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February 28, 2019 10 CFR 50.55a
NRC-19-0016

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Fermi 2 Power Plant
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Submittal of the Inservice Inspection / Nondestructive Examination Program Relief Requests for the Fourth Ten-year Interval

Pursuant to 10 CFR 50.55a, "Codes and Standards," paragraphs (z)(1) and (z)(2), DTE Electric Company (DTE) hereby requests NRC approval of the following relief requests for the fourth ten-year interval of the Inservice Inspection (ISI) / Nondestructive Examination (NDE) program at Fermi 2 which starts on May 2, 2019 and is projected to end December 31, 2029, using the provisions of IWA-2430 to extend the interval to include the Fall 2029 refueling outage.

- RR-A36, Alternative Pressure Testing Requirements for the Reactor Pressure Vessel Flange Leak-Off Piping
- RR-A39, Use of Boiling Water Reactor Vessel Internals Project Guidelines in Lieu of ASME Code Requirements
- RR-A40, Proposed Alternative to Utilize Code Case N-513-4
- RR-A41, Use of ASME Code Case N-864 for the Elimination of Reactor Pressure Vessel – Threads in Flange Examination

DTE additionally requests NRC approval of relief request RR-A37, Alternative Requirements for Examination of Boiling Water Reactor Nozzle Inner Radius Sections and Nozzle-to-Shell Welds, for the remainder of the current operating license.

The enclosure to this letter provides details of these relief requests. Relief requests RR-A36, RR-A37, RR-A39, and RR-A40 have been previously approved by the NRC for use at Fermi 2.

DTE requests NRC approval of these relief requests by February 28, 2020 to support planned testing for the fourth ten-year ISI/NDE Interval during the next refuel outage scheduled in the spring of 2020.

No new commitments are being made in this submittal.

Should you have any questions or require additional information, please contact Mr. Scott A. Maglio, Manager – Nuclear Licensing, at (734) 586-5076.

Sincerely,

Paul Fessler
Senior Vice President and CNO

Enclosures:
1. RR-A36 – Alternative Pressure Testing Requirements for the RPV Flange Leak-Off Piping
2. RR-A37 – Alternative Requirements for Examination of Boiling Water Reactor Nozzle Inner Radius Sections and Nozzle-to-Shell Welds
3. RR-A39 – Use of Boiling Water Reactor Vessel Internals Project Guidelines in Lieu of ASME Code Requirements
4. RR-A40 – Proposed Alternative to Utilize Code Case N-513-4
5. RR-A41 – Use of ASME Code Case N-864 for the Elimination of Reactor Pressure Vessel – Threads in Flange Examination

cc: NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 5, Region III
Regional Administrator, Region III
Michigan Public Service Commission
Regulated Energy Division (kindschl@michigan.gov)

Enclosure 1 to
NRC-19-0016

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

RR-A36
Alternative Pressure Testing Requirements for the RPV Flange Leak-Off Piping

10 CFR 50.55a Request Number
RR-A36
Fourth Interval Relief Request

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(2)

-Hardship or Unusual Difficulty
without a Compensating Increase in Level of Quality or Safety-

1. ASME Code Components Affected

ASME Code Class:	Code Class 1
References:	ASME Section XI, Table IWB-2500-1 and IWB-5222
Examination Category:	B-P (All Pressure Retaining Components)
Item Number:	B15.10 / B15.20
Description:	Alternative Pressure Testing Requirements for the RPV Flange Leak-Off Piping
Components:	NPS 1 RPV Flange Seal Leak-Off Piping

2. Applicable Code Edition and Addenda

ASME Section XI, 2013 Edition

3. Applicable Code Requirement

IWB-2500, Table IWB-2500-1, Code Category B-P, Item Number B15.10 requires that all Class 1 pressure retaining components be visually examined (VT-2) each refueling outage. The required system pressure test can be either a hydrostatic test or a system leakage test. The system leakage test is performed at a pressure not less than the pressure corresponding to 100% rated reactor power. Per IWB-5222(a), the pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity. Per IWB-5222(b), the pressure retaining boundary not pressurized with the valves in their normal startup lineup shall be pressurized and examined during the test conducted at or near the end of the interval and shall extend to all Class 1 pressure retaining components within the system boundary.

4. Reason for Request

As discussed in Section 3, "Applicable Code Requirements", ASME Section XI, 2013 Edition requires that Class 1 pressure boundary piping shall be pressurized and examined during the test performed at or near the end of the interval. The Reactor Pressure Vessel (RPV) head flange seal leak detection piping is separated from the reactor coolant pressure boundary by one passive membrane, which is an O-ring located on the inner vessel flange as shown in Attachment 1. A second O-ring is located on the outside of the tap in the vessel flange. Failure of the inner O-ring is the only condition under which this line is pressurized. Therefore, the line is not expected to be pressurized during the system pressure test required in IWB-2500, Category B-P and IWB-5222(b).

The configuration of this piping precludes system pressure testing while the vessel head is removed because the configuration of the vessel tap coupled with the high test pressure prevents the tap in the flange from being temporarily plugged or connected to other piping. The opening in the flange is smooth walled, making the effectiveness of a temporary seal very limited. Failure of a temporary test seal could possibly cause ejection of the device used for plugging or connecting to the vessel flange. To perform the system leakage test in accordance with the Code requirements, the RPV head flange seal detection piping would have to be redesigned, fabricated, and installed. This would impose hardship or unusual difficulty without a compensating increase in level of quality or safety.

The configuration also precludes pressurizing the line with the head installed because the seal prevents complete filling of the piping, which has no vent available. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips that are housed in recessed cavities in the flange face. If a pressure test were to be performed with the head on, the inner O-ring would be pressurized in a direction opposite to its design function. This test pressure would result in a net inward force on the inner O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The thin O-ring material would very likely be damaged by the inward force. The design of this line makes the ASME Code required system leakage test impractical either with the vessel head installed or removed.

5. Proposed Alternative and Basis for Use

In accordance with 10 CFR 50.55a(z)(2), DTE Electric Company (DTE) is requesting a proposed alternative to the requirements of IWB-5222(b). In lieu of the requirements, a VT-2 visual examination will be performed on the subject piping during vessel flood-up each refueling outage. The hydrostatic head developed due to the water above the vessel flange during flood-up will allow for the detection of any gross indications in the piping.

The flange seal leak-off line is essentially a leakage collection and detection system. The line would only function as a Class 1 pressure boundary if the inner O-ring fails thereby pressurizing the line. During this time, the control room annunciator would be in alarm. If the annunciator ceases to be in alarm, this would indicate that the outer O-ring or seal leak-off line had failed and resulted in a reactor coolant pressure boundary leak. This would

require immediate plant shutdown. Per IWA-5241 when components are not accessible, examination may be performed remotely or by installed leakage detection systems.

Fermi 2 has an installed leakage detection system. DTE has implemented a periodic Preventive Maintenance Event (PM Event B564) to pressurize an isolable section of the leak detection line to verify that the pressure switch is operative and in calibration. Performance of this event satisfies the intent of IWA-5241-(c).

This line is also inspected during the VT-2 system leakage test. Additionally, the line is filled at static head pressure during the outage while the reactor cavity is flooded. Any gross leakage would be detected during drywell entries. The VT-2 is considered to be met by PM Event B564 along with a visual inspection of the line during flood-up.

Pursuant to 10 CFR 50.55a(z)(2), approval is requested to use the proposed alternative described above in lieu of the Code requirement on the basis that the Code requirements establish a hardship or unusual difficulty without a compensating increase in the level of quality or safety.

6. <u>Duration of Proposed Alternative</u>

This relief is requested for the duration of the Fourth Inservice Inspection Interval, which begins on May 2, 2019 and is scheduled to end on December 31, 2029.

7. <u>Precedents</u>

1. Fermi 2 Relief Request RR-A36, “Evaluation of the 3rd 10-Year Interval Inservice Inspection Request for Relief No. RR-A36 on end of Interval System Pressure Test” (TAC No. ME3118) as approved by the NRC in a letter dated October 1, 2010 (ML102630094).

Reactor Pressure Vessel Seal Leak-Off Details

10 CFR 50.55a Request Number
RR-A36
Fourth Interval Relief Request
Attachment 1

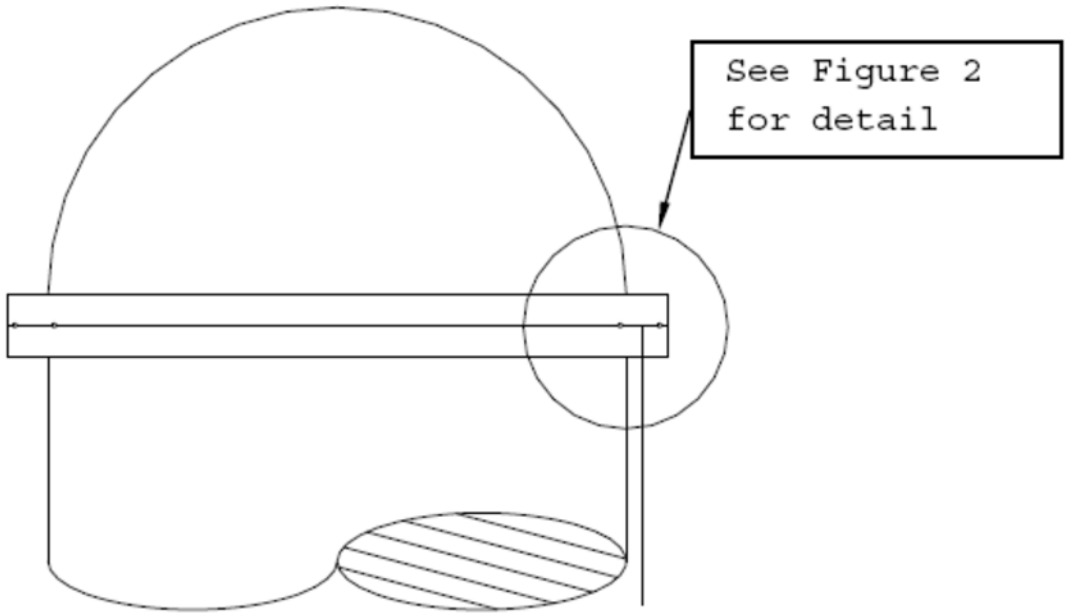

Figure 1:
REACTOR PRESSURE VESSEL HEAD FLANGE
LEAK-OFF LINE CONFIGURATION

10 CFR 50.55a Request Number
RR-A36
Fourth Interval Relief Request
Attachment 1

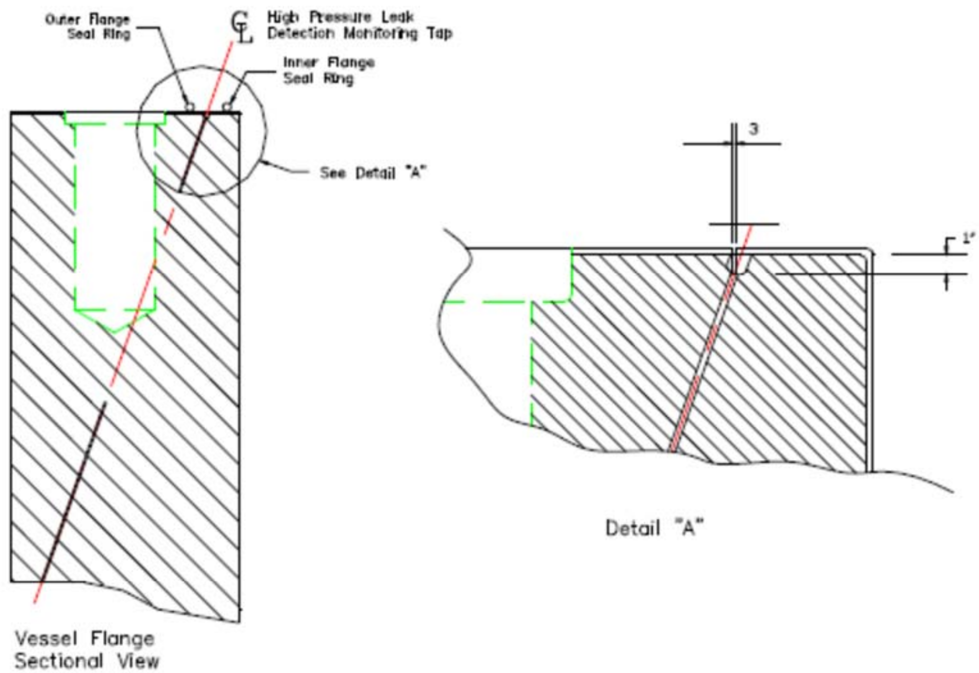

Figure 2
REACTOR PRESSURE VESSEL HEAD FLANGE
LEAK-OFF LINE DETAILS

Enclosure 2 to
NRC-19-0016

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

RR-A37
Alternative Requirements for Examination of Boiling Water Reactor Nozzle Inner Radius Sections and Nozzle-to-Shell Welds

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

-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Components Affected

ASME Code Class:	Code Class 1
References:	ASME Section XI, 2013 Edition
	Code Case N-702
	BWRVIP-108: BWR Vessel and Internals Project, “Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,” EPRI Technical Report 1003557, October 2002
	BWRVIP-241: BWR Vessel and Internals Project, “Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,” EPRI Technical Report 1021005, October 2010
Examination Category:	B-D
Item Numbers:	B3.90, B3.100
Description:	Alternative Requirements for Examination of Boiling Water Reactor (BWR) Nozzle Inner Radius Sections and Nozzle-to-Shell Welds
Components:	N1, N2, N3, N5, N6, N7, and N8 Nozzles (see Attachment 2 for specific nozzle identifications)

2. Applicable Code Edition and Addenda

ASME Section XI, 2013 Edition.

3. Applicable Code Requirement

The applicable Code requirements are contained in Table IWB-2500-1, Examination Category B-D, “Full Penetration Welded Nozzles in Vessels.” Class 1 nozzle-to-vessel weld and nozzle inner radii examination requirements are delineated in Item Number B3.90, "Nozzle-to-Vessel Welds," and B3.100, "Nozzle Inside Radius Section." All nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles require a volumetric examination to be performed each interval. Additionally, for ultrasonic

examinations, ASME Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," is required.

All of the nozzle assemblies identified in Attachment 2 are full penetration welds.

4. Reason for Request

Regulatory Guide 1.147 (Reference 2) conditionally accepts the use of Code Case N-702 (Reference 3). This code case provides an alternative to performing examination of 100% of the nozzle-to-vessel welds and inner radii for examination category B-D nozzles with the exception of the Feedwater and Control Rod Drive Return Line (CRDRL) nozzles. The alternative is to perform examination of a minimum of 25% of the nozzle inner radii and nozzle-to-shell welds, including at least one nozzle from each system and nominal pipe size, excluding the Feedwater and CRDRL Nozzles. Regulatory Guide 1.147, Table 2 conditionally accepts the use of Code Case N-702 with the following condition:

> *"The technical basis supporting the implementation of this Code Case is addressed by BWRVIP-108: BWR Vessel and Internals Project, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," EPRI Technical Report 1003557, October 2002 (ML023330203) and BWRVIP-241: BWR Vessels and Internals Project, "Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," EPRI Technical Report 1021005, October 2010 (ML11119A041). The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of NRC Safety Evaluation regarding BWRVIP-108 dated December 18, 2007 (ML073600374) or Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 dated April 19, 2013 (ML13071A240) are met. The evaluation demonstrating the applicability of the Code Case shall be reviewed and approved by the NRC prior to the application of the Code Case."*

5. Proposed Alternative and Basis for Use

In accordance with 10 CFR 50.55a(z)(1), DTE Electric Company (DTE) requests an alternative from performing the required examinations on 100% of the nozzle assemblies identified in Table 1 below (see Attachment 2 for the list of Reactor Pressure Vessel (RPV) nozzles applicable to this relief request). As an alternative, for all welds and inner radii identified in Table 1, DTE proposes to examine a minimum of 25% of the nozzle-to-vessel welds and inner radii sections, including at least one nozzle from each system and nominal pipe size, in accordance with Code Case N-702. For the nozzle assemblies identified in Attachment 2, this would mean 25% from each of the groups identified below:

Table 1
Fermi 2 Nuclear Power Plant Summary

Identification Number Group	Description	Size Inches	Total Number	Minimum Number to be Examined
N1	Recirculation Outlet	28	2	1
N2	Recirculation Inlet	12	10	3
N3	Main Steam Outlet	26	4	1
N5	Core Spray	10	2	1
N6	Spare Nozzles	6	2	1
N7	Closure Head Vent	4	1	1
N8	Jet Pump Instrumentation	4	2	1

The minimum number of listed examinations in Table 1 will be scheduled in accordance with Table IWB-2411-1.

The initial use of Code Case N-702 was supported by EPRI Technical Report 1003557, BWRVIP-108 (Reference 5) as approved by the NRC (Reference 6). The evaluation contained in BWRVIP-108 found that failure probabilities at the nozzle blend radius region and nozzle-to-vessel shell weld due to a low temperature overpressure event are very low (i.e., $<1 \times 10^{-6}$ for 40 years) with or without inservice inspection. The report concludes that inspection of 25% of each nozzle type is technically justified.

In the NRC safety evaluation report (SER) of BWRVIP-108, the NRC indicated that licensees may reference BWRVIP-108 as the technical basis in their request for alternative to use Code Case N-702, however the NRC required each licensee to demonstrate the plant-specific applicability of the BWRVIP-108 using the criteria contained in the NRC SER.

The BWRVIP then performed supplemental analysis for BWR reactor pressure vessel recirculation inlet and outlet nozzle-to-shell welds and nozzle inner radii. The results of the supplemental analysis were published in BWRVIP-241 (Reference 7) and have been approved by the NRC (Reference 8).

The NRC SER for BWRVIP-241 concluded that each licensee who requests relief from the ASME Code, Section XI requirements for RPV nozzle-to-vessel shell welds and nozzle inner radius sections may reference the BWRVIP-241 report as the technical basis for the use of ASME Code Case N-702 as an alternative. However, each licensee should demonstrate the plant-specific applicability of the BWRVIP-241 report. With NRC approval of BWRVIP-241, licensees may use the criteria of BWRVIP-108 or BWRVIP-241 as noted in Regulatory Guide 1.147.

The analyses in BWRVIP-108 and BWRVIP-241 are based on the initial 40 years of plant operation. In addition, the BWRVIP published BWRVIP-241, Appendix A (Reference 9) to

demonstrate that the content of BWRVIP-241 provides the necessary information to comply with the technical information requirements of the license renewal rule (Reference 10).

