March 30, 2000

Mr. Douglas R. Gipson
Senior Vice President
Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI 2 - RELIEF REQUESTS FOR THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION (ISI) NONDESTRUCTIVE EXAMINATION (NDE) PROGRAM (TAC NO. MA6391)

Dear Mr. Gipson:

By letter dated August 19, 1999, as supplemented February 4 and March 15, 2000, the Detroit Edison Company (the licensee) requested relief from certain ISI requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, for the Fermi 2 plant. The requests for relief are related to inspections under the ISI NDE program for the second 10-year interval.

The staff has completed its review of the requests for relief and its Safety Evaluation is enclosed. The staff finds the licensee's proposed alternatives to the Code requirements in Requests for Relief Nos. RR-A18, RR-A26, RR-A28, RR-A29, and RR-C3 provide an acceptable level of quality and safety. Therefore, the proposed alternatives documented in RR-A18, RR-A26, RR-A28, RR-A29, and RR-C3 are authorized pursuant to 10 CFR 50.55a(a)(3)(i). The staff finds that compliance with the specified requirement in Requests for Relief Nos. RR-A19 and RR-C4 would result in hardship without a compensating increase in the level of quality and safety. Therefore, the proposed alternatives in RR-A19 and RR-C4 are authorized pursuant to 10 CFR 50.55a(a)(3)(ii). For Requests for Relief Nos. RR-A1, RR-A6 and RR-A23 the staff concludes that the Code requirements are impractical for the subject welds and that granting relief will not endanger life, property, or the common defense and security and is otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i). The staff finds that relief is unnecessary for the alternative in RR-A30.

The staff reviewed the licensee's Request for Relief No. RR-A25 and concluded that the alternative proposal provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), the inspection of the circumferential welds may be permanently deferred for the remaining term of operation under the existing, initial operating license.

Your staff pointed out a typographical error in the safety evaluation dated February 17, 2000, for the second interval inservice testing program. On page 11 of the safety evaluation, under refueling outage justification (ROJ)-016, valve E1100F046A was listed twice and E1100F046B was not listed. As discussed with your staff on March 14, 2000, the second E1100F046A should have been E1100F046B. The valves were listed correctly in ROJ-016 in your submittal dated August 19, 1999.

Sincerely,

/RA by T.J. Kim Acting For/

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUESTS FOR RELIEF FOR THE SECO0ND 10-YEAR INTERVAL INSERVICE

INSPECTION NONDESTRUCTIVE EXAMINATION PROGRAM

DETROIT EDISON COMPANY

FERMI 2

DOCKET NO. 50-341

1.0 INTRODUCTION

The inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (the Code) and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The regulation at 10 CFR 50.55a(a)(3) states, in part, that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. For Fermi 2, the applicable edition of Section XI of the ASME Code for the second 10-year inservice inspection interval is the 1989 Edition.

Pursuant to 10 CFR 50.55a(g)(5)(iii), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility, information shall be submitted to the Commission in support of that determination and a request made for relief from the ASME Code requirement. After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and may impose requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed.

By letter dated August 19, 1999, as supplemented February 4 and March 15, 2000, the Detroit Edison Company (the licensee) requested relief from certain ISI requirements of the ASME Code.

2.0 EVALUATION

The information provided by the licensee in support of the requests for relief from Code requirements has been evaluated and the basis for disposition is documented below. Section 2.1 discusses requests for relief related to snubbers. Section 2.2 discusses requests for relief related to inspection of piping and welds.

2.1 Snubbers

2.1.1 Relief Requested

In its August 19, 1999, letter, Detroit Edison requested relief from ASME Section XI requirements to perform examinations and tests of snubbers in accordance with the 1988 Addenda to Part 4 of ASME/ANSI OM-1987 (OM-4), for the second 10-year ISI interval (request for relief RR-C3). OM-4 is referenced by ASME Code Section XI, Article IWF-5000. The licensee requested the use of the technical requirements provided in Section 5.1 of the Fermi 2 Technical Requirements Manual (TRM), instead of ASME Code Section XI, for the required snubber visual examination and functional testing, pursuant to 10 CFR 50.55a(a)(3)(i).

By the August 19 letter, the licensee also requested that the staff authorize the use of ASME Code Case N-508-1, “Rotation of Serviced Snubbers and Pressure Relief Valves for the Purpose of Testing,” for Fermi 2, as an alternative to the ASME Code, Section XI, Article IWA-7000, for snubber replacement (request for relief RR-C4).

2.1.2 Basis of Requests

In its August 19, 1999, letter, the licensee stated that it had imposed examination and testing requirements in accordance with Technical Specification (TS) Surveillance Requirement 4.7.5 for all safety-related snubbers, including ASME Classes 1, 2, and 3. Functional testing provides a 95 percent confidence level that at least 90 percent of the snubbers operate within the specified acceptance limits. The visual examination is a separate process that complements the functional testing program and provides additional confidence in snubber operability. Visual examination requirements are based on NRC Generic Letter (GL) 90-09, “Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions,” dated December 11, 1990.

With implementation of the Fermi 2 improved Technical Specifications (TS), the snubber surveillance requirements are relocated in their entirety to the Fermi 2 TRM (TRM Section 5.1.1, “Augmented Inservice Inspection Program for Snubbers”).

Snubbers require periodic testing. To reduce system out-of-service time, testing is often accomplished by removing an existing snubber from service, installing a replacement, then testing the removed snubber at a later time. The current ASME rules stipulate that each snubber rotation comply with Section XI, Article-7000 requirements. The IWA-7000 requirements impose extensive administrative and documentation controls. IWA-7250(a)(8) requires that replacements be documented on the Owner's Report for Repairs and Replacements, Form NIS-2. Therefore, even if an operable snubber was replaced with a rebuilt item because the original one was nearing the end of its service life, the subject activity would be documented on an NIS-2 form. Code Case N-508-1, item (g), on the other hand, does not require the use of the NIS-2 form unless the replacement was required due to the original snubber being deficient or inoperable.

2.1.3 Evaluation

The licensee stated in its August 19, 1999, letter that, in lieu of using OM-4 (which is referenced by Article IWF-5000), the alternative examination and testing program, in accordance with TRM 5.1.1 requirements (formerly TS 4.7.5, which incorporates the visual examination schedule of GL 90-09), are designed to demonstrate the functional integrity of the snubbers and are, at least, equivalent to the requirements of Article IWF-5000. These alternative requirements were previously reviewed and approved by the staff in the Fermi 2 TS. The staff finds the alternative examination and testing program to be acceptable.

Currently, the ASME Code, Section XI, Article IWA-7000, requires that snubber rotation be performed in compliance with a repair/replacement program. The program requires the preparation of a replacement plan, completion and submittal of a Code Form NIS-2, and an evaluation, review and concurrence by an authorized nuclear inspector. Code Case N-508-1 provides an alternative to the ASME Code, Section XI, Article IWA-7000 requirement to generate a replacement program when removing snubbers from a system for testing. The Code Case allows snubbers to be rotated from stock and installed on components and piping systems within the Section XI boundary, provided all the requirements stated in the Code Case are met. For normal rotation of operable snubbers with those items from stock, therefore, it is the Owner's (i.e., the licensee's) responsibility to maintain traceability of the affected snubbers. But no Code-required documentation (i.e., NIS-2 Forms) is required.

The licensee's proposed alternative to use the Code Case for the purpose of snubber rotation associated with testing will eliminate unnecessary administrative and documentation requirements, minimize the time during which the affected system is out of service, and conserve resources. The staff finds that the same level of quality and safety is maintained when component rotation and testing is performed in accordance with IWA-7000 or Code Case N-508-1. The staff concludes that the application of the Code requirement to generate a replacement program when removing snubbers from a system for testing would be a hardship without a compensating increase in the level of quality and safety.

2.1.4 Conclusion

Based on the information provided by the licensee, the staff determined that the licensee has presented an adequate justification for relief from the requirements of OM-1988 Addenda to the OM-1987 Edition, Part 4 (which is referenced by ASME Code 1989 Edition, Section XI, Article IWF-5000), with regard to visual examination and functional testing of Fermi 2 snubbers. The staff has determined that the proposed alternative use of TRM 5.1.1 for Fermi 2 snubber activities will provide an acceptable level of quality and safety. Therefore, pursuant to

10 CFR 50.55a(a)(3)(i), the licensee's request for relief for the second 10-year interval of the Fermi 2 ISI program is authorized.

The staff also concludes that the licensee's proposed alternative use of Code Case N-508-1 for rotation of serviced snubbers for the purpose of testing in lieu of ASME Code, Section XI, Article IWA-7000 requirements may be authorized, pursuant to 10 CFR 50.55a(a)(3)(ii). This is based on the determination that the alternative provides reasonable assurance of operational readiness and that compliance with the Code requirements would result in hardship without a compensating increase in the level of quality and safety.

2.2 Piping and Welds

2.2.1 Request for Relief No. RR-A1 Examination Category B-A, Item B1.21, Circumferential Head Welds, Item B1.22, Meridional Head Welds, and Item B1.30, Shell to Flange Welds

2.2.1.1 ASME Section XI Requirements

Subsection IWB, Table IWB 2500-1, Examination Category B-A, Item Nos. B1.10 through B1.40, require volumetric examination of reactor pressure vessel (RPV) weld and base material regions described in figures IWB-2500-1 through 2500-3 for pressure retaining welds in the reactor pressure vessel each inspection interval.

2.2.1.2 Licensee's Code Relief Request

In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100 percent volumetric examination coverage for the welds listed below.

