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Detroit Edison

A DTE Energy Company

10CFR50.55a

August 16, 2001
NRC-01-0054

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington D C 20555-0001

Reference: 1) Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

2) Detroit Edison Letter to NRC, "Submittal of Inservice Inspection (ISI) Nondestructive Examination (NDE) Relief Request, RR-A31, Regarding Inner Radius Examination of Class 1 Reactor Vessel Nozzles," NRC-01-0047, dated June 11, 2001

Subject: Re-Submittal of Inservice Inspection (ISI) Nondestructive Examination (NDE) Request for Relief Regarding Inner Radius Examination of Class 1 Reactor Vessel Nozzles

In Reference 2, Detroit Edison submitted ISI-NDE Relief Request RR-A31 which described an alternative method to the requirements of the American Society of Mechanical Engineers (ASME) Section XI for the examination of the reactor vessel nozzle inner radius regions. The alternative examination was proposed in accordance with the provisions of 10CFR50.55a(a)(3)(i) for the remainder of the Second ISI-NDE 10-year Interval at the Fermi 2 Power Plant.

As a result of comments discussed in telephone conversations between Detroit Edison and the NRC staff on July 11, July 19 and August 2, 2001, the requested relief has been divided into two Relief Requests, RR-A31 and RR-A32. RR-A31 proposes an alternative examination in accordance with 10CFR50.55a(a)(3)(i) for nozzles where plant configuration is such that visual examination of the inner radius may be performed on essentially 100% of the inner radius. RR-A32 requests relief based on

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hardship in accordance with 10CFR50.55a(a)(3)(ii) for nozzles where visual examination of the inner radius is limited by physical obstructions.

Detroit Edison would appreciate NRC approval of the two enclosed relief requests by September 28, 2001 to allow for implementation in the upcoming eighth refueling outage, RFO8, scheduled to start on October 26, 2001.

Should you have any questions or require additional information, please contact Mr. Norman K. Peterson of my staff at (734) 586-4258.

Sincerely,

William J. O'Connor

Enclosure

cc: T. J. Kim
M. A. Ring
NRC Resident Office
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

RELIEF REQUEST
RR-A31 (REVISION 1)

COMPONENT FUNCTION/DESCRIPTION
Reactor Pressure Vessel Nozzle Inner Radius Section

SYSTEM
Reactor Vessel Head Nozzles (3)
Main Steam Nozzles (4)
CRD Return Nozzle - non-functional / capped (1)
Reactor Recirculation Loop Suction Nozzles (2)

ASME CODE CLASS
Class 1

ASME SECTION XI REQUIREMENTS
ASME Section XI, 1989 Edition, Table IWB-2500-1 for Examination Category B-D, requires a volumetric examination of the inner radius section of all RPV nozzles welded with full penetration welds as shown in Figures IWB-2500-7(a) through (d).

BASIS FOR RELIEF
Pursuant to 10 CFR 50.55a(a)(3)(i) Detroit Edison is requesting relief from ASME Section XI requirements to perform the volumetric examination described above. Detroit Edison is proposing to implement a visual examination alternative. This option is similar to the inspection alternative proposed in ASME Section XI Code Case N-648. The visual examination will cover the same inspection surface as specified for the volumetric examination. Visual examination has the additional benefit of reducing personnel radiation exposure, consistent with the Fermi ALARA program.

All nozzle forgings were nondestructively examined during fabrication and have previously been examined using inservice ultrasonic techniques specific to the nozzle configuration. No indication of fabrication defects or service related cracking has been detected by these examinations.

Nozzle inner radius examinations are the only non-welded area requiring examination on the RPV. This requirement was deterministically made early in the development of ASME Section XI, and applied to 100% of nozzles welded with full penetration welds. Fatigue cracking is the only applicable degradation mechanism for the nozzle inner radius region. For all nozzles other than Feedwater, there is no significant thermal cycling during operation. Therefore, from a risk perspective there is no need to perform volumetric examination on any nozzles other than feedwater and operational CRD returns. No service related cracking has ever been discovered in any of the BWR fleet plant nozzles other than on feedwater or operational CRD returns. The six

feedwater nozzle inner radius sections will continue to be examined with ultrasonic techniques developed and qualified in accordance with GE-NE-523-A71-0594-A, Revision 1 (the NRC has approved this report under TAC No. MA6787). The feedwater nozzles alone represent 20% of all nozzles currently requiring volumetric inner radius examination, which is more than industry accepted risk sampling requirements for similar items. Detroit Edison believes that application of a visual examination alternative for the listed nozzle inner radius regions ensures an acceptable level of quality and safety.