For the aspects of the analysis contained in BWRVIP-108 and BWRVIP-241 that may be affected by the extended operating period (neutron fluence and thermal cycles), BWRVIP-241 Appendix A concludes:

1. Neutron Fluence

   "The effect of neutron fluence on material fracture toughness is the only aging effect for the nozzles that requires aging management review for license renewal. This aging effect will be managed by an inspection program incorporating the recommendations described in Section 1 and the resulting plant-specific strategy (i.e., plant-specific analysis and reinspection schedule). The inspection methods and implementation guidance addresses the:

   - *Location that requires inspection.*
   - *Extent of baseline inspection for each location.*
   - *Extent of reinspection for each location.*
   - *Methodology for scope expansion should flaws be detected.*
   - *Analysis methods to determine the need for corrective action and establish a reinspection schedule if flaws are detected.*

   Implementation of the inspection recommendations in an inspection program and the resulting plant-specific strategy will provide verification of the structural integrity of the nozzles. Therefore, there is reasonable assurance that crack initiation and growth will be adequately managed so that the intended functions of the nozzles will be maintained consistent with the CLB in the extended operating period."

2. Thermal Cycles

   "*It should be noted that the actual number of thermal cycles experienced by a plant might exceed those assumed in BWRVIP-108NP. However, studies conducted by BWRVIP have concluded that stress corrosion cracking initiation and growth is a much more dominant contributor to probability of failure when compared to fatigue crack growth from thermal cycles. Therefore, the number of actual plant cycles can greatly exceed those assumed in the BWRVIP reports without having a significant effect on the results."*

The NRC SER of BWRVIP-241, Appendix A (Reference 11) concluded, in part, that use of BWRVIP-108NP and BWRVIP-241 are acceptable as the technical basis for requesting use of Code Case N-702 for the extended operating period provided the conditions of Regulatory Guide 1.147 are met.

Fermi 2 License Renewal Application (LRA) (Reference 12), Section 4.2.1 indicates that the only reactor vessel nozzles included in the reactor vessel beltline during the extended operating period (defined as that portion of the reactor shell projected to exceed an end of life fluence greater than 1E+17 n/cm^2) are the N16 nozzles (water level instrumentation). Reactor vessel nozzles subject of this request are not projected to exceed the fluence threshold making them part of the beltline region. The methods used for this determination are accepted by the NRC in Section 4.2.2.2 of NUREG-2210 (Reference 13).

NUREG-2210, Section 4.3.1.1.4 concludes that Fermi's actions to address the effects of cumulative fatigue damage on the intended functions of the RPV components listed in LRA Table 4.3-2 will be adequately managed by the Fatigue Monitoring Program for the period of extended operation. Additionally, BWRVIP-241, Appendix A, states that the dominant contributor to the probability of failure (POF) is stress corrosion crack initiation and growth, not fatigue. Therefore, the additional cycles that may be accumulated during the extended operating period have no significant effect on the POF determined by BWRVIP-108NP and BWRVIP-241.

For this request, Fermi 2 is applying the NRC criteria for plant-specific applicability from Section 5.0 of the SER for BWRVIP-241, which requires the following:

(1) The maximum Reactor Pressure Vessel (RPV) heatup/cooldown rate is limited to less than 115ºF per hour.

- Per TS SR 3.4.10.1, Fermi 2 limits the maximum heatup and cooldown rates to less than or equal to 100°F in any one hour period and thus meets the requirement of Criterion 1.

<u>For recirculation inlet nozzles</u> (Reference Attachment 1)

(2) $(pr/t)/C_{RPV} \leq 1.15$

- The calculation for the Fermi 2 N2 Nozzle results in 0.89, which is less than 1.15.

(3) $[p(r_o^2+r_i^2)/(r_o^2-r_i^2)]/C_{NOZZLE} \leq 1.47$

- The calculation for the Fermi 2 N2 Nozzles results in 1.23, which is less than 1.47.

Therefore, these criteria are met for the Recirculation Inlet Nozzles.

<u>For recirculation outlet nozzles</u> (Reference Attachment 1)

(4) $(pr/t)/C_{RPV} \leq 1.15$

- The calculation for the Fermi 2 N1 Nozzles results in 1.07, which is less than 1.15.

(5) $[p(r_o^2+r_i^2)/(r_o^2-r_i^2)]/C_{NOZZLE} \leq 1.59$

- The calculation for the Fermi 2 N1 Nozzles results in 1.07, which is less than 1.59.

Therefore, the criteria are met for the Recirculation Outlet Nozzles.

Based on meeting Criterion 1 through 5, the analysis contained in BWRVIP-108NP and BWRVIP-241 is applicable to the Fermi 2 reactor vessel nozzles identified in Table 1.

Additionally, Code Case N-702 permits a VT-1 examination to be used in lieu of the volumetric examination for the inner radii (i.e., Item No. B3.100, "Nozzle Inside Radius Section"). This VT-1 examination is outlined in Code Case N-648-1 (Reference 4). Fermi 2 will perform either volumetric examination or VT-1 examination of the inner radius as required by Code Case N-702. The VT-1 visual examination, if performed, will be in accordance with Code Case N-648-1 or a later approved revision subject to the conditions of Regulatory Guide 1.147.

6. Conclusion

Pursuant to the provisions of 10 CFR 50.55a(z)(1), Fermi 2 requests NRC approval to implement the requirements of Code Case N-702 as an alternative to the requirements of ASME Section XI, Examination Category B-D for the nozzles identified in Table 1. Based on information contained in this request, BWRVIP-108NP, BWRVIP-241, and BWRVIP-241, Appendix A, as approved by the NRC, the alternative examination requirements of Code Case N-702 will provide an acceptable level of quality and safety for the remainder of the current operating license including the extended operating period.

In the event Fermi 2 elects to perform a VT-1 of the nozzle inner radii as permitted by Code Case N-702, the VT-1 will be performed in accordance with Code Case N-648-1 including the conditions contained in Regulatory Guide 1.147 or a later revision, as approved and subject to the condition(s) of Regulatory Guide 1.147. Fermi will not use the VT-1 examination option unless coverage greater than 90% of the exam surface area is demonstrated or visual exam surface area is demonstrated to be greater than achievable by volumetric exam.

7. Duration of Proposed Alternative

Typically, the duration for requests approved under the provisions of 10 CFR 50.55a(z) are limited to the active inservice inspection ten-year interval for which they are submitted. However, there are instances where the NRC approved requests under the provisions of 10 CFR 50.55a(z) for durations longer than the 10 years associated with an inservice inspection interval. For example, licensee’s requests to reduce examinations of BWR reactor vessel circumferential welds based on BWRVIP-05 (Reference 14).

In an NRC internal memorandum (Reference 15), among other topics related to Code Case N-702, the NRC concluded that their approval for multiple intervals is appropriate if the licensee uses bounding values for fluence up to 60 years. The Vessels and Internals Integrity Branch, Division of Engineering, Office of Nuclear Reactor Regulation (EVIB) indicated a high degree of confidence in this technical position and believes there is little likelihood that new information would be discovered that would question the frequency of inspections included in Code Case N-702.

As stated herein, the bounding fluence for up to 60 years for the reactor vessel nozzles and inner radii subject to this request is less than 1E+17 n/cm^2. Therefore, Fermi 2 requests that this relief be approved for the remainder of the current operating license (ends on midnight March 20, 2045).

8. Precedents

1. Fermi 2 – Relief from the Requirements of the ASME Code, dated June 5, 2018 (EPID L-2017-LLR-0090)

2. River Bend Station, Unit 1 – Request for Alternative RBS-ISI-015 from the Requirements of the ASME Code Regarding Reactor Pressure Vessel Weld Inspection, dated May 17, 2018 (EPID L-2018-LLR-0009)

3. Edwin I. Hatch Nuclear Plant, Unit NOS. 1 and 2 – Proposed Inservice Inspection Alternative HNP-ISI-ALT-05-05, dated April 16, 2018 (CAC NOS. MF9812 and MF9813; EPID L-2017-LLR-0053)

4. Duane Arnold Energy Center – Request for Relief No. RR-03 RE: Proposed use of Alternative Requirements for Nozzle Inner Radius and Nozzle-to-Shell Weld Inspections for Fifth 10-Year Inservice Inspection Interval, dated April 16, 2018 (CAC NO. MF9374; EPID L-2017-LLR-0110)

5. Nine Mile Point Nuclear Station, Units 1 and 2 – Alternative to the Requirements of the American Society of Mechanical Engineers Code, dated December 5, 2017 (CAC NOS. MF9381 and MF9382; EPID L-2017-LLR-0015)

6. LaSalle County Station, Units 1 and 2, Relief from the Requirements of the ASME Code and OM Code RE: Relief Requests I4R-02, I4R-03, I4R-06, I4R-07, and I4R-09, Proposed Alternatives to Various Inservice Inspection Interval (ISI) Requirements of the American Society of Mechanical Engineers (ASME Code), Section XI 2007 Edition with the 2008 Addenda for the Fourth 10-Year ISI Interval, dated November 17, 2017 (EPID NOS. L-2017-LLR-0038 (CAC NOS. MF9760 and MF9761), L-LR-2017-0076 (CAC NOX. MF9762 and MF9763), L-2017-LLR-0033 (CAC NOS. MF9766 and MF9767), L-2017-LLR-0035 (CAC NOS. MF9770 and MF9771), and L-2017-LLR-0037 (CAC NOS. MF9768 and MF9769))

7. Grand Gulf Nuclear Station, Unit 1 – Relief Request GG-ISI-021 Proposing an Alternative for the Fourth 10-Year Inservice Inspection Program, dated October 30, 2017 (CSC NO. MF9752; EPID L-2017-LLR-0031)

9. <u>References</u>

1. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2013 Edition

2. NRC Regulatory Guide 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Revision 18

3. Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1"

4. Code Case N-648-1, "Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles Section XI, Division 1"

5. Electric Power Research Institute (EPRI) Technical Report 1003557, BWRVIP-108: "BWR Vessel and Internals Project Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii"

6. Letter dated December 19, 2007 from Matthew A. Mitchell, Chief Vessels & Internals Integrity Branch Division of Component Integrity Office of Nuclear Reactor Regulation to Rick Libra, BWRVIP Chairman, Safety Evaluation of Proprietary EPRI Report, "BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP-108)"

7. Electric Power Research Institute (EPRI) Technical Report 1021005, BWRVIP-241: "BWR Vessel and Internals Project Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii"

8. Letter dated April 19, 2013 from Sher Bahadur, Deputy Director Division of Policy and Rulemaking Office of Nuclear Reactor Regulation to Dennis Madison, Chairman, BWR Vessel and Internals Project, Final Safety Evaluations of the Boiling Water Reactor Vessel Internals Project (BWRVIP)-241 Report, "Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii" (TAC No. ME6328)

9. BWRVIP-241, Appendix A, "BWR Nozzle Radii and Nozzle-to-Shell Weld Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)"

10. Title 10 of the Code of Federal Regulations, Part 54, "Requirements for License Renewal of Operating Licenses for Nuclear Power Plants"

11. Letter dated April 26, 2017 from Kevin Hsueh, Chief Licensing Processes Branch Division of Policy and Rulemaking Office of Nuclear Reactor Regulation to Tim Hanley, Chairman, BWR Vessel and Internals Project , Revised Final Safety Evaluation for the License Renewal Appendix A for "BWRVIP-241-A: BWR Vessel and Internals Project, Probabilistic Fracture Mechanics Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii", and "BWRVIP-108NP-A: BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-To-Vessel Shell Welds and Nozzle Blend" (TAC NO. MF4638)

12. Letter dated April 24, 2014 from J. Todd Conner, Site Vice President, DTE Energy Company to U.S. Nuclear Regulatory Commission, "Fermi 2 License Renewal Application" (ML14121A532)

13. NUREG 2210, Safety Evaluation Report Related to the License Renewal of Fermi 2, Docket Number 50-341, DTE Electric Company (ML16356A234)

14. Electric Power Research Institute (EPRI) Technical Report 105697, BWRVIP-05: "BWR Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations"

15. Memorandum dated August 21, 2017 from Michael L. Marshall, Jr., Senior Project Manager Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation to James G. Danna, Chief Plant Licensing Branch I Division of Operating Reactor Licensing Office of Nuclear Reactor Regulation, "Summary of Internal NRC Staff Meetings Concerning ASME Code Case N-702, Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division"

Evaluation of Fermi 2 against Criteria 2, 3, 4, and 5 in NRC SER for BWRVIP-241

Note: C_{RPV} and C_{NOZZLE} values are taken from BWRVIP-108 ANSYS Finite Element Model

Recirculation Inlet Nozzles (N2)		**Recirculation Outlet Nozzles (N1)**	
Criterion 2: $(pr/t)/C_{RPV} \leq 1.15$		**Criterion 4:** $(pr/t)/C_{RPV} \leq 1.15$	
p = RPV normal operating pressure	1045 psig	p = RPV normal operating pressure	1045 psig
r = RPV inner radius	127.125”	r = RPV inner radius	127.125”
t = RPV wall thickness	7.6875”	t = RPV wall thickness	7.6875”
$C_{i\text{-}RPV}$	19332	$C_{o\text{-}RPV}$	16171
0.89 < 1.15		**1.07 < 1.15**	
Criterion 3: $[p(r_o^2 + r_i^2) / (r_o^2 - r_i^2)]/C_{NOZZLE} \leq 1.47$		**Criterion 5:** $[p(r_o^2 + r_i^2) / (r_o^2 - r_i^2)]/C_{NOZZLE} \leq 1.59$	
p = RPV normal operating pressure	1045 psig	p = RPV normal operating pressure	1045 psig
r_o = nozzle outer radius	11”	r_o = nozzle outer radius	22.5625”
r_i = nozzle inner radius	6.19”	r_i = nozzle inner radius	13.125”
$C_{i\text{-}NOZZLE}$	1637	$C_{o\text{-}NOZZLE}$	1977
1.23 < 1.47		**1.07 < 1.59**	

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Applicable Fermi 2 Nuclear Power Plant Nozzles				
Nozzle ID Nozzle-to-Vessel (NV) Inner Radius (IR)	Identification	Category Number	Item Number	System
N1A (NV)	5-314A	B-D	B3.90	Recirculation (Outlet)
N1A (IR)	5-314A IRS	B-D	B3.100	Recirculation (Outlet)
N1B (NV)	5-314B	B-D	B3.90	Recirculation (Outlet)
N1B (IR)	5-314B IRS	B-D	B3.100	Recirculation (Outlet)
N2A (NV)	13-314A	B-D	B3.90	Recirculation (Inlet)
N2A (IR)	13-314A IRS	B-D	B3.100	Recirculation (Inlet)
N2B (NV)	13-314B	B-D	B3.90	Recirculation (Inlet)
N2B (IR)	13-314B IRS	B-D	B3.100	Recirculation (Inlet)
N2C (NV)	13-314C	B-D	B3.90	Recirculation (Inlet)
N2C (IR)	13-314C IRS	B-D	B3.100	Recirculation (Inlet)
N2D (NV)	13-314D	B-D	B3.90	Recirculation (Inlet)
N2D (IR)	13-314D IRS	B-D	B3.100	Recirculation (Inlet)
N2E (NV)	13-314E	B-D	B3.90	Recirculation (Inlet)
N2E (IR)	13-314E IRS	B-D	B3.100	Recirculation (Inlet)
N2F (NV)	13-314F	B-D	B3.90	Recirculation (Inlet)
N2F (IR)	13-314F IRS	B-D	B3.100	Recirculation (Inlet)
N2G (NV)	13-314G	B-D	B3.90	Recirculation (Inlet)
N2G (IR)	13-314G IRS	B-D	B3.100	Recirculation (Inlet)
N2H (NV)	13-314H	B-D	B3.90	Recirculation (Inlet)
N2H (IR)	13-314H IRS	B-D	B3.100	Recirculation (Inlet)
N2J (NV)	13-314J	B-D	B3.90	Recirculation (Inlet)
N2J (IR)	13-314J IRS	B-D	B3.100	Recirculation (Inlet)
N2K (NV)	13-314K	B-D	B3.90	Recirculation (Inlet)
N2K (IR)	13-314 K IRS	B-D	B3.100	Recirculation (Inlet)
N3A (NV)	8-316A	B-D	B3.90	Main Steam

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Applicable Fermi 2 Nuclear Power Plant Nozzles				
Nozzle ID Nozzle-to-Vessel (NV) Inner Radius (IR)	Identification	Category Number	Item Number	System
N3A (IR)	8-316A IRS	B-D	B3.100	Main Steam
N3B (NV)	8-316B	B-D	B3.90	Main Steam
N3B (IR)	8-316B IRS	B-D	B3.100	Main Steam
N3C (NV)	8-316C	B-D	B3.90	Main Steam
N3C (IR)	8-316C IRS	B-D	B3.100	Main Steam
N3D (NV)	8-316D	B-D	B3.90	Main Steam
N3D (IR)	8-316D IRS	B-D	B3.100	Main Steam
N5A (NV)	14-316A	B-D	B3.90	Core Spray
N5A (IR)	14-316A IRS	B-D	B3.100	Core Spray
N5B (NV)	14-316B	B-D	B3.90	Core Spray
N5B (IR)	14-316B IRS	B-D	B3.100	Core Spray
N6A (NV)	4-318A	B-D	B3.90	Head Spare
N6A (IR)	4-318A IRS	B-D	B3.100	Head Spare
N6B (NV)	4-318B	B-D	B3.90	Head Spare
N6B (IR)	4-318B IRS	B-D	B3.100	Head Spare
N7 (NV)	2-318	B-D	B3.90	Head Vent
N7 (IR)	2-318 IRS	B-D	B3.100	Head Vent
N8A (NV)	19-314A	B-D	B3.90	Jet Pump Instrumentation
N8A (IR)	19-314A IRS	B-D	B3.100	Jet Pump Instrumentation
N8B (NV)	19-314B	B-D	B3.90	Jet Pump Instrumentation
N8B (IR)	19-314B IRS	B-D	B3.100	Jet Pump Instrumentation

Enclosure 3 to
NRC-19-0016

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

RR-A39
Use of Boiling Water Reactor Vessel Internals Project Guidelines
in Lieu of ASME Code Requirements

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

-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Component(s) Affected

ASME Code Class:	Code Class 1
References:	ASME Code Section XI, 2013 Edition
Examination Category:	B-N-1 and B-N-2
Items Numbers:	B13.10, B13.20, B13.30 and B13.40
Description:	Use of Boiling Water Reactor Vessel Internals Project (BWRVIP) Guidelines in Lieu of ASME Code Requirements
Components:	Code Item Numbers: B13.10-Vessel interior (B-N-1), B13.20-Interior attachments, within beltline region (B-N-2), B13.30-Interior attachments beyond beltline Region (B-N-2) and B 13.40-Core support structure (B-N-2)

2. Applicable Code Edition and Addenda

ASME Code Section XI, 2013 Edition

3. Applicable Code Requirements

ASME Code Section XI requires the examination of components within the Reactor Pressure Vessel. These examinations are included in Table IWB-2500-1, Examination Categories B-N-1 and B-N-2 and identified with the following Item Numbers:

B 13.10 Examine accessible areas of the reactor vessel interior each inspection period (B-N-1) by the VT-3 method as defined in IWA-2213 of ASME Code Section XI.