WELD	ITEM	DESCRIPTION	COVERAGE	LIMITATION
5-306	B1.21	Head circ. weld	0%	Bottom head CRD [control rod drive] penetrations and skirt attachment weld (dollar plate)
1-319A	B1.22	Head merid. weld	~73.6%	Top head lifting lugs
1-319C	B1.22	Head merid. weld	~70%	Top head lifting lugs
1-319E	B1.22	Head merid. weld	~72%	Top head lifting lugs
1-319G	B1.22	Head merid. weld	~71.3%	Top head lifting lugs
2-306A	B1.22	Head merid. weld	0%	Bottom head CRD penetrations and skirt attachment weld
2-306B	B1.22	Head merid. weld	0%	Bottom head CRD penetrations and skirt attachment weld
2-306C	B1.22	Head merid. weld	0%	Bottom head CRD penetrations and skirt attachment weld

2-306D	B1.22	Head merid. weld	0%	Bottom head CRD penetrations and skirt attachment weld
2-306E	B1.22	Head merid. weld	0%	Bottom head CRD penetrations and skirt attachment weld
2-306F	B1.22	Head merid. weld	0%	Bottom head CRD penetrations and skirt attachment weld
2-306G	B1.22	Head merid. weld	0%	Bottom head CRD penetrations and skirt attachment weld
13-308	B1.30	Shell to flange	54%	RPV flange configuration (coverage augmented by scan from flange seal surface)

2.2.1.3 Licensee's Basis for Requesting Relief (as stated):

Pursuant to 10 CFR 50.55a(g)(5)(iii) Detroit Edison is requesting relief from ASME Section XI requirements to examine essentially 100% of accessible Category B-A weld lengths because within the limits of RPV design it is impractical to do so. Detroit Edison believes that the alternatives specified provide an acceptable level of quality and safety.

Relief Request RR-A1 documented limitations based on both the installed ultrasonic examination system, which used pole tracks for scanning, and part geometry. During RF02 Fermi implemented the use of an automated examination system that uses a magnetic wheel scanning device which maximizes coverage to the extent possible using current technology. Limitations to automated scanning of RPV shell welds due to the examination system have been eliminated. Current limitations are based only on RPV configuration or interference from other components as described in the "Alternatives" section below.

Reactor Vessel Ultrasonic Examination techniques meet the requirements of ASME Section XI; ASME Section V, Article 4; and Regulatory Guide 1.150. Detroit Edison believes that the alternative examinations proposed satisfy the intent of the ASME Code within the limits of accessibility for examination inherent to the BWR [boiling water reactor] design. Table I [see licensee's submittal for this relief request dated August 19, 1999] identifies the welds with limitations and the cause of the limitation (see also attached figures) [in the licensee's submittal for this relief request dated August 19, 1999]. The extent of examination is reported in accordance with ASME Section V.

<u>ALTERNATIVES</u>:
Welds 1-319A, 1-319C, 1-319E, & 1-319G

The four listed top head weld exams were examined for most of the weld length during the first interval. They are limited because of a lifting lug positioned on each weld. Because of the physical access limitations it is impractical to examine the full volume of these welds for their entire length.

The Fermi proposed alternative for the ASME Code exam performance is partial examination for these welds. For the weld volume that is partially scanned, the ultrasonic examination covers the most critical area at the inside surface of the head. The areas of highest stress on the outside surface in the area of the limitation (lifting lug attachment welds) receive a surface examination per Category B-H. The alternative of partial examination combined with the surface exam yields similar results to a full examination.

Because of the substantial coverage obtained by the partial ultrasonic examination and the surface examination of the interfering lug/welds, along with the low empirical probability of reactor vessel weld failure, Detroit Edison considers the proposed alternative examination to provide an acceptable level of quality and safety.

Inaccessible Bottom Head Welds
Welds 5-306 and 2-306A through 2-306G
The access restrictions caused by the CRD penetrations and RPV support skirt make it impractical to perform a meaningful ultrasonic examination of these welds with current technology. For the inaccessible RPV bottom head welds, the proposed alternatives include a combination ASME Section XI Code required leakage inspections and monitoring of drywell leakage during operation.

Reasonable assurance of structural integrity is maintained because the welds received volumetric and surface NDE [nondestructive examination] to verify that no deleterious material or processing defects were present at the time of fabrication. The welds are physically located at the bottom of the reactor vessel, below the withdrawn control rod blades. There is also more than 170 inches of water from the bottom of the active fuel height to the weld location. This physical arrangement reduces the neutron fluence and the coincident material degrading impacts significantly, when compared to RPV beltline welds that are inspectable. The same CRD penetrations that prevent the examination of the welds would also serve to prevent rapid propagation of a large defect by providing a crack arrest point.

Because of the visual inspections (VT-2) and leakage monitoring performed, physical access limitations, reasonable assurance of structural integrity for these welds, and the low empirical probability of reactor vessel weld failure, Detroit Edison considers the proposed alternative to provide an acceptable level of quality and safety.

Weld 13-308
The RPV shell to flange weld exam is limited due to vessel flange configuration, which makes it impractical to examine the full volume of the weld. The Code allowed alternative exam of ASME Section V, Article 4, T441.5.1 (Longitudinal exam from the flange) was performed during RF06 but this exam was also limited because of the RPV stud holes. Even with this Code allowed alternative, it is not possible to obtain full volume coverage even when scanning is performed from both sides of the weld for 360 degrees.

The Fermi proposed alternative is a partial exam from the shell side combined with the longitudinal wave exam from the flange surface. As shown in Figure 3 [see licensee's submittal dated August 19, 1999], the proposed alternative partial exam performed from the shell side provides significant coverage of the ID surface where flaws would be most likely to originate. A significant portion of full weld volume is also covered by the longitudinal exam from the flange surface. Based on physical limitations, the coverage achievable by the alternative examinations, and the low empirical probability of reactor vessel weld failure, Detroit Edison considers the proposed alternative examination to provide an acceptable level of quality and safety.

2.2.1.4 Staff Evaluation and Conclusion

Examination Category B-A, Item Nos. B1.21 and B1.22 require 100 percent volumetric examination of the accessible length of all RPV circumferential and meridional head welds. Complete examination coverage of the subject B1.21 and B1.22 welds is restricted by physical obstructions including control rod drives, vessel support skirt attachments and top head lifting lugs. Examination Category B-A, item number B1.30 requires 100 percent volumetric examination of the RPV flange-to-shell weld. Complete examination coverage is restricted by the flange geometry (flange radius and stud holes interfere with scanning from the flange surface). These conditions make 100 percent volumetric examination impractical for the subject weld. To gain additional access for examination of the subject welds, the RPV would require design modifications. Imposition of this requirement would impose a significant burden on the licensee.

The licensee has examined these welds to the extent practical; examination volumes achieved range from 0 - 73.6 percent of each weld. The subject welds are outside of the highly irradiated core belt-line region of the RPV. In addition, other RPV shell welds have been examined to the extent required by the Code. Therefore, based upon the volumetric coverage obtained on the accessible portion of some of the subject welds, volumetric examinations that meet Code requirements on other RPV welds, and VT-2 visual examinations that are performed in conjunction with the pressure testing each refueling outage, the staff concludes that existing patterns of degradation, if present, would have been detected and reasonable assurance of the structural integrity of the subject welds has been provided. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the licensee's second 10-year inservice inspection interval subject to the alternative proposed by the licensee.

2.2.2 Request for Relief No. RR-A6 Examination Category B-D, Full Penetration Welds of Nozzles in Vessels

2.2.2.1 ASME Section XI Requirements

Subsection IWB, Table IWB 2500-1, Examination Category B-D, Item Nos. B3.90 and B3.100 require volumetric examination of RPV nozzle-to-shell welds and base material regions as shown in figure 2500-7(b).

2.2.2.2 Licensee's Code Relief Request

In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100 percent volumetric examination of the reactor vessel nozzle welds listed below.

WELD	ITEM	DESCRIPTION	COVERAGE	LIMITATION
8-316A-D	B3.90	Nozzle weld	69.1%	Nozzle blend radius
4-316A, D	B3.90	Nozzle weld	~60%	Nozzle blend radius and instrumentation nozzles
4-316B,C,E,F	B3.90	Nozzle weld	64.1%	Nozzle blend radius
14-316A,B	B3.90	Nozzle weld	68.9%	Nozzle blend radius
15-315	B3.90	Nozzle weld	68%	Nozzle blend radius
13-314A-K	B3.90	Nozzle weld	66.7%	Nozzle blend radius
5-314A,B	B3.90	Nozzle weld	65.6%	Nozzle blend radius and bottom head to shell taper
19-314A,B	B3.90	Nozzle weld	63.1%	Nozzle blend radius and bottom head to shell taper
2-318	B3.90	Nozzle weld	61.4%	Nozzle blend radius
4-318A,B	B3.90	Nozzle weld	62%	Nozzle blend radius
19-314A,B	B3.100	Nozzle weld inner radius	80.2%	Bottom head to shell taper

2.2.2.3 Licensee's Basis for Relief (as stated)

> Pursuant to 10 CFR 50.55a(g)(5)(iii) Detroit Edison is requesting relief from ASME Section XI requirements to examine essentially 100% of accessible Category B-D nozzle welds, because within the limits of design and accessibility it is impractical to do so.

Relief Request RR-A6 only documented ultrasonic examination limitations based on interference caused by proximity to other nozzles. Other limitations have been identified during the performance of examinations during the first interval.

The primary limitation to full ASME Code volumetric coverage is nozzle configuration. The nozzle type used in the Fermi 2 reactor is a flanged nozzle as shown in Figure 1 [in licensee's submittal dated August 19, 1999]. This type of nozzle provides the best access for inspection of the nozzle types permitted in the ASME Code as shown in the Figures of IWB-2500-7. The Code required volume ($t_S/2$) extends into the nozzle outside blend radius. The curve of the radius section hinders the ability of transducers to maintain contact with the nozzle and also changes the effective beam angle. This limitation results in a typical maximum composite coverage of all Code required scans (0 {Longitudinal}, 45, and 60 {Parallel & Transverse} degree) between 60% and 70% depending on nozzle diameter and thickness. The maximum obtainable coverage is achieved by the 60-degree transverse (T) scan. Essentially all of the weld and heat affected zones are covered by this angle beam scan for the entire weld circumference on most nozzles. Typical scan limitations are shown in Figures 2A through 2C [in licensee's submittal dated August 19, 1999]. The estimated volumetric coverage obtained is reported in Table 1 [see table in Licensee's Code Relief Request above].