ALTERNATIVE EXAMINATION

Detroit Edison proposes to perform a visual examination per the requirements of the approved Fermi ISI NDE Program and ASME Section XI. Required coverage will include essentially 100% of the surface M-N as shown in Figures IWB-2500-7(a) through (d) in lieu of the volumetric examination required by Table IWB-2500-1, Examination Category B-D, Item B3.100, for inservice examination of reactor vessel nozzles listed above.

The resolution sensitivity for remote in-vessel exams will be established using a 1-mil wire standard similar to that used for other RPV internal examinations intended to detect cracking. Any direct exams, such as exams of the RPV head vent and spare nozzles, will be performed in accordance ASME Section XI VT-1 requirements.

Crack-like surface flaws exceeding the acceptance criteria of Table IWB-3512-1 are unacceptable for continued service unless the reactor vessel meets the requirements of IWB 3142.2, IWB 3142.3 or IWB-3142.4.

APPLICABLE TIME PERIOD

Relief is requested for the remainder of the Second 10-year Interval.

RELIEF REQUEST
RR-A32

COMPONENT FUNCTION/DESCRIPTION
Reactor Pressure Vessel Nozzle Inner Radius Section

SYSTEM
Core Spray Nozzles A and B (2)
Reactor Recirculation Loop Inlet Nozzles A, B, C, D, E, F, G, H, J, and K (10)
Jet Pump Instrumentation Nozzles A and B (2)

ASME CODE CLASS
Class 1

ASME SECTION XI REQUIREMENTS
ASME Section XI, 1989 Edition, Table IWB-2500-1 for Examination Category B-D, requires a volumetric examination of the inner radius section of all RPV nozzles welded with full penetration welds as shown in Figures IWB-2500-7(a) through (d).

BASIS FOR RELIEF
Pursuant to 10 CFR 50.55a(a)(3)(ii) Detroit Edison is requesting relief from ASME Section XI requirements to perform the volumetric examination described above. Performance of the volumetric examination results in significant personnel radiation exposure and does not result in a significant increase in the level of plant quality or safety. Detroit Edison is proposing to implement a visual examination of the accessible portions of the nozzle inner radius region for the specified nozzles.

Performance of the volumetric examination requires the examiner to enter and remain inside the biological shield penetration area around the nozzle (see photo) for the duration of the ultrasonic examination, which takes approximately 1 hour. Dose rates for the specified RPV nozzles are in the range of 180mr/hr to 200mr/hr, with shielding in place. Performance of these examinations results in an estimated personnel exposure of about 3 Rem per inspection interval. Performance of a visual examination using remote cameras essentially eliminates any personnel exposure.

Visual examination of the inner radius region for the subject nozzles is limited because reactor internal piping configuration prevents placement of the camera in all positions necessary to examine surface M-N over the full circumference (see Figure 1). The specific nozzle limitations and estimated coverage are as follows.

Nozzle Type/(No.)	Limitation	Estimated Coverage
Core Spray (2)	Thermal Sleeve and Sparger	40%
Recirculation Inlet (10)	Thermal-sleeve/Jet-pump riser	50%
Jet-pump instrumentation (2)	Instrumentation lines	60%

The limited visual examination does not significantly reduce the level of plant quality and safety for the following reasons:

