September 30, 2002

Mr. Lew W. Myers
Chief Operating Officer
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 - REQUESTS FOR RELIEF FOR THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN (TAC NO. MB1607)

Dear Mr. Myers:

By letter dated September 19, 2000 (Serial Number 2672), FirstEnergy Nuclear Operating Company submitted the Third 10-Year Interval Inservice Inspection Program for the Davis-Besse Nuclear Power Station, Unit 1. Included in the submittal were 32 requests for relief from conformance with certain requirements of Section XI of the 1995 Edition and Addenda through the 1996 Addenda, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Additional information was provided in your letters dated September 7, 2001 (Serial Number 2729), November 27, 2001 (Serial Number 2736), February 6, 2002 (Serial Number 2762), August 1, 2002 (Serial Number 2798), August 9, 2002 (Serial Number 1-1281), September 3, 2002 (Serial Number 2805), and September 26, 2002 (Serial Number 2808), and in an electronic transmission dated February 14, 2002 (ADAMS Accession Number ML022410047).

Your letter dated August 1, 2002, resubmitted relief request (RR) RR-A2 in its entirety. RR-A2 will be evaluated in a separate safety evaluation under TAC No. MB5849. Your letter dated August 9, 2002, stated that previously submitted RR-A8 and RR A-12 were determined to remain applicable for the replacement reactor vessel head. Finally, your letter dated September 3, 2002, withdrew RR-A9.

The staff's evaluations for the remaining 30 relief requests (i.e., RR-A2 will be handled separately and RR-A9 was withdrawn as discussed above) are included in Enclosures 1 through 4 of this letter. Due to an administrative error, a letter issued September 27, 2002, addressed the same items as Enclosures 2 through 4 of this letter, but contained preliminary versions of the associated safety evaluations. This letter replaces the September 27 letter in its entirety, and the previous letter should be disregarded.

ENCLOSURE 1

Enclosure 1 addresses RR-A1, RR-A3 through RR-A8, RR-A10 through RR-A12, RR-A14 through RR-A17, RR-B1 through RR-B4, and RR-C1.

For Requests for Relief RR-A1, RR-A6, RR-A7 and RR-C1, the staff concludes that the Code requirements are a hardship without a compensating increase in quality and safety.

Furthermore, the licensee's proposed alternatives provide reasonable assurance of structural integrity of the subject components contained in the licensee's requests for relief. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii), for the third 10-year inservice inspection interval, which expires September 20, 2010.

For RR-A3, the staff concludes that the removal of insulation at elevated pressures and temperatures would result in a hardship without a compensating increase in the level of quality and safety. Furthermore, the staff concludes that the licensee's proposed alternative provides reasonable assurance of structural integrity of the bolted connections. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year interval, which expires September 20, 2010, or until such time as Code Case N-616 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-616, it must follow all provisions in the subject code case with the limitations or conditions (if any) listed in RG 1.147.

For Requests for Relief RR-A8, RR-A16, and RR-B2, the staff concludes that the licensee's proposed alternatives provide reasonable assurance of quality and safety and are authorized pursuant to 10 CFR 5.55a(a)(3)(i) for the third 10-year interval, which expires September 20, 2010.

For Requests for Relief RR-A4, RR-A5, RR-B1, RR-B3 and RR-B4, the staff concludes that the Code-required examinations are impractical and to require the licensee to perform the Code-required examinations would be a burden on the licensee. The subject components contained in the request for relief would be required to be redesigned in order for the licensee to perform the Code-required examinations. Therefore, relief is granted for RR-A4, RR-A5, RR-B1, RR-B3, and RR-B4 pursuant to 10 CFR 50.55a(g)(6)(i). Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

For RR-A15, the staff determined that the Code requirements are impractical. To impose the Code requirements would cause a burden on the licensee, because the subject components would have to be redesigned in order to perform the Code required examinations. However, by November 22, 2002, it is expected that there will be a qualified procedure to examine dissimilar metal welds and austenitic stainless steel welds from the pipe bore. Therefore, for dissimilar metal welds and austenitic stainless steel welds, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval, but not beyond November 22, 2002. For welds connecting cast stainless steel components, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval, which expires September 20, 2010. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

For RR-A17, the staff concludes that the licensee's proposed alternative to use Code Case N-546 with the 1995 Edition vision test requirements 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 third 10-year interval, which expires September 20, 2010, or until such time as Code Case N-546 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-546, it must follow all provisions in the subject code case with the limitations or conditions listed in RG 1.147, if any.

For RR-A10, RR-A11, RR-A12, and RR-A14, the licensee has proposed to use Code Cases N-566-1, N-598, N-623, and N-639, respectively, as proposed alternatives to the Code requirements. The staff concludes that the proposed alternatives provide an acceptable level of quality and safety. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year interval, which expires September 20, 2010, or until such time as Code Cases N-598, N-623, and N-639 are published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Cases N-598, N-623, and N-639, it must follow all provisions in the subject code cases with the limitations or conditions listed in RG 1.147, if any.

ENCLOSURE 2

Enclosure 2 includes the staff's evaluation of RR-A13. RR-A13, which implements Code Case N-528, provides an alternative to certain administrative requirements of Section III, when material is purchased, exchanged, or transferred between nuclear plant sites. The staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative to use Code Case N-528 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year interval, which expires September 20, 2010, or until such time as Code Case N-528 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-528, it must follow all provisions in the subject code case with the limitations or conditions listed in RG 1.147, if any.

ENCLOSURE 3

Enclosure 3 includes the staff's evaluation of RR-A18 and RR-A19. The staff concludes the licensee's proposed use of Davis-Besse Unit 1 Technical Specification 3/4.7.7 as an alternative to the ASME Code, Section XI, Subsection IWF-5200(a) and (b) and IWF-5300(a) and (b) for the examination and testing requirements for snubbers provides an acceptable level of quality and safety (RR-A18). In addition, the staff concludes that the requirements of IWA-2317 of the 1998 Edition of ASME Section XI as an alternative to the provisions of IWA-2313 and IWA-2314 for visual examination personnel performing VT-3 snubber examination provides an acceptable level of quality and safety (RR-A19). Therefore, the licensee's proposed alternatives with regard to the examination and testing of snubbers and VT-3 examination personnel qualifications are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year interval, which expires September 20, 2010.

ENCLOSURE 4

Enclosure 4 includes the staff's evaluation of RR-E1 through RR-E8. The staff concludes that, for RR-E2 and RR-E4, the licensee's proposed alternatives will provide an acceptable level of quality and safety. Therefore, the proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for Davis-Besse's third 10-year containment inservice inspection (ISI) interval, which expires September 20, 2010.

The staff concludes that for RR-E1, RR-E3, RR-E5 and RR-E7, compliance with the code requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the licensee's proposed alternatives will provide reasonable assurance of containment pressure integrity. Therefore, these proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for Davis-Besse's third 10-year containment ISI interval, which expires September 20, 2010.

For RR-E6, which implements Code Case N-604, the staff concludes that compliance with the code requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the licensee's proposed alternatives will provide reasonable assurance of containment pressure integrity. Therefore, these proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for Davis-Besse's third 10-year containment ISI interval, which expires September 20, 2010, and use of the code case is authorized until such time as the code case is published in a future version of RG 1.147. At that time, if the licensee intends to continue implementing this code case, it must follow all provisions of Code Case N-604 with limitations or conditions specified in RG 1.147, if any.

For RR-E8, which implements Code Case N-605, the staff concludes that the licensee's proposed alternatives will provide an acceptable level of quality and safety. Therefore, the proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for Davis-Besse's third 10-year containment inservice inspection (ISI) interval, which expires September 20, 2010, and use of the code case is authorized until such time as the code case is published in a future version of RG 1.147. At that time, if the licensee intends to continue implementing this code case, it must follow all provisions of Code Case N-605 with limitations or conditions specified in RG 1.147, if any.

This completes the staff's activities associated with TAC No. MB1607.

Sincerely,

**/RA/**

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: As stated

cc w/encl: See next page

The staff concludes that for RR-E1, RR-E3, RR-E5 and RR-E7, compliance with the code requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the licensee's proposed alternatives will provide reasonable assurance of containment pressure integrity. Therefore, these proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for Davis-Besse's third 10-year containment ISI interval, which expires September 20, 2010.

For RR-E6, which implements Code Case N-604, the staff concludes that compliance with the code requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the licensee's proposed alternatives will provide reasonable assurance of containment pressure integrity. Therefore, these proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for Davis-Besse's third 10-year containment ISI interval, which expires September 20, 2010, and use of the code case is authorized until such time as the code case is published in a future version of RG 1.147. At that time, if the licensee intends to continue implementing this code case, it must follow all provisions of Code Case N-604 with limitations or conditions specified in RG 1.147, if any.

For RR-E8, which implements Code Case N-605, the staff concludes that the licensee's proposed alternatives will provide an acceptable level of quality and safety. Therefore, the proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for Davis-Besse's third 10-year containment inservice inspection (ISI) interval, which expires September 20, 2010, and use of the code case is authorized until such time as the code case is published in a future version of RG 1.147. At that time, if the licensee intends to continue implementing this code case, it must follow all provisions of Code Case N-605 with limitations or conditions specified in RG 1.147, if any.

This completes the staff's activities associated with TAC No. MB1607.

Sincerely,
**/RA/**
Anthony J. Mendiola, Chief, Section 2
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Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: As stated
cc w/encl: See next page
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| SCoffin | TChan | GGrant, RIII | | | | |

*See TQuay to SBajwa memorandum dated 9/25/01
**See KManoly to AMendiola memorandum dated 2/25/02
***See DTerao to AMendiola memorandum dated 10/09/01
****See SCoffin to AMendiola memorandum dated 9/23/02 and TChan to AMendiola memorandum dated 5/30/02
*****See previous concurrence

**ADAMS ACCESSION NO. ML022700279**

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|---|---|---|---|---|
| NAME | BMoroney | THarris | TQuay* | KManoly** |
| DATE | | 09/ /02 | 09/25/01 | 02/25/02 |

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**OFFICIAL RECORD COPY**

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# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## THIRD 10-YEAR INTERVAL INSERVICE INSPECTION

## RELIEF REQUEST (RR) NOS. RR-A1, RR-A3 THROUGH RR-A8,

## RR-A10 THROUGH RR-A12, RR-A14 THROUGH RR-A17,

## RR-B1 THROUGH RR-B4, AND RR-C1

## DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

## FIRSTENERGY NUCLEAR OPERATING COMPANY

## DOCKET NO. 50-346

### 1.0 INTRODUCTION

By letter dated September 19, 2000, FirstEnergy Nuclear Operating Company (the licensee) submitted the Third 10-Year Interval Inservice Inspection (ISI) Program for the Davis- Besse Nuclear Power Station, Unit 1. Included in the submittal were 32 requests for relief from conformance with certain requirements of Section XI of the 1995 Edition and Addenda through the 1996 Addenda, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Additional information was provided in letters dated September 7 and November 27, 2001, February 6, August 1, August 9, September 3 and September 26, 2002, and electronic transmission of February 14, 2002. This safety evaluation will address Relief Request (RR) Nos. RR-A1,RR-A3 through RR-A8, RR-A10 through RR-A12, RR-A14 through RR-A17, RR-B1 through RR-B4, and RR-C1.

### 2.0 REGULATORY EVALUATION

Inservice inspection of the ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the Nuclear Regulatory Commission (NRC), if the licensee demonstrates that: (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 pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) 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. The Code of record for the Davis-Besse Nuclear Power Station, Unit 1, third 10-year ISI interval is the 1995 Edition through the 1996 Addenda of the ASME B&PV Code.

## 3.0 TECHNICAL EVALUATION

The information provided by the licensee in support of the request for relief from Code requirements has been evaluated and the basis for disposition is documented below.

### 3.1 Request for Relief No. RR-A1:

Code Requirement: ASME Code, Section XI, 1995 Edition and 1996 Addenda (the Code), Subsection IWB, Table 2500-1, Examination Category B-F and B-J, Item Numbers B5.10 and B9.11 require a volumetric and surface examination of welds.

Licensee's Code Relief Request: Examination Categories B-F and B-J of Table IWB-2500-1 of the 1995 Edition, 1996 Addenda of ASME Section XI requires the examination of Code Items B5.10 and B9.11 be in accordance with Figure IWB-2500-8. Figure IWB-2500-8 requires both a volumetric examination of the inner 1/3 T [thickness] and a surface examination from the outside (OD) surface. The volumetric examination can be performed from the inside (ID) of the nozzle using an automated remote inspection tool. However, the surface examination requires a manual magnetic particle examination from the OD surface. In accordance with 10 CFR 50.55a(a)(3)(i) and because of the high radiation levels at this location, the licensee requested that the required surface examination be replaced with an ultrasonic examination from the ID which is capable of detecting opposite side surface flaws. The system/component(s) for which relief is requested include:

*Reactor Vessel Inlet and Outlet Nozzles to Pipe Welds (Code Item B9.11):*

36" Outlet Reactor Nozzle (Z) to Pipe Weld (FW111B)
36" Outlet Reactor Nozzle (X) to Pipe Weld (FW111A)
28" Inlet Reactor Nozzle (Z/W) to Pipe Weld (FW56B)
28" Inlet Reactor Nozzle (Y/Z) to Pipe Weld (FW113B)
28" Inlet Reactor Nozzle (X/Y) to Pipe Weld (FW56A)
28" Inlet Reactor Nozzle (W/X) to Pipe Weld (FW113A)

*Reactor Vessel Nozzle to Core Flood Safe End Welds (Code Item B5.10):*

14" Core Flood Nozzle Safe-End Weld (WR-53-Y)
14" Core Flood Nozzle Safe-End Weld (WR-53-W)

Licensee's Basis for Requesting Relief: (as stated)
The surface examination would require approximately 40 man-hours for preparation of each of the reactor vessel inlet and outlet nozzle to pipe welds and an additional 10 man-hours of inspection time. Because radiation levels are anticipated to result in a

total exposure of 60 to 90 manrem and since shielding is impractical in this area, the requirements are, therefore, impractical for use at Davis-Besse. Because of these high radiation levels, the licensee requests that the required surface examination be replaced with an ultrasonic examination from the ID which is capable of detecting opposite side surface flaws.

A demonstration of opposite surface flaw detection capabilities was performed at Framatone [sic] Technologies' (FTI) Lynchburg, Virginia facility on August 9, 1989. The purpose of this demonstration was to define the capabilities of detecting the minimum size (through wall dimension) reflector originating at the opposite (OD) surface that could be detected during an actual automated remote reactor vessel examination from the ID of the component.

In order to determine the opposite surface flaw detection capabilities a test block was fabricated that contained a series of axial and circumferential OD notches ranging in depth from .024" to .353". The test block reflector design was based on ASME Section XI acceptance criteria for surface planar flaws. All notch sizes used for the purpose of this demonstration are considered as acceptable surface planar flaws per Section XI acceptance criteria.

The block was submerged in water to simulate the actual field application and was subsequently scanned with a contact 45 degree L-wave dual element transducer.

Examination criteria such as scan direction, scan motions, and transducer were duplicated as closely as possible to the actual examination which will be performed with an automated remote reactor vessel inspection tool.
Scanning was performed in both the circumferential and axial directions to detect the referenced notches.

The essential examination parameters required to reproduce these results will be incorporated, as required, into the technical procedure which will be utilized for the on site examination. These parameters include a 45 degree, medium damped, 2.25 mHz, 1 1/4" x 1 1/4", dual element, L-wave transducer focused at a 2" depth; scanned both parallel and perpendicular to the weld with scanning speed not to exceed 4"/sec.; indexing performed using a maximum of .2" index increment and a sample interval of at least .050".

Data was acquired and analyzed using the FTI data acquisition and imaging system (ACCUSONEX). The ultrasonic system was calibrated using the side drilled holes to establish a calibrated sweep range with a Distance Amplitude Correction (DAC) Curve.

