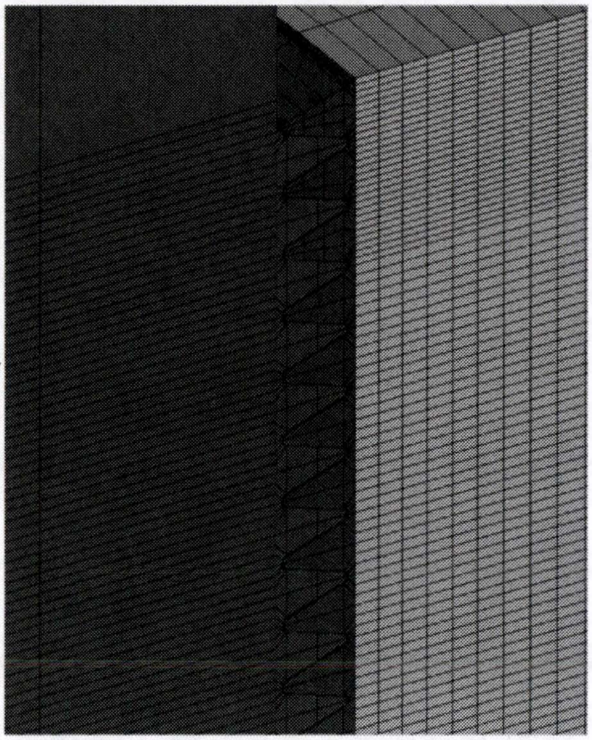


Figure 3
Finite Element Model Mesh with Detail at Thread Location

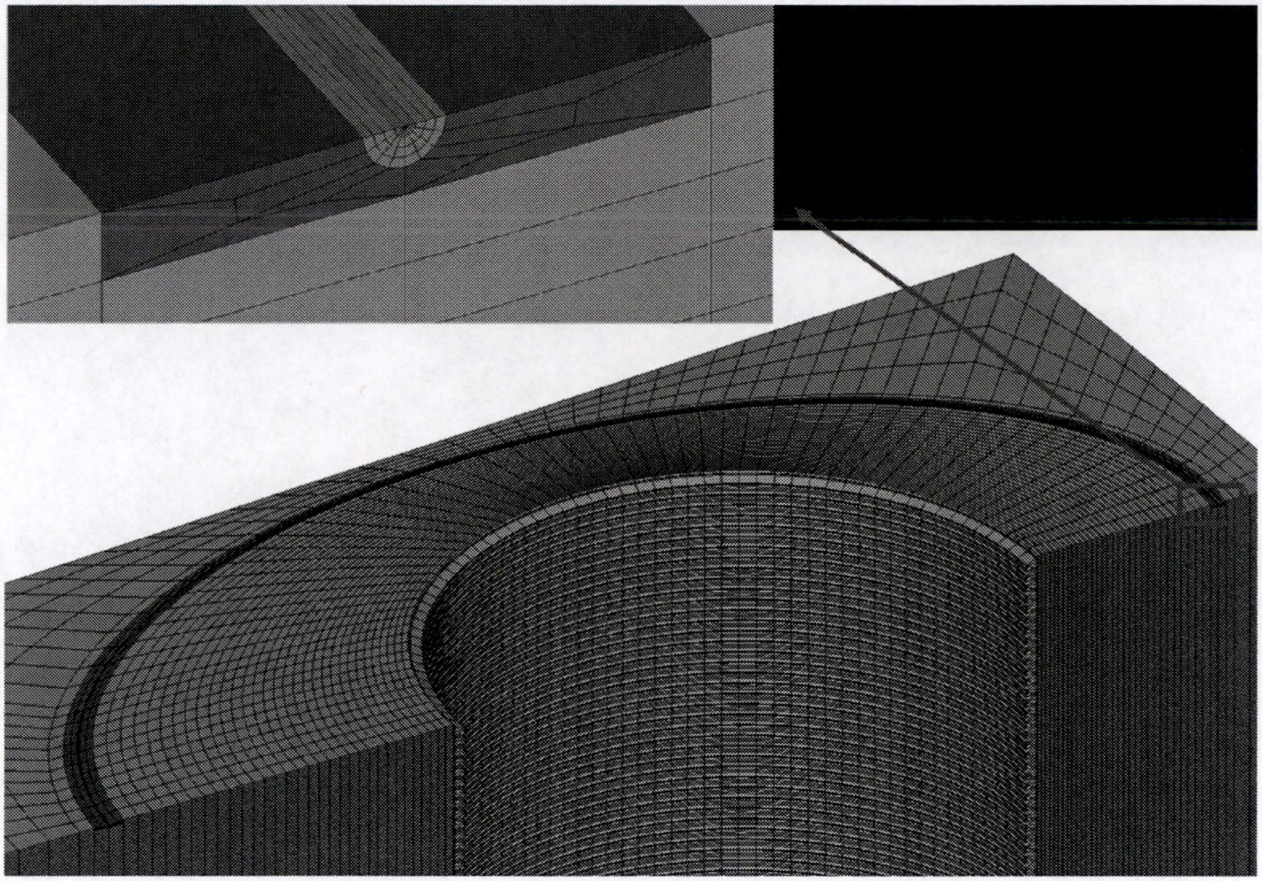


Figure 4
Cross Section of Circumferential Flaw with Crack Tip Elements Inserted after 10th Thread from Top of Flange

ATTACHMENT 2

ALTERNATIVE REQUEST IR-3-30
PROPOSED ALTERNATIVE TO ASME SECTION XI FOR ELIMINATION OF REACTOR VESSEL – THREADS IN FLANGE EXAMINATION

MILLSTONE POWER STATION UNIT 3
DOMINION NUCLEAR CONNECTICUT, INC.

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(1)

-- Proposed Alternative Provides an Acceptable Level of Quality and Safety --

1. ASME Code Components Affected

ASME Code Class:	Code Class 1
References:	ASME Section XI, Paragraph IWB-2500
Examination Category:	B-G-1
Item Number:	B6.40
Description:	Reactor Vessel – Threads in Flange
Components:	Pressure retaining bolting greater than 2 inches

2. Applicable Code Edition and Addenda

ASME Section XI, 2004 Edition (No Addenda).

3. Applicable Code Requirement

The Reactor Pressure Vessel (RPV) threads in flange 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 Figure IWB-2500-12.

4. Background and Reason for Request

Dominion Nuclear Connecticut, Inc. (DNC) has worked with the industry to evaluate eliminating the RPV threads in flange examination requirement. Licensees in the United States (US) and internationally have worked with the Electric Power Research Institute (EPRI) to produce a technical report (Reference 1) which provides the basis for elimination of the requirement. The report evaluates potential degradation mechanisms and includes a stress analysis / flaw tolerance evaluation. The report also includes a review of operating experience (OE) based on a survey of inspection results associated with RPV flange/bolting and related RPV assessments for over 168 units. The evaluation concludes that the safety benefit of the current examination requirements are not commensurate with the associated impact on

worker exposure, personnel safety, radwaste, and increased time at reduced RCS inventory. 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 included in the EPRI report. Potential degradation types 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 notes a general conclusion from Reference 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 inservice inspections do not provide additional value going forward. As discussed in the OE review summary below, the RPV threads in flange have received over 10,000 inservice inspections, 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, 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 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 and documented 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 units (ten PWRs and six Boiling Water Reactors (BWRs)) were considered in the analysis. The evaluation was performed using a geometrical configuration that bounds the sixteen units considered in this effort. The details of the RPV parameters for MPS3 as compared to the bounding values used in the evaluation are shown in Table 1. As shown in the table, the diameter of the stud used in the analysis is smaller than that at MPS3. Hence, the

stresses from the analyzed configuration would be conservative in application to MPS3. Dimensions of the analyzed geometry are shown in Figure 1.