- There are no mechanisms of damage other than fatigue for the nozzle inner radius, and for other than feedwater nozzles, there is no cause for significant thermal cycling. Therefore, the primary flaw of concern would be a flaw that was not detected during the manufacturing process[1]. All of the nozzles at Fermi were examined during and after manufacturing by surface and volumetric techniques. Additionally, preservice and inservice ultrasonic examinations have detected no flaws. It is very unlikely that flaws will be initiated by the fatigue mechanism.
- After approximately 25 years of operation (over 1000 reactor years), no cracking of any kind in the subject nozzle inner radius regions has been found.
- Fracture toughness tests performed at Oakridge National Laboratories indicate there is a large flaw tolerance for BWR nozzle inner radius regions. Even if flaw propagation was assumed, test results indicate a leak before break scenario would occur, which would not result in a significant increase in core damage frequency[2]. Additionally, pressure testing continues to be performed each refueling outage, and during plant operation the containment is monitored for changes in unidentified leakage.
- More than 50% of the total nozzle population receives a complete nozzle inner radius examination.
- Visual examination of the accessible nozzle inner radius surface (zone M-N) provides reasonable assurance that deep flaws are not present. Additionally, when flaws are initiated by the fatigue mechanism, they typically are encountered over a significant portion of the nozzle circumference as was the case for cracking of feedwater nozzles addressed in NUREG-0619.

In summary, fatigue cracking is the only applicable degradation mechanism for the nozzle inner radius region, and for all nozzles other than Feedwater there is no significant thermal cycling during operation. Therefore, from a risk perspective, there is no need to perform volumetric examination on any nozzles other than feedwater and operational CRD returns. This is supported by the fact that no service related cracking has ever been discovered in any of the BWR fleet plant nozzles other than on feedwater or operational CRD returns. The six feedwater nozzle inner radius sections will continue to be examined with ultrasonic techniques developed and qualified in accordance with GE-NE-523-A71-0594-A, Revision 1 (the NRC has approved this report under TAC No. MA6787). The feedwater nozzles alone represent 20% of all nozzles currently requiring volumetric inner radius examination, which is more than industry accepted risk sampling requirements for similar items. Additionally, Relief Request RR-A31 provides for a full visual examination of 10 additional nozzles, resulting in a complete examination of more than 50% of the total nozzle population. Detroit Edison believes that the partial visual examination alternative for the nozzle inner radius regions above results in a significant reduction in personnel dose and still ensures an acceptable level of quality and safety.

1 & 2: Conclusions made in ASME NDE subcommittee report ISI-99-26 , "Technical Basis for Elimination of Reactor Vessel Nozzle Inner Radius Inspections."

ALTERNATIVE EXAMINATION

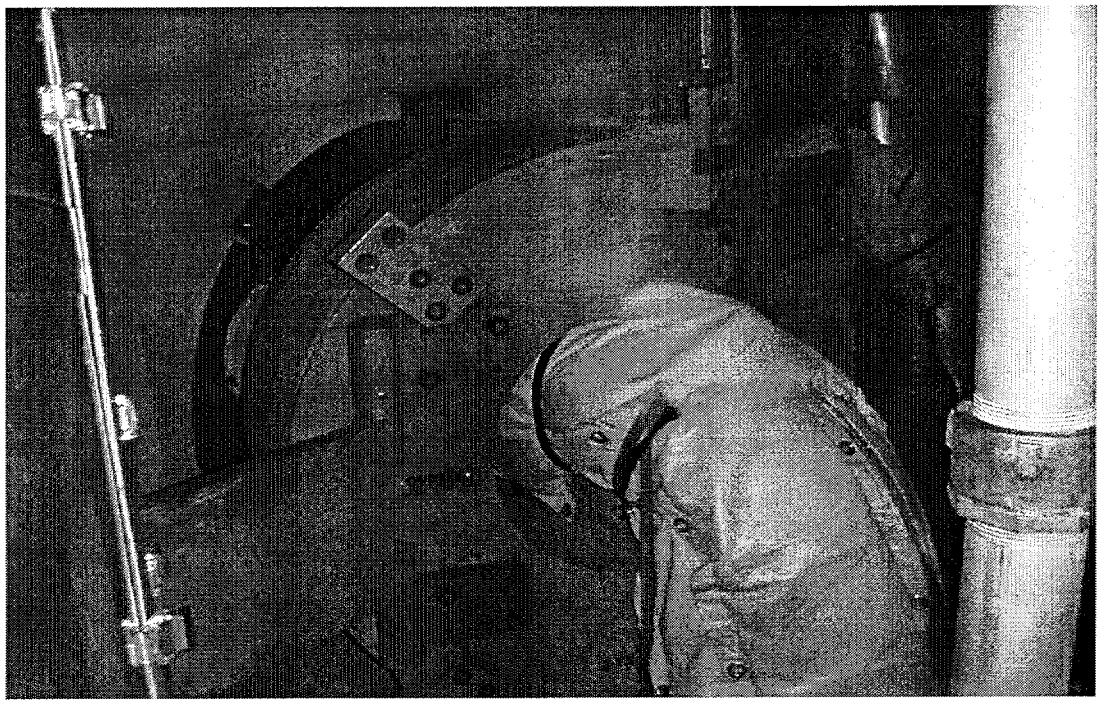
Detroit Edison proposes to perform a visual examination of the accessible surface M-N as shown in Figures IWB-2500-7(a) through (d) in lieu of the volumetric examination required by Table IWB-2500-1, Examination Category B-D, Item B3.100, for inservice examination of reactor vessel nozzles identified in this relief request.