B 13.20 Examine interior attachment welds within the beltline region each inspection interval (B-N-2) by the VT-1 method as defined in IWA-2211 of ASME Code Section XI.

B 13.30 Examine interior attachment welds beyond the beltline region each inspection interval (B-N-2) by the VT-3 method as defined in IWA-2213 ASME Code Section XI.

B 13.40 Examine surfaces of the welded core support structure each inspection interval (B-N-2) by the VT-3 method as defined in IWA-2213 of ASME Code Section XI.

These examinations are performed to assess the structural integrity of the reactor vessel interior, its welded attachments and the welded core support structure within the boiling water reactor pressure vessel.

The components/welds listed in Table 1, "Comparison of ASME Code Section XI Table IWB-2500-1 Examination Category B-N-1 and B-N-2 Requirements to BWRVIP Guidance Requirements," are subject to this request for alternative. Table 1 provides only an overview of the requirements. For more details, refer to ASME Section XI, Table IWB-2500-1 and the appropriate Boiling Water Reactor Vessel and Internals Project (BWRVIP) document.

4. Reason for Request

DTE Electric Company (DTE) is requesting NRC approval of this proposed alternative to the ASME Code Section XI requirements provided above for Fermi 2 on the basis that the use of the BWRVIP Guidelines discussed below will provide an acceptable level of quality and safety.

The BWRVIP Inspection and Evaluation (I&E) Guidelines have recommended aggressive specific inspections by BWR operators to completely identify material condition issues with BWR components. A wealth of inspection data has been gathered during these inspections across the BWR industry. The BWRVIP I&E Guidelines focus on specific and susceptible components, specify appropriate inspection methods capable of identifying real anticipated degradation mechanisms, and require re-examination at conservative intervals. In contrast, the ASME Code Section XI inspection requirements were prepared before the BWRVIP initiative and have not evolved with BWR inspection experience. The scope of the BWRVIP guidelines meet or exceed that of ASME Section XI and in many instances include components that are not part of the ASME Section XI jurisdiction.

5. Proposed Alternative

Pursuant to 10 CFR 50.55a(z)(1), DTE requests authorization to utilize the alternative requirements of the BWRVIP Guidelines in lieu of the requirements of ASME Code Section XI, examination requirements for Examination Categories B-N-1 and B-N-2.

DTE will satisfy the Examination Category B-N-1 and B-N-2 requirements as described in Table 1 in accordance with BWRVIP Guideline requirements. The proposed alternative is detailed in Table 1, "Comparison of ASME Code Section XI Table IWB-2500-1

Examination Category B-N-1 and B-N-2 Requirements to BWRVIP Guidance Requirements," and shows a comparison between the existing ASME Code Section XI and BWRVIP requirements that will be used under this alternative. Specifically, DTE will satisfy the Examination Category B-N-1 and B-N-2 requirements at Fermi 2 as described in Table 1 in accordance with BWRVIP Guidelines in lieu of the associated ASME Code Section XI requirements, including examination method, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting.

Not all of the components addressed by the BWRVIP Guidelines are components that require ASME Code Section XI examinations, but the particular Guidelines that are applicable to ASME Code Section XI components are listed below along with the BWRVIP-94NP Administrative Guide that Fermi 2 will use to implement this alternative:

BWRVIP Guidelines Used for Section XI Code Examinations (Part of this Request)	
BWRVIP-03NP	"BWR Vessel and Internals Project, Reactor Pressure Vessel and Internal Examination Guidelines"
BWRVIP-06-R1-A	"BWR Vessel and Internals Project, Safety Assessment of BWR Internals"
BWRVIP-14-A	"BWR Vessel and Internals Project, BWR Evaluation of Crack Growth in BWR Stainless Steel RPV Internals"
BWRVIP-18-A-R2	"BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines," (Licensee Renewal (LR) Safety Evaluation Report (SER) updated to Revision 2 in the annual update of May 9, 2016)
BWRVIP-25(1)	"BWR Core Plate Inspection and Flaw Evaluation Guidelines"
BWRVIP-26-A	"BWR Top Guide Inspection and Flaw Evaluation Guidelines"
BWRVIP-27-A	"BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines"
BWRVIP-38	"BWR Shroud Support Inspection and Flaw Evaluation Guidelines"
BWRVIP-41-R3	"BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"
BWRVIP-47-A	"BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
BWRVIP-48-R1(2)	"Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"
BWRVIP-49-A	"Instrument Penetration Inspection and Flaw Evaluation Guidelines"
BWRVIP-74-A	"BWR Reactor Vessel Inspection and Flaw Evaluation Guidelines"
BWRVIP-76-R1-A(3)	"BWR Core Shroud Inspection and Flaw Evaluation Guidelines" (LR) SER updated to Revision 1-A in the annual update of May 9, 2016)
BWRVIP-94NP	"BWR Vessel and Internals Project "Program Implementation Guide"

BWRVIP-100-A(3)	"Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds"
Notes: (1) Based on the renewed license for Fermi 2 *Enhancement 3* of LRA Section B.1.10 and SER Section 3.0.3.2.3, BWRVIP-25 shall be met by submittal of an analysis justifying the elimination of inspections for the core plate bolting or an analysis determining acceptance criteria for continued examination per BWRVIP-25. The analysis is to be submitted to the NRC no later than 2 years prior to the period of extended operation). (2) Currently, there are no existing BWRVIP Guidelines or ASME Code Section XI requirements regarding the feedwater spargers except for BWRVIP-48-A which governs inspection of reactor vessel internal attachment welds, namely the feedwater sparger brackets. Fermi 2 will continue to use inspections modeled after the guidance of NUREG-0619 on the feedwater spargers outside of this request. (3) If flaw evaluations are required for BWRVIP-76-R1-A examinations, the fracture toughness values of BWRVIP-100NP-A will be utilized.	

Table 1 compares current ASME Code Section XI, Table IWB-2500-1, Examination Category B-N-1 and B-N-2 requirements with the above current BWRVIP Guideline requirements, for BWR/4s as applicable, to Fermi 2.

In addition to the items in Table 1, a detailed Table 2, "Vessel Attachment Welds – Fabricated either from E-308/E-309 (Furnace Sensitized) Austenitic Stainless Steel or Inconel 182 Material," is included in this request. Table 2 lists inspection results for specific vessel attachment welds, fabricated with these materials, that have an increased concern for cracking. These welds are currently examined under the inspection strategy found in Section 3 of BWRVIP-48-A and BWRVIP-38.

When a BWRVIP Guideline refers to ASME Section XI, the technical requirements of ASME Section XI as described by the BWRVIP Guideline will be met, but the examination is under the auspices of the BWRVIP Program as defined by BWRVIP-94NP, "BWRVIP Vessel and Internals Project Program Implementation Guide." When implementing the guidance of BWRVIP-94, Fermi 2 will meet the following:

> "When BWRVIP Guidelines are approved by the Executive Committee and are initially distributed, or subsequently revised, each utility shall modify their vessel and internals program documentation to reflect the new requirements and shall implement the guidance within two refueling outages, unless a different schedule is identified by the BWRVIP at the time of document distribution. Implementation is to be based on the date of the distribution/notification letter to the members. Implementation means not only incorporating the requirements into the utility program, but also performing the initial or baseline inspection and evaluation requirements.

> However, if new guidance approved by the Executive Committee includes revisions to NRC approved guidance that are less conservative than those approved by the NRC, this less conservative guidance shall be implemented only after NRC approves the change. 'NRC approved' generally means the document was submitted to the NRC for review and approval and a final Safety Evaluation (SE) has been issued and is to be incorporated into a publication of a '-A' document or equivalent."

Therefore, where the revised version of a BWRVIP Inspection Guideline continues to also meet the requirements of the version of the BWRVIP Inspection Guideline that forms the safety basis for the NRC authorized proposed alternative to the requirements of 10 CFR 50.55a, it may be implemented. Otherwise, the revised Guidelines will only be implemented after NRC approval of the revised BWRVIP Guidelines or a plant-specific request for an alternative has been approved. Attachment 1, "Comparison of Code Examination Requirements to BWRVIP Examination Requirements," represents the most current comparison at the time of this request.

Any deviations from the referenced BWRVIP Guidelines for the duration of the proposed alternative will be appropriately documented and communicated to the NRC, per the BWRVIP Deviation Disposition Process. Currently, DTE has an active deviation for the core plate bolting under BWRVIP-25. This deviation was resubmitted to the BWRVIP and the NRC (Reference 1) specifically to extend its interval of applicability until 1) the revised BWRVIP-25 is approved by the NRC or 2) some other NRC approved solution is implemented.

Inspection services, by an Authorized Inspection Agency, will be applied to the proposed alternative actions of this alternative request.

In the event that conditions are identified that require repair/replacement activities and the component is within the jurisdiction of ASME Code Section XI (welded attachments to the RPV or Core Support Structure), the repair/replacement activities will be performed in accordance with ASME Code Section XI, Article IWA-4000. Subsequent examinations will be in accordance with the applicable BWRVIP Guideline.

As part of the BWRVIP initiative, the BWR reactor internals and attachments were subjected to a safety assessment to identify those components that provide a safety function and to determine if long-term actions were necessary to ensure continued safe operation. The safety functions considered are those associated with (1) maintaining a coolable geometry, (2) maintaining control rod insertion times, (3) maintaining reactivity control, (4) assuring core cooling, and (5) assuring instrumentation availability. The results of the safety assessment are documents in BWRVIP-06-R1-A, "BWR Vessel and Internals Project, Safety Assessment of BWR Internals," which has been approved by the NRC. As a result of BWRVIP-06-R1-A, component specific BWRVIP guidelines were developed providing appropriate examination and evaluation requirements to address the specific component safety function and potential degradation mechanism.

6. Basis for Use

The BWRVIP Guidelines to be used at Fermi 2 that are listed in this request for ASME Code Section XI components have been designated for use in accordance with the LR SER, (Reference 2). Revisions to the listed BWRVIP Guidelines will be implemented as applicable in accordance with latest revision BWRVIP-94NP.

BWRs now examine reactor internals in accordance with BWRVIP Guidelines. These Guidelines have been written to address the safety significant vessel internal components and to examine and evaluate the examination results for these components using appropriate methods and reexamination frequencies. The BWRVIP has established a reporting protocol for examination results and deviations. The NRC has agreed with the BWRVIP approach in principal and has issued SERs for many of these Guidelines (References 3 - 18).

As additional justification, Attachment 1, "Comparison of ASME Code Section XI Examination Requirements to BWRVIP Examination Requirements," provides specific examples that compare the inspection requirements of ASME Code Section XI Item Numbers B13.10, B13.20, B13.30, and B13.40 in Table IWB-2500-1, to the inspection requirements in the BWRVIP documents. Specific BWRVIP documents are provided as examples. This comparison also includes a discussion of the inspection methods and where they are applied. Results of specific BWRVIP examinations are provided in Attachment 2, "Fermi 2 – Reactor Internals Inspection History through RF19," and the respective outage dates and durations are provided in Attachment 3, "Past Refuel Outage Dates."

Therefore, based on the Safety Evaluations of many of the BWRVIP Guidelines and the comparisons performed demonstrating the use of these Guidelines above, DTE concludes that this alternative request to the ASME Code Section XI requirements will avoid unnecessary inspections, while in some cases conserving radiological dose, because the inspections will then be focused on the most recent BWR experience available. Thus, this request, when authorized, will provide an acceptable level of quality and safety and will not adversely impact the health and safety of the public.

7. Duration of Proposed Alternative

Upon authorization by the NRC, this request for an alternative to use the BWRVIP Guidelines in lieu of ASME Code Section XI requirements will be implemented during the Fourth Ten-Year ISI Interval beginning on May 2, 2019 and ending on December 31, 2029, (which will include a portion of the period of extended operation beginning March 21, 2025).

8. <u>Precedents</u>

The NRC Staff has authorized similar requests for the following licensees including Fermi 2, during the Third Ten-Year ISI Interval:

- Peach Bottom Atomic Power Station Units 2 and 3, SER Docket No. 50-277 and 50-278, EPID NO. L-2018-LLR-0056, July 18, 2018, (ADAMS Accession No. ML18179A394)

- Cooper Nuclear Station, SER Docket No. 50-298, CAC No. ME6366, February 16, 2016, (ADAMS Accession No. ML16034A479)

- River Bend Nuclear Station, SER Docket No. 50-458, TAC No. MF1867, May 30, 2014, Supplemented with SER Correction on November 13, 2014, (ADAMS Accession No. ML14183A086)

- Grand Gulf Nuclear Station, Unit 1, SER Docket No. 50-416, TAC No. MF3678, August 1, 2014, (ADAMS Accession No. ML14184A782)

- Fermi, Unit 2, SER Docket No. 50-341, TAC No. ME6765, February 17, 2012, (ADAMS Accession No. ML120370286)

9. <u>References</u>

1. DTE Electric Company Letter, NRC-15-0104, “Notification of Revision to BWRVIP Core Plate Deviation Disposition,” dated December 22, 2015 (ML15356A601)

2. Safety Evaluation Report Related to the License Renewal of Fermi 2, Docket No. 50-341, DTE Electric Company, dated July 2016, (ADAMS Accession No. ML16190A241)

3. Letter BWRVIP to USNRC, dated February 8, 2017, (BWRVIP-03NP, Revision 19), “BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines," EPRI Technical Report 3002008095. Provided to USNRC for INFORMATION ONLY as this report is updated periodically, (ADAMS Accession No. ML17054C666)

4. U.S. Nuclear Regulatory Commission Approval Letter for Technical Report BWRVIP-06, Revision 1-A, “BWR (Boiling Water Reactor) Vessels and Internals Project, Safety Assessment of BWR Reactor Internals,” Electric Power Research Institute Technical Report 1019058 (TAC NO. ME4044)

5. Letter BWRVIP to USNRC, dated January 31, 2009, Final Report, (BWRVIP-14NP-A), "BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals," (ADAMS Accession No. ML101880724) and Re-transmitted to USNRC on May 12, 2009, as BWRVIP-14-A, (ADAMS Accession No. ML091390008)

6. Letter USNRC to BWRVIP, dated February 22, 2016, Final Safety Evaluation for Electric Power Research Institute Boiling Water Reactor Vessel and Internals Project Technical Report 1016568, (BWRVIP-18-A-R2), "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines," (TAC No. MF8809) and (ADAMS Accession No. ML16011A190)

7. Letter USNRC to BWRVIP, dated December 19, 1999, Safety Evaluation of "BWR Vessel and Internals Project, BWR Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-25)," (ADAMS Accession Nos. ML993620267 and ML993620274)

8. Letter USNRC to BWRVIP, dated August 29, 2005, NRC Approval Letter of (BWRVIP-26-A), "BWR Vessel and Internals Project Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines," (ADAMS Accession No. ML052490550)

9. Letter USNRC to BWRVIP, dated June 9, 2004, Non-Proprietary Version of NRC Staff Review of (BWRVIP-27-A), "BWR Standby Liquid Control System/Core Plate DP Inspection and Flaw Evaluation Guidelines," (ADAMS Accession No. ML041700446)

10. Letter USNRC to BWRVIP, dated July 24, 2000, Final Safety Evaluation of the "BWR Vessel and Internals Project, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38), EPRI Report TR-108823 (TAC NO. M99638) and (ADAMS Accession No. ML003735498)

11. Letter USNRC to BWRVIP, dated February 4, 2001, Final Safety Evaluation of the "BWR Vessel and Internals Project, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)," (TAC NO. M99870)," (ADAMS Accession No. ML010460111)

12. Letter USNRC to BWRVIP, dated September 1, 2005, NRC Approval Letter of (BWRVIP-47-A), "BWR Vessel and Internals Project Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines," (ADAMS Accession No. ML052490537)

13. Letter USNRC to BWRVIP, dated July 25, 2005, NRC Approval Letter of (BWRVIP-48-A), "BWR Vessel and Internals Project Vessel ID Attachment Weld Inspection and Flaw Evaluation Guideline," (ADAMS Accession No. ML052130284)

14. Letter BWRVIP to USNRC, dated May 24, 2002, Notes the (-A) version of (BWRVIP-49-A), "BWR Vessel and Internals Project, Instrument Penetrations Inspection and Flaw Evaluation Guidelines," EPRI Technical Report 1006602 was submitted with this letter and contains the SER for this Guideline, (ADAMS Accession No. ML021510018)

15. Letter BWRVIP to USNRC, dated June 18, 2003, Notes the (-A) version of (BWRVIP-74-A), "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal," EPRI Technical Report 1008872 was submitted with this letter and contains the SER for this Guideline, (ADAMS Accession No. ML031710343)

16. Letter USNRC to BWRVIP, dated December 28, 2015, Notes the (-A) version of (BWRVIP-76, Revision 1A), "BWR Vessel and Internals Project, BWR Core Shroud Inspection and Flaw Evaluation Guidelines," are approved in future licensing actions as specified in the final safety evaluation, (CAC No. ME8317), (ADAMS Accession No. ML15307A468)

17. Letter BWRVIP to USNRC, dated September 22, 2011, (BWRVIP-94NP, Revision 2), "BWR Vessel and Internals Project, Program Implementation Guide," EPRI Technical Report 1024452. Provided to USNRC for INFORMATION ONLY as this report is updated periodically, (ADAMS Accession No. ML11271A058)

18. Letter USNRC to BWRVIP, dated November 1, 2007, "NRC Approval Letter with Comment for BWRVIP-100-A, BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds," (ADAMS Accession No. ML073050135)

TABLE 1 – Comparison of ASME Code Section XI Table IWB-2500-1 Examination Category B-N-1 and B-N-2 Requirements to BWRVIP Guidance Requirements (1)