Table 1: Comparison of MPS Unit 3 Parameters to Bounding Values Used in Analysis

Plant	No. of Studs	Stud Nominal Diameter (inches)	RPV Inside Diameter at Stud Hole (inches)	Flange Thickness at Stud Hole (inches)	Design Pressure (psia)
MPS3	54	7	173	16	2500
Range for 16 Units Considered	54 - 60	6.0 – 7.0	157 - 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 the 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 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.

Flaw Tolerance Evaluation

A flaw tolerance evaluation was performed using the results of the stress analysis 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.

At four flaw depths of a 360° inside-surface-connected, partial-through-wall circumferential flaw, stress intensity factors (Ks) 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 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 2 for the four crack depths. Because the crack tip varies in depth around the circumference, the maximum K from the 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√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:

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

Where,

K_I = Allowable stress intensity factor (ksi√in)
K_{Ic} = Lower bound fracture toughness at operating temperature (220 ksi√in)

As shown in Table 2, the allowable K is not exceeded for 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.

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 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 load case 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 does 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 show that the threads in flange component at MPS3 is very flaw tolerant and can operate for 80 years without violating ASME Code, Section XI safety margins. This demonstrates that the threads 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 MPS3 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, data from a total of 94 units has been obtained to date, with no degradation identified. As shown in Table 3 below, the 94 units represent data from 33 BWRs and 61 PWRs. For the BWR units, a total 3,793 examinations were conducted and for the PWR units a total of 6,869 examinations were conducted. No service-induced degradation was identified. The response data includes information from BWR and PWR plant designs in operation in the US. 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 ATWS event. In particular, the reactor coolant system 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 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 a concern that steam generator tubes might fail before other RCS components, with a resultant bypass of containment. With respect to this alternative request, in these studies 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 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 and generally not in contact with primary water at plant operating temperatures/pressures). The robustness of the design has been affirmed by analysis at several plants that supported continued plant operation even with a bolt/stud assumed to be out of service.

5. Proposed Alternative and Basis for Use

In lieu of the requirements for a volumetric ultrasonic examination, DNC proposes eliminating the requirement for the RPV threads in flange examination for MPS3.

DNC has confirmed that MPS3 plant-specific parameters (e.g. vessel diameter, number of studs, ISI findings) are consistent with or bounded by the EPRI report.

Since there is reasonable assurance that the proposed alternative is an acceptable alternate approach to the performance of the ultrasonic examinations, DNC 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, DNC uses a detailed procedure for the removal and care of the RPV studs. After the studs are removed, the stud threads and stud hole threads are inspected for damage. RPV stud hole plugs are installed to protect the flange threads from damage. Prior to reinstallation, the studs and stud holes are cleaned as necessary and lubricated. The studs are then replaced and tensioned into the RPV. This activity is performed each refueling outage 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.

In addition, other inspection activities, including the system leakage test (examination category B-P), which is conducted each refueling outage, will continue to be performed.

6. Duration of Proposed Alternative

DNC requests approval of this alternative for the remainder of the third 10-year inservice inspection interval for MPS3 that began on April 23, 2009 and is scheduled to end on April 22, 2019.

7. References

1. EPRI Nondestructive Evaluation Report- Reactor Pressure Vessel Threads in Flange Examination Requirements. 3002007626; dated March 2016 Electric Power Research Institute (EPRI). ADAMS Accession Number ML16221A068