Another limitation to full ASME Code volumetric coverage is the vessel taper at the bottom head to lower shell course weld. This geometric condition prevents full coverage of the bottom side of the two jet pump instrumentation nozzles and the two recirculation suction nozzles. Composite coverage for these welds remains above 60%. This limitation also impacts the nozzle inner radius coverage for the two core spray nozzles as reported in Table 1 [in licensee's submittal dated August 19, 1999].

The limitation originally described in RR-A6 of this relief request indicated a limitation of 46 degrees or 12.8% of the full circumference for 2 of 6 feedwater nozzles based on automated examination equipment accessibility. The examinations were performed manually and the limitation was less than originally described and accepted (see Figure 3) [in licensee's submittal dated August 19, 1999]. A part of the scan path was able to be performed for the full circumference. Additionally, Fermi examines these feedwater nozzles as specified in NUREG 0619 to detect cracking in the nozzle inner radius and bore areas where cracks have previously been detected in other BWRs. These exams were fully completed and no service related flaws have been detected.

All nozzle forgings received ultrasonic examination during manufacture and the nozzle to shell welds were subject to radiographic examination during fabrication of the reactor pressure vessel. All of the nozzle welds requiring volumetric examination by ASME Section XI have been completed during the first ten-year inspection interval and no service related defects have been detected. The nozzle inner radius ultrasonic examination techniques used at Fermi performed scanning from the blend radius; however, since this technique was designed to detect internal surface defects no credit has been taken for those exams.

Reactor Vessel Ultrasonic Examination techniques meet the requirements of ASME Section XI; ASME Section V, Article 4; and Regulatory Guide 1.150. Detroit Edison believes that the extent of examinations completed satisfy the intent of the ASME Code and 10 CFR 50.55a(g)(4) within the limits of accessibility for examination inherent to BWR pressure vessel design.

2.2.2.4 Licensee's Proposed Alternative (as stated)

Perform examination of the ASME Code volume to the extent practical.

2.2.2.5 Staff Evaluation and Conclusion

The Code requires 100 percent volumetric examination of the subject RPV nozzle-to-vessel welds and inside radius sections. However, complete examination is restricted by the proximity to other nozzles, outside blend radius of nozzles and the vessel taper at the bottom head to lower shell course weld. These limitations make the 100 percent volumetric examination impractical. To gain access for examination, the RPV nozzles would require design modifications. Imposition of this requirement would create an undue burden on the licensee.

The licensee has examined a significant portion of these welds, obtaining 60 - 69 percent coverage for each of the nozzle-to-vessel welds and 89 percent coverage for the subject nozzle inside radius section. Based on the coverages obtained the staff concludes that any existing patterns of degradation would have been detected by the examinations that were completed and reasonable assurance of the structural integrity has been provided.

Based on the impracticality of meeting the Code coverage requirements for the subject nozzle-to-vessel welds and inside radius sections, and the reasonable assurance provided by the examinations that were completed on these and other Class 1 nozzles, the staff grants relief pursuant to 10 CFR 50.55a(g)(6)(i).

2.2.3 Request for Relief No. RR-A18, Use of Code Case N-546, Alternative Requirements for Qualification of VT-2 Examination Personnel

2.2.3.1 Components for which Relief is Requested

Class 1, 2, and 3 Pressure Retaining Piping and Components

2.2.3.2 ASME Section XI Requirements

ASME Section XI, 1989 Edition, Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1 require the performance of a VT-2 examination during the specified pressure tests. IWA-2300 requires that personnel performing the VT-2 examinations be qualified by the owner or the owner's agent in accordance with the owner's qualification program having levels of competency comparable to SNT-TC-1A as defined in ANSI N45.2.6.

2.2.3.3 Licensee's Proposed Alternative (as stated)

Code Case N-546 provides the following alternative qualification rules for personnel such as licensed and nonlicensed operators, local leak rate personnel, system engineers, and inspection and nondestructive examination personnel.

(a) The individual must have at least 40 hours plant walkdown experience such as that gained by licensed and nonlicensed operators, local leak rate personnel, system engineers, and inspection and nondestructive examination personnel.
(b) At least 4 hours of training on Section XI requirements and plant specific procedures for VT-2 visual examination will be completed.
(c) Vision test requirements of IWA-2231 (1995 Edition) will be satisfied.

In addition, the following actions will ensure consistent quality in the performance of examinations.

1. Records of the training and qualifications specified in Code Case N-546 will be provided and maintained in accordance with the Fermi written practice.
2. Visual examination will be conducted in accordance with specific written procedures.
3. Visual examination procedures will provide for a documented independent review and evaluation of test results.

2.2.3.4 Licensee's Basis for Relief (as stated)

Pursuant to 10 CFR 50.55a(a)(3)(i) Detroit Edison is requesting relief from ASME Section XI requirements to certify VT-2 examiners in accordance with IWA-2300. Detroit Edison is proposing to use the alternatives specified in ASME Code Case N-546 (copy attached). This will eliminate the need to qualify VT-2 examination personnel in the same manner as NDE personnel. VT-2 requires no special knowledge of technical principles; it is simply an inspection for evidence of leakage. No special skills or technical training are required in order to observe water dripping from a component or bubbles forming on a surface wetted with a leak detection solution. Therefore, qualification in accordance with the provisions of the Code Case will not present any reduction in quality or safety. In fact, it will facilitate the qualification of those personnel most familiar with the walkdown of plant systems.

The Code Case is ASME approved indicating the ASME Code Committee members reached a consensus that the alternative provides essentially equivalent results to the requirements of IWA-2300. Detroit Edison agrees with the Code Committee that use of the alternative described in this Code Case will provide an acceptable level of quality and safety.

2.2.3.5 Staff Evaluation and Conclusion

The Code requires that VT-2 visual examination personnel be qualified and certified in accordance with SNT-TC-1A. The Code also requires that the examination personnel be qualified for near and far distance vision acuity. In lieu of the Code requirements, the licensee proposed to implement Code Case N-546 for personnel performing VT-2 visual examinations. This Code Case includes the following requirements:

1. At least 40 hours plant walkdown experience, such as gained by licensed and nonlicensed operators, local leak rate personnel, system engineers, and inspection and nondestructive examination personnel.

2. At least 4 hours of training on Section XI requirements and plant-specific procedures for VT-2 visual examination.

3. Vision test requirements of IWA-2321, 1995 Edition.

The qualification requirements in Code Case N-546 are not significantly different from those for VT-2 visual examiner certification. Licensed and nonlicensed operators, local leak rate personnel, system engineers, and inspection and nondestructive examination personnel typically have a sound working knowledge of plant components and piping layouts. This knowledge makes them acceptable candidates for performing VT-2 visual examinations.

The NRC staff has determined that in order to find this Code Case acceptable for use, the licensee must meet the following provisions:

A. Develop procedural guidelines for obtaining consistent quality VT-2 visual examinations,

B. Document and maintain records to verify the qualification of persons selected to perform VT-2 visual examinations, and

C. Implement independent review and evaluation of detected leakage by persons other than those that performed the VT-2 visual examinations.

The staff has noted that the licensee has made provisions to meet the above conditions. Based on a review of Code Case N-546 the staff believes that the proposed alternative to the Code qualification requirements in Code Case N-546 in conjunction with the licensee's added provisions will provide an acceptable level of quality and safety. Therefore, the licensee's proposed alternative to use Code Case N-546 with the additional provisions stated therein, is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year interval at Fermi 2.

2.2.4 Request for Relief No. RR-A19, IWA-5213, Test Condition Holding Time

2.2.4.1 The Components for Which Relief is Requested

Insulated portions of high pressure coolant injection (HPCI) turbine/exhaust lines, and associated vents and drains.

2.2.4.2 ASME Section XI Requirements

ASME Section XI, 1989 Edition, IWA-5213(d) (Test Condition Holding Time) and Code Case N-498-1 (Alternative Rules for 10-year Hydrostatic Pressure Testing), which is included in the Fermi Inservice Inspection Program, requires a 4-hour hold time after attaining nominal operating pressure conditions for insulated systems.

2.2.4.3 Licensee's Proposed Alternative (as stated)

> The system pressure test described in Code Case N-498-1 will be conducted as required, except that a 20 minute hold time will be used in lieu of the 4 hour hold time requirement. The 20 minute hold time will allow time for abnormal leaks to migrate through the insulation without challenging the Technical Specification limitation on maximum torus water temperature. Any evidence of abnormal leakage will be investigated by locally removing insulation. A similar alternative for test performance was approved at another nuclear utilities (e.g., Hope Creek). Reasonable assurance of system structural integrity is maintained through implementation of the alternative test and by the extent and frequency of other Technical Specification/ASME required system operability tests.

2.2.4.4 Licensee's Basis for Relief (as stated)

> Pursuant to 10 CFR 50.55a(a)(3)(ii) Detroit Edison is requesting relief from ASME Section XI requirements to maintain a 4-hour hold time prior to the visual examination for the pressure test described in this relief request. Fermi proposes to perform the test using an alternative hold time of 20 minutes. This alternative is necessary because the 4-hour hold time could result in system conditions outside of Technical Specification operating limits.
>
> As part of the Emergency Core Cooling System (ECCS), the HPCI system is not required to operate during normal plant operation. However, the system is periodically tested in accordance with applicable inservice testing and the Technical Specification requirements. These periodic tests are conducted to verify the operability of system components. The quarterly operability test (24.202.001) normally includes about 30 minutes of pump run time. In order to satisfy ASME Section XI hold time requirement, the test would require a HPCI pump run for greater than 4 hours (hold time plus exam time). Running the HPCI pump for this duration is not practical and represents an undue hardship on the facility without a compensating increase in the level of quality and safety.
>
> Operating the HPCI pump for this amount of time would subject the facility to excessive heat loads. Control of these heat loads would require the operation of additional ECCS subsystems to remove heat from the suppression pool.
>
> Extended operation of the HPCI pump would also challenge the Technical Specification limitation on maximum suppression pool (torus) water temperature. The Fermi Technical Specifications require the torus average water temperature to be maintained less than 105°F during testing which adds heat to the torus. Operating the HPCI pump for a period substantially longer than the system operability test could cause this temperature to be exceeded. If the torus average water temperature exceeds 110°, Technical Specifications require the reactor mode switch to be placed in the shutdown position.