The gain level for the DAC curve was used as a reference for the gain adjustment during this benchmark demonstration. To lower the recording threshold the DAC curve was adjusted to a 20 percent full screen height (FSH) flat recording threshold. The test block was then scanned several different times at increased gain levels until all of the notches were detectable. The result was all the notches were detectable at a gain level of 24 dB above reference with a recording threshold of 20 percent FSH. Due to the low material noise of the carbon steel test block excessive noise signals were not encountered allowing the use of such high gain levels.

The scanning sensitivity to be used for actual field examination will be determined in a similar manner using notches of various dimensions installed in the calibration block for the nozzle-to-pipe welds.

Table 1 shows the relationship of the notch dimensions to the maximum allowable flaw size permitted in this material. ASME Section XI acceptance standard IWB-3514 was used to make these determinations.

TABLE 1

AXIAL NOTCHES

| Notch | (l) Length | (a) Depth | a/l | a/t' | Maximum Allowable a/t |
|---|---|---|---|---|---|
| A | .380" | .125" | .33 | 5.2% | 13.8% |
| B | .375" | .062" | .17 | 2.6% | 10.9% |
| C | .376" | .024" | .06 | 1.0% | 8.6% |
| D | .623" | .251" | .40 | 10.5% | 13.8% |
| E | .623" | .124" | .20 | 5.2% | 11.7% |
| F | .623" | .050" | .08 | 2.1% | 9.1% |
| G | .826" | .175" | .43 | 14.7% | 13.8% |
| H | .826" | .175" | .21 | 7.37% | 12.0% |
| I | .826" | .043" | .05 | 1.87% | 8.6% |

CIRCUMFERENTIAL NOTCHES

| Notch | (l) Length | (a) Depth | a/l | a/t' | Maximum Allowable a/t |
|---|---|---|---|---|---|
| J | .375" | .130" | .35 | 5.4% | 13.8% |
| K | .376" | .065" | .17 | 2.7% | 10.9% |
| L | .375" | .031" | .08 | 1.3% | 9.1% |
| M | .624" | .255" | .41 | 10.6% | 13.8% |
| N | .625" | .129" | .21 | 5.4% | 12.0% |
| O | .627" | .054" | .09 | 2.3% | 9.2% |
| P | .826" | .352" | .43 | 14.7% | 13.8% |
| Q | .823" | .177" | .22 | 7.4% | 12.3% |
| R | .825" | .072" | .09 | 3.0% | 9.2% |

T' = 2.4"

MAXIMUM ALLOWABLE LENGTH FOR A FLAW DETECTED BY A SURFACE EXAMINATION IS 0.68"

Sizing scans looking for tip diffracted signals were performed on the opposite surface reflectors. The same transducer that was used for detection was used for the sizing scans.

Tip diffracted signals were detected with good reliability for sizing data on the notches equal to or greater than 10 percent through wall. Tip signals for the reflector at 5 percent through wall were detected but did not provide the resolution required to accurately size this small of a reflector.

The Performance Demonstration Initiative (PDI) does not address the examination of piping welds from the inside surface. It is The licensee's understanding that PDI will include the examination of piping from the inside surface as part of the qualification process for Supplement 12, Requirements for the Coordinated Implementation of Selected Aspects of Supplements 2, 3, 10, and 11, of Appendix VIII. Supplement 12 of Appendix VIII is required by 10 CFR 50.55a to be implemented by November 2002. Once Supplement 12 is implemented, the examination process used at Davis-Besse for examining the piping welds from the inside surface will be qualified in accordance with the PDI requirements.

Relief is request pursuant to 10 CFR 50.55a(a)(3)(i) as the proposed examination from the inside surface will provide an acceptable level of quality and safety. A relief request for these welds was previously approved for the Second 10-Year Interval in Relief Request RR-A1 (TAC Nos. M79034 and M77942).

In its letter dated November 27, 2001, in response to the NRC request for additional information the license stated:

If PDI is unsuccessful in qualifying piping examinations from the inside surface, the Davis-Besse Nuclear Power Station (DBNPS) will use the examination technique described in the relief request to perform the examinations. This examination technique was approved for the second interval (Request for Relief RR-A1 in the second interval) via TACs M79034 and M77942)

<u>Licensee's Proposed Alternative Examination</u>: (as stated)
The Reactor Vessel Nozzle to Pipe Welds and the Reactor Vessel Nozzle to Core Flood Safe-End Welds will be examined from the inside surface utilizing equipment and procedures which are capable of detecting the minimum size (through wall dimension) reflector originating at the opposite (OD) surface which could be detected during automated remote reactor vessel examination from the ID of the component. Once the PDI is capable of qualification of piping examinations from the inside surface, the process used for the examination will also be qualified in accordance with the PDI requirements.

<u>Staff Evaluation:</u>
The ASME Code Section XI examination requirements for the Reactor Vessel Inlet and Outlet Nozzle-to-Pipe and Core Flood Nozzle to Safe-End Welds includes both a volumetric examination of the inner 1/3 T (thickness) and a surface examination from the outside diameter (OD) surface.

The licensee proposed an alternative to the Code-required surface examination. The licensee proposed to replace the surface examination with an ultrasonic examination from the ID which is capable of detecting opposite side surface flaws. The licensee noted that the radiation levels are anticipated to result in a total exposure of 60 to 90 manrem, if the Code requirements are imposed.

The Performance Demonstration Initiative (PDI) at this time does not address the examination of piping welds from the inside surface. However, once Supplement 12 regarding examination of piping welds from the inside surface is implemented, the licensee will qualify its personnel in accordance with the PDI requirements.

The Code-required surface examination of the subject welds is hardship without a compensating increase in quality and safety. If required to perform the Code-required surface examination, maintenance and inspection personnel would receive excessive radiation exposure. The proposed alternative examination provides reasonable assurance that unallowable inservice flaws have not developed in the subject welds or that they will be detected and repaired prior to the return of the components to service. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval, which expires September 20, 2010.

3.2 Request for Relief No. RR-A3:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Paragraph IWA-5242(a) requires that for systems borated for the purpose of controlling reactivity, insulation shall be removed from pressure retaining bolted connections for VT-2 visual examination.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(ii), relief is requested from the removal of insulation from pressure retaining bolted connections when conducting the pressure tests and VT-2 visual examination for ASME Class 1 and 2 systems.

Licensee's Basis for Requesting Relief: (as stated)
There are approximately 60 insulated Class 1 bolted connections including the four Reactor Coolant Pumps. It is estimated that removal and replacement of the insulation will incur 20 man-rem of exposure. Since Class 1 pressure tests must be conducted with the plant in Hot Standby (Mode 3) to satisfy technical specification pressure/temperature limits, replacement of insulation must be conducted on components at their normal operating pressure (2155 psig) and temperature (532 oF). It is estimated that this would add at least two days to the critical path of a refueling outage, delaying return of the unit to service.

The intended purpose of removing insulation from bolted connections during pressure tests is to more readily detect boric acid leakage and potential corrosion of bolting. Boric acid leakage leaves a boric acid crystal residue when it evaporates. As the Class 1 leakage tests are conducted at normal operating pressure (2155 psig), the same leakage would occur during normal operation, as would be expected during the Class 1 leakage test. Leakage following normal operation would be evident due to the presence of boric acid residue. This residue would be visible during inspections conducted with the system depressurized.

Not all materials are susceptible to corrosion from boric acid leakage. ASME Code Case N-616 states that when corrosive resistant bolting material used has a chromium content greater than or equal to 10 percent, such as SA-564 Grade 630 H1100, SA-453 Grade 660, SB-637 (UNS N07718) or SB-637 (UNS N07750), it is permissible to perform the VT-2 examination without insulation removal. The similar corrosion resistance of 410 series stainless steels (such as SA-193 Grade 6) has been demonstrated in Electric Power Research Institute Reports NP-5769 and TR-104748.

Relief is requested pursuant to 10 CFR 50.55a(a)(3)(ii) as the removal of insulation at elevated pressures and temperatures would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. A relief request for performing the VT-2 visual examination of Class 1 bolted connections without the system pressurized was previously approved for the Second 10-Year Interval in RR-A7 (TAC Nos. M79034 and M77942).

Corrosion resistance bolting has been in service for many years. There have not been any incidents of corrosion of these materials noted. Removing and reinstalling insulation, including erection and removal of scaffolding when necessary to provide access, would require significant time and radiation exposure to facilitate examination for a condition which experience throughout the industry has shown to be very unlikely to occur.

The Code requires that a VT-2 visual examination be performed during system pressure tests at nominal operating pressures and temperatures for that system. The Class 1 pressure tests must be conducted with the plant in Hot Standby (Mode 3) to satisfy the technical specification pressure/temperature limits. Based on the safety implications of removing and/or reinstalling the insulation at elevated pressures and temperatures, the removal of insulation and the performance of the VT-2 visual examinations at least once per inspection period with the system depressurized is acceptable. Any leakage at the bolted connection would be evident from boric acid residue.

<u>Licensee's Proposed Alternative Examination</u>: (as stated)
Code Case N-616 will be implemented for the VT-2 examination of ASME Class 1 and 2 bolting. The VT-2 examination will be performed without insulation removal when corrosive resistant bolting material used in a pressure retaining bolted connection has a chromium content greater than or equal to 10 percent. Code Case N-616 will not be applied to:

1. A453 Grade 660 bolting that is pre-loaded to 85 percent of yield or greater.

2. Bolts made from A-193 Grade B6 material (Grade 410 stainless steel) tempered below 1100 °F.

3. Bolts made from SA-564 grade 630 material that were not hardened to H1100 condition.

For Class 1 and 2 pressure retaining connections with bolting that has a chromium content less than 10 percent, insulation will be removed to perform the VT-2 visual examination of the bolted connection for evidence of leakage. This VT-2 visual examination will be performed once each period. For Class 1 systems, the VT-2 examination will be performed without the connection being pressurized. For Class 2 systems, the VT-2 examination will be performed with the system pressurized.

At all bolting connections, included those containing corrosive resistant bolting material, if evidence of leakage is detected, either by discovery of active leakage or evidence of boric acid crystals, the insulation will be removed and the bolted connection will be reexamined and, if necessary, evaluated in accordance with the corrective measures of paragraph IWA-5250.

In the licensee's letter dated November 27, 2001, in response to NRC's Request for Additional Information the licensee stated:

The use of Code Case N-616 has been previously approved for Arkansas Nuclear One via TAC MB0665 and MB0694. This includes 410 series stainless steels. The Davis-Besse Nuclear Power Station does not intend to change the frequency of performing VT-2 examinations of Class 1 systems. In addition to using Code Case N-616, the request is to remove insulation on the components not covered by Code Case N-616 each period versus each refueling outage. If evidence of boric acid residue is found on one of the components not scheduled for insulation removal, the insulation will be removed in accordance with the Boric Acid Corrosion Control Program. Removal of insulation each period versus each refueling outage was approved in the second interval (Relief Request RR-A7 in the second interval) via TACs M79034 and M77942.

<u>Staff Evaluation</u>:
The ASME Code Section XI, 1995 Edition and 1996 Addenda, requires the removal of all insulation from pressure-retaining bolted connections in systems borated for the purpose of controlling reactivity when performing VT-2 visual examinations during system pressure tests. The Code requires this examination to be performed during each refueling outage for Class 1 systems, and each inspection period for Class 2 and 3 systems. The licensee proposed to use Code Case N-616, "Alternative Requirements for VT-2 Visual Examination of Classes 1, 2, and 3 Insulated Pressure Retaining Bolting Connections, Section XI, Division 1," in lieu of the Code requirements.

The staff has developed a position over the years on the use of AISI Type 17-4 PH stainless steel (SA-564 Grade 630), AISI Type 410 stainless steel (SA-193 Grade 6), and A-286 stainless steel (SA-453 Grade 660) fasteners. The 17-4 PH stainless steel and the 410 stainless steel are suitable for use in contact with primary water if they are aged at a temperature of 1100 °F or higher. If they are aged at a lower temperature, they become susceptible to primary water stress corrosion cracking. The hardness of these alloys should be below Rc 30 on the Rockwell hardness scale if they are properly heat treated. A-286 stainless steel is susceptible to stress corrosion cracking in primary water, particularly if preloaded above 100 thousand pounds per square inch (ksi). Bengtsson and Korhonen of ASEA-ATOM, Vasteras, Sweden, examined the behavior of A-286 in a boiling water reactor (BWR) environment, as reported in the Proceedings of the International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors, August 22-25, 1983, Myrtle Beach, South Carolina, sponsored by the National Association of Corrosion Engineers, the Metallurgical Society of American Institute of Mining, Metallurgical, and Petroleum Engineers, and the American Nuclear Society. They found that A-286, in comparison to other tested materials, was more susceptible to intergranular stress corrosion cracking in BWR water than any other material they tested. They also found that A-286 is less likely to crack as the applied stress is reduced. Piascik and Moore from Babcock & Wilcox reported a number of vessel internals bolt failures of A-286 bolts in pressurized-water reactor water in *Nuclear Technology*, Vol. 75, December 1986. They correlated the failures with bolt fillet peak stress and found that bolts preloaded below 100 ksi showed no failures.

The NRC staff position is that any 17-4 PH stainless steel or 410 stainless steel stud or bolt aged at a temperature below 1100 °F or with hardness above Rc 30 on the Rockwell hardness scale must have the thermal insulation removed for VT-2 examination during the system pressure test. For A-286 stainless steel studs or bolts, the preload must be verified to be below 100 ksi or the thermal insulation must be removed and the joint visually inspected. For nuts conforming to SA-194, experience indicates it would not be necessary to remove the thermal insulation for visual inspection.

The licensee noted that Code Case N-616 will not be applied to:

a) A453 Grade 660 bolting that is pre-loaded to 85 percent of yield or greater.

b) Bolts made from A-193 Grade B6 material (Grade 410 stainless steel) tempered below 1100 °F.

c) Bolts made from SA-564 grade 630 material that were not hardened to H1100 condition.

For Class 1 and 2 pressure retaining connections with bolting that has a chromium content less than 10 percent, the licensee will remove the insulation to perform the VT-2 visual examination of the bolted connection for evidence of leakage, and this examination will be performed once each period. For Class 1 systems, the licensee will perform VT-2 examinations without the connection being pressurized and, for Class 2 systems, the VT-2 examinations will be performed with the system pressurized.

The licensee noted that, if evidence of leakage is detected at any bolted connection, including those containing corrosion resistant bolting material, either by discovery of active leakage or evidence of boric acid crystals, the insulation will be removed and the bolted connection will be reexamined and, if necessary, evaluated in accordance with the corrective measures of paragraph IWA-5250.

Code Case N-616 has eliminated the requirement to remove the insulation from ASME Classes 1, 2, and 3 pressure retaining bolted connections when conducting a VT-2 examination if corrosion resistant bolting is used. However, the code case does not include the requirement to hold the system at operating pressure and temperature for a minimum of 4 hours. The licensee provided clarifying information regarding hold times prior to VT-2 examinations for Class 1 and 2 systems. The licensee stated that for Class 1 bolted connections, the system will have been pressurized for the entire operating cycle prior to the VT-2 examinations. For Class 2 bolted connections, the system will be either maintained at nominal operating pressure for 4 hours or the insulation will be removed for the VT-2 examinations of the bolted connections.

The staff has determined that the removal of insulation at elevated pressures and temperatures for the purpose of examining bolted connections utilizing corrosion resistant bolting material would result in a hardship without a compensating increase in the level of quality and safety. Furthermore, the staff determined that the licensee will perform the VT-2 examinations for Class 1 bolted connections after the system has been pressurized for the entire operating cycle and, for Class 2 bolted connections, the system will be maintained at nominal operating pressure for 4 hours prior to the VT-2 examinations. Significant leakage, if any, would penetrate the insulation and be detected. In addition, periodic removal of the insulation for VT-2 examination, even under cold and non-pressurized conditions, should allow for detection of even minor leakage in a timely manner via the presence of boric acid crystals or residue. Therefore, the licensee's proposed alternative provides reasonable assurance of structural integrity of the bolted connections.

For RR-A3, the staff concludes that the removal of insulation at elevated pressures and temperatures would result in a hardship without a compensating increase in the level of quality and safety. Furthermore, the staff concludes that the licensee's proposed alternative provides reasonable assurance of structural integrity of the bolted connections. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year interval, which expires September 20, 2010, or until such time as Code Case N-616 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-616, it must follow all provisions in the subject code case with the limitations or conditions listed in RG 1.147, if any.