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

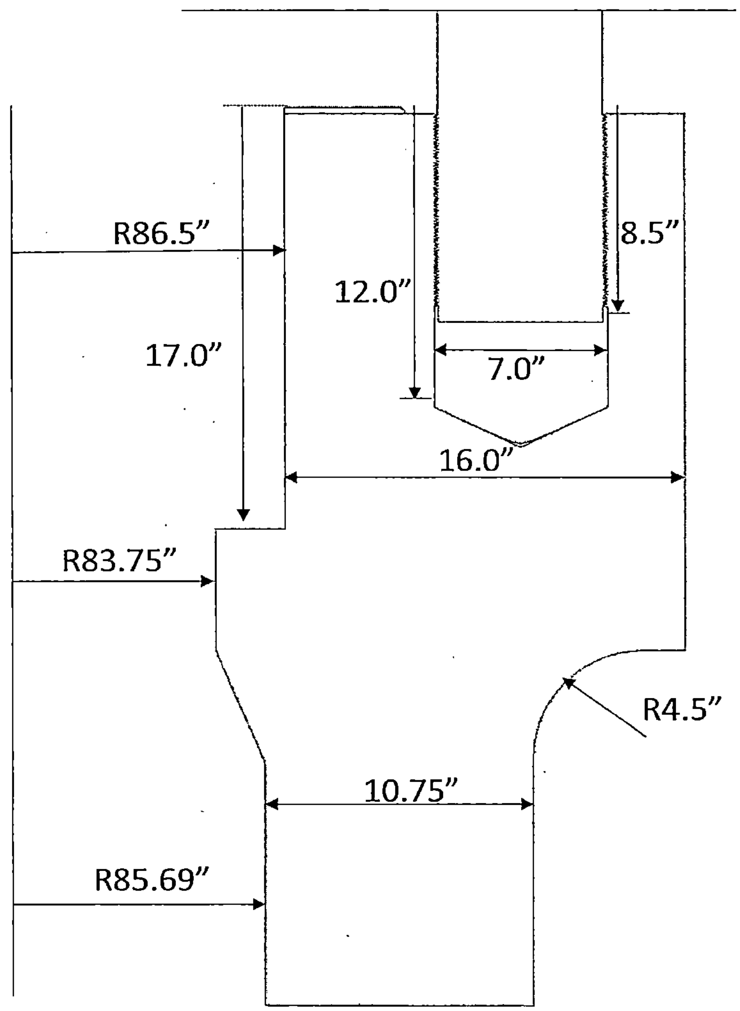


Figure 1
Modeled Dimensions

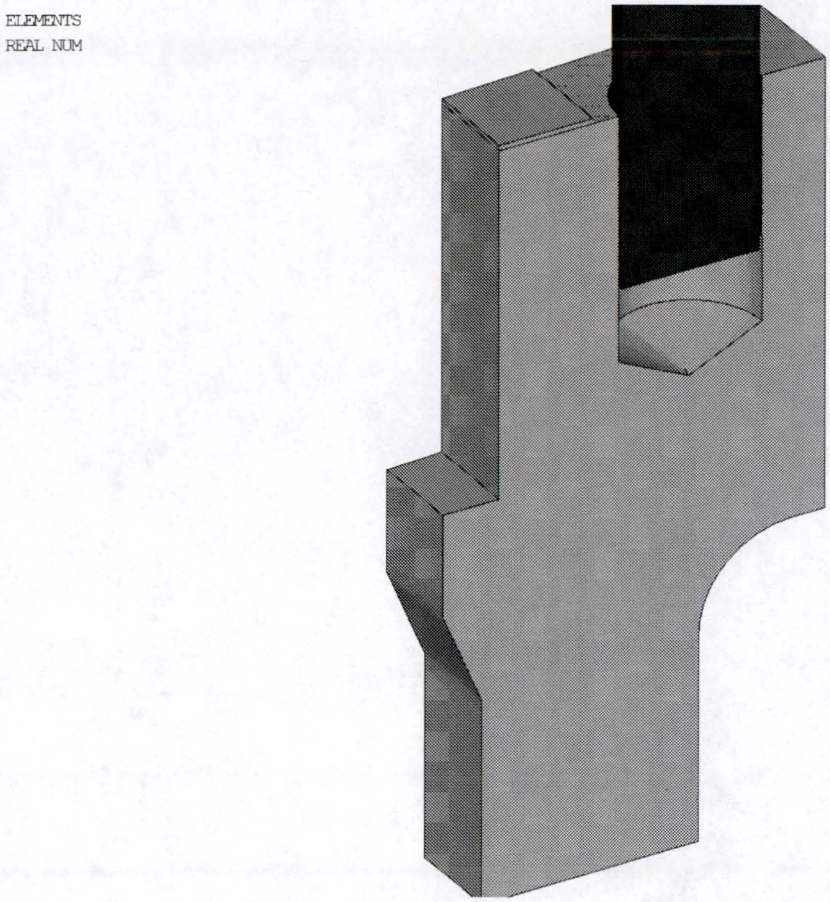