Removal of the insulation from the subject components in order to use the ten minute hold time allowed by the Code or Code Case N-498-1, would be equally burdensome. The impacts associated with insulation removal and reinstallation, include personnel radiation exposure, radwaste generation, and limited manpower resources are not justified by a compensating increase in the level of quality and safety.

Performing a HPCI system hydrostatic test per IWA 5213 (d) would also be burdensome. A hydrostatic test would require installation of blank flanges and temporary pipe supports, and gagging or removal of relief valves. System out of service time, and radiation exposure incurred in carrying out a hydrostatic test would result in a hardship without a compensating increase in the level of quality and safety.

Other inspection and test activities performed that serve to verify continued system integrity include the following:

Quarterly inservice testing of HPCI raises the pressure of the system to nominal operating conditions. Any leakage would migrate through the insulation over a period of time and would become evident.

Nondestructive examination of circumferential welds per Section XI Table IWB-2500-1, Category C-F-2. The weld selections on this line were random selections because none of the welds exceeded the moderate or high stress criteria.

Every 18 months this line is inspected in accordance with the Fermi Leakage Reduction Program per Technical Specification requirements.

2.2.4.5 <u>Staff Evaluation and Conclusion</u>

For system pressure tests of insulated components, Article IWA-5213(d) of the Code requires a 4-hour hold time at operating pressure and temperature prior to performing the VT-2 visual examination. However, maintaining the Code-required test conditions for 4 hours for the HPCI system would result in excessive heat loads on the suppression pool. By comparison, the hold time for non-insulated components is 10 minutes following pressurization to test conditions prior to performing the visual examination. To overcome the problem of high heat loads to the suppression pool, removal of insulation is a possible alternative. However, this approach would result in excess radiation exposure to plant personnel and create additional radioactive waste. The staff believes that the imposition of the Code-required 4-hour hold time is a hardship to the licensee for the pressure test of the HPCI system at Fermi 2.

As an alternative, the licensee has proposed a visual examination that will be performed following a 20-minute hold time. This hold time will maintain the suppression pool temperature within the acceptable limit while allowing discovery of gross reactor coolant leakage. In addition, this system receives inservice testing on a quarterly basis, and 25 percent of the welds receive surface examinations in accordance with the Code. The licensee's proposed alternative, in conjunction with inservice testing and the Code-required surface examinations should detect any significant patterns of degradation and will provide reasonable assurance of the continued operational readiness of the HPCI system. Therefore, the staff concludes that the Code-required hold time requirements would result in a burden without a compensating increase in the level of quality and safety at Fermi 2. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second 10-year inspection interval.

2.2.5 Request for Relief No. RR-A23, Examination Category B-J, Item 9.11, Pressure Retaining Welds in Piping

2.2.5.1 Components for Which Relief is Requested

Pressure Retaining Piping Welds in the following ASME Code Class 1 systems:

> Reactor Recirculation (B31)
> Residual Heat Removal (E11)
> Feedwater (N21)

2.2.5.2 ASME Section XI Requirements

ASME Section XI, 1989 Edition, Subsection IWB, Table IWB-2500-1, Category B-J, Item 9.11 requires a volumetric and surface examination of circumferential piping welds greater than or equal to 4-inch diameter. Note 3 of Table IWB-2500-1 requires that the examination include essentially 100% of the weld length and volume specified in Figure IWB-2500-8.

2.2.5.3 Licensee's Basis for Relief (as stated)

> During the course of inservice examination, 4 of 156 Category B-J circumferential welds have been encountered that are impractical to fully examine in accordance with ASME Section XI (> 90% of length and volume). Pursuant to 10 CFR 50.55a(g)(5)(iii) Detroit Edison is requesting relief from ASME Section XI requirements to perform complete examinations of listed piping welds, as described above.
>
> Fermi proposes to examine these welds to the extent practical within the limits of design and accessibility. Reasonable assurance of piping system structural integrity is provided by the Fermi ISI NDE Program as detailed in this relief request. Detroit Edison considers the proposed alternative examination to provide an acceptable level of quality and safety.
>
> The adjacent weld, which is also a moderate stress weld, is fully examined. Inspections completed through the sixth refueling outage (RF06) have detected no reportable service induced defects in any carbon steel piping welds subject to ISI.

2.2.5.4 Licensee's Proposed Alternative Examination (as stated)

> Partial examination of each weld to the greatest extent possible using appropriate surface and ultrasonic examination methods. Additionally, leakage inspections performed at the completion of each refueling outage per

Category B-P include all of these welds. The extent of partial examination and technical justification for each is provided below:

Reactor Recirculation (B31)

Category /Item	Weld Identification	Percentage Complete	Limitation Description	Alternative Examination
B-J/B9.11	FW-RS-2-A5	86% PT >90% UT	Pump Insulation Support Ring & Brackets	Examine accessible area

This stainless steel weld is a low stress random selection. The weld was given an IGSCC [intergranular stress corrosion cracking] mitigation treatment (Induction Heat Stress Improvement) as defined in NUREG 0313 Rev. 2, prior to service. Fermi has also implemented an augmented inspection program in accordance with Generic Letter 88-01. The combined Code and GL-88-01 selections result in greater than 50% of all Reactor Recirculation System welds being inspected each interval. The inspection sample set is sufficiently large to provide for reliable detection of representative degradation. There is no decrease in the ability to detect system degradation as a result of this limitation. Redesigning or removing the obstructions to marginally increase coverage of this weld is impractical. It would also substantially increase man-hours and radiation dose without a compensating increase in plant safety. Detroit Edison believes this alternative provides an acceptable level of quality and safety.

RHR (E11)

Category /Item	Weld Identification	Percentage Complete	Limitation Description	Alternative Examination
B-J/B9.11	FW E11-2299 -0W1	>50% UT 100% PT	Tee Configuration Limits UT Only	Examine accessible area

This stainless steel tee-to-pipe weld is a high stress weld selection. The weld was radiographed during construction and satisfied Section III acceptance criteria. There are also six other high stress locations in the RHR system that were fully examined. The surface of the weld is fully accessible for liquid penetrant examination. Ultrasonic examination is limited to effective scanning from the pipe side only because of reducing-tee configuration. The ultrasonic examination covers all of the base material on the pipe side of the weld and the weld root area. Because the examination covers the weld root area, which is also the thinnest section of this pipe-to-tee weld zone, there is adequate assurance that IGSCC or fatigue or cracking could be detected. Altering the weld design to increase exam coverage would be impractical.

Additionally, two adjacent welds on both sides of this weld are fully examined. Fermi has also implemented an augmented inspection program in accordance with Generic Letter 88-01. The combined Code and Generic Letter 88-01 selections result in greater than 50% of all susceptible welds being inspected each interval. The inspection sample set is sufficiently large to provide for reliable detection of representative degradation. There is no decrease in the ability to detect system degradation as a result of this limitation.

Radiographic examination was considered as an alternative but has the following limitations. The radiation emitted from the pipe would negatively impact the sensitivity of the examination. Performance of the examination would take approximately one shift to complete and prevent other outage activities from be performed during the radiography evolution. Radiographic examination of the weld would require draining of the recirculation loop piping and a portion of RHR. This would require plugging jet-pumps and recirc suction lines inside the vessel. RHR Shutdown cooling would not be available to remove decay heat. For these reasons radiography is not a feasible alternative for the ultrasonic examination.

Because of the acceptable initial condition, pressure test history and continued performance, the capability to complete the surface exam and greater than 50 percent of the exam volume including the root area, it is reasonable to conclude there is no significant impact on the level of plant quality and safety by the reduction in volumetric coverage of this weld. Detroit Edison believes this alternative provides an acceptable level of quality and safety.

Feedwater (N21)

Category /Item	Weld Identification	Percentage Complete	Limitation Description	Alternative Examination
B-J/B9.11	FW-N21-2336-0W1	~76%UT 100% MT	Tee to Valve Configuration	Examine accessible area

This carbon steel tee-to-pipe weld is a moderate stress weld selection category as defined in the Fermi UFSAR [Updated Final Safety Analysis Report]. The moderate stress category results in an inspection sample of 28% of all Category B9.11 circumferential welds. The increased inspection sample is comprised of welds with the highest probability of failure and results in added assurance of system integrity. This is a more conservative approach to selecting welds than a supplemental random selection to bring the examination sample to 25%, as specified in the Code. The inspection sample set exceeds ASME Code requirements and is sufficiently large to provide for reliable detection of system degradation.

The weld was radiographed during construction and satisfied Section III acceptance criteria. The valve body and weld ends were also radiographed in accordance with NB 2570. The surface of the weld is fully accessible for magnetic particle examination. Ultrasonic examination is limited because of tee-to-valve configuration. The ultrasonic examination does cover the weld and the weld root area in at least one direction. The base material on the valve side is not fully covered in two directions. Altering the weld design to marginally increase coverage is impractical.

Because of the acceptable initial condition, pressure test history and continued performance, the capability to complete the surface exam and approximately 75% of the exam volume including the root area, it is reasonable to conclude there is no significant impact on the level of plant quality and safety by the reduction in volumetric coverage of this weld. Because the inspection sample population exceeds ASME Code requirements, there is no decrease in the ability to detect system degradation as a result of this limitation. Detroit Edison believes this alternative provides an acceptable level of quality and safety.