3.3 Request for relief No. RR-A4:

Code Requirement: ASME Code Section XI, 1995 Edition, 1996 Addenda, Subsection IWB, Table IWB-2500-1, Examination Category B-J, Item No. B9.31 (Branch Pipe Connection Welds NPS [nominal pipe size] 4" or Larger) requires a surface and volumetric examination of 25 percent of the population of branch connection welds during the 10-year interval. This relief request is applicable to two Reactor Coolant Piping Branch Connection Welds, which includes one 12" Decay Heat Nozzle and one 10" Surge Line Nozzle. The examination requirements are identified in Figure No. IWB-2500-9, -10, and -11. Essentially, 100 percent of the weld length requires examination.

Code Case N-460 states that, when the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in examination coverage may be accepted provided the reduction in coverage for that weld is less than 10 percent.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from performing a volumetric examination of 100 percent of the weld length. The reduction in examination coverage of the weld length is greater than 10 percent.

Licensee's Basis for Requesting Relief: (as stated)
The volumetric examination of the identified welds is limited by the component geometry such that the reduction in coverage is greater than 10 percent. The surface examination is not limited.

The branch connection weld configuration is similar to Figure IWB-2500-9. Article III-4000 requires a total of four scans for complete examination coverage.

Scan 1 - A circumferential scan clockwise around the nozzle for reflectors traverse to the weld.
Scan 2 - A circumferential scan counter-clockwise around the nozzle for reflectors traverse to the weld.
Scan 3 - An axial scan from the pipe surface around the nozzle for reflectors parallel to the weld.

Scan 4 - An axial scan from the nozzle surface around the nozzle for reflectors parallel to the weld.

The Reactor Coolant Piping is cladded, which limits the examination path to a one-half vee sound path.

Each branch connection weld can be completely (100 percent) examined circumferentially in both directions. It can also be examined completely (100 percent) in the axial direction from the pipe surface. However, no reliable scan can be performed from the nozzle side due to the nozzle radius interfering with the examination scan. The Reactor Coolant Piping cladding also limits the ability to “bounce” the ultrasonic beam from the pipe side of the weld to obtain coverage in the fourth beam direction. This results in examination of approximately 75 percent of the required examination volume. The attached sketch provides an illustration of the typical nozzle configuration and examination coverage.

The examination volume is examined in at least one direction to detect reflectors in both the parallel and traverse directions to the weld. This should detect any defects that may exist.

Additional welds exist for this examination category item, but are in less critical locations or are subject to less severe service conditions. However, these welds are similar in configuration and would also require relief if they were selected for examination during the present 10-year interval.

Relief is request [sic] pursuant to 10 CFR 50.55a(a)(3)(i) as the proposed examination will provide an acceptable level of quality and safety as reflectors oriented both parallel

and traverse to the weld can be located. A relief request for these welds was previously approved for the Second 10-Year Interval in Relief Request RR-A10 (TAC No. M93310).

<u>Licensee’s Proposed Alternative Examination</u>: (as stated)
Each weld will be examined in the circumferential direction in accordance with ASME Code requirements. Each weld will be examined in the axial direction from the pipe surface only. The surface examination will be performed as required by the ASME Code.

<u>Staff Evaluation:</u>
To meet Code requirements, the subject welds must be examined in four directions, including two axial directions, or from one side using extended beam paths to obtain two-directional coverage. The nozzle geometry for these branch connection welds precludes axial scanning from the nozzle side of the weld. Extending the beam path to obtain two-directional coverage from one side is not possible, due to the stainless steel cladding on the inside surface of the reactor coolant piping causing interference with the “bounce” capability of the ultrasonic beam from the pipe side of the welds. Significant design modifications would be required to allow access for examination of these two welds. Imposition of the required design modifications would create a significant burden for the licensee. Therefore, the Code requirements are impractical.

Review of the licensee’s proposed alternative examination shows that a significant portion (approximately 75 percent) of each weld will receive a volumetric examination, and an axial scan of each weld will provide approximately 50 percent coverage. Consequently, any significant patterns of degradation should be detected. Therefore, performing the surface examination to ASME Code and the limited volumetric examination provides reasonable assurance of the structural integrity of these two branch connection welds. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i) relief is granted for the third 10-year ISI interval, which expires September 20, 2010.

3.4 Request for relief No. RR-A5:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Subsection IWB, Table IWB-2500-1, Examination Category B-D, Item Number B3.110 requires a volumetric examination of five pressurizer nozzle to vessel welds (one pressurizer spray nozzle to upper pressurizer head weld, three pressurizer relief nozzle to upper pressurizer head welds, and one pressurizer surge line nozzle to lower head weld).

Subsection IWB, Table IWB-2500-1, Examination Category B-D, Item Number B3.130 requires a volumetric examination of six steam generator (primary side) nozzle to vessel welds (four steam generator outlet nozzle to lower head welds and two steam generator inlet nozzle to upper head welds).

The requirements for these examinations are identified in Figure No. IWB-2500-7(a).

Code Case N-460 states that when the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in examination coverage may be accepted provided the reduction in coverage for that weld is less than 10 percent.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from performing a volumetric examination of essentially 100 percent of the weld length. The reduction in examination coverage of the weld is greater than 10 percent.

Licensee's Basis for Requesting Relief: (as stated)
Paragraph T-441.1, Article 4, Section V of the 1995 Edition, 1996 Addenda of ASME Section XI requires the examination volume be scanned with angle beam search units directed both at right angles to the weld axis and along the weld axis. Wherever feasible, each examination shall be performed in two directions. The adjacent base metal in the examination volume must be completely scanned by two angle beams, but need not be completely scanned by both angle beams from both directions. The examination volume for this weld is defined in Figure IWB-2500-5. The pressurizer and steam generator heads are cladded which limits the examination path to a one-half vee sound path.

The base material coverage for each nozzle examination volume requires two beam angles from at least two directions (parallel and perpendicular to the weld axis). Examination of the weld material requires two beam angles from four directions (parallel and perpendicular to the weld axis in opposing directions). In addition to the angle beam pair, a near surface search unit will be used to examine the near surface. Maximum coverage is obtained by scanning to the extent possible from the head, blend radius, and nozzle barrel. The limiting conditions are due to the nozzle configuration and internal cladding which limits the ability to "bounce" the ultrasonic beam from the inside surface of the vessel. The approximate examination coverage obtained for each of the nozzle-to-vessel welds are as noted in the following table.

| Component | Examination Coverage |
|---|---|
| Pressurizer Spray Nozzle to Upper Pressurizer Head Weld | 72% |
| Pressurizer Relief Nozzle to Upper Pressurizer Head Weld | 60% |
| Pressurizer Surge Nozzle to Lower Pressurizer Head Weld | 69% |
| Steam Generator Outlet Nozzle to Lower Head Weld | 75% |
| Steam Generator Inlet Nozzle to Upper Head Weld | 72% |

Relief is requested pursuant to 10 CFR 50.55a(a)(3)(i) as the proposed examination will provide an acceptable level of quality and safety as reflectors orientated both parallel and transverse to the weld can be located. A relief request for these welds was previously approved for the Second 10-Year Interval in Relief Request RR-A11 (TAC No. M93310).

In its letter dated November 27, 2001, in response to the NRC request for additional information the license stated:

Relief Requests RR-A2 and RR-A5 request relief pursuant to 10 CFR 50.55a(a)(3)(i) as greater than 90 percent of the examination volume cannot be examined.

The components for which relief is requested are carbon steel vessels with stainless steel cladding on the inside surface. Due to this cladding, the ultrasonic beam cannot be "bounced" from the inside clad surface to increase the examination coverage. The ultrasonic examination is conducted in accordance with Section XI, Appendix I of the ASME- Code, 1995 Edition through the 1996 Addenda. Section XI, Appendix I states that for vessels other than the Reactor Vessel greater than 2" in thickness, the ultrasonic examination shall be conducted in accordance with Article 4 of Section V. The following discussion illustrates how The licensee determines the volume examined during ultrasonic examinations.

Article 4, Section V of the ASME Code, 1995 Edition, 1996 Addenda requires the weld and adjacent base metal to be examined using nominal angles of 45 and 60 degrees, (deviation is permitted if geometry limits the coverage, however, separation of angles must be a least 10 degrees) and a straight beam. Four basic scan directions are required for the angle beams. Two perpendicular to the weld axis (axial scan) from opposite directions and two parallel to the weld axis (circumferential scan) from opposite directions. These requirements apply for each of the angle beams used (i.e. 45 and 60 degrees). Each of the 45 and 60 degree angle beams is required to pass through all of the weld volume in the four basic scan directions. However, the adjacent base metal scanning requirements allow the two beam angles to pass through, in only one direction each for the axial and circumferential scans.

The following methodology is used to determine the extent of examination coverage.

1. A scaled cross sectional drawing of the component configuration, extent of coverage and the area of interest is drawn using a Computer Aided Design Drafting (CADD) program. The examination area is divided into 3 zones. Zones 1 and 3 are the base material on either side of the weld. Zone 2 is the weld material.

2. As noted above, Zones 1 and 3 require 5 scans (45 and 60 degrees from 1 axial and 1 circumferential and a straight beam minimum) while Zone 2 requires 9 scans (45 and 60 degrees from 2 axial and 2 circumferential directions and a straight beam). Each scan is assigned a weighting factor to be used in the determination of the overall examination coverage. For example, the axial scan of the Zone 1 base material for reflectors parallel to the weld (axial scan) consists of 2 angle beam scans from one direction. This represents 2 of the 5 (40 percent) base metal scans in the Zone 1 area. Therefore, the axial scan in Zone 1 is assigned a weighting factor of 0.40. Similarly, weighting factors for the other scans are determined as follows:

| Zone 1 (5 Scans) | Zone 2 (9 Scans) | Zone 3 (5 Scans) |
|---|---|---|
| Axial = 40% (0.40) | Axial = 44% (0.44) | Axial = 40% (0.40) |
| Circ = 40% (0.40) | Circ = 44% (0.44) | Circ = 40% (0.40) |
| 0 degrees = 20% (0.20) | 0 degrees = 12% (0.12) | 0 degrees = 20% (0.20) |

3. The examination coverage (i.e., the amount of the sound beam that passes through each zone) is plotted on the CADD drawing for each of the ASME Code required scans. The area covered in each zone by the axial, circumferential, and straight beam examinations is then measured by CADD. If the area covered received all the required scans, it is considered 100 percent complete. If it received one-half of the required scans, it is considered 50 percent complete, etc. This area is then multiplied by the weld length to determine the examination volume covered.

4. The examination volume covered in each zone by the axial, circumferential, and straight beam scans is multiplied by the weighting factor. After applying a weighting to each scan, they are added together and divided by the total area for that zone to determine the percent complete for the zone. Then all the 3 zones are added together and divided by 3 to determine the total examination coverage.

5. When the total examination coverage is less than 90 percent, additional angles, such as 70 degrees and 35 degrees are plotted to determine if they will increase the examination coverage. Based on this determination and the principals of as low as reasonably achievable (ALARA), additional scans beyond those required by the ASME Code are performed to increase examination coverage when considered necessary.

The configuration of the nozzle to vessel welds is similar to that shown in Figure IWB-2500-7(a). Attachment 5 [of the licensee's letter dated November 27, 2001] shows the weld profiles and limitations for the Pressurizer Spray Nozzle. The scans and limitations shown for this nozzle are typical of the other nozzles covered in Relief Request RR-A5.

Licensee's Proposed Alternative Examination: (as stated)
Each weld will be examined in two opposing directions from the circumferential and axial directions to the extent possible.

Staff Evaluation:
To meet the Code requirements, the subject welds must be examined in four directions, including two axial directions, or from one side using extended beam paths to obtain two-directional coverage. The nozzle geometry for these vessel-to-nozzle welds precludes axial scanning from the nozzle side of the weld. Extending the beam path to obtain two-directional coverage from one side is not possible, due to stainless steel cladding on the inside surface of the Reactor Coolant System Piping causing interference with the "bounce" capability of the ultrasonic beam from the pipe side of the welds. These conditions make the Code coverage requirements impractical for these welds. Significant design modifications would be required to allow access for examination of these eleven welds. Imposition of the necessary design modifications would create a significant burden for the licensee.

Review of the licensee's proposed alternative examination shows that, in addition to the Code required surface examinations, a significant portion (approximately 60 percent) of each weld will receive a volumetric examination. Consequently, any significant patterns of degradation should be detected. As a result the limited volumetric examination provides reasonable assurance of the structural integrity of the subject vessel-to-nozzle welds. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval, which expires September 20, 2010.

3.5 Request for relief No. RR-A6:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Subsection IWB-2500, Table IWB2500- 1, Examination Category B-P, Item No. B15.50 requires a system leakage test in accordance with IWB-5220. IWB-5222(b) requires the pressure retaining boundary during the system leakage test conducted at or near the end of the inspection interval to extend to all Class 1 pressure retaining components within the system boundary.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(ii), relief is requested from performing the system leakage test at or near the end of the interval for a segment of Class 1 piping approximately 4 feet in length located between isolation valves DH 11 and DH 12. Valves DH 11 and DH 12 are installed in the normal cooldown line from the Reactor Coolant System to the Decay Heat Removal System.

Licensee's Basis for Requesting Relief: (as stated)
Valves DH 11 and DH 12 are installed in the normal cooldown line from the Reactor Coolant System to the Decay Heat Removal System. DH 12 is the first isolation valve off of the Reactor Coolant System while DH 11 is the second isolation valve. The ASME Class 1 boundary extends through DH 12 to DH 11. The piping between DH 11 and DH 12 is a four foot section of 12 inch seamless schedule 140 piping with a 1 inch drain line and a 3/4 inch by-pass line around DH 12. Two 12-inch pipe welds, which connect the piping to the valves, are the only large bore welds within the boundary. This piping is installed within a watertight enclosure (Decay Heat Pit) which protects DH 11 and DH 12 from flooding following a Loss-of-Coolant Accident.

Valves DH 11and DH 12 are motor operated gate valves with flexible wedges. NRC Information Notice 92-26 "Pressure Locking of Motor Operated Flexible Wedge Gate Valves" provides information regarding flexible wedge gate valves which may be subject to pressure locking under certain operating conditions. The licensee's evaluation of this Information Notice identified that DH 11 and DH 12 were susceptible to pressure locking. Subsequently, DH 11 and DH 12 were modified by drilling a small vent hole in the upstream disc of each valve's wedge to eliminate the pressure locking concern.

The required system leakage test pressure of the piping between DH 11 and DH 12 is 2155 psig. During the eleventh refueling outage, the piping between DH 11 and DH 12 was pressurized during Mode 5 to determine if this test pressure could be achieved. At 800 psig, the pressure locking vent hole allowed leakage past the DH 12 upstream disc. This confirmed that in order to perform the system leakage test of the piping between DH 11 and DH 12, the Reactor Coolant System must be pressurized to prevent flexing of the DH 12 wedge. Compliance with the pressure/temperature limits of DBNPS Technical Specification 3.4.9.1 requires the plant enter Mode 3 prior to attaining the Reactor Coolant System pressure of 2155 psig required to perform the system leakage test.

Three options are available to perform the system leakage test of the piping between valves DH 11 and DH 12.

Option one is to open valve DH12 while in Mode 3 at full temperature and pressure to pressurize the piping between DH 11 and DH 12 to 2155 psig. The piping could then be examined remotely through the Decay Heat Pit Inspection Port. Opening DH 12 while in Mode 3 is not considered viable as Technical Specification 3.5.2 prohibits DH 11 or DH 12 from being open when the Reactor Coolant System pressure is greater than 328 Psig. Therefore, DH 12 may not be opened to pressurize the piping during the normal Reactor Coolant System hydrostatic test.