ASME Item No. Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam Type	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam Type	BWRVIP Frequency
B13.10	Reactor Vessel Interior	Areas of the RPV above and below the core made accessible during a normal refuel.	VT-3	Each Period	None	While there is not a specific BWRVIP Guideline that addresses the scope of B-N-1, the examinations performed by BWRVIP-18, 25, 26, 27, 41, 47, 138 provide a general overview of the reactor interior which may be considered representative of the B-N-1 scope.		
B13.20	Interior Attachments within Beltline - Riser Braces	Accessible Welds	VT-1	Each 10-year Interval	BWRVIP-48-R1, Table 3-2	Riser Brace Attachment	EVT-1	25% during each subsequent 6 years
	Lower Surveillance Specimen Holder Brackets				BWRVIP-48-R1, Table 3-2	Bracket Attachment	VT-1	Each 10-Year Interval
B13.30	Interior Attachments beyond Beltline - Steam Dryer Hold-down Brackets	Accessible Welds	VT-3	Each 10-year interval	BWRVIP-48-R1, Table 3-2	Bracket Attachment	VT-3	Each 10-Year Interval
	Guide Rod Brackets				BWRVIP-48-R1, Table 3-2	Bracket Attachment	VT-3	Each 10-Year Interval
	Steam Dryer Support Brackets				BWRVIP-48-R1, Table 3-2	Bracket Attachment	EVT-1	Each 10-Year Interval
	Feedwater Sparger Brackets				BWRVIP-48-R1, Table 3-2	Bracket Attachment	EVT-1	Each 10-Year Interval
	Core Spray Piping Brackets				BWRVIP-48-R1, Table 3-2	Bracket Attachment	EVT-1	Each 10-Year Interval
	Upper Surveillance Specimen Holder Brackets				BWRVIP-48-R1, Table 3-2	Bracket Attachment	VT-3	Each 10-Year Interval

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ASME Item No. Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam Type	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam Type	BWRVIP Frequency
	Shroud Support (Weld H9)				BWRVIP-38, 3.1.3.2, Figures 3-2 and 3-5	Weld H-9	EVT-1 or UT	Maximum of 6 years for EVT-1, Maximum of 10 years for UT
B13.40	Integrally Welded Core Support Structure	Accessible Surfaces	VT-3	Each 10-year interval	BWRVIP-38, 3.1.3.2, Figures 3-2 and 3-5	Shroud support welds H8 and H9 (2) including gussets	EVT-1 or UT	Based on as-found conditions, to a maximum 6 years for one side EVT-1, 10 years for UT where accessible
	Core Shroud Horizontal Welds				BWRVIP-76-R1-A, 2.2	Welds H1-H7 as applicable	UT or EVT-1	Based on as-found conditions, to a maximum of 10 years for UT when inspected from both sides of the welds
	Core Shroud Vertical Welds				BWRVIP-76-R1-A, 2.3	Vertical Welds as applicable	EVT-1 or UT	Maximum 10 years for UT based on inspection of horizontal welds
	Core Shroud Repairs (2)				BWRVIP-76-R1-A, 3.5	Tie-Rod Repair	VT-3	In accordance with designer recommendations per BWRVIP-76 R1

Notes:

(1) This table provides only an overview of the requirements. For more details, refer to ASME Code Section XI, Table IWB-2500-1 and the appropriate BWRVIP Document.

(2) No repairs have been performed on the core shroud.

TABLE 2 – Vessel Attachment Welds – Fabricated Either from E-308/E-309 (Furnace Sensitized) Austenitic Stainless Steel or Inconel 182 Material

Vessel Attachment Welds (Weld No.'s)	Description	Weld Filler Material	Inspection Performed to date Code B-N-2 / BWRVIP-48-R1 and -38 (1)	Results (2)
5-327	Shroud Support to RPV	Inconel 182	B-N-2 until RF14/BWRVIP since RF06	NRI
6-327	Shroud Support Gussets to RPV	Inconel 182	B-N-2 until RF14/BWRVIP since RF06	NRI
2-323-A&B	Guide Rod Brackets to Inlays	Inconel 182	B-N-2 until RF14/BWRVIP since RF06	NRI
103-305-A thru D	Steam Dryer Support Lugs to Inlays	Inconel 182	B-N-2 until RF14/BWRVIP since RF06	NRI
104-305-A thru H	Core Spray Brackets to Inlays	Inconel 182	B-N-2 until RF14/BWRVIP since RF06	NRI
101-305-A thru C, 102-305-A thru C	Surveillance Specimen Brackets to Inlays	Inconel 182	B-N-2 until RF14/BWRVIP since RF06	NRI
105-305-A thru M	Feedwater Sparger Brackets to Inlays	Inconel 182	B-N-2 until RF14/BWRVIP since RF06	NRI
16-308-A thru V	Buildup for Jet Pump Riser Pads	Inconel 182	B-N-2 until RF14/BWRVIP since RF06	NRI

Notes:

(1) Since RF06 and until RF14, Fermi 2 implemented the requirements of ASME Section XI and the guidance of the BWRVIP. Since RF14, Fermi has implemented the BWRVIP as previously approved by the NRC via Relief Request RR-A39 for the Third Interval.

(2) "NRI" indicates No Relevant Indications were identified during examination.

Comparison of Code Examination Requirements to BWRVIP Examination Requirements

The following discussion provides a comparison of the examination requirements provided in ASME Code Item Numbers B13.10, B13.20, B13.30, and B13.40 in Table IWB-2500-1, to the examination requirements in the BWRVIP Guidelines. Specific BWRVIP Guidelines are provided as examples for comparisons. This comparison also includes a discussion of the examination methods.

1. Code Requirement - B13.10 - Reactor Vessel Interior Accessible Areas (B-N-1)

   The ASME Code Section XI requires a VT-3 examination of reactor vessel accessible areas, which are defined as the spaces above and below the core made accessible during normal refueling outages. The frequency of these examinations is specified as the first refueling outage, at intervals of approximately 3 years during the first inspection interval, and each ASME inspection period during each successive 10-year Inspection Interval. Typically, these examinations are performed every other refueling outage of the Inspection Interval. This examination requirement is a non-specific requirement that is a departure from the traditional Section XI examinations of welds and surfaces. The purpose of the examination is to identify relevant conditions such as distortion or displacement of parts; loose, missing, or fractured fasteners; foreign material, corrosion, erosion, or accumulation of corrosion products; wear; and structural degradation.

   Portions of the various examinations required by the applicable BWRVIP Guidelines listed in this request require entry to accessible areas of the reactor vessel during each refueling outage. Examination of Core Spray Piping and Spargers (BWRVIP-18-A-R2), Top Guide (BWRVIP-26-A), Jet Pump Welds and Components (BWRVIP-41-R3), Interior Attachments (BWRVIP-48-R1), Core Shroud Welds (BWRVIP-76-R1-A), Shroud Support (BWRVIP-38) and Lower Plenum Components (BWRVIP-47-A) provides such access. Locating and examining specific welds and components within the reactor vessel areas above, below (if accessible), and surrounding the core (annulus area) entails access by remote camera systems that essentially performs equivalent VT-3 examination of these areas or spaces as the specific weld or component examinations are performed. This provides an equivalent method of visual examination on a more frequent basis than that required by the ASME Code Section XI. Evidence of wear, structural degradation, loose, missing, or displaced parts, foreign materials, and corrosion product buildup can be, and has been observed during the course of implementing these BWRVIP examination requirements. Therefore, the specified BWRVIP Guideline requirements meet or exceed the subject Code requirements for examination method and frequency of the interior of the reactor vessel. Accordingly, these BWRVIP examination requirements provide an acceptable level of quality and safety as compared to the subject Code requirements.

2. Code Requirement - B 13.20 - Interior Attachments Within the Beltline (B-N-2)

The ASME Code Section XI requires a VT-1 examination of accessible reactor interior surface attachment welds within the beltline each 10-year interval. In the boiling water reactor, this includes the jet pump riser brace welds-to-vessel wall and the lower surveillance specimen support bracket welds-to-vessel wall. In comparison, the BWRVIP requires the same examination method and frequency for the lower surveillance specimen support bracket welds, and requires an Enhanced VT-1 (EVT-1) examination on the remaining attachment welds in the beltline region in the first 12 years, and then 25% during each subsequent 6 years.

The jet pump riser brace examination requirements are provided below to show a comparison between the Code and the BWRVIP examination requirements.

Comparison to BWRVIP Requirements - Jet Pump Riser Braces (BWRVIP-41-R3) and BWRVIP-48-R1)

- The ASME Code requires a 100% VT-1 examination of the jet pump riser brace-to reactor-vessel wall pad welds each 10-year interval

- The BWRVIP requires an EVT-1 examination of the jet pump riser brace-to-reactor vessel wall pad welds the first 12 years and then 25% during each subsequent 6 years

- BWRVIP-48-R1 specifically defines the susceptible regions of the attachment that are to be examined

The ASME Code Section XI, VT-1 examination is conducted to detect discontinuities and imperfections on the surfaces of components, including such conditions as cracks, wear, corrosion, or erosion. The BWRVIP EVT-1 is conducted to detect discontinuities and imperfections on the surface of components and is additionally specified to detect potentially very tight cracks characteristic of fatigue and Inter-Granular Stress Corrosion Cracking (IGSCC), the relevant degradation mechanisms for these components. General wear, corrosion, or erosion although generally not a concern for inherently tough, corrosion resistant stainless steel material, would also be detected during the process of performing a BWRVIP EVT-1 examination.

The ASME Code Section XI, 2013 Edition, VT-1 visual examination method requires that a letter character with a height of 0.044 inches can be read. The BWRVIP EVT-1 visual examination method requires the same 0.044-inch resolution on the examination surface and additionally the performance of a cleaning assessment and cleaning as necessary. While the jet pump riser brace configuration varies depending on the vessel manufacturer, BWRVIP-48-R1 includes diagrams for each configuration and prescribes examination for each configuration including Fermi 2 (Combustion Engineering).

The calibration standards used for BWRVIP EVT-1 exams utilize the same Code characters, thus assuring at least equivalent resolution compared to the Code. Although the BWRVIP examination may be less frequent, it is a more comprehensive method. Therefore, the enhanced flaw detection capability of an EVT-1, with a less frequent examination schedule provides an acceptable level of quality and safety to that provided by the ASME Code.

3. <u>Code Requirement - B13.30 - Interior Attachment Beyond the Beltline Region (B-N-2)</u>

The ASME Code Section XI requires a VT-3 examination of accessible reactor interior surface attachment welds beyond the beltline each 10-year interval. In the boiling water reactor, this includes the core spray piping primary and supplemental support bracket welds-to-vessel wall, the upper surveillance specimen support bracket welds-to-vessel wall, the feedwater sparger support bracket welds-to-reactor vessel wall, the steam dryer support and hold down bracket welds-to-reactor vessel wall, the guide rod support bracket weld-to-reactor vessel wall, the shroud support plate-to-vessel wall, and shroud support gussets. BWRVIP-48-R1 requires as a minimum the same VT-3 examination method as the Code for some of the interior attachment welds beyond the beltline region, and in some cases specifies an enhanced visual examination technique EVT-1 for these welds. For those interior attachment welds that have the same VT-3 method of examination, the same scope of examination (accessible welds), the same examination frequency (each 10-year interval) and ASME Code Section XI flaw evaluation criteria, the level of quality and safety provided by the BWRVIP requirements are equivalent to that provide by the ASME Code.

For the Core Spray support bracket attachment welds, the steam dryer support bracket attachment welds, the feedwater sparger support bracket attachment welds, and the shroud support plate-to-vessel welds, as applicable, the BWRVIP Guidelines require an EVT-1 examination at the same frequency as the Code, or at a more frequent rate. Therefore, the BWRVIP requirements provide the same level of quality and safety to that provided by the ASME Code.

The Core Spray piping bracket-to-vessel attachment weld is used as an example for comparison between the Code and BWRVIP examination requirements as discussed below.

<u>Comparison to BWRVIP Requirements - Core Spray Piping Bracket Welds (BWRVIP-48-R1)</u>

- The ASME Code examination requirement is a VT-3 examination of each weld every 10 years

- The BWRVIP examination requirement is an EVT-1 inspection of 100% of the primary and supplemental core spray piping bracket to vessel ID attachment welds and heat-affected zones on both the vessel and bracket sides of the welds every 10 years. The BWRVIP examination method EVT-1 has superior flaw detection and sizing capability,

the examination frequency is greater than the Code requirements, and the same flaw evaluation criteria are used

- The ASME Code VT-3 examination is conducted to detect component structural integrity by ensuring the components general condition is acceptable. An enhanced EVT-1 is conducted to detect discontinuities and imperfections on the examination surfaces, including such conditions as tight cracks caused by IGSCC or fatigue, the relevant degradation mechanisms for BWR internal attachments

Therefore, with the EVT-1 examination method, the same examination scope (accessible welds), the same flaw evaluation criteria (Section XI), the level of quality and safety provided by the BWRVIP criteria is superior than that provided by the Code.

4. <u>Code Requirement - B13.40 - Integrally Welded Core Support Structures (B-N-2)</u>

The ASME Code Section XI requires a VT-3 examination of accessible surfaces of the welded core support structure each 10-year interval. In the boiling water reactor, the welded core support structure has primarily been considered the shroud support structure, including the shroud support plate (annulus floor) the shroud support ring, the shroud support welds, the shroud support gussets. In later designs, the shroud itself is considered part of the welded core support structure. Historically, this requirement has been interpreted and satisfied differently across the industry. The proposed alternate examination replaces this ASME requirement with specific BWRVIP Guidelines that examine susceptible locations for known relevant degradation mechanisms.

- The Code requires a VT-3 of accessible surfaces each 10-year interval

- The BWRVIP requires as a minimum the same examination method (VT-3) as the Code for integrally welded Core Support Structures, and for specific areas, requires either an enhanced visual examination technique (EVT-1) or volumetric examination (UT)

BWRVIP recommended examinations of integrally welded core support structures are focused on the known susceptible areas of this structure, including the welds and associated weld heat affected zones. As a minimum, the same or superior visual examination technique is required for examination at the same frequency as the code examination requirements. In many locations, the BWRVIP Guidelines require a volumetric examination of the susceptible welds at a frequency identical to the Code requirement.

For other integrally welded core support structure components, the BWRVIP requires an EVT-1 or UT of core support structures. The core shroud is used as an example for comparison between the Code and BWRVIP examination requirements as shown below.

Comparison to BWRVIP Requirements - BWR Core Shroud Examination and Flaw Evaluation Guideline (BWRVIP-76)

- The ASME Code requires a VT-3 examination of accessible surfaces every 10 years

- The BWRVIP requires an EVT-1 examination from the inside and outside surface where accessible or ultrasonic examination of each core shroud circumferential weld that has not been structurally replaced with a shroud repair at a calculated "end of interval" (EOI) that will vary depending upon the amount of flaws present, but not to exceed ten years

The BWRVIP recommended examinations specify locations that are known to be vulnerable to BWR relevant degradation mechanisms rather than "all surfaces." The BWRVIP examination methods (EVT-1 or UT) are superior to the Code required VT-3 for flaw detection and characterization. The BWRVIP examination frequency is equivalent to or more frequent than the examination frequency required by the Code. The superior flaw detection and characterization capability, with an equivalent or more frequent examination frequency and the comparable flaw evaluation criteria, results in the BWRVIP criteria providing a level of quality and safety equivalent to or superior to that provided by the Code requirements.

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Fermi 2 - Reactor Internals Inspection History through RF19

Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
Core Shroud (BWRVIP-07/76)	RF04	VT-1 (1mil wire)	Inspected 100% ID welds H2, H3, and, H4; 100% OD welds H1-H7; accessible areas H8 & H9.
		VT-1/VT-3	The only indications identified were two <1" vertical in orientation above the H2 weld at azimuth 125 degrees. These were evaluated against established flaw screening criteria and found acceptable.
	RF05	EVT-1 (1/2mil wire)	Inspected approximately 60-70 degrees arc on the core shroud in area of previous indications. H2-H4 inspected on shroud ID, H1-H7 inspected on shroud OD. No new indications, no change observed in previous indications above H2 weld.
	RF06	UT	Performed focused phased array UT examination of the H3, H4, H5, and H7 welds utilizing GE's universal carousel. No indication of cracking was identified.
		EVT-1	A cursory exam was performed on H-3 weld to confirm UT results for information only. No new indications and no change was observed in the previous indication above H2 weld.
	RF07	EVT-1	Re-inspected the indication above the H2 weld on the inside of the shroud. No change in appearance. The control rod blade was withdrawn to perform the examination.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF08	N/A	No inspections performed on the Core Shroud. Inspections were performed on the Shroud Support.
	RF09	N/A	No inspections performed on the Core Shroud. Inspections were performed on the Shroud Support.
	RF10	N/A	No inspections performed on the Core Shroud. Inspections were performed on the Shroud Support.
	RF11	N/A	No inspections performed on the Core Shroud. Inspections were performed on the Shroud Support.
	RF12	UT	Performed phased array UT examination of the H3, H4, H5, and H7 welds from both sides utilizing AREVA's demonstrated technique. No indication of cracking was identified. Inspection coverage exceeded 60% for all welds with coverage spaced around the entire circumference.
	RF13	N/A	No inspections performed on the Core Shroud. Inspections were performed on the Shroud Support.
	RF14 (10/10)	N/A	No inspections performed on the Core Shroud. Inspections were performed on the Shroud Support.
	RF15 (04/12)	N/A	No inspections performed on the Core Shroud. Inspections were performed on the Shroud Support.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF16 (2014)	N/A	No inspections performed on the Core Shroud. Inspections were performed on the Shroud Support.
	RF17 (2015)	N/A	No inspections performed.
	RF18 (2017)	UT	Performed phased array UT examination of the H3, H4, HS, H6 and H7 welds from both sides utilizing GEH's demonstrated technique. No indication of cracking was identified. Inspection coverage exceeded 85% for all welds with coverage spaced around the entire circumference. Off-axis UT examinations performed on a high fluence zone (-l.36E21 n·cm2, 57.67° to 80.35°) and low fluence zone (-2.33E20 n·cm2, 74.18° to 98.87°) near H4 with 5.2% (of207" diameter) coverages. The GEH technique had been demonstrated but not published in a BWRVIP document at the time of the exam. No relevant indications.
	RF19	N/A	No inspections performed
Shroud Support (BWRVIP-38/*104) Access Hole Cover (BWRVIP-180)	RF03	VT-3	Inspected shroud support gusset welds and H8/H9 in conjunction with jet pump inspections. No indications.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF04	VT-1/VT-3	Inspected areas in conjunction with jet pumps, included were gusset welds H8 and H9. H8 and H9 welds inspected at 0 and 180 degrees with 1 mil wire. No indications.
	RF05	EVT-1 (1/2mil wire)	Inspected sample area 60-70 degree arc plus 180 degrees location on H8, H9, and gussets. No indications.
	RF06	VT-3	Inspection performed in conjunction with jet pump inspections. Approximately 50% of the gussets and H8 and H9 welds were inspected. This was a best effort exam which ranged from MVT-1 to VT-3 depending on camera angle and lighting. No cleaning was performed. No indications identified.
	RF07	EVT-1	Inspection performed in conjunction with jet pump inspections. Remaining 50% of the gusset welds were inspected. This was a best effort exam which ranged from EVT-1 to VT-3 depending on camera angle and lighting. (Credited as an EVT-1 exam) No cleaning was performed or needed. No indications identified. The H8 and H9 welds were inspected in detail at 0 and 180 Deg. Azimuth to EVT-1 standards where there were no obstructions.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF08	EVT-1	The H8 and H9 welds were re-inspected to achieve required coverage. 22% of both welds were inspected and included the areas at 0 and 180 degrees as well as adjacent to Jet Pumps 2 and 3. Accessible areas on Gussets 1, 3, 11, 12, and 22 were inspected. No indications of cracking identified.
	RF09	EVT-1/ VT-1	The H8 and H9 welds were inspected adjacent to Jet Pumps 3 and 4 (Coverage obtained 1% and 8.3%). Accessible areas on Gussets 2 and 15 inspected (90% coverage on each obtained). Both access hole covers were inspected (VT-1). No indications identified.
	RF10	EVT-1/ VT-1	The H8 and H9 welds were inspected adjacent to Jet Pump 5(Coverage obtained 1% and 8.3%). Accessible areas on Gussets 7 and 8 inspected (70/90% coverage obtained @ VT-1 quality, EVT-1 not credited). No indications identified.
	RF11	EVT-1	The H8 and H9 welds were inspected at 0 and 180 degrees as well as several other locations. Coverage obtained was 24% for H8 and 30% for H9. Accessible areas on Gussets 5, 6, 7, 8, 9, 10, 18, and 21 were inspected with 50% to 80% coverage obtained @ EVT-1. No indications identified.
		UT	A portion of the H9 weld was examined from the vessel outside diameter using a manual technique as required by BWRVIP-104. Approximately 19.6% of weld was examined with no indications.

Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF12	EVT-1	Accessible areas on Gussets 4 and 13 were inspected with 55% to 80% coverage obtained using EVT-1. No indications identified. Both Access Hole covers were inspected per draft BWRVIP -180 requirements. Cracking identified on 0 degree cover. Reference OE 25794.
	RF13	EVT-1	Accessible areas on Gussets 5 and 6 were inspected with 75% coverage obtained using EVT-1. No indications identified. The 0 Degree Access Hole cover was re-inspected and no additional cracking was identified. No repair installed.
	RF14 (10/10)	EVT-1	Accessible areas on Gussets 1, 21, and 22 were inspected with 50% - 60% coverage obtained using EVT-1. No indications identified. All 3 welds on the 0 Degree Access Hole Cover were re-inspected and no additional cracking was identified. No repair installed.
	RF15 (04/12)	EVT-1	Accessible areas on Gussets 4, 5, 6, 13, 16, and 17 were inspected with 50% to 70% EVT-1 coverage obtained. No indications identified. The H8 and H9 welds were EVT-1 visually inspected from the annulus side with combined coverage at several locations of 15 .9% for the H8 weld and 20.5% for the H9 weld. No indications identified.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF16 (2014)	EVT-1	Accessible areas on Gussets 11 and 12, as well as the 180° Access Hole Cover were inspected with 90% coverage and no indications were identified. All 3 welds on the 0° Access Hole Cover were re-inspected. No additional cracking was identified and the component was evaluated to be acceptable without repair.
	RF17 (2015)	EVT-1	Accessible areas on Gussets 14, 15, and 18, were inspected with 35% to 65% coverage and no indications were identified.
	RF18 (2017)	EVT-1	Accessible areas on Gussets 7, 8, 9, and 10 were inspected with 3 0% to 60% coverage and no indications were identified. The 0° Access Hole Cover was inspected with 80% coverage with no changes in relevant indication status.
	RF19	EVT-1	Accessible areas on Gussets 2, 3, 19, and 20 were inspected with 30% to 60% coverage. NRI.
Core Spray Piping (BWRVIP-18/ 18-A)	Each outage RF01 thru RF04	VT-1 (1mil wire)	During RF01 two small arc strikes were identified on loop piping. These have been re-inspected each outage. No change in condition. Inspections performed per IEB 80-013 and SIL 289. No indication of cracking.
	RF05	EVT-1 (1/2mil wire)	All welds brushed prior to inspection using 1/2 mil wire. Remainder of loop piping inspected without brushing. No indication of cracking.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF06	EVT-1	Inspected all welds on both loops of core spray to EVT-1 standards as opposed to BWRVIP-18 requirements of MVT-1. Cleaning assessment was performed; cleaning was not necessary. No indication of cracking.
	RF07	EVT-1	Inspected all welds on both loops of core spray to EVT-1 standards. Cleaning assessment was performed; cleaning was not necessary. No indication of cracking.
	RF08	EVT-1	Inspected all welds on both loops of core spray to EVT-1 standards. Cleaning assessment was performed, cleaning was not necessary. No indication of cracking.
	RF09	EVT-1	Inspected all target welds on both loops of core spray and sample welds on Div 2 to EVT-1 standards. Cleaning assessment was performed, cleaning was not necessary. No indications of cracking.
	RF10	EVT-1	Inspected all target welds on both loops of core spray and rotating sample welds on Div 2 to EVT-1. Cleaning assessment was performed, cleaning was necessary for selected locations and welds were brushed. No indications of cracking. Inspection coverage reported separately but generally >80%.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF11	EVT-1	Inspected all target welds on both loops of core spray and rotating sample welds on Div 1 to EVT-1. Cleaning assessment was performed, cleaning was necessary for selected locations and welds were brushed. No indications of cracking. Inspection coverage reported separately but generally >80%.
	RF12	EVT-1	Inspected all target welds on both loops of core spray and rotating sample welds on Div 1 to EVT-1. Cleaning assessment was performed, cleaning was necessary for selected locations and welds were brushed. No indications of cracking. Inspection coverage reported separately but generally >55%.
	RF13	EVT-1	Inspected all target welds on both loops of core spray and rotating sample welds on Div 2 to EVT-1. Cleaning assessment was performed, cleaning was necessary for selected locations and welds were brushed. No indications of cracking. Inspection coverage reported separately but generally >55%.
	RF14 (10/10)	EVT-1	Inspected all target welds on both loops of core spray and rotating sample welds on Div 2 to EVT-1. Cleaning was performed for all locations and welds were hydrolazed or brushed. No indications of cracking. Inspection coverage reported separately but generally >60%.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF15 (4/12)	EVT-1	Inspected all target welds on both loops of core spray and rotating sample welds on Div 1 to EVT-1. Cleaning was performed for all locations and welds were brushed. No indications of cracking. Inspection coverage reported separately but generally >60%.
	RF16 (2014)	EVT-1	Inspected all target welds on both loops of core spray and a rotating sample welds on Div 1 with no indications of cracking observed. Brushing was performed on all locations. Inspection coverage is reported separately in Att. 2, but averaged 58%.
	RF17 (2015)	EVT-1	Inspected all target welds on both loops of core spray and a rotating sample welds on Div 2 with no indications of cracking observed. Brushing was performed on all locations. Inspection coverage is reported separately in Att. 2, but averaged 60%.
	RF18 (2017)	EVT-1	Inspected a sample of welds on both loops with no indications of cracking observed. Brushing was performed on all locations. Inspection coverage averaged 70%.
	RF19	EVT-1	Inspected a sample of welds on both loops with no indications of cracking observed. Brushing was performed on all locations. Inspection coverage averaged 50%.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
Core Spray Sparger (BWRVIP-18/18-A)	Each outage RF01 to RF04	VT-1 (1mil wire)	During RF01 one arc strike identified on upper CS sparger. Re-inspections have not identified any changes. No indication of cracking.
	RF05	VT-1/ EVT-1 (1/2mil)	1/2 mil wire used for junction box remainder utilized l mil wire. No indication of cracking.
	RF06	EVT-1, MVT-1	Inspected per BWRVIP-18 using EVT-1 for sparger T-box and end caps and MVT-1 for remaining locations. No indications of cracking.
	RF07	EVT-1/ VT-1	Inspected per BWRVIP-18 using EVT-1 for sparger T-box welds, end cap welds, drain plug welds, and support brackets and welds, and VT-1 for flow nozzles and tack welds. No indications of cracking identified.
	RF08	EVT-1/ VT-1	Inspected per BWRVIP-18 using EVT-1 for SI, S2 and S4 welds. Selected S3a, S3b welds inspected using VT-I. Selected S3c welds as well as selected SB bracket welds were inspected using EVT-1 technique. A best effort exam was performed on all accessible areas. No indications of cracking identified.
	RF09	EVT-1/ VT-1	Inspected per BWRVIP-18 using EVT-1 for 50% of the Sl, S2 and S4 welds and VT-1 for 50% of the S3a, S3b and S3c welds on the same spargers. 9 SB bracket welds were inspected using EVT-1 technique. Coverage for specific welds will be reported separately. No indications of cracking were identified.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF10	EVT-1/ VT-1	Inspected per BWRVIP-18 using EVT-1 for 50% of the Sl, S2 and S4 welds and VT-1 for 50% of the S3a, S3b and S3c welds on the same spargers. 6 SB bracket welds were inspected using EVT-1 technique. Coverage for specific welds will be reported separately but was > 60% for welds and >85% for brackets. No indications of cracking were identified.
	RF11	EVT-1/ VT-1	Inspected per BWRVIP-18-A using EVT-1 for 50% of the Sl, S2 and S4 welds on the same spargers. 6 SB bracket welds were inspected using VT-1 technique. Coverage for specific welds will be reported separately but was> 50% for welds and >75% for brackets. No indications of cracking were identified.
	RF12	EVT-1/ VT-1	Inspected per BWRVIP-18-A using EVT-1 for 50% of the Sl, S2 and S4 welds on the same spargers. 6 SB bracket welds were inspected using EVT-1 technique. Coverage for specific welds will be reported separately but was > 40% for welds and >75% for brackets. No indications of cracking were identified.
	RF13	EVT-1/ VT-1	Inspected per BWRVIP-18-A using EVT-1 for 50% of the Sl, S2 and S4 welds on the same spargers. 6 SB bracket welds were inspected using EVT-1 technique. Coverage for specific welds will be reported separately but was> 50% for welds and >70% for brackets. No indications of cracking were identified.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF14 (10/10)	EVT-1/ VT-1	Inspected per BWRVIP-18-A using EVT-1 for 50% of the Sl, S2 and S4 welds on the C and D spargers. 6 SB bracket welds and S3 nozzle welds were inspected using VT-1 technique. Coverage for specific welds will be reported separately but was > 40% for welds and >60% for brackets. No indications of cracking were identified.
	RF15 (4/12)	EVT-1/ VT-1	Inspected per BWRVIP-18-A using EVT-1 for 50% of the Sl, S2 and S4 welds on the A and B spargers. 6 SB bracket welds inspected using VT-1 technique. Coverage for specific welds will be reported separately but was > 40% for welds and >60% for brackets. No indications of cracking were identified.
	RF16 (2014)	EVT-1/ VT-1	Inspected per BWRVIP-18-A using EVT-1 for 50% of the Sl, S2 and S4 welds on the A and B spargers. 6 SB bracket welds were inspected using VT-1. Coverage for specific welds is reported separately in Attachment 2 but was > 40% for most welds and> 60% for most brackets. No indication of cracking was identified.
	RF17 (2015)	EVT-1/ VT-1	Inspected 50% of the Sl, S2 and S4 welds on the A and B spargers using EVT-1 and 6 sparger bracket welds using VT-1. Coverage for specific welds is reported separately in Attachment 2 but averaged 46% for the welds and 50% for the brackets. No indication of cracking was identified.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF18 (2017)	VT-1	Inspected 3 sparger bracket welds using VT-1 with 50% coverage each. No indication of cracking was identified.
	RF19	EVT-1/VT-3	Inspected 50% of the S1, S2, S3, and S4 welds on the lower spargers only. EVT-1 coverage averaged 50%. NRI.
Top Guide (Rim, etc.) Beams (BWRVIP-26) (BWRVIP-183)	RF01 & RF02	VT-3	Inspected rim each outage. No indications.
	RF03	VT-1/VT-3	Inspected 6 locations (RICSIL 059) and rim area 0° - 180°. No indications.
	RF04	VT-1/VT-3	Inspected 6 locations (SIL 554) and rim area 0° - 360°. No indications.
	RF05	VT-1	Inspected 15 locations (SIL 554). No indications.
	RF06	VT-1	Inspected bottom edge of beams at 11 core locations per SIL 554. No indication of cracking.
	RF07	VT-1	Inspected bottom edge of beams at 8 core locations per SIL 554. No indication of cracking.
	RF08	VT-1	Inspected bottom edge of beams at 5 core locations per SIL 554. No indication of cracking.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF09	VT-1	Inspected bottom edge of beams at 6 core locations per SIL 554. No indication of cracking.
	RF10	VT-1/VT-3	Inspected bottom edge of beams at 2 core locations per (SIL 554) and rim area 0° - 90°. No indication of cracking.
	RF11	VT-1/VT-3	Inspected bottom edge of beams at 2 core locations per SIL 554. No indication of cracking. Inspected 90 degree segment of top guide rim (90° - 180°) and no indications were identified.
	RF12	VT-1/VT-3	Inspected intersection and bottom edge of beams at 5 core locations per SIL 554. No indication of cracking.
	RF13	EVT-1	Inspected intersection and bottom edge of beams at 5 core locations per BWRVIP-183 utilizing a new visual inspection tool and rim area 0° - 90°. No indication of cracking.
	RF14 (10/10)	VT-3	Inspected rim area 0° - 180° with no indications identified.
	RF15 (4/12)	EVT-1	Inspected intersection and bottom edge of beams at 5 core locations per BWRVIP-183 utilizing a new visual inspection tool. No indication of cracking. Fabrication related conditions identified on the bottom surface of the plate material at 3 cell locations. Inspected the rim area 180° - 360° with no indications.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF16 (2014)	N/A	No inspections performed in RF16.
	RF17 (2015)	EVT-1	Inspected intersection and bottom edge of beams at 9 core cell locations per BWRVIP-183 with no indication of cracking.
	RF18 (2017)	N/A	No inspections performed in RF18.
	RF19	EVT-1	No inspections performed in RF19.
Core Plate Rim Bolts, etc. (BWRVIP-25)	RF05	VT-1 (1mil wire)	Inspected 6 core plate bolts located between 100 and 160 degrees and adjacent area. No indications.
	RF06	VT-3	Inspected tops of approximately 20 bolts per SIL 588. No indications identified.
	RF07	VT-3	Inspected tops of approximately 20 bolts per SIL 588. No indications identified.
	RF08	VT-3	Inspected tops of approximately 20 core plate bolts (VT-3) per SIL 588. Did not meet BWRVIP requirements. No indications identified.
	RF09	N/A	No inspections performed. BWRVIP analysis concluded that inspections are not required. (Reference BWRVIP 2003-117 and TJ-2003-01).

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF10	N/A	No inspections performed. BWRVIP analysis concluded that inspections are not required. (Reference B WR VIP 2003-117 and TJ-2003-01).
	RF11	N/A	No inspections performed. BWRVIP analysis concluded that inspections are not required. (Reference BWRVIP 2006-041 and DD-2006-01).
	RF12	N/A	No inspections performed. BWRVIP analysis concluded that inspections are not required. (Reference BWRVIP 2006-041).
	RF13	N/A	No inspections performed. BWRVIP analysis concluded that inspections are not required. (Reference BWRVIP 2006-041).
	RF14 (10/10)	N/A	No inspections performed. BWRVIP analysis concluded that inspections are not required. (Reference BWRVIP 2006-041) BWRVIP 2010- 243 now requires preparation of a Deviation Disposition by 3/31/2011.
	RF15 (4/12)	N/A	No inspections performed. BWRVIP analysis concluded that inspections are not required. Deviation Disposition DD-2011-01 was submitted to BWRVIP 3/30/2011.
	RF16 (2014)	N/A	No inspections performed in RF16, as justified by Deviation Disposition DD-2011-01.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF17 (2015)	N/A	No inspections performed in RF 17, as justified by Deviation Disposition DD-2011-01 Revision 1.
	RF18 (2017)	N/A	No inspections performed in RF18, as justified by Deviation Disposition DD-2011-01 Revision 1.
	RF19	N/A	No inspections performed in RF19, as justified by Deviation Disposition DD-2011-01 Revision 1.
SLC (BWRVIP-27)	RF04	VT-3	Performed a visual inspection from Reactor penetration to shroud support when access was provided during jet pump beam replacement. No indications.
	RF05 – RF07	N/A	No inspections performed as access was not provided.
	RF08	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, but did not remove mirror insulation box from safe-end. No leakage observed.
	RF09	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.
	RF10	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF11	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.
	RF12	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.
	RF13	VT-2*/UT	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed. Performed a manual POI qualified ultrasonic inspection of the nozzle to safe end weld as well as additional base material of bored material. No indications identified.
	RF14 (10/10)	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.
	RF15 (4/12)	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.
	RF16 (2014)	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF17 (2015)	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.
	RF18 (2017)	VT-2*	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed.
	RF19	VT-2*/UT/PT	Performed enhanced inspection on nozzle area from inside skirt area, and removed cover on the mirror insulation box for the safe-end for direct inspection. No leakage observed. Performed augmented surface PT and manual PDI qualified ultrasonic inspection of the nozzle to safe end weld as well as additional base material of bored material. NRI.
Jet Pump Assembly (BWRVIP-41)	Each outage examine at least 50% thru RF05	VT-1, VT-3	Jet pump assemblies are inspected each outage from top to bottom. During RF-04 all (20) hold down beams were replaced as a preventative measure and to avoid performing UT's on the old style/original beams. Inspections are performed to the recommendations of SIL 551,574,465 S-1, and RICSIL 078. During RF05 one of the 80 restrainer screw tack welds was found to be cracked. This was evaluated and was not repaired during RF05.