Feedwater (N21)

Category /Item	Weld Identification	Percentage Complete	Limitation Description	Alternative Examination
B-J/B9.11	FW-N21-2336-1W03	50% UT 100% MT	Sweepolet to Valve Configuration	Examine accessible area

This carbon steel reducer-to-valve weld is a high stress weld selection. The weld was radiographed during construction and satisfied Section III acceptance criteria. The valve body and weld ends were also radiographed in accordance with NB 2570. There are also eleven other high stress locations (includes terminal ends) in the Feedwater System that will be fully examined. The surface of the weld is fully accessible for magnetic particle examination. Ultrasonic examination is limited to effective scanning from the crown of the weld. The ultrasonic examination covers most of the base material on both sides of the weld in one direction. The entire weld and root was scanned in the circumferential direction. Additionally, the high stress weld directly adjacent to this weld was fully examined.

There are over 50 high stress carbon steel weld selections spread among the systems subject to inservice inspection. The Fermi Class 1 inspection population for all systems exceeds ASME Code requirements by 15 welds because moderate stress welds are included in the selection basis. The welds that were selected are the most probable locations for stress related failure. The selection methodology used was more stringent than required by Code. Because of the selection methodology and sample size there is no reduction in capability to detect system degradation as compared to Code requirements. Through the sixth refueling outage (RF06) there were no service induced defects detected. Industry experience does not indicate cracking of carbon steel butt welds to be a problem. All of these reasons indicate that it is impractical to alter the weld design to increase exam coverage for this weld.

Radiographic examination was considered as an alternative but is undesirable for the following reasons. Draining the feedwater line to perform the examination would make reactor water clean up unavailable and would negatively impact reactor vessel clarity potentially affecting refueling and inspection activities. It would also prevent drywell and steam tunnel outage activities from be performed during the radiography evolution adding critical path time to the outage schedule. The benefit of increasing the coverage of this weld by radiographic examination has only a small potential of increasing plant safety margin and a disproportionate impact on other plant activities. Because of these impacts and since the Fermi inspection program exceeds ASME Code requirements for the sampling program this alternative is not considered to be practical.

Because of the acceptable initial condition, pressure test history and continued performance, the capability to complete the surface exam and approximately 50 percent of the Code exam volume, it is reasonable to conclude there is no significant impact on the level of plant quality and safety by the reduction in volumetric coverage of this weld. Detroit Edison believes this alternative provides an acceptable level of quality and safety.

2.2.5.5 Staff Evaluation and Conclusion

The Code requires 100 percent volumetric and surface examination of the subject welds. Complete volumetric or surface examinations cannot be performed due to component configurations (tee configuration, tee to valve configuration, sweepolet-to-valve configuration), and interference from pump insulation, and support rings and brackets. Therefore, the Code volumetric or surface examination requirements for the subject welds are impractical. To meet the Code requirements, the subject welds and/or adjoining components would require significant re-design and modifications. Imposition of this requirement would place a considerable burden on the licensee.

The licensee has completed 50-76 percent and greater than 90 percent of the Code-required volumetric and surface examinations, respectively, of three of the subject welds during the first interval. The remaining weld included in the licensee’s request received 100 percent volumetric examination and was limited to 86 percent of the Code-required surface examination. Furthermore, the subject welds are part of a larger population (156 welds) of Examination Category B-J circumferential welds that will be examined during the second interval. Based on the volumetric and surface examinations of the subject welds completed and the examinations performed on the remaining population of circumferential B-J welds, the staff concludes that patterns of degradation, if present, would be detected. Consequently, reasonable assurance of the structural integrity of the subject welds will be provided. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the licensee’s second inservice inspection interval. Should less coverage be obtained on the subject welds during the second 10-year inspection interval, the licensee will need to submit a request for relief.

2.2.6 Request for Relief No. RR-A25, Examination Category B-A, Item B1.11, Pressure Retaining Reactor Pressure Vessel Circumferential Shell Welds

2.2.6.1 The components for which relief is requested

Pressure Retaining RPV Circumferential Shell Welds
(Welds 4-308A, 4-308B, 1-313, and 9-307)

2.2.6.2 ASME Section XI Requirements

ASME Section XI, 1989 Edition, Subsection IWB, Table IWB 2500-1, Examination Category B-A, Item No. B1.11, and the augmented examination requirement of 10 CFR 50.55a(g)(6)(ii)(A)(2) requires volumetric examination of essentially 100 percent of RPV circumferential weld and base material regions in the reactor pressure vessel each inspection interval.

2.2.6.3 Licensee's Basis for Relief (as stated)

> Pursuant to 10CFR55.55a(a)(3)(i), and consistent with information contained in NRC Generic Letter 98-05, Detroit Edison is requesting an alternative from ASME Section XI requirements to examine essentially 100% of accessible Category B-A circumferential welds and is proposing permanent relief (for the remaining portion of the initial license period) from these examinations.
>
> The basis for this request for inspection relief is documented in the report "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," that was transmitted to the NRC in September 1995. The BWRVIP-05 report provides the technical basis for eliminating inspection of BWR RPV circumferential shell welds. The BWRVIP-05 report concludes that the probability of failure of the BWR RPV circumferential shell welds is orders of magnitude lower than that of the longitudinal shell welds. The NRC staff has conducted an independent risk-informed assessment of the analysis contained in BWRVIP-05. This assessment also concluded that the probability of failure of the BWR RPV circumferential welds is orders of magnitude lower than that of the longitudinal shell welds. Additionally, the NRC assessment demonstrated that inspection of BWR RPV circumferential welds does not measurably affect the probability of failure.
>
> As discussed during the ACRS [Advisory Committee on Reactor Safeguards] Full Committee meeting on July 9, 1998 the Staff has completed its evaluation of the BWR Vessel and Internals Project (BWRVIP) recommendations for reduced inspections of the reactor pressure vessel shell welds as described in the BWRVIP-05 report. Based on the Staff s review, it has been concluded that inservice inspection (ISI) of the BWR RPV circumferential welds is not necessary during the current license term since these welds have low failure frequencies. The NRC issued a Final Safety Evaluation documenting acceptance of the BWRVIP-05 report on July 28, 1998.
>
> The NRC Staff issued Generic Letter 98-05 regarding the use of the BWRVIP-05 report as the basis for BWR licensees to request relief from the requirements to conduct volumetric examinations of the BWR RPV circumferential welds. This independent NRC assessment utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. Three key assumptions in the PFM analysis are: the neutron fluence was estimated to be end-of-license mean fluence, the chemistry values are mean values based on vessels types and the potential for beyond design basis events is considered.

Although BWRVIP-05 provides the technical basis supporting this relief request, the following information is provided to show the conservatism of the NRC analysis for the Fermi 2 Nuclear Power Plant. For plants with RPVs fabricated by Combustion Engineering the mean end-of-license neutron fluence use in the NRC PFM analysis was 20 x 10^{17} n/cm^2. However, at Fermi 2 the highest fluence anticipated at the end of the requested relief period (end of the initial license period) is 6.5 x 10^{17} n/cm^2. Thus, embrittlement due to fluence effects is much lower, and the NRC analysis is conservative for Fermi 2 in this regard. Therefore, there is significant conservatism in the already low circumferential weld failure probabilities as related to Fermi 2. Other Fermi 2 RPV shell weld information that the NRC staff has requested (GL 98-05) be included in requests for relief is provided in attached Table 1. The data in Table 1 indicates that Fermi 2 upper bound adjusted reference temperature (ART) remains within acceptable limits as defined in the NRC Final Safety Evaluation of the BWRVIP-05 report.

At an August 8, 1997 meeting with industry, the NRC staff indicated that the potential for, and consequences of, nondesign basis events (not addressed in the BWRVIP-05 report) should be considered. In particular, the NRC staff stated that nondesign basis cold over-pressure transients should be considered. It is highly unlikely that a BWR would experience a cold overpressure transient. For a BWR to experience such an event multiple operator errors would be required. At the August 8, 1997 meeting, the NRC staff described several types of events that could be precursors to BWR RPV cold over pressure transients. These were identified as precursors because no cold overpressure event has occurred at an U.S. BWR. Also at the August 8 meeting, the NRC staff identified one actual cold overpressure event that occurred during shutdown at a non-U.S. BWR. This event apparently included several operator errors that resulted in a maximum RPV pressure of 1150 psi [pounds per square inch] with a temperature range of 79°F to 88°F.

As provided in the following discussion, Fermi 2 has in place procedures and Technical Specifications which monitor and control reactor pressure, temperature, and water inventory during all aspects of cold shutdown which would minimize the likelihood of a Low Temperature Over-Pressurization (LTOP) event from occurring. Additionally, these procedures are reinforced through operator training.

The Pressure Test procedures, which are used at Fermi 2, have sufficient procedural guidance to prevent a cold, over-pressurization event. Pressure testing is performed at the conclusion of each outage. The system leakage tests include requirements for operations management to perform a "pre-job briefing" with all essential personnel. This briefing details the anticipated testing evolution with special emphasis on: conservative decision making, plant safety awareness, lessons learned from similar in-house or industry operating experiences, the importance of open communications, and finally, the process in which the test would be aborted if plant systems responded in an adverse manner. Vessel temperature and pressure are required to be monitored throughout these tests to ensure compliance with the Technical Specification pressure-temperature curve.

Additionally, to ensure a controlled, deliberate pressure increase, the rate of pressure increase is administratively limited throughout the performance of the test. If the pressurization rate exceeds this limit, direction is provided to remove the CRD [control rod drive] pumps which are used for pressurization, from service.

With regard to inadvertent system injection resulting in an LTOP condition, the high pressure make-up systems (High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems, as well as the normal feedwater supply (via the Reactor Feedwater Pumps) at Fermi 2 are all steam driven. During reactor cold shutdown conditions, no reactor steam is available for the operation of these systems. Therefore, it is not possible for these systems to contribute to an over-pressure event while the unit is in cold shutdown.

The Standby Feed Water (SBFW) system is an available high pressure electric driven make up system. The SBFW system does not automatically inject water into the RPV. The SBFW system requires deliberate operator action to open the injection isolation valve. Procedures are in place to administratively control the use of the SBFW system.