The second option requires entry into the Decay Heat Pit while in Mode 3 with the Reactor Coolant System at full temperature and pressure to pressurize the piping between DH 11 and DH 12 with a hydrostatic pump. Entering the Decay Heat Pit while in Mode 3 requires removal of the Decay Heat Pit cover to provide access to the Decay Heat Pit with the Reactor Coolant System at full temperature (532 °F) and pressure (2155 psig). The system leakage test would be performed by pressurizing the piping between DH 11 and DH 12 with a hydrostatic pump to obtain the required test pressure. Technical Specification 3.5.2 requires the Decay Heat Pit be sealed and closed in Modes 1, 2, and 3. Opening this pit in Mode 3 requires an intentional entry in Technical Specification 3.0.3 as it would make both Low Pressure Injection pumps inoperable. As Technical Specification 3.0.3 is not intended to be used as an operational convenience to permit redundant safety systems to be out of service, this is not an acceptable option.

The licensee considers these two options a hardship and a detriment to the quality and safety of the Reactor Coolant System.

The third option requires the reactor vessel to be defueled and the RCS drained down to disassemble DH12 and perform a temporary modification to the disk to establish test conditions. The licensee considers this option to be a hardship and unusually difficult without a compensatory increase in the level of quality and safety.

Testing of the piping between DH11 and DH 12 during the system leakage test would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. A relief request for this piping between DH 11 and DH 12 was previously approved for the Second 10-Year Interval in Relief Request RR-A16 (TAC No. MA4549).

In its letter dated November 27, 2001, in response to the NRC request for additional information the license stated:

The upstream portions of this system that contains similar materials and more severe environmental and operating conditions are examined during the Class I leakage test. This Relief Request RR-A6 does contain a proposed alternative examination (page 221) A similar relief request was previously approved for the second interval (Relief Request RR-A16 in the second interval) via TAC MA4549.

<u>Licensee's Proposed Alternative Examination:</u> (as stated)
The required system leakage test pressure for the piping between DH 11 and DH 12 is 2155 psig. In lieu of the system leakage test at the required pressure, the segment of piping between DH 11 and DH 12 will be VT-2 examined, with the insulation removed, for boric acid residue which would be indicative of leakage. The piping will have undergone a 4-hour hold at greater than 200 psig and several days of operation at Decay Heat Removal pressures (approximately 45 psig). Since the Decay Heat System contains borated water, any pressure boundary leak would be identified by the formation of boric acid crystals at the location of leakage.

As noted above, the piping between DH 11 and DH 12 is a four foot section of seamless piping connected to DH 11 and DH 12 by field welds. In addition to the VT2 examination, each of these two field welds will be subjected to a surface examination and a volumetric examination during each In-service Inspection Interval. The volumetric examination is a limited examination as the valve taper prevents ultrasonic scanning from the valve side.

<u>Staff Evaluation</u>: Based on the requirements of the technical specifications, the system leakage test of the segment of piping between valves DH 11 and DH 12 would require the Reactor Vessel to be defueled and the Reactor Coolant System drained below the level of valve DH 12 to perform the necessary modification to the valve disk prior to pressurizing for testing purposes the 4-foot section of seamless stainless steel piping between valves DH 11 and DH 12. This option would present significant hardship for the licensee without a compensating increase in quality or safety level.

Review of the licensee's proposed alternative examination shows both pressure boundary welds between the valves and the section of pipe will be subjected to a limited volumetric examination and a surface examination in accordance with the ASME Code Section XI. Although this is a limited examination, examination in three of the four required examination directions should detect any significant degradation or flaw propagating through the weld. This examination provides reasonable assurance of the structural integrity of the welds.

During operation of the Decay Heat System, valves DH 11 and DH 12 are normally opened before the Reactor Coolant System pressure reaches 260 psig during a Reactor Coolant System cooldown in order to meet Technical Specification 3.4.2 requirements for Low Temperature Overpressure Protection. During the cooldown, the pressure of the piping between valves DH 11 and DH 12 will be greater than 200 psig for at least 4 hours. This is followed by long term cooling at Decay Heat Removal pressure of approximately 45 psig. Walkdown of the Decay Heat System section of piping between DH 11 and DH 12 after a 4-hour cooldown period for boric acid residue provides a sufficient means of detecting any leakage.

Therefore, for the licensee to perform the ASME Code Section XI requirements would be a hardship without a compensating increase in the level of quality and safety. The licensee's proposed alternative provides reasonable assurance of structural integrity of the subject components. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval, which expires September 20, 2010.

3.6 Request for relief No. RR-A7:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, IWB-5222(b) requires the pressure retaining boundary during the system leakage test conducted at or near the end of each inspection interval be extended to all Class 1 pressure retaining components within the system boundary.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(ii), relief is requested from performing the system leakage test of the Class 1 piping and valves downstream of the first isolation valve of small diameter ( $\leq$ 1 inch) vent, drain and instrument piping during the system leakage test conducted at or near the end of the inspection interval.

Licensee's Basis for Requesting Relief: (as stated)
Vent, drain, and instrument piping segments consist of two manually operated isolation valves separated by a short pipe nipple which is connected to the Reactor Coolant System by another short pipe nipple. Manually operated isolation valves provide double isolation of the Reactor Coolant System and are closed during normal operating conditions.

The System Leakage Test is performed in Mode 3 with the Reactor Coolant System at full temperature and pressure. Performance of the System Leakage Test conducted at or near the end of the interval requires the first isolation valve be opened to pressurize the piping between the first and second isolation valves. Following completion of the test, the first isolation valve must then be closed to restore the double isolation of the Reactor Coolant System. The licensee proposes to perform the system leakage test of the Reactor Coolant System conducted at or near the end of the interval with the first isolation valve in its normal closed position. This will still provide an acceptable level of quality and safety based on the following:

1. ASME Section XI paragraph IWA-4540 provides the requirements for the hydrostatic pressure testing of piping and components following repair/replacement. IWA-4540(b)(5) exempts component connections, piping, and associated valves that are nominal pipe size (NPS) 1" and smaller from system hydrostatic tests following repair/replacement.

2. The non-isolable portion of the Reactor Coolant System drain and vent connections will be pressurized and visually examined as required. Only the isolated portion of the small diameter drain and vent connections will not be pressurized.

3. The vent and drain piping and valves are nominally heavy wall (Schedule 160 pipe and 1500# valves) installed to the requirements of Subsection NB of ASME Section III.

The Davis-Besse Nuclear Power Station Technical Specifications (TS 3.4.6) requires Reactor Coolant System leakage monitoring during normal operation (Modes 1, 2, 3, and 4). Should any of the Technical Specification limits be exceeded, corrective actions, including plant shutdown, are required to identify the source of leakage and restore the Reactor Coolant System boundary integrity.

Personnel safety and ALARA issues are also associated with pressurizing these connections. These issues are as follows:

1. Pressure testing these connections to the outboard isolation valve requires the inboard isolation valve be opened to subject the isolable portion of the piping to Reactor Coolant System nominal operating pressure and temperature. Opening this inboard isolation valve under Reactor Coolant System full temperature and

2. temperature conditions is contradictory to the 10 CFR 50.55a(c)(2)(ii) requirement for double isolation of the Reactor Coolant System and thus creates the possibility for safety concerns for personnel performing the visual examination of the connections.

3. Performing the system leakage test with the inboard isolation valves open requires several man-hours to position the valves for the test and then to restore them after the test is complete. It is estimated that the dose associated with this valve alignment and realignment is approximately 0.4 man-rem.

The system leakage test is performed near the end of the outage at full temperature and pressure following completion of all Reactor Coolant System work. This system leakage test is a critical path activity. To minimize the time the Reactor Coolant System does not have double isolation, the alignment and realignment of the isolation valves is performed immediately preceding and following the test. This activity directly adds to the time necessary to perform the system leakage test and the duration of the outage.

The licensee considers the requirement to pressurize the downstream portions of small diameter vent, drain, and instrument piping a burden that is not compensated by a significant increase in quality and safety. Therefore, relief from this requirement is requested in accordance with 10 CFR 50.55a(a)(3)(ii). A relief request for not pressurizing small diameter ( $\leq$ 1 inch) vent, drain, and instrument piping during the system hydrostatic test was previously submitted for the Second 10-Year Interval in Relief Request RR-A18.

In its letter dated November 27, 2001, in response to the NRC request for additional information the license stated:

The upstream portions of these components that contain similar materials and more severe environmental and operating conditions are examined during the Class 1 leakage test. A VT-2 examination will extend to and include the outboard closed valve in the RCS boundary. This relief request was recently (June 4, 2001) approved for the second interval (Relief Request RR-A18 in the second interval) via TAC MA7210.

Licensee's Proposed Alternative Examination: (as stated)
The System Leakage Test conducted at or near the end of the interval will be conducted with the small diameter ($\leq$ 1 inch) vent, drain, and instrument piping in its normal operating condition.

Staff Evaluation: Boundaries for the system leakage test conducted at or near the end of each inspection interval have been extended to all Class 1 pressure retaining components within the system boundary, per the 1996 Addenda of ASME Section XI. This extends the boundary out to the second isolation valve on the Reactor Coolant System. In order to comply with the pressure and temperature of the Davis-Besse Unit No. 1 technical specifications, the plant must be in Mode 3 to obtain the necessary pressure, which requires valves to be in their normal position during heat-up to Mode 3. Once the system

leakage pressure is obtained, the inboard valves, which are normally closed, would have to be opened to extend the examination boundary to the second isolation valve.

The licensee noted that opening the inboard isolation valve is contrary to the double isolation valve philosophy of 10 CFR 50.55a. Since the small diameter vent, drain, and instrument piping have been installed in accordance with the requirements of Subsection NB of ASME Section III, the licensee felt that these requirements provide assurance of the pressure boundary integrity of this piping, and failure of this piping is highly unlikely. The staff concurs with the licensee that opening the inboard isolation valve is contrary to the double isolation valve philosophy of 10 CFR 50.55a and based on industry experience that failure of the subject piping is highly unlikely. Therefore, the staff has determined that the requirement to pressurize the downstream portions of small diameter vent, drain, and instrument piping is a hardship without a compensating increase in the level of quality or safety.

The licensee's proposed alternative to perform a system leakage test at or near the end of the interval with the small diameter ($\leq$1 inch) vent, drain, and instrument piping in its normal operating condition provides reasonable assurance of structural integrity of the subject components. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval, which expires September 20, 2010.

3.7 Request for relief No. RR-A8:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Appendix I, Paragraph I-2110 requires the examination of the Reactor Vessel-to-Flange Weld be conducted in accordance with Article 4 of Section V, as supplemented by Table I-2000-1.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested to use the requirements of Appendix VIII of the 1995 Edition, 1996 Addenda of ASME Section XI using the PDI protocol during the 10-Year examination of the Reactor Vessel.

Licensee's Basis for Requesting Relief: (as stated)
Paragraph I-2110 of Appendix I of the 1995 Edition, 1996 Addenda of ASME Section XI requires ultrasonic examination procedures, equipment, and personnel used for reactor vessel shell welds be qualified by performance demonstration in accordance with Appendix VIII. However, this paragraph excludes reactor vessel to flange welds from the qualification requirements of Appendix VIII.

The configuration of the Davis-Besse Nuclear Power Station Reactor Vessel Flange to Shell Weld permits full examination coverage of the weld from both sides of the weld using the same ultrasonic scanning equipment and techniques as used for the vessel shell to shell welds. This equipment is qualified to examine the Reactor Vessel Flange to Shell Weld as well as the reactor vessel shell welds to the requirements of Supplements 4 and 6 of Appendix VIII using the PDI protocol.

Examination utilizing Appendix VIII in lieu of Article 4 of ASME Code Section V provides a more effective examination that has been proven through the PDI qualification process to detect flaws which could affect the integrity of the reactor vessel. This is substantiated in the Backfit Analysis for the Federal Register (FR-99-24256) amendment to 10 CFR 50.55a, which states that examinations performed to Appendix VIII, as modified by the PDI program greatly increases the reliability of the detection and sizing of cracks and flaws.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). The examination of the Reactor Vessel Flange to Shell Weld to the requirements of Appendix VIII using the PDI protocol will provide an increase in the level of quality and safety. A relief request to utilize Appendix VIII for the Reactor Vessel flange-to-shell weld was previously submitted for the Second 10-Year Interval in Relief Request RR-A18.

Licensee's Proposed Alternative Examination: (as stated)
The Reactor Vessel Flange-to-Shell Circumferential Weld (Weld Number RC-RPV-WR-19) will be examined in accordance with the requirements of Supplements 4 and 6 of Appendix VIII of the 1995 Edition, 1996 Addenda of ASME Section XI using the PDI protocol.

Staff Evaluation: The licensee noted that Appendix I of the 1995 Edition (1996 Addenda) of ASME Section XI excludes the reactor vessel to flange welds from the qualification requirements of Appendix VIII. However, the configuration of the Davis-Besse Nuclear Power Station Reactor Vessel Flange to Shell Weld permits examination of the weld with ultrasonic scanning equipment and techniques qualified to the requirements of Supplements 4 and 6 of Appendix VIII using the PDI protocol.

The licensee proposed as an alternative to use the PDI qualification process for the examination of the Reactor Vessel Flange-to-Shell Circumferential Weld (Weld Number RC-RPV-WR-19). The PDI qualification process will enhance the overall level of assurance of the reliability of ultrasonic examination techniques in detecting and sizing flaws in the subject weld, because use of the PDI protocol results in more accurate detection and characterizing of flaws than when using Article 4 of ASME Code

Section V. As a result the licensee's proposed alternative 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 third 10-year ISI interval, which expires September 20, 2010.

3.8 Relief Request No. RR-A10:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, IWA-5250(a)(2) requires that if leakage occurs at a bolted connection on other than a gaseous system, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100. The bolt selected shall be the one closest to the source of leakage.

Licensee's Code Relief Request: (as stated)
Relief is requested from removing bolting and performing VT-3 examination of bolts when leakage is detected during system pressure tests. The requirements of Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections," will be implemented. A VT-1 visual examination will be used to provide the visual evidence of corrosion at the bolted connection pursuant to item (c)(6) of the Code Case.

Licensee's Basis for Requesting Relief: (as stated)
The removal of bolts for VT-3 visual examination is not always the most prudent action when leakage is discovered at a bolted connection. Leakage at bolted connections is typically identified during system leakage tests. For Class 1 systems, this leakage test is conducted prior to plant startup following each refueling outage. This test is performed at full operating pressure (2155 psig) and temperature. When leakage is discovered during this test, the corrective action (i.e. removal of bolts) must be performed with the system at full temperature and pressure or the plant must be cooled down. The removal of a bolt under full temperature and pressure conditions can be extremely physically demanding due to the adverse heat environment. Cooling down the plant subjects the plant to additional heatup and cooldown cycles and can add 3-4 days to the duration of a refueling outage. Bolted connections associated with pumps and valves are typically studs threaded into the body of the component. Removal of these studs is typically very difficult and time consuming due to [the] length of time they have been installed and are often damaged during the removal process. This difficulty is compounded when the removal must be performed under heat stress conditions.

The requirements of IWA-5250(a)(2) must be applied regardless of the significance of the leakage or the corrosion resistance of the materials used in the bolted connection. Implementation of Code Case N-566-1 permits engineering judgment to be used to evaluate the need for corrective action when leakage is discovered at a bolted connection. This code case permits factors such as the number and service age of the bolts, the bolting materials, the corrosiveness of the system fluid, the leakage location and system function, leakage history at the connection or at other system components, and visual evidence of corrosion at the bolted connection to be used to evaluate the need for corrective measures.

Relief is requested pursuant to 10 CFR 50.55a(a)(3)(i) as the application of Code Case N-566-1, with a VT-1 examination of the bolted connection to satisfy item (c)(6) of the Code Case, will provide an acceptable level of quality and safety as any leakage at mechanical connections will be thoroughly evaluated for acceptability for continued service.

Code Case N-566-1 provides alternatives to the removal of bolting from mechanical connections when leakage is discovered during a system pressure test. Factors such as the number and service age of the bolts, the bolting materials, the corrosiveness of the system fluid, the leakage location and system function, leakage history at the connection or at other system components, and visual evidence of corrosion at the bolted connection are used to determine the integrity of the bolted connection. A VT-1 visual examination will be used to provide the visual evidence of corrosion at the bolted connection pursuant to item (c)(6) of the Code Case. These alternatives provide assurance that the integrity of the mechanical joint will be maintained.