Figure 2
Finite Element Model Showing Bolt and Flange Connection

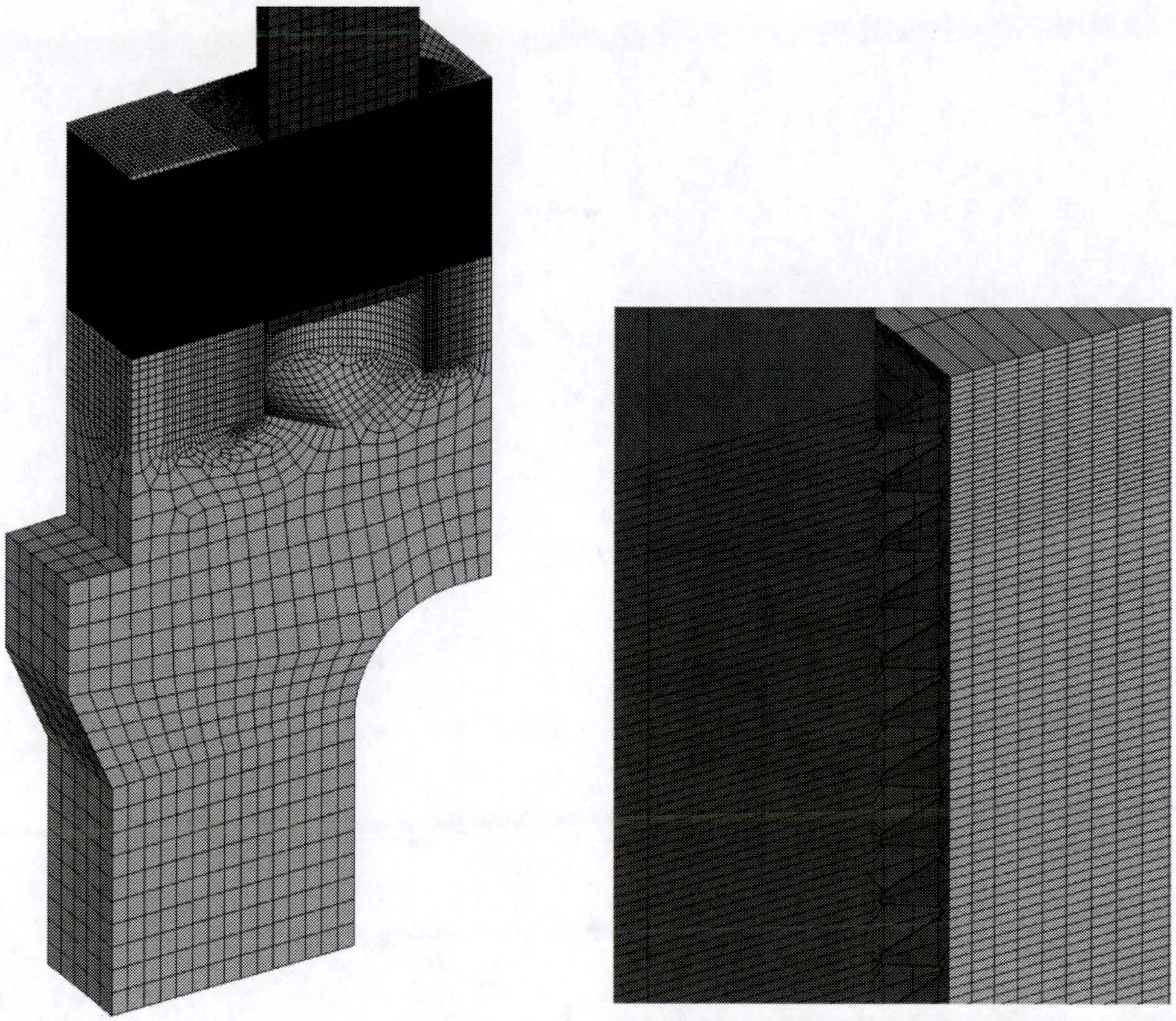


Figure 3
Finite Element Model Mesh with Detail at Thread Location