In the case of low pressure system initiation, the Fermi 2 pressure-temperature limit curves for hydrostatic testing as provided in Fermi Technical Specifications, permit pressures up to 312 psig [pounds per square inch gauge] at temperatures from 71°F up to 100°F. Above 100°F, the permissible pressure increases immediately to near 600 psig and increases rapidly with increasing temperature. The shutoff head for the Core Spray and Residual Heat Removal Pumps are both below 400 psig. Therefore, the potential for an over-pressurization event which would exceed the pressure-temperature limits, due to an inadvertent actuation of this system is very low.

Procedural control is also in place to respond to an unexpected or unexplained rise in reactor water level, which could result from a spurious actuation of an injection system. Actions specified in this procedure included preventing condensate pump injection, securing ECCS system injection, tripping CRD pumps, terminating other injection sources, lowering RPV level via the RWCU [reactor water cleanup] system, and the steam line drains.

In addition to procedural barriers, Licensed Operator Training is given which further reduces the possibility of the occurrence of LTOP events. During Initial Licensed Operator Training the following topics are covered: Brittle fracture and vessel thermal stress; Operational Transient (OT) procedures, including the OT on reactor high level; Technical Specifications training, including discussion of Pressure/Temperature (P/T) Limits; and Simulator Training of plant heatup and cooldown including performance of surveillance tests which ensure pressure-temperature curve compliance.

In addition to the above, continuous review of industry operating plant experiences is conducted to ensure that the Fermi 2 procedures consider the impact of actual events, including potential LTOP events. Appropriate adjustments to the procedures and associated training are then implemented to preclude similar situations from occurring at Fermi 2.

Based on the above, the probability of a cold over-pressure transient is considered to be highly unlikely.

The NRC staff transmitted a Request for Additional Information (RAI) regarding the BWRVIP-05 report to the BWR Vessel and Internals Project (BWRVIP). The BWRVIP provided a response to the RAI that included additional information on the BWRVIP PFM analysis, comparisons to the NRC Staff PFM analysis, and additional information regarding beyond design basis cold overpressure transients. We believe the BWRVIP-05 report and the NRC Final Safety Evaluation Report analysis provide sufficient basis to support this relief request.

Based on the documentation in BWRVIP-05, the risk-informed independent assessment performed by the NRC staff and the discussion above, permanent relief (for the remaining portion of the initial license period) from completing inspection of the RPV circumferential shell welds at Fermi 2 is justified.

TABLE 1 Comparison of Licensee's Probabilistic Fracture Mechanics Assessment (PFMA) with the Staff's PFMA for the Limiting CEOG Case Study for Circumferential Welds

[The data from the licensee's table is displayed in Table 1 under Section 2.2.6.5, "Staff Evaluation and Conclusion."]

2.2.6.4 Licensee's Proposed Alternative (as stated)

The beltline circumferential weld (1-313) was partially examined during the first inspection interval (approximately 54% complete, RF02, Spring 1991). Additionally, Detroit Edison will perform examination of approximately 5% of the Fermi 2 RPV circumferential weld areas only at the intersection of longitudinal seams.

2.2.6.5 Staff Evaluation and Conclusion

Staff Review of BWRVIP-05 Report

By letter dated September 28, 1995, as supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, June 13, and December 18, 1997, the Boiling Water Reactor Vessel and Internals Project (BWRVIP), a technical committee of the BWR Owners' Group (BWROG), submitted the proprietary report, "BWR Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," which proposed to reduce the scope of inspection of the BWR RPV welds from essentially 100 percent of all RPV shell welds to 50 percent of the axial welds and 0 percent of the circumferential welds. By letter dated October 29, 1996, the BWRVIP modified their proposal to increase the examination of the axial welds to 100 percent from 50 percent while still proposing to inspect essentially 0 percent of the circumferential RPV shell welds, except that the intersection of the axial and circumferential welds would have included approximately 2-3 percent of the circumferential welds.

On May 12, 1997, the NRC staff and members of the BWRVIP met with the Commission to discuss the NRC staff's review of the BWRVIP-05 report. In accordance with guidance provided by the Commission in Staff Requirements Memorandum (SRM) M970512B, dated May 30, 1997, the staff initiated a broader, risk-informed review of the BWRVIP-05 proposal, and issued a final safety evaluation related to the review of BWRVIP-05 on July 28, 1998, which generically approved the reduction in inspection of circumferential RPV welds. In SECY-98-219, the staff provided the Commission with its methods and acceptance criteria for considering both partial and permanent requests for relief from the augmented reactor vessel examinations required by 10 CFR 50.55a(g)(6)(ii)(A)(5).

In GL 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998, the staff informed licensees owning BWR designs that the NRC staff had completed its review of BWRVIP-05. In the GL, the staff also informed BWR licensees that they could request periodic or permanent (i.e., for the remaining term of operation under the existing initial license) relief from the inspection of BWR circumferential welds if the licensee meets the following criteria:

1. If at the expiration of the license for the plant, the circumferential welds in the vessel are shown to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 30, 1998, final safety evaluation; and

2. If it is demonstrated that the licensee for a facility has implemented operator training and established procedures that limit the frequency of cold overpressure events to the degree specified in the staff's July 28, 1998, final safety evaluation. In the GL, the staff also informed BWR licensees that they would still need to perform their required inspections of "essentially 100 percent" of all longitudinal RPV welds.

Technical Report BWRVIP-05 provides the technical basis for permanently deferring the augmented inspections of circumferential welds in BWR RPVs. In the report, the BWRVIP concluded that the probabilities of failure for BWR RPV circumferential welds are orders of magnitude lower than that of the longitudinal welds. The NRC conducted an independent risk-informed, probabilistic fracture mechanics assessment (PFMA) of the analysis presented in the BWRVIP-05 report.[1] The staff conservatively calculated the probability that an RPV shell weld would catastrophically fail during the licensed operating term for a BWR nuclear plant. During the review, the staff used the FAVOR Code to perform the PFMA. The final failure probability for an RPV weld was calculated as the product of the frequency for the critical (limiting) transient event and the conditional failure probability for the weld using the limiting conditions from that event.

The staff determined the conditional probabilities of failure for longitudinal and circumferential welds in vessels fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering (CE), and Babcock and Wilcox (B&W). The analysis identified pressures and temperatures resulting from a cold overpressure event in a foreign reactor as the limiting event for BWR RPVs. The staff estimated that the probability for the occurrence of the limiting overpressurization transient

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[1] The staff's PFMA of BWRVIP-05 is documented in a letter dated July 28, 1998, to Mr. Carl Terry, Chairman of the BWRVIP.

was 1 x 10^{-3} per reactor year. Table 2.6-4 of the staff's PFMA identifies the conditional failure probabilities for the bounding reference cases for longitudinal and circumferential welds in CB&I, CE, and B&W fabricated vessels. The materials and neutron irradiation parameters used by the staff in calculating the conditional probability failures for the reference cases were also identified in Table 2.6-4 of the staff's PFMA.

For CE-fabricated vessels, the conditional probability of failure for circumferentially oriented flaws was determined to be 6.34 x 10^{-5} per reactor year. The corresponding mean RT_{NDT} value used to calculate the conditional probability of failure for the CE reference case was 98.1°F. Using this data, the staff calculated the best-estimate failure probability for CE-fabricated circumferential welds to be 6.34 x 10^{-8} per reactor year.[2] Vessels with RT_{NDT} values less than those resulting from the staff's assessment are considered to have less embrittlement than the vessels simulated in the review. Therefore, these vessels should have a conditional probability of failure less than or equal to the values in the staff's final safety evaluation.

Staff Review of Licensee's Request

The staff confirmed that the RT_{NDT} values for the circumferential welds through the projected end-of-license are less than the values in the reference case for the CE-fabricated vessels (see Table 1). RT_{NDT} is a measure of the amount of irradiation embrittlement. Since the RT_{NDT} values are less than the values in the reference case for CE fabricated vessels, the Fermi 2 RPV has less embrittlement than the reference case and is considered to have a conditional probability of vessel failure less than or equal to that estimated in the staff's final safety evaluation.

TABLE 1 Comparison of Licensee's Probabilistic Fracture Mechanics Assessment (PFMA) with the Staff's PFMA for the Limiting CE Owners Group Case Study for Circumferential Welds

	Fermi 2 RPV Shell Weld Bounding Circ. Weld	NRC's Limiting PFMA Analysis
Neutron fluence at the end of the requested relief period (upper bound value)	6.5 x 10^{17} n/cm^2	20 x 10^{17} n/cm^2
Initial (unirradiated) reference temperature	-50°F	0°F
Weld Chemistry Factor	236°F	172.2°F
Weld Copper content	0.23%	0.183%
Weld Nickel content	1.0%	0.704%

[2] This value is the product of the conditional probability of failure for the CE Owners Group reference case (6.34 x 10^{-5} per reactor year) and the estimated frequency for the limiting event (1 x 10^{-3} per reactor year).

	Fermi 2 RPV Shell Weld Bounding Circ. Weld	NRC's Limiting PFMA Analysis
Increase in reference temperature due to irradiation (ΔRT_{NDT})	79.3°F	98.1°F
Margin term	56°F	0°F
Mean adjusted reference temperature (ART)	29.3°F	98.1°F
Upper bound adjusted reference temperature (ART)	85.3°F	98.1°F

The staff reviewed the information provided by the licensee regarding the Fermi 2 high-pressure injection sources, operator training, and established plant-specific procedures to prevent RPV cold overpressurization. The licensee evaluated the potential for and consequences of cold overpressure transients. The licensee has assessed the systems that could lead to a cold overpressurization of the Fermi 2 RPV. These included the HPCI and RCIC systems, normal feedwater supply, SBFW, core spray (CS), residual heat removal (RHR), CRD and RWCU.