Licensee's Proposed Alternative Examination: (as stated)
When leakage is discovered at a bolted connection, the provisions of Code Case N-566-1, with a VT-1 visual examination of the bolted connection to satisfy item (c)(6) of the Code Case, will be implemented.

Staff Evaluation: Code Case N-566-1 provides alternatives to the removal of bolting from mechanical connections when leakage is discovered during a system pressure test. Factors such as the number and service age of the bolts, the bolting materials, the corrosiveness of the system fluid, the leakage location and system function, leakage history at the connection or at other system components, and visual evidence of corrosion at the bolted connection are used to determine the integrity of the bolted connection. These alternatives provide assurance that the integrity of the mechanical joint will be maintained.

The licensee proposes to implement the requirements of Code Case N-566-1 throughout the third 10-year inspection interval whenever leakage is discovered at a mechanical joint during the performance of system pressure tests. These requirements will include one of the following: stop the leakage and inspect the bolting and component material for joint integrity or, if the leakage is not stopped, evaluate the joint in accordance with IWB-3142.4 for joint integrity. The evaluation will include consideration of the number and condition of bolts, leaking medium, bolt and component material, system function, and leakage monitoring.

The Code requires that all bolts be removed from leaking bolted connections and that the bolts be VT-3 visual examined for corrosion and evaluated in accordance with IWA-3100. The Code requirements provide assurance that bolting corroded by system leakage will be detected and that corrective actions will be taken. However, application of the Code requirements may sometimes be unnecessary since corrosion is dependent on other factors beyond system leakage. Additionally, removal and examination of all bolts may not be necessary to assure continued integrity of the bolted connection.

The licensee noted that when an evaluation of the above elements is concluded and the evaluation determines that the leaking condition has not degraded the fasteners, then no further action is necessary. In addition, the licensee will take reasonable attempts to stop the leakage.

If the evaluation determines that an additional examination is required, the licensee proposed that the bolt closest to the leak be removed and VT-1 examined. The bolt will be evaluated per IWA-3100, which requires that the evaluation of flaws are in accordance with IWB-3000, IWC-3000, and IWD-3000 for Class 1, 2, and 3 pressure retaining components, respectively. The staff determined that removal and VT-1 examination of the bolt closest to the leak is a reasonable alternative since degradation of this bolt is most likely, and would be representative of the worst case condition of the other bolts in the subject connection. The licensee stated that if the leakage is identified when the bolted connection is in service, and the information in the evaluation is supportive, the removal of the bolt for VT-1 examination may be deferred to the next refueling outage.

Based on the items included in the evaluation process, the staff concludes that the evaluation proposed by the licensee presents a sound engineering approach. In addition, if the initial evaluation indicates the need for a more detailed analysis, the bolt closest to the source of leakage will be removed, VT-1 visually examined, and evaluated in accordance with IWA-3100(a). The VT-1 examination criteria are more stringent than the simple corrosion evaluation described in IWA-5250.

For RR-A10, the staff concludes that the proposed alternative 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) for the third 10-year interval, which expires September 20, 2010, or until such time as Code Case N-566-1 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-566-1, it must follow all provisions in the subject code case with the limitations or conditions specified in RG 1.147, if any.

3.9 Request for Relief No. RR-A11:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, IWE-2412-1, and IWF-2410-2 specify the percent of examinations which must be completed during each inspection period when using Inspection Program B.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from the maximum percent of examinations which must be completed during each inspection period as specified in Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, IWE-2412-1, and IWF-2410-2 of the 1995 Edition, 1996 Addenda of ASME Section XI.

Licensee's Basis for Requesting Relief: (as stated)
The DBNPS is on 2-year refueling cycles. Over the 10-Year inspection interval there would be a total of 5 refueling outages in which to accomplish the required examinations. During each of the 5 refueling outages it is desirable to perform approximately 20 percent of the examinations in order to distribute the examinations evenly between outages. For the Third 10-Year Inspection Interval, the DBNPS will have 2 outages during the first inspection period, 1 outage during the second inspection period, and 2 outages during the third inspection period. Applying the 20 percent completion percentage for each outage, 40 percent of the total inspection interval examinations would be completed during the first inspection period, 60 percent would be completed during the second inspection period and 100 percent would be completed by the end of the third inspection period. The maximum percentages which must be completed for each period as specified in Tables IWB -2412-1, IWC-2412-1, IWD-2412-1, IWE-2412-1, and IWF-2410-2 are 34 percent in the first period, 67 percent in the second period, and 100 percent at the end of the third period. Applying the 20 percent completion percentage would exceed the maximum completion percentage for the first inspection period. Code Case N-598 provides alternative requirements for the

maximum percent of examinations that may be credited during inspection periods. Code Case N-598 permits completion of 50 percent of the examinations by the end of the first inspection period.

The intent of Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, IWE-2412-1, and IWF-2410-2 is to ensure that examinations are evenly distributed among the inspection periods and are not concentrated in any one inspection period.

As the total number of examinations conducted during the 10-year inspection interval is unchanged, the overall level of quality and safety will be unaffected.

Licensee's Proposed Alternative Examination: (as stated)
Code Case N-598 will be used in lieu of the requirements of Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, IWE-2412-1, and IWF-2410-2. Code Case N-598 allows 50 percent of the required examinations to be completed by the end of the first period and 75 percent by the end of the second inspection period.

Staff Evaluation:
The intent of Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, IWE-2412-1, and IWF-2410-2 is to ensure that examinations are evenly distributed among the inspection periods and are not concentrated in any one inspection period. Application of the alternative requirements of Code Case N-598 will permit examinations to be distributed across the available refueling outages, which meets the intent of the Code, and provides an equivalent level of protection and quality. Examinations performed during the Third 10-Year Inspection Interval will be scheduled to comply with the requirements of Code Case N-598. The licensee's proposed alternative to use Code Case N-598 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 third 10-year ISI interval, which expires September 20, 2010, or until such time as Code Case N-598 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-598, it must follow all provisions in the subject code case with the limitations or conditions specified in RG 1.147, if any.

3.10 Request for Relief No. RR-A12:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Table IWB-2500-1, Examination Category B-A, Note 3 requires that when using Inspection B, the shell-to-flange weld examination may be performed during the first and third periods, in which case 50 percent of the shell-to-flange weld shall be examined by the end of the first period, and the remainder by the end of the third period. During the first period, the examination need only be performed from the flange face, provided this same portion is examined from the shell during the third period.
Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from performing 50 percent of the Reactor Vessel shell-to-flange weld (RC-RPV-WR-19) by the end of the first period.

Licensee's Basis for Requesting Relief: (as stated)
The Reactor Vessel Shell-to-Flange Weld (RC-RPV-WR-19) was examined in April 2000, during the second 10-Year Reactor Vessel Examination. This examination was conducted from the Reactor Vessel shell with angles of 45° shear wave and a 45° longitudinal wave. Additionally, a 70° longitudinal wave was used for examining the near

surface region. The weld was examined by scanning from four opposing beam directions such that all of the angle beams passed through the weld metal from each direction. The adjacent base metal was scanned from one direction perpendicular to the weld and from two directions parallel to the weld. This examination was in accordance with Supplements 4 and 6 of Appendix VIII to the 1995 edition, 1996 Addenda of ASME Section XI as amended by 10 CFR 50.55a. There was no indications recorded during this examination. No welded repair/replacement activities have been performed on this weld during its inservice history.

The ASME Code Committee has approved Code Case N-623. This code case permits the deferral of the shell-to-flange weld without conducting partial examinations from the flange face if the following conditions are met:

1. No welded repair/replacement activities have ever been performed on the shell-to-flange weld.

2. The shell-to-flange weld contains no identified flaws or relevant conditions that currently require successive inspections in accordance with IWB-2420(b).

3. The vessel is not in the first inspection interval.

Davis-Besse meets the conditions noted in Code Case N-623.

Licensee's Proposed Alternative Examination: (as stated)
The examination of 50 percent of the weld from the flange face in the first inspection period will not be performed. The full volume of the Reactor Vessel Shell-to-Flange weld will be examined during the 10-year reactor vessel examination near the end of the interval.

Staff Evaluation:
Section XI of the ASME Code, Table IWB-2500-1, requires that the RPV shell-to-flange weld be volumetrically examined once each inspection interval and the RPV head-to-flange weld be surface and volumetrically examined once each inspection interval. The footnotes to Table IWB-2500-1 provide partial deferrals for both of these welds. Footnote 3 specifies that, during the first and second period, the examination may be performed from the flange face, and the remaining volumetric examinations required to be conducted from vessel wall may be performed at or near the end of the inspection interval. Footnote 4 provides deferral of the shell-to-flange welds, stating that the examinations may be performed during the first and third periods, provided at least 50 percent of the shell-to-flange welds are examined by the end of the first period, and the remainder by the end of the third inspection period.

The licensee proposes to follow the requirements of Code Case N-623. The staff finds the licensee meets the requirements listed in Code Case N-623. Deferral of the weld examinations to the end of the inspection interval is supported by the operating history of the industry. The industry experience to date indicates that examinations performed on the reactor pressure vessels shell-to-flange have not identified any detrimental flaws or relevant conditions and that changing the schedule for examining these welds to the end of the licensee's 10-year inservice inspection interval will provide a suitable frequency for verifying the integrity of the subject weld. The subject weld will still receive the same examinations that have been required by the ASME Code Section XI since the reactor was placed in commercial service. The only change is that the RPV shell-to-flange weld examination will be deferred to the end of the inspection interval without conducting partial examinations from the flange face earlier in the inspection interval. No changes are being made to the volumes or areas of material that are examined, nor to the nondestructive examination (NDE) personnel qualifications. This relief request does not involve changes to NDE methods or acceptance criteria.

The staff has determined that the licensee's proposed alternative to use Code Case N-623 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 third 10-year ISI interval, which expires September 20, 2010, or until such time as Code Case N-623 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-623, it must follow all provisions in the subject code case with the limitations or conditions specified in RG 1.147, if any.

3.11 <u>Request for Relief No. RR-A14:</u>

<u>Code Requirement:</u> ASME Section XI, 1995 Edition, 1996 Addenda, Appendix I, Article I-2120 requires the ultrasonic examination of vessels, other than the reactor vessel, greater than 2 inches in thickness be conducted in accordance with Article 4 of Section V, as supplemented by Table I-2000-1, which specifies that Supplement 1 to Appendix I is applicable to vessels greater than 2 inches in thickness.

Appendix I, Supplement 1(a) requires the material from which calibration blocks are fabricated be one of the following:

(1) a nozzle dropout from the component;
(2) a component prolongation; or
(3) material of the same material specification, product form, and heat treatment condition as one of the materials being joined.

<u>Licensee's Code Relief Request</u>: (as stated)
Relief is request [sic] in accordance with 10 CFR 50.55a(a)(3)(i). The use of calibration blocks fabricated from material of similar chemical analysis, tensile properties, and metallurgical structure as the material being examined as permitted by Code Case N-639 will provide an acceptable level of quality and safety.

<u>Licensee's Basis for Requesting Relief:</u> (as stated)
The purpose of calibration blocks is to provide notches and holes of known sizes in materials which are acoustically similar to the material being examined. The calibration blocks are used to configure the ultrasonic examination system for the required examination and provide for repeatability of examinations.

The requirements of Appendix I, Supplement 1(a)(3) requires material of the same material specification, product form, and heat treatment condition as one of the materials being joined for the calibration block material. This requirement does not address the grade or type of material nor the heat treatment of the material. It only addresses the material specification and the heat treatment condition, e.g. quenched and tempered. The material grade or type and heat treatment are more important in determining the acoustic properties of a material than the materials specification and heat treatment condition. This is recognized in ASME Section V, Article V, T542.2.1.1 which requires the basic calibration block be fabricated from the same product form and material specification or equivalent P-Number grouping as the material being examined. ASME Section XI, Appendix III, III-3411(d) also permits the use of calibration blocks of similar chemical analysis, tensile properties, and metallurgical structure when material of the same specification is not available. The use of materials of similar chemical analysis, tensile properties, and metallurgical structure when calibration blocks of the same material specification, product form, and heat treatment condition as the material being examined is not available will provide materials with acoustic properties similar to those being examined.

Calibration blocks made from material of similar chemical analysis, tensile properties, and metallurgical structure as the material being examined will provide calibration blocks which have acoustic properties similar to the material being examined. Ultrasonic examinations conducted using calibration blocks which are acoustically similar to the material being examined will provide equivalent examinations to those conducted using calibration blocks meeting the requirements of ASME Section XI, Appendix I, Supplement 1(a). The use of the alternative material requirements contained in Code Case N-639 will provide an acceptable level of quality and safety.

<u>Licensee's Proposed Alternative Examination:</u> (as stated)
Relief is requested to use Code Case N-639, Alternative Calibration Block Material, when calibration blocks of the same material specification, product form, and heat treatment condition as the material being examined are not available.

<u>Staff Evaluation:</u>
The licensee has proposed as an alternative to ASME Code Section XI to use Code Case N-639, Alternative Calibration Block Material, when calibration blocks of the same material specification, product form, and heat treatment condition as the material being examined are not available. The licensee noted that its calibration blocks will be made from material of similar chemical analysis, tensile properties, and metallurgical structure as the material being examined. In addition, the licensee will provide calibration blocks which have acoustic properties similar to the material being examined. The staff has determined that ultrasonic examinations conducted using calibration blocks which are described by the licensee provide equivalent examinations to those conducted using calibration blocks meeting the requirements of ASME Section XI, Appendix I, Supplement 1(a). The staff determined that the licensee's proposed alternative to use Code Case N-639 with the calibration blocks described by the licensee provides an acceptable level of quality and safety. Therefore, the licensee's proposed alternative to use Code Case N-639 with calibration blocks that will be made from material of similar chemical analysis, tensile properties, and metallurgical structure as the material being examined is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year ISI interval, which expires September 20, 2010, or until such time as Code Case N-639 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-639, it must follow all provisions in the subject code case with the limitations or conditions specified in RG 1.147, if any.

3.12 Request for Relief No. RR-A15:

Code Requirement: Subsection IWB, Table IWB-2500-1, Examination Category B-J, Item No. B9.11 (Circumferential Welds NPS 4 or Larger) of the 1995 Edition, 1996 Addenda of ASME Section XI requires examination of essentially 100 percent of the weld. Figure IWB-2500-8 establishes the examination volume for Code Category B-J circumferential piping welds.

Subsection IWC, Table IWC-2500-1, Examination Category C-F-1, Pressure Retaining Welds in Austenitic Stainless Steel or Higher Alloy Piping, of the 1995 Edition, 1996 Addenda of ASME Section XI requires examination of essentially 100 percent of the weld. Figure IWC-2500-7 establishes the examination volume for Code Category C-F-1 circumferential piping welds.

Code Case N-460 states that when the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in examination coverage may be accepted provided the reduction in coverage for that weld is less than 10 percent.

10 CFR 50.55a(b)(2)(xvi)(B) requires examinations performed from one side of a ferritic or stainless steel pipe weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate proficiency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and 10 CFR 50.55a(b)(xv)(A).

10 CFR 50.55a(b)(2)(xv)(A) requires the following examination coverage when applying Supplement 2 and 3 to Appendix VIII of the 1995 Edition, 1996 Addenda of ASME Section XI:

1. Piping must be examined in two axial directions and when examination in the circumferential direction is required, the circumferential examination must be performed in two directions, provided access is available.

2. Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds, full coverage credit from a single side may be claimed only after completing a successful single sided Appendix VIII demonstration using flaws on the opposite side of the weld.

System/Component(s) for Which Relief is Requested: (as stated)
Dissimilar Metal Welds with single side access subject to ultrasonic examination with Supplement 10 to Appendix VIII of the 1995 Edition, 1996 Addenda of ASME Section XI.

The following ASME Class 1 dissimilar metal welds will be examined from one side only.