The resolution sensitivity for remote in-vessel exams will be established using a 1-mil wire standard similar to that used for other RPV internal examinations intended to detect cracking.

Crack-like surface flaws exceeding the acceptance criteria of Table IWB-3512-1 are unacceptable for continued service unless the reactor vessel meets the requirements of IWB 3142.2, IWB 3142.3 or IWB-3142.4.

APPLICABLE TIME PERIOD

Relief is requested for the remainder of the Second 10-year Interval.



Typical Biological Shield Opening

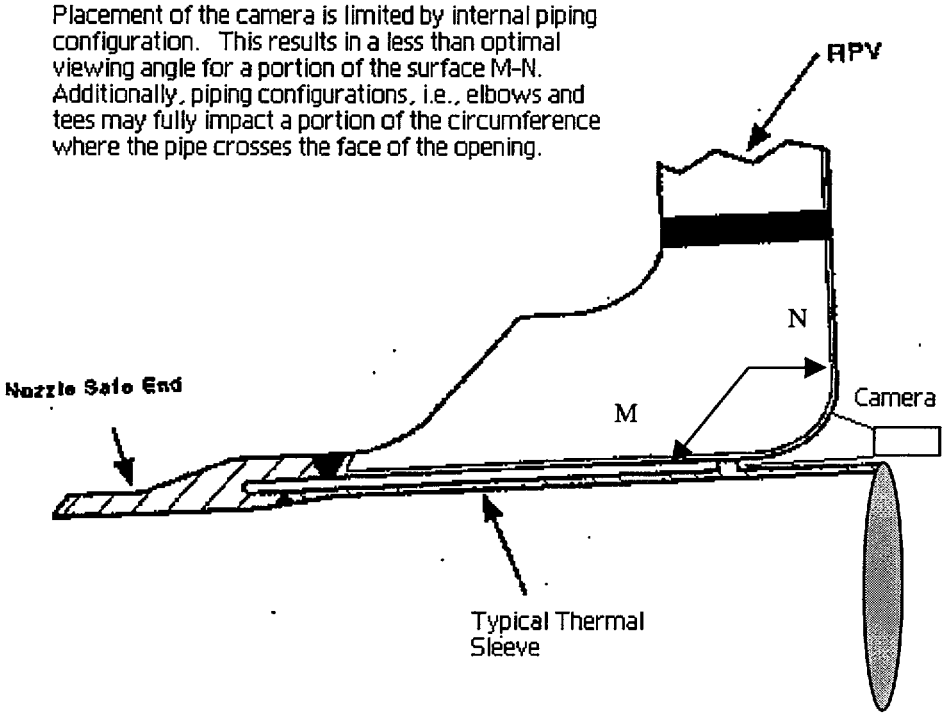


Figure 1
Typical Cross Section of BWR Nozzle with Internal Piping