The HPCI and RCIC pumps are steam driven and do not function during cold shutdown. The reactor feedwater pumps are the high pressure make-up system during normal operations. The reactor feedwater pumps are also steam driven and, therefore, cannot be operated during cold shutdown. The SBFW system requires deliberate operator action to initiate injection. The licensee stated that procedures are in place to administratively control the use of the SBFW system. Although not addressed in the licensee's submittal, the staff notes that generally there are no automatic starts associated with standby liquid control system (SLCS). Operator initiation of SLCS should not occur during shutdown. However, the SLCS injection rate of approximately 40 gpm [gallons per minute] would allow operators sufficient time to control reactor pressure if manual initiation occurred.

The CS and RHR systems are low pressure ECCS systems with shutoff heads below 400 psig. If either one of these systems were manually or inadvertently initiated during cold shutdown, the resulting reactor pressure and temperature would be below the pressure-temperature limits. The CRD and RWCU systems use a feed and bleed process to control RPV level and pressure during normal cold shutdown conditions. Plant procedures are in place to respond to any unexpected or unexplained rise in reactor water level which could result from spurious actuation of an injection system. The procedure actions include preventing condensate pump injection, securing ECCS injection, tripping CRD pumps, terminating other injection sources, and lowering RPV level via the RWCU system and the steam line drains.

In all cases, the operators are trained in methods of controlling water level within specified limits in addition to responding to abnormal water level conditions during shutdown. The licensee also stated that procedural controls for reactor temperature, level, and pressure are an integral part of operator training. Plant-specific procedures have been established to provide guidance to the operators regarding compliance with the Technical Specification pressure-temperature limits. On the basis of the pressure limits of the operating systems, operator training, and established plant-specific procedures, the licensee determined that a non-design basis cold overpressure transient is unlikely to occur.

The information provided sufficient basis to support approval of the alternative examination request. The staff concludes that a non-design basis cold overpressure transient is unlikely to occur at Fermi 2, which is consistent with the staff's analysis.

Staff Conclusion

Based upon its review the staff reached the following conclusions:

(1) Based on the licensee's assessment of the materials in the circumferential welds in the Fermi 2 RPV, the conditional probability of vessel failure is considered to be less than or equal to that estimated from the staff's analysis.

(2) Based on the licensee's high pressure injection sources, operator training, and established plant-specific procedures, the staff concludes that a non-design basis cold overpressure transient is unlikely to occur at Fermi 2.

(3) Based on the above, the staff concludes that the licensee has proposed a reasonable alternative for permanently deferring the augmented inspections of the circumferential welds required by 10 CFR 50.55a(g)(6)(ii)(A)(2). This includes the successive and the additional examination of flaws required by IWB-2420 and IWB-2430, respectively, of the ASME Section XI Code. The staff has also determined that the alternative program provides an acceptable level of quality and safety.

Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), the augmented inspections of the circumferential welds in the Fermi 2 RPV may be permanently deferred for the remaining term of operation under the existing initial operating license.

2.2.7 Request for Relief No. RR-A26

2.2.7.1 The Components for Which Relief is Requested

Code Class 2 piping classified as extensions of containment (that is part of the containment system or which penetrates or is attached to the containment vessel).

2.2.7.2 Systems Affected

E11-Residual Heat Removal
E21-Core Spray
E41-High Pressure Coolant Injection
E51-Reactor Core Isolation Cooling
G51-Torus Water Management
P11-Demineralized Service Water
P44-Emergency Equipment Cooling Water
T46-Standby Gas Treatment
T48-Containment Atmosphere Control

2.2.7.3 ASME Section XI Requirements

ASME Section XI, 1989 Edition, Subsection IWE-1220(d) provides an exemption from IWE-required examinations for piping that is part of the containment system or which penetrates or is attached to the containment vessel. The exemption subsequently requires this piping to be examined in accordance with IWB or IWC as appropriate.

2.2.7.4 Licensee's Basis for Relief (as stated)

> Pursuant to 10 CFR 50.55a(a)(3)(i) Detroit Edison is requesting relief from ASME Section XI requirements to perform the surface or volumetric examinations specified in the 1989 Edition of ASME Section XI for piping classified as extensions of containment. The proposed alternative of visual examination is consistent with the examination of other containment items that only require visual examination.
>
> Detroit Edison has identified a new subset of piping which is considered an *extension of containment* (penetrates containment and within the outboard isolation valve) where the only reason for its selection is the containment function. This is because the system function is either not safety related (e.g., RHR containment/suppression pool spray lines) or that if the rules of IWC were applied, the piping would be exempt from examination per IWC 1220. This is because the piping is either open ended beyond the last shut off valve or the line process conditions are less than or equal to 275 psig and at a temperature equal to or less than 200°F. Since the piping selection is based solely on the containment function, it would not make sense to exempt the piping based on the lack of a safety related system/line function or configuration and design parameters of the process stream.
>
> If the selected piping was subject to IWE requirements for Category E-B, Item E3.10 (containment penetration welds) the examination method would be a visual examination. Additionally, 10 CFR 55a(b)(2)(x)(c) [actually 10 CFR 50.55a(b)(2)(x)(C)] has made containment weld inspection optional because there has been no degradation mechanism specific to containment welds. Application of the IWC rules for Category C-F-2 as specified in IWE-1220(d) would result in surface and volumetric examination as required depending on the nominal pipe wall thickness. As stated previously, considering only the IWC selection requirements, the piping could be exempted from examination. Since the only reason for selecting the subject piping is the containment function, it seems appropriate to apply the IWE inspection methodology rather than the IWC inspection methodology.

2.2.7.5 Licensee's Proposed Alternative Examination (as stated)

> Detroit Edison proposes to perform a visual examination (VT-1) of all selected welds, except those selected based on the high or moderate stress categories defined in UFSAR 5.2.8.8, that will be examined as specified by Table IWC-2500-1 for category C-F. The sample size will be at least 7.5% of the total number of pressure retaining extension of containment welds subject to

examination requirements. This percentage meets the 1989 Section XI selection rate requirements for Category C-F-2 pressure retaining welds. This alternative is equivalent to the IWE methodology for examination of penetration welds.

In addition to the visual examination, the extension of containment piping will be subject to 10 CFR 50 Appendix J leakage rate testing.

2.2.7.6 Staff Evaluation and Conclusion

The licensee requests relief to perform examinations that would not be required by the Code, but are appropriate to ensure that the condition of the subject piping is properly monitored. The licensee identified a subset of piping which is considered an extension of containment (penetrates containment and within the outboard isolation valve). The licensee identified this piping in the following systems: Residual Heat Removal, Core Spray, High Pressure Coolant Injection, Reactor Core Isolation Cooling, Torus Water Management, Demineralized Service Water, Emergency Equipment Cooling Water, Standby Gas Treatment, and Containment Atmosphere Control. The 1992 Edition of Section XI, IWE-1220(d) identifies components exempt from IWE-required examination by listing "piping, pumps, and valves that are part of the containment system, or which penetrate or are attached to the containment vessel" and stating that "these components shall be examined in accordance with the rules of IWB or IWC, as appropriate to the classification defined by the Design Specifications." Following the rules of IWC in this case, the licensee finds that these items may be exempt from examination under the rules in IWC-1221 or IWC-1222.

The licensee proposes to perform a (VT-1) examination on a sampling (7.5 percent) of these extension-of-containment items that are exempted from IWC inspection requirements as allowed by IWC-1221 or IWC-1222. The licensee also proposes to examine selected welds that are high or moderate stress categories as defined in the Fermi 2 UFSAR 5.2.8.8 in accordance with Table IWC-2500-1 for Category C-F.

The staff finds that the licensee's proposed alternative provides an acceptable approach to ensuring the integrity of the subset of piping which is considered an extension of containment (penetrates containment and within the outboard isolation valve) that may be exempt from examination under the rules of IWC-1221 or IWC-1222.

The staff concludes that the licensee's proposed alternative contained in Request for Relief No. RR-A26, provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second interval at Fermi 2.

2.2.8 Request for Relief No. RR-A28 Pressure Retaining Bolted Connection Leakage

2.2.8.1 The Components for Which Relief is Requested

Code Class 1, 2, and 3 systems included in the ISI NDE Program Plan, pressure-retaining bolted connections.

2.2.8.2 ASME Section XI Requirements

ASME Section XI, 1989 Edition, IWA-5250(a)(2) requires the following corrective measures if leakage is observed during VT-2 examinations during the system pressure test at bolted mechanical joints: 1) remove all the bolting material associated with that joint; 2) perform a VT-3 examination for corrosion; and 3) evaluate the conditions in accordance with IWA-3100.

2.2.8.3 Licensee's Basis for Relief (as stated)

> The 1989 Code is too restrictive and does not allow for evaluation and application of prudent engineering judgement. Satisfying the Code requirement for removing bolting may require significant planning and scheduling due to operational concerns and personnel safety. In cases of unisolatable or non-redundant piping, the requirement to remove the bolting in order to conduct the visual examination may necessitate shutdown of the plant and result in unnecessary plant transient cycles.
>
> Pursuant to 10 CFR 50.55a(a)(3)(i) Detroit Edison is requesting relief from ASME Section XI requirements to remove bolting for visual examination when leakage is noted at a bolted connection. This request for relief is based on the alternative provided in Code Case N-566-1. This Code Case is ASME approved indicating Code Committee consensus that the alternative evaluation will provide essentially equivalent results. Removal of bolting is not often necessary because Fermi 2 is a boiling water reactor (BWR) and the reactor coolant system and associated systems do not experience the corrosive environment from boric acid residues as would a pressurized water reactor (PWR). Therefore, there is no reason to suspect degradation of bolting caused solely by the chemistry of leaking coolant.
>
> The purpose of IWA-5250(a)(2) is to determine if inservice leakage has degraded the bolting material. Therefore bolting that is new or was visually examined during joint disassembly would not warrant removal. Additionally, bolting that is in air or gas service should also be excluded.
>
> Bolted flange connections such as those on the control rod drive (CRD) housings have a history of leaking upon return to service but decrease over time. This bolting is a chrome alloy material that is resistant to general corrosion. CRDs are rebuilt periodically and bolting is VT-1 examined and reinstalled or replaced as necessary.
>
> Bolting in flanged joints are often partially visible because of the space between the flanges. While flange or valve bonnet leakage is normally not acceptable the prudent corrective measure may be to verify torque and re-tighten bolting as necessary rather than remove the bolting.
>
> The reasons provided above demonstrate the need for evaluation of leakage and application of engineering judgement.