- RC-MK-A67-1-FW105A - Reactor Coolant Pump 2-1 Inlet Nozzle to 28 inch Elbow Weld
- RC-MK-A67-3-FW105B - Reactor Coolant Pump 1-2 Inlet Nozzle to 28 inch Elbow Weld
- RC-MK-A67-2-FW134A - Reactor Coolant Pump 2-2 Inlet Nozzle to 28 inch Elbow Weld
- RC-MK-B67-1-FW134B - Reactor Coolant Pump 1-1 Inlet Nozzle to 28 inch Elbow Weld

Austenitic stainless steel components with single side access subject to ultrasonic examination with Supplement 2 to Appendix VIII of the 1995 Edition, 1996 Addenda of ASME Section XI.

The following ASME Class 1 stainless steel welds will be examined from one side only.

- DH-33A-CCA-4-F6A-FW5 - Decay Heat Valve DH 12 to 12 inch Pipe Weld
- DH-33A-CCA-4-F6A-FW6 - Decay Heat Valve DH 11 to 12 inch Pipe Weld
- CF-33B-CCA-6-3-FW29 - Core Flood Valve CF 30 to 14 inch Elbow Weld
- CF-33B-CCA-6-5B-FW15 - Core Flood Valve CF 31 to 14 inch Elbow Weld

The following ASME Class 2 stainless steel welds will be examined from one side only.

- HP-33C-CCB-2-41-FW31 - High Pressure Injection Valve HP 49 to 2½ 2 inch Pipe Weld
- HP-33C-CCB-2-35-FW22A - High Pressure Injection Valve BP48 to 2½ inch Pipe Weld
- *DH-33B-GCB-10-21-FW66 - Decay Heat Valve DH 830 to 8 inch Pipe Weld
- *CS-34-GCB-5-2-FW6 - Containment Spray Valve CS 1531 to 8 inch Elbow Weld
- *DH-33A-GCB-7-5-FW20 - Decay Heat Valve DH 1517 to 12 inch Tee Weld
- *DH-33A-GCB--7-6-FW17-Decay Heat Valve DH 1518 to 12 inch Pipe Weld

* These welds have a wall thickness less than 3/8 inch and are being examined as required by Relief Request RR-B2.

<u>Licensee's Code Relief Request</u>: (as stated)
Relief is request in accordance with 10 CFR 50.55a(a)(3)(ii). Technology is not currently available to qualify examination procedures for cast stainless steel and austenitic stainless steel welds from one side only.

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

Reactor Coolant Pump to Nozzle Welds
The Reactor Coolant Pump nozzle to elbow welds are dissimilar metal welds. The piping elbows are manufactured from ferritic steel while the Reactor Coolant Pumps are manufactured from cast stainless steel. Appendix VIII, Supplement 10 addresses ferritic to austenitic materials, but does not address ferritic to cast stainless steel welds. The examination of cast stainless steel is not addressed in Appendix VIII. Current technology is not capable of reliably performing ultrasonic examination of cast stainless steels. Therefore, credit can not be taken for examination from the cast side of the weld which limits examination credit to that obtained from the single sided examination from the ferritic piping. As a result, the weld is examined by only 3 of the 4 required directions as no scans can be credited from the cast stainless steel side of the weld. As the area of interest within the cast stainless steel is not interrogated, the examination coverage is approximately 40 percent of the required examination volume.

Valve to Piping Welds
The valve to piping welds addressed in this relief request are austenitic stainless steel welds. The valve taper prevents scanning from the valve side of the weld which results in a single side examination from the pipe side of the valve. For single sided examinations, 10 CFR 50.55a(b)(2)(xvi)(B) requires a procedure be qualified using flaws from the opposite side of the weld. There are currently no qualified single side examination procedures that demonstrate equivalency to two-sided examination procedures on austenitic piping welds. Current technology is not capable of reliably detecting or sizing flaws on the far side of an austenitic weld for configurations common to United States nuclear applications. As a result, the weld is examined by only 3 of the 4 required directions as no scans can be credited from the valve side of the weld. As the area of interest within the valve is not interrogated, the examination coverage is approximately 40 percent of the required examination volume.

The PDI Program conforms to 10 CFR 50.55a regarding single side access for piping. The PDI Performance Demonstration Qualification Summary certificates for austenitic piping list the limitation that single side examination is performed on a best effort basis. The best effort qualification is provided in place of a complete single side qualification to demonstrate that the examiners qualification and the subsequent weld examination is based on application of the best available technology.

There are currently no qualified PDI single side examination procedures that demonstrate equivalency to two-sided examination procedures on austenitic piping or dissimilar ferritic to cast stainless steel welds. Current technology is not capable of reliably detecting or sizing flaws on the far side of an austenitic weld for configurations common to United States nuclear applications making examination from one side of the weld impractical. Examination [in] 3 of the 4 required directions ensures that a portion the examination volume is interrogated which should detect any gross degradation of the weldment.

Licensee's Proposed Alternative Examination: (as stated)
The best available techniques, as qualified through the PDI, will be used from the accessible side of the weld on a best effort basis.

Staff Evaluation:
The ASME Code Section XI requires examination of essentially 100 percent of the subject welds. 10 CFR 50.55a(b)(2)(xvi)(B) requires examinations performed from one side of a ferritic or stainless steel pipe weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single-side examinations. To demonstrate proficiency to two-sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and 10 CFR 50.55a(b)(xv)(A).

10 CFR 50.55a(b)(2)(xv)(A) requires the following examination coverage when applying Supplement 2 and 3 to Appendix VIII of the 1995 Edition (1996 Addenda) of ASME Section XI:

1. Piping must be examined in two axial directions and when examination in the circumferential direction is required, the circumferential examination must be performed in two directions, provided access is available.

2. Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds, full coverage credit from a single side may be claimed only after completing a successful single sided Appendix VIII demonstration using flaws on the opposite side of the weld.

The Reactor Coolant Pump nozzle to elbow welds are dissimilar metal welds. The piping elbows are manufactured from ferritic steel while the Reactor Coolant Pumps are manufactured from cast stainless steel. Current technology is not capable of reliably performing ultrasonic examination of cast stainless steel. The licensee noted that credit cannot be taken for examination from the cast side of the weld which limits examination credit to that obtained from the single sided examination from the ferritic piping. The licensee examined the weld by only 3 of the 4 required directions as no scans were credited from the cast stainless steel side of the weld. The licensee obtained approximately 40 percent of the required examination volume.

The valve to piping welds are austenitic stainless steel welds and the taper of the valve prevents scanning from the valve side of the weld which results in a single side examination from the pipe side of the valve. Current technology is not capable of reliably detecting or sizing flaws on the far side of an austenitic weld for the subject configuration. The licensee examined the subject weld by only 3 of the 4 required directions as scans from the valve side of the weld were not credited. The licensee obtained an examination coverage of approximately 40 percent of the required examination volume.

The staff determined that the Code requirements are impractical. To impose the Code requirements would be a burden on the licensee, because the subject components would have to be redesigned in order to perform the Code required examinations. The licensee obtained approximately 40 percent of the required examination volume and 100 percent of the Code required surface examination. Therefore, based on the examinations obtained, the licensee's proposed alternative provides reasonable assurance of structural integrity of the subject components.

However, in accordance with 10 CFR 50.55a(g)(6)(ii)(C), PDI is required to have a qualified procedure to examine dissimilar metal welds and austenitic stainless steel welds from the pipe bore by November 22, 2002. Therefore, for dissimilar metal welds and austenitic stainless steel welds, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval, but not beyond November 22, 2002. For welds connecting cast stainless steel components relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval, which expires September 20, 2010. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

3.13 Relief Request No. RR-A16:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Appendix VII, Paragraph VII-4240 requires a minimum of 10 hours of annual training.

10 CFR 50.55a(b)(2)(xiv) requires that all personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on-training on specimens that contain cracks. This training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from the provisions of 10 hours of annual training contained in Paragraph VII-4240, Appendix VII of ASME Section XI.

Licensee's Basis for Requesting Relief: (as stated)
Paragraph VII-4240 requires 10 hours of annual training. 10 CFR 50.55a(b)(2)(xiv) requires that all personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on training on specimens that contain cracks. This training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

Paragraph 2.4.1.1.1 in the Federal Register (Volume 64, No. 183 dated September 22, 1999) contained the following statement:

> The NRC had determined that this requirement (10 hours of training on an annual basis) was inadequate for two reasons. The first reason being that the training does not require laboratory work and examination of flawed specimens. Signals can be difficult to interpret and, as detailed in the regulatory analysis for the rulemaking, experience and studies indicate that the examiner must practice on a frequent basis to maintain the capability for proper interpretation. The

second reason is related to the length of training and its frequency. Studies have shown that an examiner's capability begins to diminish within approximately 6 months if skills are not maintained. Thus, the NRC had determined that 10 hours of annual training is not sufficient practice to maintain skills, and that an examiner must practice on a more frequent basis to maintain proper skill level. The PDI program has adopted a requirement for 8 hours of training, but it is required to be hands-on practice. In addition, the training must be taken no earlier than 6 months prior to performing examinations at a licensee's facility. PDI believes that 8 hours will be acceptable relative to an examiner's abilities in this highly specialized skill area because personnel can gain knowledge of new developments, material failure modes, and other pertinent technical topics through other means. These changes are reflected in 50.55a(b)(2)(xiv).

Implementation of the requirements contained in both paragraph VII-4240 of ASME Section XI and 10 CFR 50.55a will result in redundant systems. The use of the 10 CFR 50.55a requirements only will simplify record keeping, satisfy needs for maintaining skills, and provide an acceptable level of quality and safety.

<u>Licensee's Proposed Alternative Examination</u>: (as stated)
Annual ultrasonic training shall be conducted in accordance with 10 CFR 50.55a(b)(2)(xiv) in lieu of the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix VII, Paragraph VII-4240.

<u>Staff Evaluation</u>:
Subarticle VII-4240, Appendix VII of Section XI of the Code requires 10 hours of annual training to impart knowledge of new developments, material failure modes, and any pertinent technical topics as determined by the licensee. No hands-on training or practice is required to be included in the 10 hours of training. This training is required of all ultrasonic test (UT) personnel qualified to perform examinations of ASME Code Class 1, 2, and 3 components. Independent of the ASME Code, 10 CFR 50.55a(b)(2)(xiv) imposes the requirement for Appendix VIII qualification that 8 hours of hands-on training with flawed specimens containing cracks be performed no earlier than 6 months prior to performing examinations at a licensee's facility. The licensee contends that maintaining two separate UT annual training programs for Appendix VIII and non-Appendix VIII qualifications create redundancies in training programs.

As part of the staff's rulemaking effort to revise 10 CFR 50.55a(b)(2), the issue of UT annual training requirements was reviewed. This review was included in the summary of comments to the rule that was published in the Federal Register on September 22, 1999, (64 FR 51370). In the review, the staff determined that the 10 hours of annual training requirement specified in the ASME Code was inadequate for the two reasons quoted in the licensee's basis for relief (Section 2.3 above). In resolving public comment to the rulemaking, the staff adopted a recommendation advanced by the nuclear power industry which proposed 8 hours of hands-on practice with specimens containing cracks. This practice would occur no earlier than 6 months prior to performing examinations at a licensee's facility. These recommendations were accepted by NRC and are reflected in 10 CFR 50.55a(b)(2)(xiv). The staff has determined that the proposed alternative to use 10 CFR 50.55a(b)(2)(xiv) in lieu of

Subarticle VII-4240 will maintain the skill and proficiency of UT personnel at or above the level provided in "the Code for annual UT training," thereby, providing an acceptable level of quality and safety.

For RR-A16, the staff concludes that the proposed alternative 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 third 10-year interval, which expires September 20, 2010.

3.14 Request for Relief No. RR-A17:

Code Requirement:
IWA-2313 of the 1995 Edition, 1996 Addenda of ASME Section XI requires personnel performing visual examinations or using other NDE methods not addressed in ANSI/ASNT CP-189 be qualified and certified to comparable levels of qualification as defined in ANSI/ASNT CP-189 and the Employer's written practice.

IWA-2314 of the 1995 Edition, 1996 Addenda of ASME Section XI requires personnel be certified in accordance with ANSI/ASNT CP-189, except that the ASNT Level III Certificate is not required. Level I and Level II personnel shall be recertified by qualification examinations every 3 years. Level III personnel shall be recertified by qualification examinations every 5 years.

Licensee's Code Relief Request: (as stated)
Relief is requested from the provisions of IWA-2313 and IWA-2314 for visual examination personnel performing VT-2 examinations. The requirements of Code Case N-546 will be supplemented with the conditions as cited in Draft Regulatory Guide DG-1091 (December 2001) for the use of the Code Case.

Licensee's Basis for Requesting Relief: (as stated)
The 1995 Edition, 1996 Addenda of ASME, Section XI requires personnel conducting VT-2 examinations be qualified and certified to comparable levels of qualification as defined in ANSI/ASNT CP-189 and the Employer's written practice.

VT-2 examination is a straightforward examination for leakage. VT-2 examination does not require any special knowledge of technical principals underlying its performance. No special skills or technical training are required to observe leakage from a component or bubbles forming on a joint wetted with leak detection solution. As such, there is no need to subject VT-2 examination personnel to the same qualification and certification requirements as imposed on other nondestructive examination techniques.

Code Case N-546 provides alternative requirements for the qualification of VT-2 examination personnel. Code Case N-546 permits VT-2 examination personnel be qualified in accordance with the following requirements:

(a) At least 40 hours of plant walkdown experience, such as that gained by licensed and non-licensed 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.

(c) Vision test requirements of IWA-2331,1995 Edition.

These requirements will be supplemented by conditions for use of Code Case N-546 as stated in Draft Regulatory Guide DG-1091 (December 2001)

The requirements of Code Case N-546 are less burdensome than qualifying and maintaining the VT-2 certification program required by IWA-2313. Code Case N-546 makes it feasible to train and certify more highly qualified personnel to perform VT-2 examinations. Furthermore, it permits experienced personnel who are already required to perform other functions in the plant to perform the VT-2 examination during their normal duties which already includes the identification of leaking components that require maintenance. This would reduce the number of people who are required to enter radiological restricted areas, resulting in fewer plant workers exposed to potential radiation dose and keeping radiation exposure as low as reasonably achievable.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). Implementation of Code Case N-546, as supplemented with the conditions cited in Draft Regulatory Guide DG-1091 (December 2001) for use of the Code Case, will ensure that experienced and competent personnel examine systems for leakage. This will provide an acceptable level of quality and safety comparable to that which would be obtained by using personnel qualified to the comparable levels of qualification as defined in ANSI/ASNT CP-189.

Code Case N-546 requires a minimum of 40 hours of plant walkdown experience. Experience in identifying equipment problems during walkdowns and knowledge of operating conditions will enhance the ability of plant personnel to locate leakage during VT-2 examinations. With the specified 4 hours of training and testing on Section XI requirements and plant-specific procedures for VT-2 examinations, the designated plant personnel will understand how leaks should be identified and documented and be fully capable of performing VT-2 examinations. The use of the alternative requirements of Code Case N-546, supplemented with the conditions cited in Draft Regulatory Guide DG-1091 (December 2001) for use of the Code Case, for the qualification of VT-2 personnel will provide an acceptable level of quality and safety.

Licensee's Proposed Alternative Examination: (as stated)
The requirements of Code Case N-546, supplemented with the conditions cited in Draft Regulatory Guide DG-1091 (December 2001) for use of the Code Case, will be used to qualify VT-2 examination personnel.

Staff Evaluation:
The ASME Code Section XI, IWA-2300, requires that personnel performing VT-2 visual examinations be qualified and certified using a written, approved procedure prepared in accordance with SNT-TC-1A and the additional requirements of Division 1 of ASME Section XI. The Code also requires that the examination personnel be qualified for near and far distance vision acuity. Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee

proposed to use Code Case N-546 with the conditions cited in Draft Regulatory Guide DG-1091 (December 2001) in lieu of the requirements of IWA-2300 for VT-2 visual examination personnel.