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	RF06	MVT-1, VT-3	Performed inspections to the intent of BWRVIP-41 as well as augmented VT-3 of selected areas on jet pumps 1-10. Inspections included all High, Medium and Low Priority locations. Inspected RS-1 and RS-2 welds on jet pumps 11-20. One indication identified on RS-1 weld, 1.75" long. JCO performed prior to startup. No other new indications identified.
	RF07	EVT-1	Performed inspections to the intent of BWRVIP-41 including EVT-1's as well as augmented VT-1 and VT-3's of selected areas on jet pumps 11-20. Inspections included all High, Medium and Low Priority locations. Re-inspected previously identified indication on RS-1 weld, 1.75" long that was identified in RF06. No change in indication length or appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. No other indications or changes in previous indications identified.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF08	EVT-1	Performed re-inspections to the intent of BWRVIP-41 including EVT-1's as well as augmented VT-1 and VT-3's of selected areas on jet pumps l & 2. Inspections included all High, Medium and Low Priority locations. Re-inspected previously identified 1.75" long indication on RS-1 weld for Jet Pumps 7 & 8 that was identified in RF06. No change in indication length or appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. Inspected all 20 jet pumps per recommendations of SIL 629 and verified no wedge damage (WD-1) as well as full contact with restrainer screws. No damage identified on any location. Re-inspected all restrainer screw tack welds with no changes observed.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF09	EVT-1	Performed re-inspections to BWRVIP-41 including EVT-1's as well as augmented VT-1 and VT-3's of selected areas on Jet Pumps 3&4. Inspections included all High, Medium and Low Priority locations. Re-inspected previously identified 1.75" long indication on RS-1 weld for Jet Pumps 7 & 8 that was identified in RF06. No change in indication length or appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. Inspected all 20 Jet Pump Hold Down Beams by UT for BBl, BB2, and the transition area BB3 using the latest available technique from General Electric. No indications identified on the beams. Re-inspected all restrainer screw tack welds, contact area, and wedges after both tack welds on Jet Pump 15 were found cracked. No other damage or indications identified on any location. Jet Pump 15 permanently repaired by the installation of an auxiliary spring wedge. (Reference CARD 03-16929).
	RF10	EVT-1	Performed re-inspections to BWRVIP-41 including EVT-1's as well as augmented VT-1 and VT-3's of selected welds on Jet Pumps 4, 5, 6, 7, & 8. Re-inspected previously identified 1.75" long indication on RS-1 weld for Jet Pumps 7 & 8 that was identified in RF06. No change in indication length / appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. Re-inspected auxiliary spring wedge on Jet Pump 15. No other damage or indications identified on any location.

Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF11	EVT-1	Performed re-inspections to BWRVIP-41 including EVT-1's as well as augmented VT-1 and VT-3's of selected welds on Jet Pumps 7, 8, 9, & 10. Re-inspected previously identified 1.75" long indication on RS-1 weld for Jet Pumps 7 & 8 that was identified in RF06. No change in indication length / appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. Inspected all Jet Pump wedges after wear was identified on JP2 restrainer bracket. Performed inspection of other welds on Jet Pump 2 as required by BWRVIP-41. Auxiliary spring wedges installed on Jet Pumps 1 and 2 and a slip joint clamp was installed on Jet Pump 2 to restore integrity. No other damage or indications identified.
	RF12	EVT-1	Performed re-inspections to BWRVIP-41 including EVT-1's as well as augmented VT-1 and VT-3's of selected welds on Jet Pumps 7, 8, 9, 10, 11, & 12. Re-inspected previously identified 1.75" long indication on RS-1 weld for Jet Pumps 7 & 8 that was identified in RF06. No change in indication length / appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. Inspected all 20 Jet Pump Hold Down Beams. Inspected 12 Jet Pump wedges including the wedges and hardware (auxiliary spring wedges and slip joint clamp) installed in RFl1. No other damage or indications identified.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF13	EVT-1	Performed re-inspections to BWRVIP-41 including EVT-1's as well as augmented VT-1 and VT-3's of selected welds on Jet Pumps 7, 8, 9, 10, 13, 14, 15, and 16. Re-inspected previously identified indication on RS-1 weld for Jet Pumps 7 & 8 identified in RF06. No change in indication length or appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. Inspected 9 Jet Pump wedges. No other damage or indications identified.
	RF14 (10/10)	EVT-1	Performed re-inspections to BWRVIP-41 including EVT-1's as well as augmented VT-1 and VT-3's of selected welds on most Jet Pumps including RS-8/9 welds on all pumps. Re-inspected previously identified indication on RS-1 weld for Jet Pumps 7 & 8. No change in indication length or appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. Inspected all 20 Jet Pump wedges. Minor movement noted but no other damage or indications identified.
	RF15 (4/12)	EVT-1	Performed re-inspections to BWRVIP-41 including EVT-1's as well as augmented VT-1's of selected welds on several Jet Pumps. Re-inspected previously identified indication on RS-1 weld for Jet Pumps 7 & 8. No change in indication length or appearance. Existing Flaw Evaluation on hand prepared by GE referenced as acceptance limit. Inspected all 20 Jet Pump wedges. No movement noted and no damage or indications identified.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF16 (2014)	EVT-1/ VT-1	Performed EVT-1 exams of selected welds in accordance with BWRVIP-41 Rev. 3 with no indications identified. VT-1 exams performed on all 20 main wedge assemblies. Wedge wear identified on Jet Pump 06; scope expansion performed with no further relevant indications observed and wedge was evaluated to be acceptable without repair. Growth identified during re-inspection of indication on RS-1 weld for Jet Pumps 7 & 8. Indication was evaluated to be acceptable for two cycles without repair. Ultrasonic examination of all 20 Jet Pump Hold Down Beams (BBl, BB2 and BB3). No indications identified on the beams.
	RF17 (2015)	EVT-1/ VT-1	Performed EVT-1 exams of selected welds with no indications identified. VT-1 exams performed on all 20 main wedge assemblies. Minor wedge rod wear identified on Jet Pump 1; evaluated to be acceptable without repair. Minor wear identified on Jet Pump 2 Auxiliary Wedge, scope expansion performed with no further relevant indications observed and evaluated to be acceptable without repair. Mitigating clamp installed on the RS-1 weld for Jet Pumps 7 & 8.

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	RF18 (2017)	EVT-1	Performed EVT-1 exams of selected welds with no indications identified. VT-1 exams performed on all 20 main wedge assemblies. Jet Pump 1 Wedge Rod wear obscured by wedge position shift; scope expansion performed with no further relevant indications observed and evaluated to be acceptable without repair.
	RF19	EVT-1	Performed EVT-1 exams of selected welds with no indications identified. VT-1 exams performed on all 20 main wedge assemblies. Minor wear identified on Jet Pump 11 Wedge Rod; evaluated to be acceptable without repair.
Jet Pump Diffuser (BWRVIP-41)	Each outage	VT-3	Diffusers will be sample inspected during refueling outages.
	RF06	MVT-1	BWRVIP-41 on Jet Pumps 1-10 except inaccessible areas. No cracking.
	RF07	EVT-1	BWRVIP-41 on Jet Pumps 11-20 except inaccessible areas. No cracking identified. Welds DF-3, AD-1, and AD-2 are inaccessible for inspection.
	RF08	EVT-1	BWRVIP-41 reinspection on Jet Pumps 1 and 2 except inaccessible areas. No cracking identified. Welds DF-3, AD-1, and AD-2 are inaccessible for inspection.

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	RF09	EVT-1	BWRVIP-41 reinspection on Jet Pumps 3 and 4 except inaccessible areas. No cracking identified. Welds DF-3, AD-1, and AD-2 are inaccessible for EVT-1 visual inspection, VT-3 performed. (TJ-2003-02 prepared as justification).
	RF10	EVT-1	BWRVIP-41 reinspection of selected DF-1 and DF-2 welds on Jet Pumps 5, 6, 7, & 8. Performed access study for future performance of UT examinations of welds DF-3, AD-1, and AD-2. These welds are inaccessible for visual inspection. VT-3 performed. No indications identified (Reference TJ-2003-02).
	RF11	EVT-1	BWRVIP-41 reinspection of selected DF-2 welds on Jet Pumps 9 & 10.
		UT	Performed of UT examinations on a portion of a total of 17 DF-3, AD-1, and AD-2 welds using specialized tooling. These welds are inaccessible for visual inspection. No indications identified (Reference DD-2006-02).
	RF12	EVT-1	BWRVIP-41 reinspection of selected DF-1 and 2 welds on Jet Pumps 6, 11, & 12.
		UT	No UT examinations performed during RF12 due to tooling failures. These welds are inaccessible for visual inspection. (Reference DD-2006-02).

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	RF13	EVT-1	BWRVIP-41 reinspection of selected DF-1 and 2 welds on Jet Pumps 7, 13, & 14.
		UT	No UT examinations performed during RF13 due to tooling failures. These welds are inaccessible for visual inspection. (Reference DD-2006-02).
	RF14 (10/10)	EVT-1	BWRVIP-41 reinspection of selected DF-1 and 2 welds on Jet Pumps 7, 8, 9, and 13-18. No indications identified.
		UT	Completed baseline UT examinations on all 20 Jet Pumps Diffuser/Adapter DF-3, AD-1 and AD-2 welds, (60 welds) since these welds are inaccessible for visual inspection. Deviation Disposition is no longer needed.
	RF15 (4/12)	EVT-1	BWRVIP-41 reinspection of selected DF-1 and 2 welds on Jet Pumps 10, 19, and 20. No indications identified.
	RF16 (2014)	EVT-1	BWRVIP-41 re-inspection of selected DF-1 and 2 welds on Jet Pumps 01, 02, and 11. No indications identified. ·
	RF17 (2015)	EVT-1	BWRVIP-41 re-inspection of selected DF-1 and 2 welds on Jet Pumps 03, 04, and 12. No indications identified.
	RF18 (2017)	EVT-1	BWRVIP-41 re-inspection of selected DF-1 and 2 welds on Jet Pumps 05, 06, 07, 08, 13, and 14. No indications identified.

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		UT	UT performed on DF-3, AD-1, and AD-2 on Jet Pumps 01, 02, 03, 04, 07, 08, 17, 18, 19, 20 with no relevant indications observed.
	RF19	EVT-1	BWRVIP-41 re-inspection of selected DF-1 and 2 welds on Jet Pumps 09, 10, & 15. No indications identified.
CRD Guide Tube (BWRVIP-47)	RF04	VT-3	Inspected lower portion of peripheral guide tubes and stub tubes when access was provided during jet pump hold down beam replacement. No indications identified.
	RF07	EVT-1 and VT-3	Performed best effort exam on CRGT-3 as weld was not visible on inside of tube. CRGT-2 not accessible due to flow and ARPIN was not felt to be accessible. No indications identified.
	RF08	EVT-1 and VT-3	Performed best effort exam on CRGT-3 as weld was not visible on inside of tube. CRGT-2 not accessible due to flow and PS/GT-ARPIN was not felt to be accessible. No indications identified.
	RF09	EVT-1 and VT-3	Performed exams on CRGT-1, CRGT-2, CRGT-3, and PS/GT-ARPIN at 10 Control Rod Guide Tubes/locations. No indications identified.
	RF10	N/A	No inspection performed in RF10.
	RF11	N/A	No inspection performed in RF11.

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF12	VT-3	Performed exams on CRGT-1 and PS/GT-ARPIN at 5 Control Rod Guide Tubes/locations. CRGT-2 and CRGT-3 not performed or credited due to high flow conditions. No indications identified.
	RF13	N/A	No inspections performed in RF13.
	RF14 (10/10)	EVT-1 and VT-3	Completed all remaining baseline inspections on the Control Rod Guide Tubes. Inspections performed on (4) CRGT-1, and PS/GT-ARPIN locations and on (9) CRGT-2 and CRGT-3 locations. One manufacturing flaw identified that did not impact the functionality of the component.
	RF15 (4/12)	N/A	No BWRVIP required inspections performed in RF15.
	RF16 (2014)	N/A	No BWRVIP required inspections performed in RF16.
	RF17 (2015)	N/A	No BWRVIP required inspections performed in RF17.
	RF18 (2017)	N/A	No BWRVIP required inspections performed in RF18.
	RF19	EVT-1	Performed opportunistic inspections of CRGTs 30-27 and 30-31. NRI.

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CRD Stub Tube * (BWRVIP-47)	RF04	VT-3	Inspected lower portion of peripheral guide tubes and stub tubes when access was provided during jet pump hold down beam replacement. No indications identified.
	RF19	VT-3	Performed opportunistic inspections of Stub Tubes at cell locations 30-27 and 30-31. NRI.
In-Core Housing * (BWRVIP-47)	RF04	VT-3	Small portion visible during jet pump beam replacement. No indication of degradation.
	RF19	VT-3	Performed opportunistic inspections of ICMH 32-29. NRI.
Dry Tube (BWRVIP-47)	Each outage	VT-1	9 of 12 tubes found not completely seated. Performed all inspections per SIL 409 and RICSIL 073. No indications of cracking.
	RF06	VT-1	Re-inspected 12 dry tubes. No change from previous condition. No cracking.
	RF07	VT-1	Inspected all 12 original design Dry Tubes. No change from previous conditions identified. No cracking identified.
	RF08	VT-1	Inspected all 12 original design Dry Tubes from two sides. No change from previous conditions identified. No cracking identified.
	RF09	N/A	No inspections performed in RF09.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF10	VT-1	Inspected all 12 original design Dry Tubes from two sides. Linear indications identified on 7 tubes in the collar region above the pressure boundary weld. Evaluated as acceptable for one cycle of operation. Plan to replace in RF 11. (Reference CARD 04-25703).
	RF11	VT-1	Replaced all 12 Dry Tubes in RF11. Performed baseline VT-1 and verified proper engagement in Top Guide.
	RF12	N/A	No inspections performed in RF12.
	RF13	N/A	No inspections performed in RF13.
	RF14 (10/10)	N/A	No inspections performed in RF14.
	RF15 (4/12)	N/A	No inspections performed in RF15.
	RF16 (2014)	N/A	No inspections performed in RF16.
	RF17 (2015)	N/A	No inspections performed in RF17.
	RF18 (2017)	N/A	No inspections performed in RF18.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF19	N/A	No inspections performed in RF19.
Instrument Penet. * (BWRVIP-49 & 41)	Each outage	VT-3	Inspected jet pump sensing lines and brackets each outage.
	RF04	VT-3	SLC and peripheral bottom head penetrations inspected. No indications.
	RF06	VT-3	Inspected JP sensing lines for pumps 1-10. No indications.
	RF07	VT-3	Inspected JP sensing lines for pumps 11 thru 20 only. No indications.
	RF08	VT-3	Inspected JP sensing lines for Pumps 1 & 2 only. No indications.
	RF09	VT-3	Inspected JP sensing lines for Pumps 3 & 4 only. No indications.
	RF10	VT-1	Inspected JP sensing lines for Pumps 5, 6, 7, 16, & 17. No indications.
	RF11	VT-1	Inspected JP sensing lines for Pumps 6, 7, 16, & 17. No indications.
	RF12	VT-1	Inspected JP sensing lines for Pumps 6, 7, 11, 12, 16, & 17. No indications.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF13	VT-1	Inspected JP sensing lines for Pumps 6, 7, 13, 14, 16, & 17. No indications.
	RF14 (10/10)	VT-1	Inspected JP sensing lines for Pumps 6, 7, 15, 16, 17, & 18. No indications.
	RF15 (4/12)	VT-1	Inspected JP sensing lines for Pumps 6, 7, 16, 17, 19, & 20. No indications.
	RF16 (2014)	VT-1	Inspected JP sensing lines for Pumps 1, 2, 6, 7, 16, & 17. No indications.
	RF17 (2015)	VT-1	Inspected JP sensing lines for Pumps 3, 4, 6, 7, 16, & 17. No indications.
	RF18 (2017)	VT-1	Inspected JP sensing lines for Jet Pumps 05, 06, 07, 08, 16, & 17 with no relevant indications noted.
	RF19	VT-1	Inspected JP sensing lines for Jet Pumps 06, 07, 09, 10, 16, & 17. NRI.
Vessel ID Brackets (BWRVIP-48)	Each outage	VT-1/3	Inspect sample population each outage. We have inspected most brackets each outage (core spray, feedwater). Jet pump riser brace, steam dryer support lugs, guide rod brackets and specimen holder; brackets are sample inspected. No indications of cracking identified.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF06	MVT-1	6 Feedwater brackets. All Core Spray piping brackets. 4 Steam Dryer brackets. 1 Guide Rod bracket. 1 Specimen bracket. No indication of cracking.
	RF07	EVT-1	6 Feedwater brackets. All Core Spray piping brackets. 4 Steam Dryer brackets. 1 Guide Rod bracket. No indication of cracking.
	RF08	EVT-1	6 Feedwater brackets. All Core Spray piping brackets. 4 Steam Dryer brackets. 1 Guide Rod bracket. Surveillance Holder and Brackets @ 30 az. No indication of cracking.
	RF09	EVT-1	6 Feedwater brackets. 4 Core Spray piping brackets. 1 Jet Pump riser brace (Jet Pump 3 and 4) No indication of cracking.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF10	EVT-1	6 Feedwater brackets. 3 Core Spray piping brackets. 1 Surveillance Holder bracket. 4 Steam Dryer Support brackets. 4 Steam Dryer Hold Down 1 Guide Rod Bracket. 1 Jet Pump riser brace (Jet Pump 5 and 6) No indication of cracking.
	RF11	EVT-1/ VT-1	No inspections performed in RF-11.
	RF12	EVT-1/ VT-1	6 Feedwater Sparger bracket sets. 1 Surveillance Holder bracket 4 Steam Dryer Support brackets 1 Guide Rod Bracket 2 Jet Pump riser braces (Jet Pumps 7, 8, 9, & 10) No indication of cracking identified.
	RF13	EVT-1/ VT-1	No inspections performed in RF-13.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF14 (10/10)	EVT-1/ VT-1	3 Feedwater Sparger bracket sets. 2 Core Spray Piping Brackets 1 Surveillance Holder bracket 4 Steam Dryer Support brackets I Guide Rod Bracket 2 Jet Pump riser braces (Jet Pumps 1/ 2, and 11/12) No indication of cracking identified.
	RF15 (4/12)	EVT-1/ VT-1	Inspections performed on 3 Feedwater Sparger bracket sets and 1 Guide Rod Bracket. No indications identified.
	RF16 (2014)	EVT-1/ VT-1	Inspection performed on 1 Surveillance Sample Holder Bracket. No indications identified.
	RF17 (2015)	EVT-1/ VT-1	2 Feedwater Sparger bracket sets (four individual brackets) 4 Core Spray Piping Brackets 4 Steam Dryer Support Brackets 4 Steam Dryer Holddown Brackets 1 Guide Rod Bracket 1 Jet Pump Riser Brace. No indication of cracking identified.
	RF18 (2017)	EVT-1/ VT-1	3 Feedwater Sparger bracket sets (six individual brackets) 1 Surveillance Holder Bracket 1 Guide Rod Bracket 1 Jet Pump Riser Brace. No indication of cracking identified.

Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF19	EVT-1/VT-1	4 Steam Dryer Support Brackets 1 Feedwater Sparger Bracket (two individual brackets) 3 Surveillance Sample Holder Brackets (2018-043) No relevant indications.
LPCI Coupling	N/A	N/A	Fermi does not have a LPCI Coupling
Shroud Head Bolts/Shroud Head	RF04	UT/VT	16 had indications, 17 replaced during RF04.
	RF05	N/A	Remaining bolts replaced (31) duringRF05 as a preventative measure. All 48 are now new style.
	RF06	VT-3	Bolts 1-24 (of 48). No indication of cracking.
	RF07	VT-3	Bolts 25-48 (of 48). No indication of cracking or damage. Springs were left compressed on 20 of the 24 inspected.
	RF08	VT-3	Bolts 1-24 (of 48). No indication of cracking or damage.
	RF09	VT-3	Bolts 23 and 25-48 (of 48). No indication of cracking or damage. All retainer springs verified to be functioning properly.
	RF10	VT-3	Bolts 1-24 (of 48). Inspected North 1/3rd of Shroud Head/Separator and 2 lifting lugs. No indication of cracking or damage.

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF11	VT-3	Inspected Bolts 25-48 (of 48) and inspected Center 1/3rd of Shroud Head/Separators. No indication of cracking or damage.
	RF12	VT-3	Bolts 1-24 (of 48). Inspected South 1/3rd of Shroud Head/Separator and 2 lifting lugs. All mid support ring gussets were inspected and small short cracks were identified on 3 of the 24 gussets. No repairs were required. Ref. OE 25795.
	RF13	VT-3	Bolts 25-48 (of 48). Inspected North 1/3rd of Shroud Head/Separator and 2 lifting lugs. No changes identified in previous indications identified in RF12. No other indications identified.
	RF14 (10/10)	VT-3	Bolts 1-24, 27, 30, & 33 (of 48). Inspected Center 1/3rd of Shroud Head/Separator. No changes identified in previous indications and no new indications identified.
	RF15 (4/12)	VT-3	Inspected Bolts 25-48 and 2 (of 48). Inspected South 1/3rd of Shroud Head/Separator. No changes identified in previous gusset indications and no new indications identified.
	RF16 (2014)	VT-3	Inspected Bolts 1-12 (of 48) and the North 1/3rd of Shroud Head/Separator. No new indications were identified.

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Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF17 (2015)	VT-3	Inspected bolts 13-24 (of 48). Minor pin & window wear identified on bolts 21 & 23, evaluated to be acceptable without repair. Replaced bolts 2 & 33 due to their inability to latch. Inspected the Center I/3rd of the Separator and identified one tie bar with a severed attachment weld on one end. The tie bar was removed to preclude generation of a loose part (technical justification from OEM obtained to support acceptance of one missing tie bar).
	RF18 (2017)	VT-3	Re-inspected bolts 21 & 23 with no change in condition recorded. Normal inspections included bolts 25-36. Minor pin wear on bolts 31 & 35; evaluated acceptable without repair. Pin and window wear observed on bolts 34 & 36; evaluated acceptable without repair. No other indications.
	RF19	VT-3	Reinspected bolt 36; no change in condition. Normal inspections included bolts 37-48; NRI. Examined North 1/3rd of Separator; NRI.
Steam Dryer (RF01–RF08 not previously reported)	RF09	VT-3	Inspected approximately 1/3 of dryer including hood welds and cover plate welds. (Ref. SIL 644). No indications of additional cracking identified.

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Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF10	VT-1/VT-3	Inspected approximately 50% of dryer including all inner hood vertical welds as recommended in SIL 644, Supplement 1, and Revision 1. Several new indications were identified near welds due to new locations being inspected and the change in technique. Indications were noted at base of inner hood vertical welds. Reference CARD 04-25416 and also OE #17600. No changes were identified on previously recorded indications.
	RF11	VT-1/VT-3	Inspected approximately 50% of dryer including all inner hood vertical welds as recommended in SIL 644, Revision 1 and BWRVIP-139. Several new indications were identified near welds due to new locations being inspected and the change in technique. Indications previously noted on hood welds in RF10 were re-inspected and no changes were noticed.
	RF12	VT-1/VT-3	Inspected approximately 50% of dryer including inner hood vertical welds as recommended in BWRVIP-139. Several new small indications were identified near welds due to new locations being inspected and the change in technique and camera angles used. Indications previously noted on hood welds were re-inspected and no changes were noticed.
	RF13	VT-1/VT-3	Inspected approximately 20% of dryer including "F" Bank welds and a sampling of other locations following re-inspection guidelines contained in NRC SE to BWRVIP-139. One new indication identified in support ring.

Components in BWRVIP Scope	**Date or Frequency of Inspection**	**Inspection Method Used**	**Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections**
	RF14 (10/10)	VT-1/VT-3	Inspected approximately 20% of dryer including "E" Bank welds and a sampling of other locations following re-inspection guidelines contained in BWRVIP-139-A. No new indications identified.
	RF15 (4/12)	VT-1/VT-3	Inspected approximately 20% of dryer including "D" Bank welds and a sampling of other locations following re-inspection guidelines contained in BWRVIP-139-A. No new indications identified.
	RF16 (2014)	VT-1/VT-3	Inspected approximately 20% of dryer including "C" Bank welds and a sampling of other locations following re-inspection guidelines contained in BWRVIP-139-A. Indication newly identified on interior vane bank weld HE-C-2-1; evaluated to be acceptable without repair.
	RF17 (2015)	VT-1/VT-3	Inspected approximately 20% of dryer including "B" Bank welds and a sampling of other locations following re-inspection guidelines contained in BWRVIP-139-A. Indication on interior vane bank weld HEC-2-1 identified in RF16 re-inspected with no changes observed. 24 capture plate assemblies installed to cover all tie rod nut washer locations. Vertical drain channel welds preemptively increased from 1/8" to 1/4".

Components in BWRVIP Scope	Date or Frequency of Inspection	Inspection Method Used	Summarize the Following Information: Inspection Results, Repairs Replacements, Re-inspections
	RF18 (2017)	VT-1/VT-3	Inspected approximately 20% of dryer including "A" Bank welds and a sampling of other locations following re-inspection guidelines contained in BWRVIP-139-A. Indication on interior vane bank weld HEC-2-1 identified in RF16 re-inspected with no changes observed. 24 capture plate assemblies re-inspected with no new indications. New wear at 94° Seismic Support Block at interface with RPV Support Block; evaluated as acceptable without repair.
	RF19	VT-1/VT-3	Performed VT-1inspections of approximately 20% of dryer including “F” Bank welds and a sampling of other locations following reinspection guidelines contained in BWRVIP-139-A. Upper support ring indication newly identified near DC-C-3 weld; evaluated as acceptable without repair along with revisited flaws. Additional VT-3 exams performed on inner vane bank end panels and all of vane bank A (including cover plate) as part of Moisture Carryover investigation - NRI.
*VT-2 leakage inspections have been and are performed on all RPV Instrumentation Nozzles and Piping Nozzles each refuel outage. An enhanced leakage inspection is performed on all locations to ensure no pressure boundary leakage. Inspections are performed in the annulus area adjacent to the vessel skirt, and are performed under vessel to ensure that any leakage identified is not from welded connections. Flange leakage from CRDM's is recorded, evaluated, and repaired if necessary. Mirror insulation is opened for SLC safe end inspection and for bottom head inspections but is not removed from other locations unless the leakage source can't be determined.			

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PAST REFUEL OUTAGE DATES

Refueling Outage	Duration and Dates
RF01	102 DAYS SEPTEMBER 5, 1989 - DECEMBER 16, 1989
RF02	72 DAYS MARCH 30, 1991 - JUNE 10, 1991
RF03	56 DAYS SEPTEMBER 12, 1992 - NOVEMBER 7, 1992
RF04	389 DAYS DECEMBER 25, 1993 - JANUARY 18, 1995
RF05	98 DAYS SEPTEMBER 27, 1996 - JANUARY 3, 1997
RF06	54 DAYS SEPTEMBER 4, 1998 - OCTOBER 29, 1998
RF07	52 DAYS APRIL 1, 2000 - MAY 23, 2000
RF08	33 DAYS OCTOBER 27, 2001 - NOVEMBER 30, 2001
RF09	42 DAYS MARCH 28, 2003 – MAY 10, 2003
RF10	27 DAYS NOVEMBER 6, 2004 – DECEMBER 3, 2004
RF11	41 DAYS MARCH 25, 2006 – MAY 5, 2006
RF12	50 DAYS SEPTEMBER 29, 2007 – NOVEMBER 18, 2007
RF13	34 DAYS MARCH 28, 2009 – MAY 1, 2009
RF14	41 DAYS OCTOBER 24, 2010 – DECEMBER 5, 2010
RF15	40 DAYS MARCH 26, 2012 – MAY 5, 2012
RF16	53 DAYS FEBRUARY 10, 2014 – April 5, 2014
RF17	61 DAYS SEPTEMBER 27, 2015 – NOVEMBER 28, 2015
RF18	33 DAYS MARCH 18, 2017 – APRIL 20, 2017
RF19	35 DAYS SEPTEMBER 22, 2018 – OCTOBER 27, 2018

Enclosure 4 to
NRC-19-0016

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

RR-A40
Proposed Alternative to Utilize Code Case N-513-4

10 CFR 50.55a Request Number
RR-A40
Fourth Interval Relief Request

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(2)

-Hardship or Unusual Difficulty
without a Compensating Increase in Quality and Safety-

1. ASME Code Component(s) Affected

ASME Code Class:	Code Class 2 and 3
References:	ASME Section XI, 2013 Edition
Examination Category:	C-H, D-B Pressure Retaining Components
Item Numbers:	C7.10, D2.10
Description:	Use of ASME Code Case N-513-4 for evaluation of Class 2 and 3 components within the scope of the Code Case
Components:	All American Society of Mechanical Engineers (ASME) Section XI, Class 2 and 3 moderate energy piping systems meeting Code Case N-513-4 operating limits, which are a maximum operating temperature not in excess of 200 degrees Fahrenheit and an operating pressure not in excess of 275 pounds per square inch

2. Applicable Code Edition and Addenda

ASME Section XI, 2013 Edition.

3. Applicable Code Requirement

ASME Code, Section XI, subarticle IWC-3100 applies to Class 2 components and requires that flaws exceeding the specified acceptance criteria be corrected by repair or replacement, or be deemed acceptable by analytic evaluation. ASME Code, Section XI, subarticle IWD-3100 applies to Class 3 components and requires that components exceeding the acceptance standards of IWD-3400 be subject to supplemental examination, or to a repair or replacement activity.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(2), DTE Electric Company (DTE) is requesting a proposed alternative to the Code requirements for repair and replacement activity of moderate energy piping components, due to limitations posed by ASME Code Case N-513-3,

"Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping, Section XI, Division 1," regarding the evaluation of flaws in certain locations. Under the current code, moderately degraded piping could require a plant shutdown within the required action statement timeframes to repair the observed degradation. The resulting dose accrual and plant risk would fail to provide a compensating increase in levels of quality or safety when the degraded condition is demonstrated to retain adequate margin for component functionality. While ASME Code Case N-513-3 is limited to the evaluation of flaws in straight piping, Code Case N-513-4 provides guidance for evaluating flaws located in elbows, bent pipe, reducers, expanders, and branch tees, provided those flaws are at least a minimum formulated distance from circumferential pipe welds. Code Case N-513-4 also provides for the evaluation of heat exchanger external tubing or piping.

The use of an acceptable analytic evaluation in lieu of immediate repair or replacement activities for the components covered by Code Case N-513-4 would conserve dose and decrease plant risk where shutdowns and immediate repair/replace activities on components that prove to have adequate margin can be prevented in favor of planned, orderly repairs scheduled in the longer term. Accordingly, compliance with the current code requirements represents a hardship without a compensating increase in the level of quality or safety.

5. Proposed Alternative and Basis for Relief

Pursuant to 10 CFR 50.55a(z)(2), DTE requests authorization to apply the evaluation methods of ASME Code Case N-513-4, "Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 and 3 Piping Section XI, Division 1," to Class 2 and 3 components that meet the design and operational limits described in Code Case N-513-4.

The NRC approved Code Case N-513 versions in Regulatory Guide 1.147, "Inservice Inspection Code Case Applicability, ASME Section XI, Division 1" (Reference 1), which allows the acceptance of partial through-wall or through-wall leaks for an operating cycle provided all conditions of the Code Case and the NRC conditions are met. The Code Case also requires the Owner to demonstrate system operability due to leakage. Limitations in Code Case N-513-3 related to its use on components such as elbows, bent pipe, reducers, expanders, and branch tees and external tubing or piping attached to heat exchangers have been addressed in Code Case N-513-4. The major differences between the NRC-approved Code Case N-513-3 and the Code Case N-513-4 are listed below:

1. Revised the maximum allowed time of use from no longer than 26 months to the next scheduled refueling outage.
2. Added applicability to piping elbows, bent pipe, reducers, expanders, and branch tees where the flaw is located more than $(R_o t)^{1/2}$ (where R_o is the outside pipe radius and t is the evaluation wall thickness) from the centerline of the attaching circumferential piping weld.
3. Expanded use to external tubing or piping attached to heat exchangers.
4. Revised to limit the use to liquid systems.

5. Revised to clarify treatment of Service Level load combinations.
6. Revised to address treatment of flaws in austenitic pipe flux welds.
7. Revised to require minimum wall thickness acceptance criteria to consider longitudinal stress in addition to hoop stress.
8. Other minor editorial changes to improve the clarity of the Code Case.

Code Case N-513-4 uses technical evaluation approaches that are based on principles that are accepted in other Code documents already approved by the NRC. The conservative natures of the calculations, temporary acceptance period, and leakage monitoring frequency given in Code Case N-513-4 make it no less technically rigorous in its approach than the approved Code Case N-513-3 and the NRC conditions applied to that Code Case. A technical evaluation of the significant changes in Code Case N-513-4 when compared to NRC approved Code Case N-513-3 was provided by ASME and was previously provided to the NRC in Attachment 4 of an Exelon fleet-wide relief request submittal to utilize Code Case N-513-4 (Reference 2) which was approved by the NRC. The technical information provided in Attachment 4 of Reference 2 is applicable for Fermi 2's intended use of Code Case N-513-4.

The design basis is considered for each leak and evaluated using the Fermi 2 Operability Determination Process. The evaluation process must consider requirements or commitments established for the system, continued degradation and potential consequences, operating experience, and engineering judgment. As required by the Code Case, the evaluation process considers but is not limited to system make-up capacity, containment integrity with the leak not isolated, effects on adjacent equipment, and the potential for room flooding.

Leakage rate is not typically a good indicator of overall structural stability in moderate energy systems, where the allowable through-wall flaw sizes are often on the order of inches. The periodic inspection interval defined using paragraph 2(e) of Code Case N-513-4 provides evidence that a leaking flaw continues to meet the flaw acceptance criteria and that the flaw growth rate is such that the flaw will not grow to an unacceptable size.

The effects of leakage may impact the operability determination or the plant flooding analyses specified in paragraph 1(f). For a leaking flaw, the allowable leakage rate will be determined by dividing the critical leakage rate by a safety factor of four. The critical leakage rate is determined as the lowest leakage rate that can be tolerated and may be based on the allowable loss of inventory or the maximum leakage that can be tolerated relative to room flooding, among others. The safety factor of four on leakage is based upon Code Case N-705, which is accepted without condition in Regulatory Guide 1.147 (Reference 1). Paragraph 2.2(e) of N-705 requires a safety factor of two on flaw size when estimating the flaw size from the leakage rate. This corresponds to a safety factor of four on leakage for nonplanar flaws. Although the use of a safety factor for determination of an unknown flaw is considered conservative when the actual flaw size is known, this approach is deemed acceptable based upon the precedent of Code Case N-705. Note that the alternative herein

does not propose to use any portion of Code Case N-705 and that citation of N-705 is intended only to provide technical basis for the safety factor on leakage.

During the temporary acceptance period, leaking flaws will be monitored daily as required by paragraph 2(f) of Code Case N-513-4 to confirm the analysis conditions used in the evaluation remain valid. Significant change in the leakage rate is reason to question that the analysis conditions remain valid, and would require re-inspection per paragraph 2(f) of the Code Case. Any re-inspection must be performed in accordance with paragraph 2(a) of the Code Case.

The leakage limit provides quantitative measurable limits which ensure the operability of the system and early identification of issues that could erode defense-in-depth and lead to adverse consequences.

In summary, DTE proposes to apply ASME Code Case N-513-4 to the evaluation of Class 2 and 3 components that are within the scope of the Code Case. The application of Code Case N-513-4, along with leakage limits, will maintain acceptable structural and leakage integrity while minimizing plant risk and personnel radiation exposure as compared to repairing instances of degradation in certain components under the current criteria.

6. Duration of Proposed Alternative

This relief is requested for the duration of the Fourth Inservice Inspection Interval, which begins on May 2, 2019 and is scheduled to end on December 31, 2029, or until the NRC publishes Code Case N-513-4 in a future revision of Regulatory Guide 1.147 or another document.

7. Precedent

1. Letter from U. S. Nuclear Regulatory Commission (USNRC) to Exelon, "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; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Oyster Creek Nuclear Generating Station; 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 Use ASME Code Case N-513-4 (CAC Nos. MF7301–MF7322)," dated September 6, 2016 (ML16230A237).

2. Letter from DTE Energy Company, LLC to U.S. Nuclear Regulatory Commission, "Proposed Alternative to Utilize Code Case N-513-4, 'Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1'," dated August 15, 2017 (ML18003A210), as approved by the NRC in a letter dated March 7, 2018 (ML18003A210).

8. <u>References</u>

1. NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability ASME Section XI, Division 1", Revision 18, March 2017 (ML16321A336).

2. Letter from Exelon Generation Company, LLC to U.S. Nuclear Regulatory Commission, "Proposed Alternative to Utilize Code Case N-513-4, 'Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1'," dated January 28, 2016 (ML16029A003).