2.2.8.4 Licensee's Proposed Alternative Examination (as stated)

Detroit Edison proposes to implement the alternative described in Code Case N-566-1 dated February 15, 1999 as follows.

(a) The leakage shall be stopped, and the bolting and component material shall be evaluated for joint integrity as described in (c) below, or

(b) If the leakage is not stopped, the joint shall be evaluated in accordance with IWB-3142.4 for joint integrity. This evaluation shall include the considerations listed in (c) below.

(c) The evaluation of (a) and (b) above is to determine the susceptibility of the bolting to corrosion and failure. This evaluation shall include the following:

(1) the number and service age of the bolts;
(2) bolt and component material;
(3) corrosiveness of process fluid;
(4) leakage location and system function;
(5) leakage history at the connection or other system components;
(6) visual evidence for corrosion at the assembled connection

2.2.8.5 Staff Evaluation and Conclusion

In accordance with IWA-5250(a)(2), if a leak occurs at a bolted connection, the bolting must be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100. In lieu of this requirement, the licensee has proposed to evaluate the bolting to determine its susceptibility to corrosion. The proposed evaluation will consider, as a minimum, bolting materials and visual evidence of corrosion at the assembled connection as described in Code Case N-566-1. The staff finds that the licensee's proposed alternative to use Code Case 566-1 provides an acceptable approach to ensuring the integrity of the bolted connection.

The staff concludes that the licensee's proposed alternative to the Code requirements contained in Request for Relief No. RR-A28, provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative to use Code Case N-566-1 is authorized pursuant to 10 CFR 50.55a(a)(3)(i). The alternative is authorized for the second interval at Fermi 2 or until Code Case N-566-1 is published in a future revision of Regulatory Guide (RG) 1.147. At that time, if the licensee intends to continue to implement Code Case N-566-1, the licensee should follow all the provisions in Code Case N-566-1 with the limitations issued in RG 1.147, if any.

2.2.9 Licensee's Relief Request No. RR-A29 Austenitic Stainless Steel BWR Coolant Piping Welds

2.2.9.1 ASME Section XI Requirements

Paragraph IWB-2430(a) to the 1989 Edition of Section XI of the Code states that examinations performed in accordance with Table IWB-2500-1 that reveal indications exceeding the acceptance standards of Table IWB-3410-1 shall be extended to include additional examinations at this outage. The additional examinations shall include the remaining welds, areas, or parts included in the inspection item listing scheduled for this and the subsequent period. If the examination for that inspection item is not scheduled in the subsequent period, the most immediate period containing scheduled examinations shall be taken as the subsequent period.

2.2.9.2 Licensee's Basis for Relief

Detroit Edison is requesting relief from ASME Section XI requirements for additional weld examinations for welds subject to GL 88-01 and NUREG-0313, Revision 2, because these regulatory documents provide alternative sample expansion guidance that considers IGSCC susceptibility. This methodology ensures that welds with similar risk (i.e., weld category, pipe, size, system, and locations) for cracking are examined while maintaining radiation exposure of examination personnel as low as reasonably achievable.

The Code-specified expansion methodology only considers Code item numbers and not the associated material's susceptibility to degradation. For example, Code B9.11 would include carbon steel as well as stainless steel welds. The carbon steel welds would not be subject to IGSCC. Therefore, it would not be appropriate to include those items in the sample expansion.

2.2.9.3 Licensee's Proposed Alternative

When examinations are being performed to satisfy the requirements of GL 88-01 and NUREG-0313 in addition to the Code requirements, sample expansion resulting from unacceptable IGSCC flaw indications will be performed using the methodology specified in GL 88-01 and NUREG-0313. The relief is requested for the second 10-year inspection interval.

2.2.9.4 Staff Evaluation and Conclusion

The licensee has committed to follow GL 88-01 and NUREG-0313 as part of its inservice inspection program. The GL and NUREG provide guidance for an effective inspection program that is capable of detecting IGSCC. The guidance provides inspection frequencies and qualification criteria that are more stringent than Code requirements. Included in the guidance criteria, Code-qualified inspection personnel must pass a performance demonstration test. Successfully passing the performance demonstration test qualifies inspection personnel to perform IGSCC examinations. Therefore, inspections performed by IGSCC-qualified inspection personnel may detect more flaws than inspections performed by personnel only qualified according to Code.

The GL and NUREG recommend that IGSCC-qualified inspectors using qualified procedures be used for the inspections of all boiling water reactor piping made of austenitic stainless steel that is 4 inches or larger in nominal diameter and contains reactor coolant at a temperature above 200°F during power operation, regardless of Code classifications. The recommendation is based on field failures and laboratory testing that show IGSCC is associated with these sizes of stainless steel piping. The GL and NUREG specifically state that the recommendations do not apply to piping made of carbon steel classified as P-1 by the ASME Code.

In the event that an IGSCC flaw is detected during a combination GL/NUREG and Code inspection of stainless steel piping, the flaw is considered the product of the GL/NUREG augmented inspections criteria. This is because the GL/NUREG imposes additional criteria to

Code requirements that are specifically designed for the detection of IGSCC. Accordingly, to determine the extent of IGSCC, the sample expansion criteria in the GL and NUREG would be applicable. A sample expansion according to Code is not applicable because the Code requires that both stainless steel and carbon steel piping be included in the sample expansion of an inspection. The inclusion of carbon steel, which is not susceptible to IGSCC, diminishes the meaningfulness of the expanded sample size searching for IGSCC. The alternative criteria for the expanded sample only includes piping that is susceptible to IGSCC. Therefore, the expanded sample size criteria in GL 88-01 and NUREG-0313 will provide an acceptable level of quality and safety.

Based on the above evaluation, the staff concludes that the proposed alternative will provide an acceptable level of quality and safety. Therefore, the staff authorizes the use of the proposed alternative pursuant to 10 CFR 50.55a(a)(3)(i).

2.2.10 Licensee's Relief Request Number RR-A30, Pressure-Retaining Piping Welds, Categories B-J and C-F-2

2.2.10.1 ASME Section XI Requirements

Tables IWB-2500 and IWC-2500 to the 1989 Edition of Section XI of the Code require the volumetric examination of pressure-retaining welds in piping NPS [nominal pipe size] 4 inches and larger. IWA-2232 requires that ultrasonic examinations be conducted in accordance with Appendix I. Appendix I specifies that ultrasonic examination of piping welds be performed in accordance with Appendix III.

2.2.10.2 Licensee's Basis for Relief

Detroit Edison is requesting relief from ASME Section XI amplitude-based examination requirements described in Appendix III because more effective techniques are available.

The utility Performance Demonstration Initiative (PDI) developed a program based on the 1992 Edition with the 1993 Addenda of Section XI of the Code. This program requires that ultrasonic equipment, procedures, and examiners be qualified on flawed and notched specimens with configurations similar to those found in the plant. Consequently, the PDI program provides a higher degree of reliability for detection and characterization of flaws when compared to the conventional amplitude-based ultrasonic techniques required by the 1989 Edition of the Code.

The NRC issued a letter to the Chairman of the BWR Owners' Group on March 1, 1996, discussing the transition from the IGSCC qualification program (IGSCC Coordination Plan) to the PDI program for the qualification requirements applicable to procedures and personnel for GL 88-01 and NUREG-0313 examinations. The letter stated that personnel qualification and subsequent requalification for the IGSCC program could be obtained through the PDI program. The techniques developed and qualified through the PDI program are recognized as being superior to those specified in Appendix III.

2.2.10.3 Proposed Alternative

When examinations are being performed to satisfy the requirements of GL 88-01 and NUREG-0313, PDI-qualified personnel and procedures will be used. For all other examinations, the techniques developed through the PDI program will be used by certified examination personnel trained in their use.

2.2.10.4 Staff Evaluation and Conclusion

The licensee is requesting to follow the requirements contained in the rule published on September 22, 1999, in the *Federal Register*, Volume 64, No. 183, Pages 51370 through 51400. The rule was effective November 21, 1999, and becomes mandatory for volumetric piping examinations performed using ultrasonic examination techniques on May 22, 2000. Following the rule does not require NRC approval. Therefore, relief is not necessary.

3.0 CONCLUSION

The staff finds that the licensee's proposed alternatives to the Code requirements in Requests for Relief Nos. RR-A18, RR-A26, RR-A28, RR-A29, and RR-C3 provide an acceptable level of quality and safety. Therefore, the proposed alternatives documented in RR-A18, RR-A26, RR-A28, RR-A29, and RR-C3 are authorized pursuant to 10 CFR 50.55a(a)(3)(i). The staff finds that compliance with the specified requirement in Requests for Relief Nos. RR-A19 and RR-C4 would result in hardship without a compensating increase in the level of quality and safety. Therefore, the proposed alternatives in RR-A19 and RR-C4 are authorized pursuant to 10 CFR 50.55a(a)(3)(ii). For Requests for Relief Nos. RR-A1, RR-A6, and RR-A23, the staff concludes that the Code requirements are impractical for the subject welds and that granting relief will not endanger life, property, or the common defense and security and is otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i). The staff finds that relief is unnecessary for the alternative in RR-A30.

The staff reviewed the licensee's Request for Relief No. RR-A25 and concluded that the alternative proposal provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), the inspection of the circumferential welds may be permanently deferred for the remaining term of operation under the existing, initial operating license.

Request for Relief No. RR-A27 was withdrawn by the licensee's letter of February 4, 2000.

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