The Code Case states that licensed and non-licensed operators, local leak rate personnel, system engineers, and inspection and nondestructive examination personnel are eligible to perform VT-2 visual examinations due to their plant experience. Those personnel typically have a sound working knowledge of plant components and piping layouts, making them acceptable candidates for performing VT-2 visual examinations. Furthermore, the licensee follows plant-specific procedures to obtain consistent VT-2 visual examination results. The Code Case also requires a vision test for examination personnel to that of the 1995 Edition Code. In addition to the code case requirements, it is necessary for the VT-2 visual examination personnel to demonstrate knowledge of Section XI and plant-specific procedures for VT-2 visual examinations and to demonstrate continued proficiency through periodic re-qualification in accordance with the frequency of every three years. The licensee in its letter dated February 6, 2002, included the conditions cited in Draft Regulatory Guide DG-1091 (December 2001) for the use of Code Case N-546. The conditions cited in Draft Regulatory Guide DG-1091 (December 2001) are noted below:

(1) Qualify examination personal by test to demonstrate knowledge of Section XI and plant specific procedures for VT-2 visual examinations;
(2) Requalify examination personnel by examination every three years;
(3) This Code Case is applicable only to the performance of VT-2 examinations.

Since the VT-2 examination is a check for evidence of leakage, the use of plant personnel qualified to the Code Case N-546 requirements, and who typically perform this type of examination during their daily activities, will not compromise the quality or safety of the systems examined. The staff concludes that the licensee’s proposed alternative to use Code Case N-546 with its commitments to the conditions contained in draft Regulatory Guide DG-1091 provides an acceptable level of quality and safety for the purpose of performing VT-2 visual examinations only. Therefore, the licensee’s proposed alternative to use Code Case N-546, with its commitments as noted above, is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year ISI interval, which expires September 20, 2010, or until such time Code Case N-546 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-546, it must follow all provisions in the subject code case with the limitations or conditions specified in RG 1.147, if any.

3.12 <u>Request for Relief No. RR-B1:</u>

<u>Code Requirement:</u> ASME Section XI, 1995 Edition, 1996 Addenda, Subsection IWC, Table IWC-2500-1, Examination Category C-A, Item Number C1.10 requires a volumetric examination of essentially 100 percent of the weld length at a gross structural discontinuity such as shell to flange welds of the Decay Heat Removal Heat Exchanger 27-1 and 27-2.

Code Case N-460 states that when the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in examination coverage may be accepted provided the reduction in coverage for that weld is less than 10 percent.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from performing a volumetric examination of essentially 100 percent of the weld length. The reduction in examination coverage of the weld is greater than 10 percent.

Licensee's Basis for Requesting Relief (as stated):
Paragraph T-441.1, Article 4, Section V of the 1995 Edition, 1996 Addenda of ASME Section XI [sic] requires the examination volume be scanned with angle beam search units directed both at right angles to the weld axis and along the weld axis. Wherever feasible, each examination shall be performed in two directions. The adjacent base metal in the examination volume must be completely scanned by two angle beams, but need not be completely scanned by both angle beams from both directions. The examination volume for this weld is defined in Figure IWC-2500-1.

The examination of the Decay Heat Cooler shell-to-flange weld is limited by the taper of the flange at the interface with the weld, reinforcing plates at the cooler inlet and outlet nozzles, and an angle iron which supports a platform above the cooler. Each Decay Heat Cooler has a reinforcing plate installed at the cooler inlet and outlet nozzles. These reinforcing plates extend to the edge of the weld which makes that portion of the weld (approximately 12 inches) inaccessible for volumetric examination from the shell side of the cooler. In addition, the angle iron support prohibits the scan on the weld from the shell side for approximately 14 inches. At the taper to weld interface, coupling is lost due to the transition from the flange to the weld.

As a result of these factors, approximately 80 percent of the weld volume is covered during the volumetric examination. The examination volume is examined in at least one direction to detect reflectors in both the parallel and transverse directions to the weld. This should detect any defects which may exist.

Relief is requested pursuant to 10 CFR 50.55a(a)(3)(i) as the proposed examination will provide an acceptable level of quality and safety as reflectors oriented both parallel and transverse to the weld can be located. A relief request for this weld was previously approved for the Second 10-Year Interval in Relief Request RR-B1 (TAC Nos. M79034 and M77942).

In its letter dated November 27, 2001, in response to the NRC request for additional information the license stated:

*General Information for Relief Request RR-B1*

The Decay Heat Removal Heat Exchanger shell is stainless steel while the flange is carbon steel with stainless steel cladding on the inside surface. Due to this cladding the ultrasonic beam cannot be "bounced" from the inside clad surface to increase the examination coverage. The ultrasonic examination is conducted in accordance with Appendix III to Section XI of the ASME Code, 1995 Edition through the 1996 Addenda as the shell wall thickness is less [than] 2" in thickness. Appendix III requires each weld receive 4 scans (2 axial and 2 circumferential in opposing directions). The following discussion illustrates how The licensee determines the volume examined during ultrasonic examination of Weld G.

The following methodology is used to determine the extent of examination coverage.

1. A scaled cross sectional sketch of the component configuration and the examination volume is drawn using a CADD program.

2. Each weld requires 4 scan directions (2 axial and 2 circumferential in opposing directions). Each scan provides 25 percent of the total examination coverage.

3. The ultrasonic beam paths are overlaid on the drawing. The area of coverage is measured using the CADD program for each scan direction.

4. The volume of coverage for each scan is calculated by multiplying the area of coverage by the length of the weld. This volume is then multiplied by 0.25 (25 percent) as each scan represents 25 percent of the total required scans. The products for each scan are added together to determine the combined volume scanned.

5. The combined volume is divided by the total volume of the weld and multiplied by 100. This is the percent of examination completed.

Attachment 6 [to the licensee's letter dated November 27, 2001] shows the weld profiles and limitations for the Decay Heat Removal Heat Exchanger Shell-to-Flange weld.

<u>Licensee's Proposed Alternative Examination</u>: (as stated)
The shell-to-flange weld will be examined volumetrically to the extent possible. The area of the weld inaccessible due to the nozzle reinforcing plates will receive a surface examination as part of the reinforcing plate to vessel weld examination under Code Item C2.31.

<u>Staff Evaluation:</u>
The ASME Code Section XI requires a volumetric examination of essentially 100 percent of the weld length at a gross structural discontinuity such as shell to flange welds of the Decay Heat Removal Heat Exchanger 27-1 and 27-2.

The examination of the Decay Heat Cooler shell-to-flange weld is limited by the taper of the flange at the interface with the weld, reinforcing plates at the cooler inlet and outlet nozzles, and an angle iron which supports a platform above the cooler. The staff has determined from the information provided by the licensee that the Code requirements are impractical. In order for the licensee to perform the Code required examinations it would be an excessive burden, because the component would have to be redesigned. The licensee obtained approximately 80 percent of the weld volume which should detect any defects that may exist. As a result the licensee's proposed alternative provides reasonable assurance of structural integrity of the subject component. Therefore, the licensee's relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year interval, which expires September 20, 2010. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

3.13 Request for Relief No. RR-B2:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Examination Categories C-F-1 and C-F-2 require a surface and volumetric examination for piping welds greater than or equal to 3/8 inch nominal thickness for piping greater than 4 inch nominal pipe size. For High Pressure Injection Systems, a surface and volumetric examination is required for piping welds greater than 1/5 inch nominal wall thickness for piping greater than or equal to 2 inch nominal pipe size and less than or equal to 4 inch nominal pipe size.

Per Note 2 of C-F-1 and C-F-2, welds not exempted by IWC-1220 which do not meet the above criteria do not require nondestructive examination, but are required to be included in the total weld count to which the 7.5 percent sampling rate is applied.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from the minimum wall thickness requiring examination as specified in Code Categories C-F-1 and C-F-2.

Licensee's Basis for Requesting Relief: (as stated)
The piping in the Containment Spray Discharge, the Decay Heat Discharge, the Decay Heat Suction from the Reactor Coolant System Class 1 boundary, and the Main Steam Supply for the Auxiliary Feedwater Pumps from the Main Steam lines to the first isolation valve has a wall thickness less than 3/8 in. and greater than 1/5 inch. The piping in the Containment Spray suction, the High Pressure Injection Suction, the High Pressure Injection Recirculation Line, the Decay Heat Suction from the Borated Water Storage Tank, and the Decay Heat Suction from the Emergency Sump has wall thickness less than 1/5 in. This "thin wall" piping does not require any nondestructive examination under Code Categories C-F-1 or C-F-2.

When the selection criterion of C-F-1 and C-F-2 Note 2 is applied to these systems, approximately 93 percent of the Class 2 Decay Heat Discharge welds past the Containment isolation valves, approximately 26 percent of the High Pressure Injection Discharge, and approximately 11 percent of the Main Steam System welds receive a nondestructive examination per Examination Categories C-F-1 and C-F-2. These sampling rates exceed 7.5 percent sampling rate established in ASME Section XI. In addition, the welds requiring examination in the Emergency Core Cooling Systems are concentrated in approximately 1/3 of the total welds in the systems. This distribution is such that the requirements of C-F-1 and C-F-2 Note 2 can not be met.

The licensee believes the “thin wall” portion of these systems is important enough to plant safety that appropriate nondestructive examination of their circumferential welds is warranted.

Volumetric examinations are not appropriate for all piping wall thickness. Code Case N-435-1 provides alternative ASME Section XI examination requirements for vessels with a wall thickness 2 in. or less. This Code Case states that a surface examination may be applied in lieu of volumetric examinations for vessels with a wall thickness 1/5 inch or less. The Code, therefore, has recognized that volumetric examination of welds with a wall thickness less than 1/5 inch to [sic] ASME Section XI requirements is impractical.

Relief is requested pursuant to 10 CFR 50.55a(a)(3)(i) as the proposed examination will provide an acceptable level of quality and safety as inservice examination will be conducted throughout entire systems rather than being concentrated in specific portions of these systems. A relief request for these welds was previously approved for the Second 10-Year Interval in Relief Request RR-B4 (TAC No. M87188).

In its letter dated November 27, 2001, in response to the NRC request for additional information the license stated:

This request was previously approved in the second interval (Relief Request RR-B4 in the second interval) via TAC M87188. At present, Davis-Besse is not pursuing RI-ISI for Class 2 piping welds. Should any Code actions occur which would affect surface examinations, this relief request would be revised and resubmitted as necessary. Pipe sizes with wall thickness less than 3/8" (including the group with wall thickness less than 1/5") range from 6" to 18"NPS.

<u>Licensee’s Proposed Alternative Examination</u>: (as stated)
The minimum nominal wall thickness specified in Code Categories C-F-1 and C-F-2 will not be used to exclude welds from examination in the Containment Spray, Decay Heat, High Pressure Injection, or Main Steam systems. The following requirements will be used to establish examination requirements for C-F-1 or C-F-2 category welds in these systems.

1. The 7.5 percent sampling rate will be applied to all welds not exempted by IWC-1220.

2. Welds selected which meet the nominal wall thickness requirements of Code Categories C-F-1 and C-F-2 will receive a surface and volumetric examination.

3. Welds in piping greater than NPS 4 inch with wall thickness between 1/5 inch and 3/8 inch will receive an augmented surface and volumetric examination.

4. Welds in piping with wall thickness less than 1/5 inch wall will receive an augmented surface examination.

Staff Evaluation:
The ASME Code Section XI, requires a surface and volumetric examination for piping welds greater than or equal to 3/8 inch nominal thickness for piping greater than 4 inch nominal pipe size. For High Pressure Injection Systems, a surface and volumetric examination is required for piping welds greater than 1/5 inch nominal wall thickness for piping greater than or equal to 2 inch nominal pipe size and less than or equal to 4 inch nominal pipe size.

The Code exempts 4 inch nominal pipe size or greater with a wall thickness less than 3/8 inch from Code required examination; however, the licensee has proposed to expand its 7.5 percent sample rate to include piping greater than NPS 4 inch with a wall thickness less than 3/8 inch. If the licensee did not include this inspection sample of piping greater than NPS 4 inch with a wall thickness less than 3/8 inch a number of systems and welds would be excluded from examination. The licensee also proposed that the augmented inspection sample for welds in piping greater than NPS 4 inch with wall thickness between 1/5 inch and 3/8 inch will receive a surface and volumetric examination, and welds in piping with wall thickness less than 1/5 inch wall will receive an augmented surface examination. The staff has determined that the licensee's proposed alternative 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 third 10-year ISI interval, which expires September 20, 2010.

3.13 Request for Relief No. RR-B3:

Code Requirement: Subsection IWC, Table IWC-2500-1, Examination Category C-G, Item Number C6.10 (Pump Casing Welds) requires a surface examination of all welds. In cases where multiple pumps are of similar design, size, function, and service, the welds in one pump may be selected.

Code Case N-460 states that when the entire examination volume or area cannot be examined due to interference by another component or part geometry, a reduction in examination coverage may be accepted provided the reduction in coverage for that weld is less than 10 percent.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from performing a surface examination of essentially 100 percent of the weld length. The reduction in examination coverage of the weld is greater than 10 percent.

Licensee's Basis for Requesting Relief: (as stated)
The surface examination of the identified welds is limited by the component geometry such that the reduction in coverage is greater than 10 percent.

An integrally welded attachment is welded at the point where the suction and discharge nozzles attach to the High Pressure Injection Pump casing. This welded attachment covers a portion of the nozzle-to-casing welds.

The circumference of the discharge nozzle to casing weld is approximately 21 inches. The attachment covers up approximately 4.75 inches of the discharge nozzle to casing weld. Therefore, approximately 77 percent of the examination area is available for examination.

The circumference of the suction nozzle to casing weld is approximately 14 inches. The attachment covers up approximately 4.75 inches of the suction nozzle to casing weld. Therefore, approximately 66 percent of the examination area is available for examination.

The discharge and suction nozzle welds on both High Pressure Injection Pumps are of similar design. Therefore, no other welds are available for examination.

Relief is requested pursuant to 10 CFR 50.55a(a)(3)(i) as examination of the available surface area will provide an acceptable level of quality and safety. A relief request for these welds was previously approved for the Second 10-Year Interval in Relief Request RR-B7 (TAC No. M93310).

In its letter dated November 27, 2001, in response to the NRC request for additional information the license stated:

A similar relief request was previously approved for the second interval (Relief Request RR-B7 in the second interval) via TAC M93310. As shown in the picture included with the relief request, the pump support attachment welds cover the nozzle welds. In a phone conversation with the NRC on December 20, 1995, the NRC requested that the examination include the pump support attachment welds and the pipe to nozzle welds as part of the approval process of RR-B7 in the second interval.

<u>Licensee's Proposed Alternative Examination</u>: (as stated)
The available surface area of the High Pressure Injection pump discharge and suction nozzle to casing welds will be surface examined to the maximum extent possible. The examination of pump casing-to-nozzle welds will be supplemented by the examination of the accessible surfaces of the pump support attachment welds as required by Code Category C-C.

<u>Staff Evaluation:</u>
The ASME Code Section XI, Subsection IWC, Table IWC-2500-1, Examination Category C-G, Item Number C6.10 (Pump Casing Welds) requires a surface examination of all welds. In cases where multiple pumps are of similar design, size, function, and service, the welds in one pump may be selected. A picture included with the relief request showed that the pump support attachment welds cover the nozzle welds. Therefore, the staff determined that the Code requirements are impractical, because the surface examination of the identified welds is limited by the component geometry. To perform the Code required examinations would be an extensive burden on the licensee, because the component would have to be redesigned.

The licensee was able to obtain 77 percent of the surface examination area for the discharge nozzle to casing weld and for the suction nozzle to casing weld the licensee obtained 66 percent of the surface examination. The staff determined that based on the examination surface examination coverage obtained and the Code required VT-2 visual examinations during system leakage tests the licensee's proposed alternative surface examinations should detect any defects which may exist. Based on the surface examinations performed and VT-2 visual examinations during system leakage tests the staff determined that these examinations provide reasonable assurance of structural integrity of the subject components. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval, which expires September 20, 2010. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

3.14 Request for Relief No. RR-B4

Code Requirement: Table IWC-2500-1, Examination Category C-H, Code Item No. C7.30 of the 1995 Edition, 1996 Addenda of ASME Code Section XI requires a system leakage test of piping once each inspection period using a VT-2 visual inspection.