Enclosure 5 to
NRC-19-0016

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

RR-A41
Use of ASME Code Case N-864 for the Elimination of
Reactor Pressure Vessel – Threads in Flange Examination

10 CFR 50.55a Request Number
RR-A41
Fourth Interval Relief Request

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

-Alternative Provides Acceptable Level of Quality and Safety-

1. ASME Code Component(s) Affected

ASME Code Class:	Code Class 1
References:	ASME Code Section XI, 2013 Edition
Examination Category:	B-G-1, "Pressure Retaining Bolting, Greater Than 2 in. In Diameter"
Items Numbers:	B6.40, "Threads in Flange"
Description:	Use of ASME Code Case N-864 (Reference 1) for the Elimination of Reactor Pressure Vessel – Threads in Flange Examination
Components:	Pressure retaining bolting greater than 2 inches

2. Applicable Code Edition and Addenda

ASME Code Section XI, 2013 Edition

3. Applicable Code Requirements

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 Figure IWB-2500-12.

4. Reason for Request

DTE Electric Company (DTE) 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, EPRI report (Reference 2) 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 benefits of the current examination requirements are not commensurate with the associated impact on worker exposure, personnel safety, radwaste, and increased time at reduced inventory.

In consideration of the work above, the responsible ASME Codes & Standards Committees have now taken this technical basis and evaluated it for a proposed alternative to the ASME Code Section XI requirements. As a result of this evaluation and the committee consensus process, ASME has now approved and published ASME Code Case N-864. This code case eliminates the requirement for the RPV Threads in Flange to be examined in accordance with ASME Code Section XI, but it has not been reviewed or approved for generic use by NRC and is not currently listed in the latest revision of Regulatory Guide 1.147, Revision 18 (Reference 3). Therefore, in order to eliminate this examination at Fermi 2, this alternative request to use ASME Code Case N-864 is being submitted for NRC review and approval.

5. Proposed Alternative

Pursuant to 10 CFR 50.55a(z)(1), DTE requests authorization to use the alternative requirements of ASME Code Case N-864, in lieu of those contained in the ASME Code Section XI, 2013 Edition for the examination of the RPV Threads in Flange. This request will eliminate the volumetric examination requirement of the RPV Threads in Flange at Fermi 2, while providing an acceptable level of quality and safety.

Other inspection activities, including the system leakage test required by ASME Code Section XI, Examination Category B-P, which is conducted each refueling outage, will continue to be performed.

6. Basis for Use

The technical basis provided in the EPRI report and additional plant specific information for the use of ASME Code Case N-864 at Fermi 2 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 ASME Risk-Based Inspection Development Guidelines (Reference 4), which states 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 in the EPRI report to determine the stresses at critical regions of the thread 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 Fermi 2 as compared to the values used in the evaluation of the bounding preload stress 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 Fermi 2 Parameters to Values Used in Bounding 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 (psig)	Preload Stress (psi) (Note 3)
Fermi 2	68	68	6.25	251	14	1250	32,612
Range for 16 Units Considered	N/A	54 - 60	6.0 - 7.0	157 - 173	15 - 16	2500	42,338
Values Used in Bounding Analysis	**54**	**54**	**6.0**	**173**	**16**	**2500**	**42,338**

Notes:
1. This only applies 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 Fermi 2 preload on the bounding geometry is calculated as:

$$P_{preload} = \frac{C \cdot P \cdot ID^2}{s \cdot D^2} = \frac{1.1 \cdot 1250 \cdot 251^2}{68 \cdot 6.25^2} = 32{,}612 \text{ psi}$$

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:

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

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 100ºF/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 Fermi 2 are shown in Table 2 below:

Table 2: RPV Flange Thread Geometry

Plant	Thread Specification	Nominal Bolt Hole Diameter in Flange (inches)	Pitch	Thread Depth (inches)
Fermi 2	7”- 8 SPECIAL-2B	7.00	8	0.0675
Analysis Geometry	**7”-8N-2B8**	**7.00**	**8**	**0.06500**

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:

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

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

Additionally, when determining the preload as it applies to Fermi 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_{Ic} (ksi√in) (Note 2)
	From Plant Records	From NRC RVID2 Database (Note 3)			
Fermi 2	10	N/A	≥72	62	104.85

Notes:
1. T represents the minimum permissible flange temperature required for preload to the RPV head studs.
2. K_{Ic} is determined in accordance with ASME Section XI, 2013 Edition, Nonmandatory Appendix A, A-4200. K_{Ic} is to be determined for the minimum preload temperature above.
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 the EPRI report (reproduced in Table 3 above), the maximum K is 19.8 ksi√in. The allowable K calculated in Section 6.2.2 of the report is 69.6 ksi√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√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 Fermi 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 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 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 Results – U.S. 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 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 the concern that steam generator tubes might fail before other RCS components, with a resultant bypass of containment. The key take-away for 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.

Control of Non-Service Induced Degradation

To protect against non-service related degradation, Fermi 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.

Conclusion

The EPRI report provides the technical basis for the elimination of the RPV Threads in Flange examination at Fermi 2 as described above when coupled with plant specific information used for the basis of this request. This report also indirectly supports the use of ASME Code Case N-864 to eliminate these same RPV Threads in Flange examinations at Fermi 2. 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 basis for this proposed alternative is an acceptable alternate approach to the performance of the ASME Code Section XI required volumetric ultrasonic examination, Fermi 2 is requesting approval of this alternative on the basis that use of this alternative provides an acceptable level of quality and safety.

7. Duration of Proposed Alternative

Upon authorization by NRC, this request for an alternative to use ASME Code Case N-864 specifically for Fermi 2 will be implemented during the Fourth Ten-Year ISI Interval beginning on May 2, 2019 and ending on December 31, 2029 (which will include a portion of the period of extended operation from March 21, 2025).

8. Precedents

The NRC Staff has authorized similar requests to eliminate the RPV Flange in Threads examination at varied units within the U. S. Fleet using the same basis provided in this request that is to use ASME Code Case N-864. Some of the most recent NRC Safety Evaluation Reports (SER) and some of the most recently approved requests are listed below.

- NRC SER from U. Shoop to Duke Energy, S Capps regarding Brunswick Steam Electric Plant, Unit No. 1; Catawba Nuclear Station, Unit No.2; Shearon Harris Nuclear Plant, Unit No. 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 Inservice Inspection Regarding Reactor Pressure Vessel Threads in Flange (CAC Nos. MF9513 – MF-9521; EPID L-2017-LLR-0019)" dated, December 26, 2017, (ADAMS Accession No.: ML17331A086)

- [1]NRC SER from D. J. Wrona to Exelon Generating Company, LLC, B. C Hanson, regarding Braidwood Station, Units 1and 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.: ML17170A013)

- NRC SER from J. G. Danna to Dominion Nuclear Connecticut, Inc., D. G. Stoddard regarding “Millstone Power Station, Unit Nos. 2 and 3” - Alternative Requests 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)

- 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, (ADAMS Accession No.: ML17006A109)

9. <u>References</u>

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

2. EPRI Nondestructive Evaluation Report- Reactor Pressure Vessel Threads in Flange Examination Requirements. 3002007626; dated March 2016, Electric Power Research Institute (EPRI). (ADAMS Accession No.: ML16221A068)

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[1] As stated in the second precedent, the NRC SER for Exelon Generating Company, LLC, the Staff’s evaluation contained in the fourth precedent concluded that the generic stress analysis and flaw tolerance evaluation provided in the EPRI report, are acceptable and can be used to support eliminating examination of threads in the RPV flange.

3. USNRC Regulatory Guide 1.147, Revision 18, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, dated March 2017, (ADAMS Accession No.: ML16321A336)

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.

Figure – 1: Modeled Dimensions

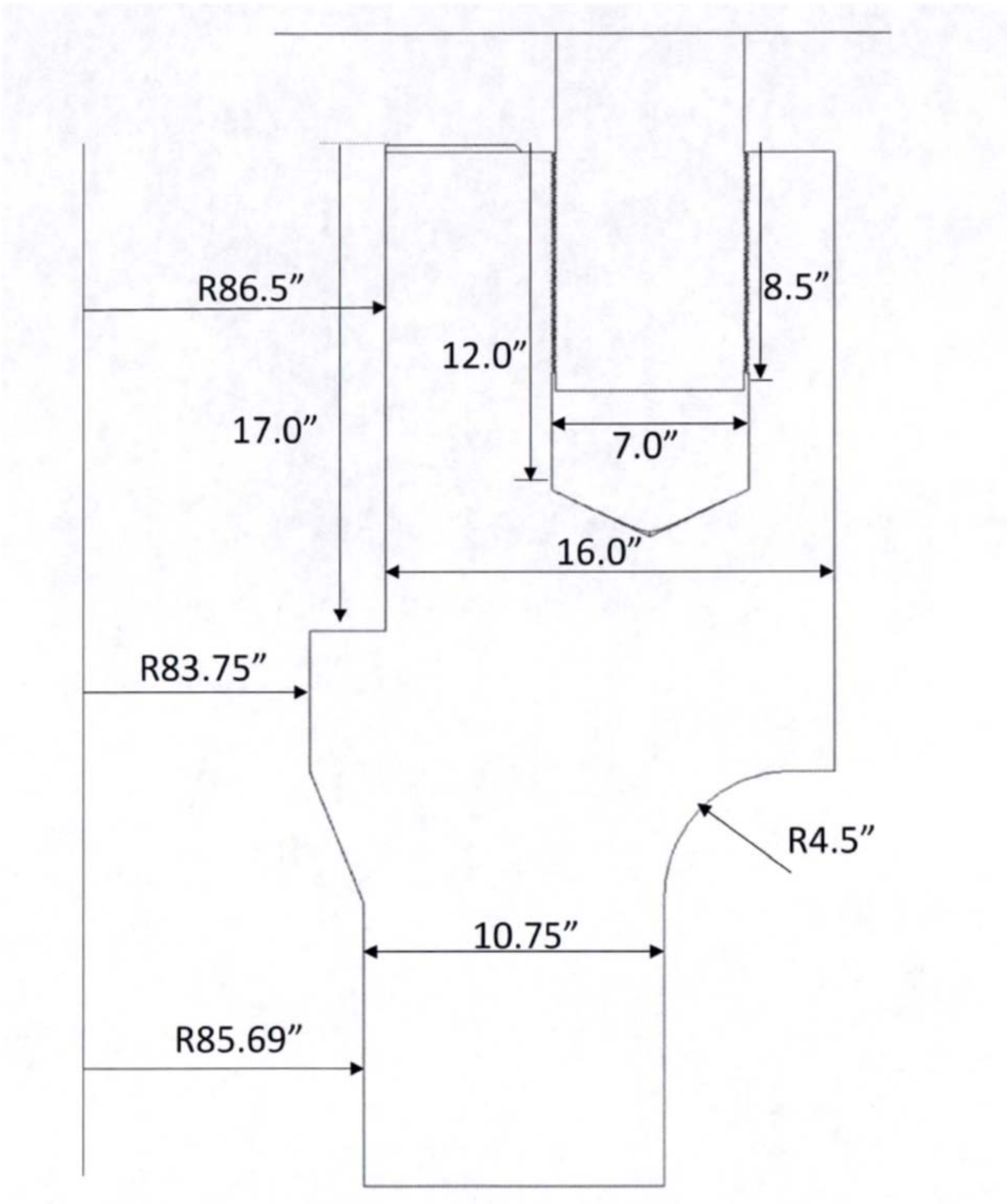

Figure – 2: Finite Element Model Showing Bolt and Flange Connection

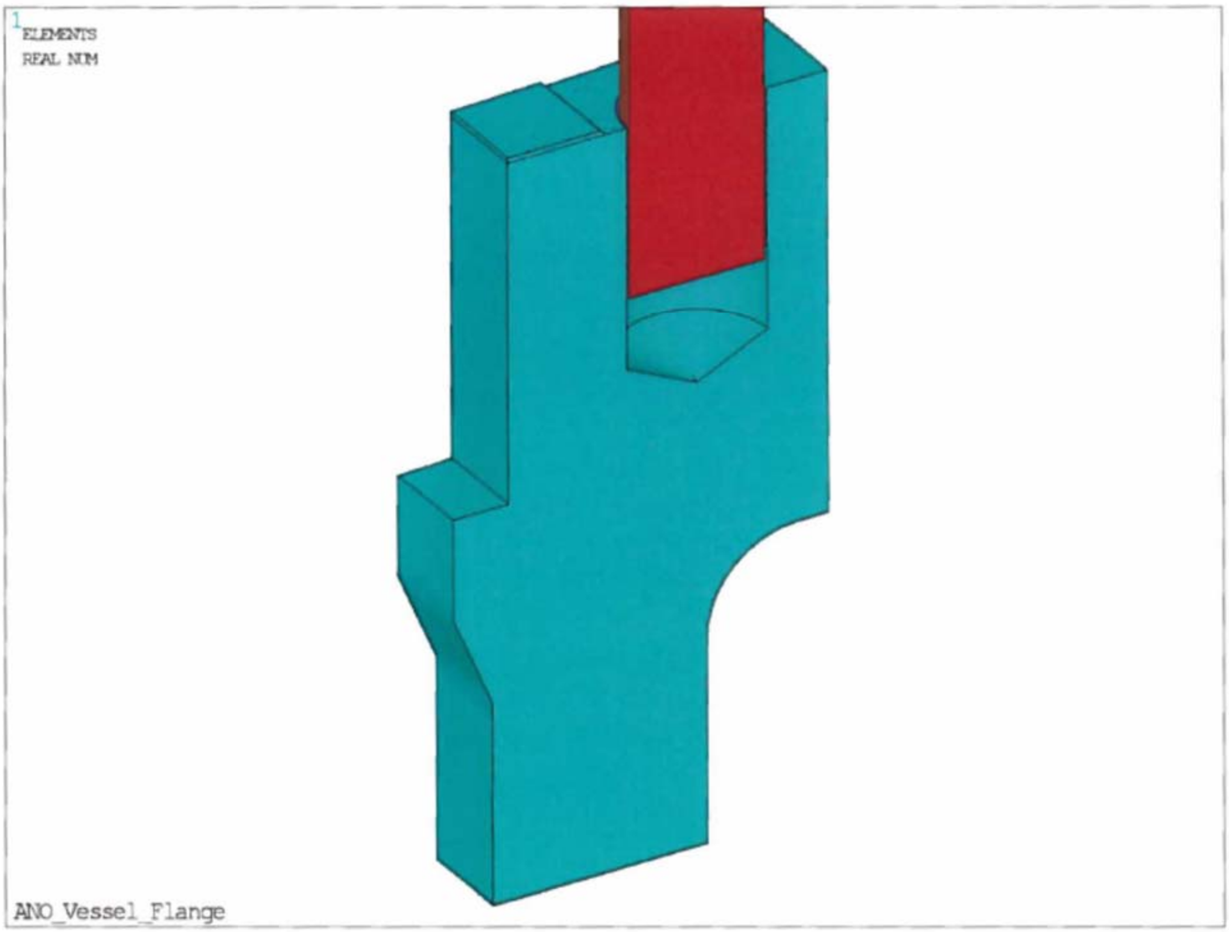

Figure – 3: Finite Element Model Mesh Detail at Thread Locations

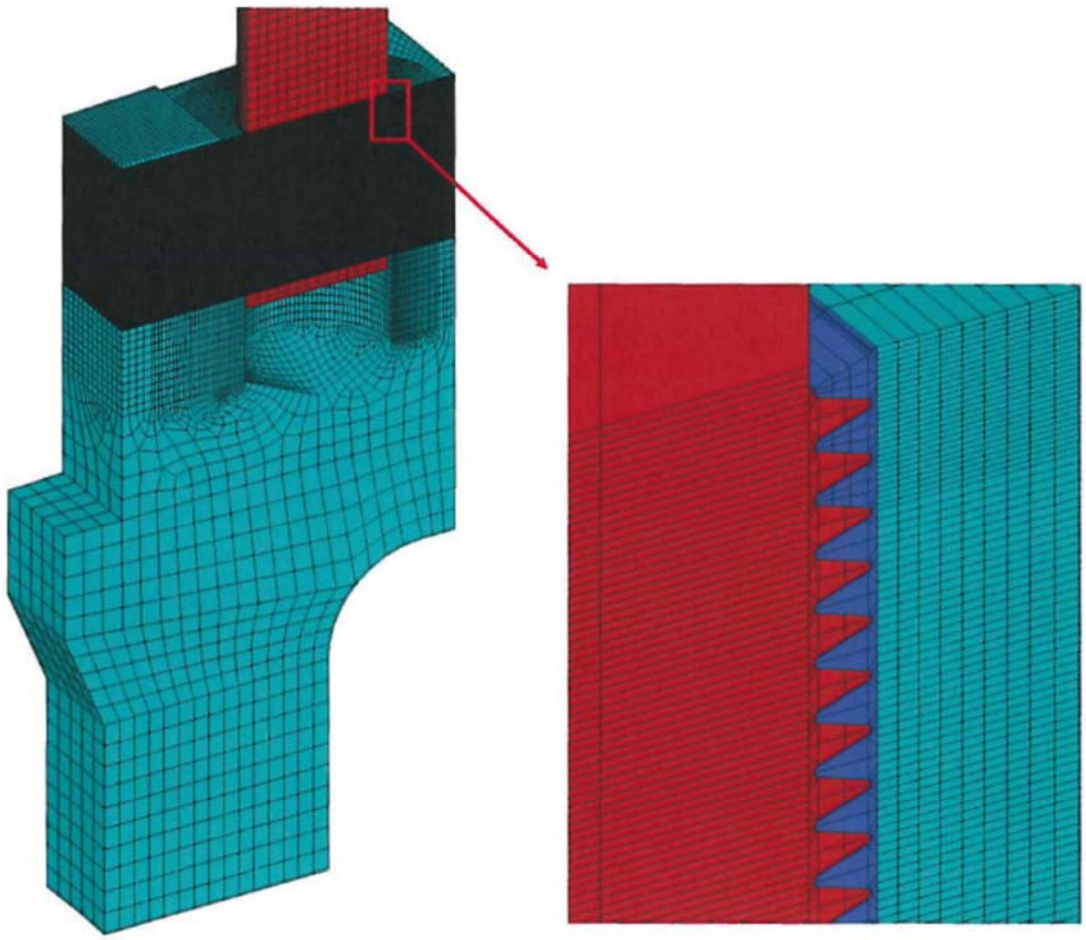

Figure – 4: Cross Section of Circumferential Flaw with Crack Tip Elements Inserted After 10th Thread from Top of Flange