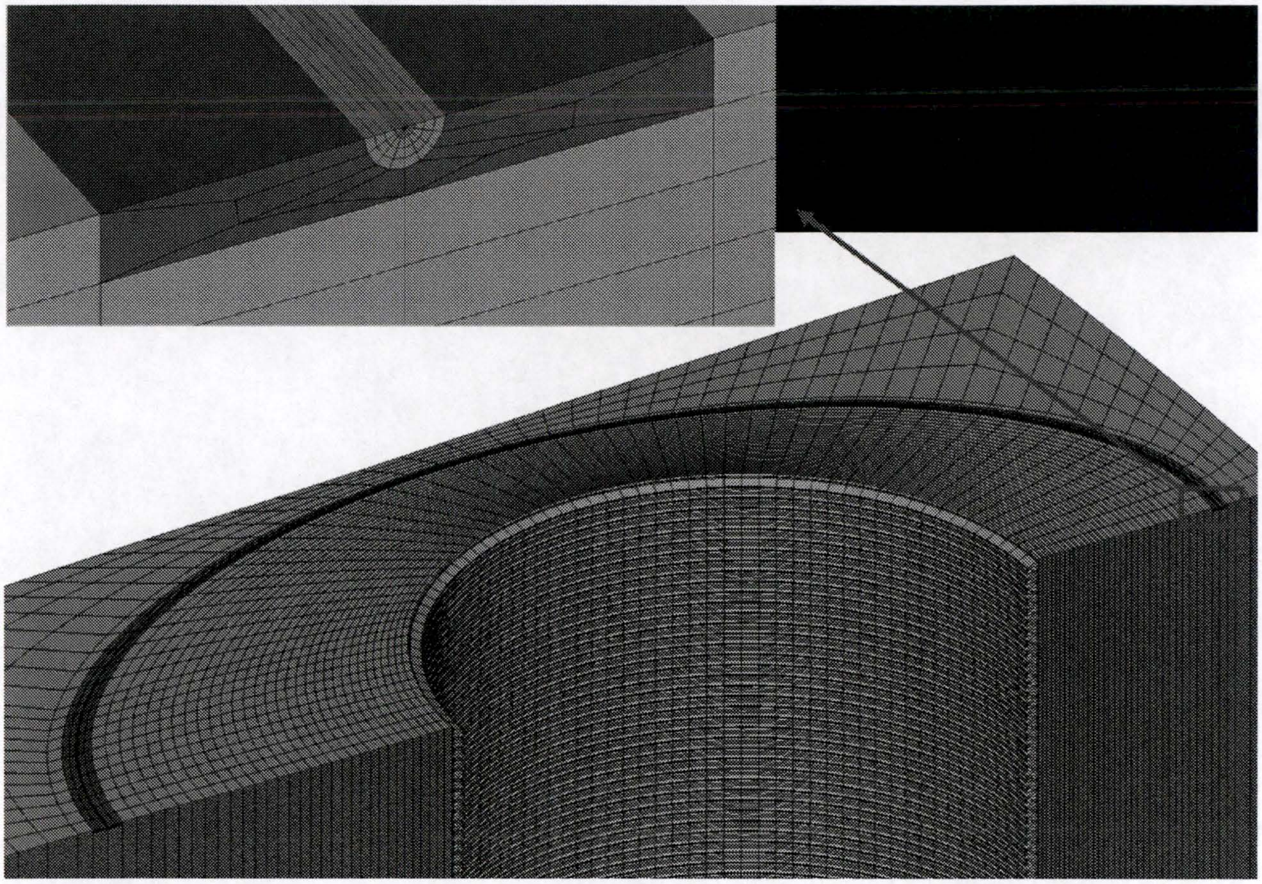


Figure 4
Cross Section of Circumferential Flaw with Crack Tip Elements Inserted after 10th Thread from Top of Flange