Licensee's Code Relief Request: (as stated)
Portions of the piping segments between the containment isolation valves for Containment Purge System Penetrations 33 and 34 are inaccessible. Relief is requested to the above referenced ASME Code Section requirement regarding the use of a VT-2 visual examination of the complete piping segment between the containment isolation valves.

Licensee's Basis for Requesting Relief: (as stated)
Containment Purge System Penetrations 33 and 34 piping are tested in accordance with 10 CFR [sic] Appendix J, Option B, Type C testing requirements. If leakage is identified during the test, the test boundary is VT-2 visually examined with the soap bubble technique to verify the source of leakage is not through-wall leakage.

Rubber foam barriers and rubber boots installed around the piping at the containment penetrations prohibit access to a portion of the pipe segment. The containment penetration sleeve also extends to within 3 inches of the containment isolation valve prohibiting access to a portion of the piping within the sleeve. Due to this configuration, it is not practical to use a soap bubble technique VT-2 visual examination along the entire length of the pipe. Even the removal of the fire penetration barriers and rubber boots would not allow for a complete soap bubble technique VT-2 visual examination due to the length of the pipe and lack of clearance between the pipe and penetration sleeve.

During normal operations, the containment purge valves are normally closed and the piping is subjected to atmospheric conditions. The inaccessible area contains only longitudinal welds. As these longitudinal welds area were made during the pipe manufacturing process under controlled conditions, it is highly unlikely that inservice flaws would develop under the piping's normal service conditions.

Accordingly, relief is requested pursuant to 10 CFR 50.55a(g)(5)(iii) as the VT-2 examination of the inaccessible portions of the Containment Purge System piping is impractical to perform. A 10 CFR 50.55a request for this piping was previously approved by the NRC for the Second 10-Year Interval Inservice Inspection Program as Relief Request RR-B5 (TAC Nos. M79034 and M77942).

Licensee's Proposed Alternative Examination: (as stated)
Containment Purge System Penetrations 33 and 34 piping will be tested in accordance with 10 CFR [sic], Appendix J, Option B, Type C testing requirements. If leakage is noted during the test, the containment isolation valves and the accessible portions of the piping segment between the containment isolation valves will be VT-2 visually examined with the soap bubble technique to verify the source of leakage is not through-wall leakage.

Staff Evaluation:
The Code requirement for the subject piping includes system pressure tests as specified in IWC-5221 and IWC-5222. Paragraph IWC-5210(b) states: "The contained fluid in the system shall serve as the pressurizing medium, except that in steam systems either water or air may be used. Where air is used, the test procedure shall permit the detection and location of through-wall leakages in components of the system tested."

The licensee noted that rubber foam barriers and rubber boots installed around the piping at the containment penetrations prohibit access to a portion of the pipe segment. The containment penetration sleeve also extends to within 3 inches of the containment isolation valve prohibiting access to a portion of the piping within the sleeve. Furthermore, the removal of the fire penetration barriers and rubber boots would not allow for a complete soap bubble technique VT-2 visual examination due to the length of the pipe and lack of clearance between the pipe and penetration sleeve. Therefore, the staff determined that the Code requirements are impractical and to require the licensee to the perform the Code required system pressure tests would be a significant burden, because the subject components would be required to be redesigned.

The licensee proposed that for the Containment Purge System Penetrations 33 and 34 piping will be tested in accordance with 10 CFR 50, Appendix J, Option B, Type C testing requirements. Furthermore, if leakage is noted during the test, the containment isolation valves and the accessible portions of the piping segment between the containment isolation valves will be VT-2 visually examined with the soap bubble technique to verify the source of leakage is not through-wall leakage. The staff determined that the licensee's proposed alternative provides reasonable assurance of structural integrity of the subject components. Therefore, the licensee's relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year interval, which expires September 20, 2010. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

The staff concludes that for RR-B4 the Code requirements are impractical and to require the licensee to perform the Code required system pressure tests would be a significant burden, because the subject components would be required to be redesigned. Furthermore, the licensee's proposed alternative provides reasonable assurance of structural integrity of the subject components. Therefore, the licensee's relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year interval, which expires September 20, 2010. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

3.15 Request for Relief No. RR-C1:

Code Requirement: ASME Section XI, 1995 Edition, 1996 Addenda, Examination Category D-B, Table IWD-2500-1 requires hydrostatic pressure testing of pressure retaining Class 3 components each inspection interval.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(i), relief is requested from performing the system hydrostatic test as required by Table IWD-2500-1, Examination Category D-B, Code Items D2.20, D2.40, D2.60, and D2.80, 1995 Edition, 1996 Addenda. Code Case N-498-1 which provides alternative rules for the 10-year hydrostatic test will be used in lieu of the Examination Category D-B requirements.

Licensee's Basis for Requesting Relief: (as stated)
The 1998 Edition, Supplement 8 of the ASME Code Cases for Nuclear Components states that Code Case N-498-1 is only applicable up to and including the 1992 Edition with the 1993 Addenda of ASME Section XI. Code Case N-498-1 provides alternative requirements for the 10-year hydrostatic test Class 1, 2, and 3 Systems. Code Case N-498-1 has been approved by the NRC without limitations in NRC Regulatory Guide 1.147 Revision 12. As the Davis-Besse Third 10-Year Inservice Inspection Program is written to comply with the 1995 Edition, 1996 Addenda of ASME Section XI, Code Case N-498-1 cannot be applied.

The examination requirements in Table IWD-2500-1 for Examination Category D-B, [sic]. All Pressure Retaining Components, in the 1992 Edition, 1993 Addenda of ASME Section XI are identical to those contained in the 1995 Edition, 1996 Addenda of ASME Section XI for Examination Category D-B. Therefore, Code Case N-498-1 which is applicable to the 1992 Edition, 1993 Addenda should also be applicable to the 1995 Edition, 1996 Addenda.

As noted in NRC Regulatory Guide 1.147, Revision 12, the NRC has found the use of Code Case N-498-1 technically acceptable. There are no limitations or exemptions noted to the Code Case in Regulatory Guide 1.147. Relief is requested pursuant to 10 CFR 50.55a(a)(3)(i). The use of Code Case N 498-1 will provide an acceptable level of quality and safety.

<u>Licensee's Proposed Alternative Examination</u>: (as stated)
Code Case N-498-1 will be used as an alternative to the system hydrostatic test requirements of Category D-B of the 1995 Edition, 1996 Addenda of ASME Section XI.

<u>Staff Evaluation:</u>
The Code requires that a system hydrostatic test be performed once per interval to include all Class 3 components. The licensee has proposed an alternative to the Code requirements for the Class 3 systems. The licensee requested that a Class 3 system leakage test be conducted in lieu of the Class 3 system hydrostatic test.

Hardships are generally encountered with the performance of hydrostatic testing in accordance with the Code. Hydrostatic pressure testing frequently requires a significant effort to set up and perform due to the need to use special equipment, such as temporary attachment of test pumps and gages, and the need for unique valve lineups. Hydrostatic testing only subjects the piping components to a small increase in pressure over the design pressure and, therefore, does not present a significant challenge to pressure boundary integrity. Accordingly, hydrostatic pressure testing is primarily regarded as a means to enhance leak detection during the examination of components under pressure, rather than as a measure of the structural integrity of the components.

Considering that the hydrostatic pressure tests rarely result in pressure boundary leaks that would not occur during system leakage tests, the staff believes that the increased assurance of the integrity of Class 3 systems that could be achieved by the performance of a hydrostatic test is not commensurate with the burden of performing such a test. It is also believed that the added assurance provided by a hydrostatic test of Class 3 welds over that provided by a system pressure test is not commensurate with the burden of hydrostatic testing.

Therefore, the Code requirement to perform a system hydrostatic test once per interval for Class 3 systems is a hardship without a compensating increase in the level of quality and safety. The licensee's proposed alternative to perform system leakage tests on Class 3 systems provides reasonable assurance of structural integrity of the subject Class 3 systems. Therefore, the licensee's proposed alternative to use Code Case N-498-1 for Class 3 systems is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval, which expires September 20, 2010. Use of the code case is authorized until such time as the code case is published in a future version of RG1.147. At that time, if the licensee intends to continue implementing this code case, it must follow all provisions of the subject Code Case with limitations or conditions specified in RG 1.147, if any.

## 4.0 CONCLUSION

For Requests for Relief RR-A1, RR-A6, RR-A7, and RR-C1 the staff concludes that the Code requirements are a hardship without a compensating increase in quality and safety. Furthermore, the licensee's proposed alternatives provide reasonable assurance of structural integrity of the subject components contained in the licensee's requests for relief. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii), for the third 10-year inservice inspection interval, which expires September 20, 2010.

For RR-A3, the staff concludes that the removal of insulation at elevated pressures and temperatures would result in a hardship without a compensating increase in the level of quality and safety. Furthermore, the staff concludes that the licensee's proposed alternative provides reasonable assurance of structural integrity of the bolted connections. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year interval, which expires September 20, 2010 or until such time as Code Case N-616 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-616, it must follow all provisions in the subject code case with the limitations or conditions listed in RG 1.147, if any.

For Requests for Relief RR-A8, RR-A16, and RR-B2, the staff concludes that the licensee's proposed alternatives provide reasonable assurance of quality and safety and are authorized pursuant to 10 CFR 5.55a(a)(3)(i) for the third 10-year interval, which expires September 20, 2010.

For Requests for Relief RR-A4, RR-A5, RR-B1, RR-B3 and RR-B4, the staff concludes that the Code-required examinations are impractical and to require the licensee to perform the Code-required examinations would be a burden on the licensee. The subject components contained in the request for relief would be required to be redesigned in order for the licensee to perform the Code-required examinations. Therefore, relief is granted for RR-A4, RR-A5, RR-B1, RR-B3, and RR-B4 pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year interval, which expires September 20, 2010. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

For RR-A15, the staff determined that the Code requirements are impractical. To impose the Code requirements would cause a burden on the licensee, because the subject components would have to be redesigned in order to perform the Code required examinations. However, by November 22, 2002, it is expected that there will be a qualified procedure to examine dissimilar metal welds and austenitic stainless steel welds from the pipe bore. Therefore, for dissimilar metal welds and austenitic stainless steel welds, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval, but not beyond November 22, 2002. For welds connecting cast stainless steel components, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval, which expires September 20, 2010. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or 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.

For RR-A17, the staff concludes that the licensee's proposed alternative to use Code Case N-546 with the 1995 Edition vision test requirements 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 third 10-year interval, which expires September 20, 2010 or until such time Code Case N-546 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-546, it must follow all provisions in the subject code case with the limitations or conditions specified in RG 1.147, if any.

For RR-A10, RR-A11, RR-A12, and RR-A14, the licensee has proposed to use Code Cases N-566-1, N-598, N-623, and N-639, respectively, as alternatives to the Code requirements. The staff concludes that the proposed alternatives provide an acceptable level of quality and safety. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year interval, which expires September 20, 2010 or until such time as Code Cases N-598, N-623, and N-639 are published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Cases N-598, N-623, and N-639, it must follow all provisions in the subject code cases with the limitations or conditions specified in RG 1.147, if any.

Principal Contributor: Thomas McLellan, NRR/EMCB

Date: September 30, 2002

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI

RELIEF REQUEST RR-A13 TO USE ASME CODE CASE N-528

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION

DOCKET NO. 50-346

## 1.0 INTRODUCTION

By letter dated September 19, 2000, FirstEnergy Nuclear Operating Company (the licensee) requested relief from the administrative requirements of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for the third 10-year interval inspection program at the Davis-Besse Nuclear Power Station Unit 1. The licensee proposes to apply ASME Code Case N-528, "Purchase, Exchange, or Transfer of Material Between Nuclear Plant Sites." The third 10-year interval began on September 21, 2001.

Relief Request (RR)-A13 proposes to adopt Code Case N-528 as an alternative means of satisfying certain requirements of Section XI, Subarticle IWA-4220, "Code Applicability," with respect to the possession of a Certificate of Authorization or Quality System Certificate (Materials). This safety evaluation addresses the acceptability of this alternative.

## 2.0 REGULATORY EVALUATION

### 2.1 Regulatory Requirements (Procurement)

Appendix B to 10 CFR Part 50 contains the Nuclear Regulatory Commission's (NRC's) regulations for procurement quality assurance and quality control for items to be used in safety-related applications. The NRC has provided further guidance in Regulatory Guides (RGs) 1.33, and 1.123 (Ref. [1], [2]). RG 1.33 and RG 1.123 respectively endorse American National Standards Institute (ANSI) N18.7-1976 and ANSI N45.2.13-1976. For replacement parts, RG 1.123 also specifically endorses Section 5.2.13 of ANSI N18.7-1976. These standards supplement the Appendix B criteria in providing further guidance for procurement of safety-related applications. This guidance, if properly implemented, provides a measure of assurance for the suitability of equipment for safety-related applications.

Criterion III of Appendix B requires licensees to select and review for suitability of application materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems, and components. Criterion IV requires that procurement documents specify the applicable requirements necessary to ensure functional performance. Criterion VII requires licensees to assure that the following are sufficient to identify whether specification requirements for the procured material and equipment have been met: source evaluation and selection, objective evidence of quality, inspection of the source, and examination of products upon delivery. The process of ensuring compliance with 10 CFR Part 50, Appendix B, must include all those activities necessary to establish and confirm the quality and suitability of the procured material and equipment for its intended safety-related application.

## 2.2 Regulatory Requirements (ASME)

Section 50.55a, “Codes and Standards,” of 10 CFR Part 50 requires, in part, that each operating license for a boiling or pressurized water-cooled nuclear power facility be subject to the conditions in paragraph 50.55a(g), “Inservice Inspection Requirements.” Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in paragraph 50.55a(b). Paragraph 50.55a(b) incorporates the 1995 Edition and addenda of Section XI, Division 1, through the 1996 addenda.

## 2.3 Alternatives to Section XI Inservice Inspection Requirements

The regulations require that inservice inspection of certain components be performed in accordance with Section XI of the ASME Code and applicable addenda, except where alternatives have been authorized or relief has been requested by the licensee and granted by the Commission pursuant to 10 CFR 50.55a, paragraph (a)(3)(i), (a)(3)(ii), or (g)(6)(i). These provisions provide for relief when the applicant demonstrates that (1) the proposed alternative would provide an acceptable level of quality and safety, (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, or (3) the Code requirements are impractical.

The ASME Boiler and Pressure Vessel Committee publishes a document entitled “Code Cases,” which is updated every three years (Ref. [3].) Generally, the individual Code Cases that make up this document explain the intent of Code rules or provide for alternative requirements under special circumstances. Most Code Cases are eventually superseded by revision of the Code and then are annulled by action of the ASME.

RG 1.147 (Ref. [4]) lists those Section XI ASME Code Cases that are generally acceptable to the NRC staff for implementation in the inservice inspection of light-water-cooled nuclear power plants. Code Cases that are not listed in RG-1.147 require supplementary provisions on an individual plant basis to attain endorsement status. The staff has not generally endorsed Code Case N-528 by inclusion in RG-1.147 and, consequently, its acceptability must be evaluated on an individual plant basis.

2.4 Affected ASME Code Requirements

Article IWA-7000 of the Section XI, 1989 edition (Article IWA-4000, subsequent to the 1991 addenda) provides the rules and requirements for the specification and construction of items to be used for replacement. Replacement includes the addition of components, such as valves, pumps and system changes, such as rerouting of piping. Subarticle IWA-7210 (IWA-4170 for the 1991 addenda through the 1995 addition, no addenda and IWA 4220 subsequent to the 1995 edition, no addenda) require that an item to be used for replacement meet the original Construction Code (Section III of the Code) and existing design requirements.

Article NCA-3000 of Section III of the Code defines the responsibilities of N Certificate Holders. Subarticle NCA-3700 defines the responsibilities of holders of Certificates of Authorization, which is generally the organization which performs the activities to place and attach components to their support structures. The responsibilities of N Certificate Holders include surveying, qualifying, and auditing suppliers of subcontracted services, including material suppliers and material manufacturers. When material suppliers or material manufacturers hold a Quality System Certificate (Materials), as defined in subarticle NCA-3800, the Certificate Holder does not need to survey or audit the supplier for work within the scope of the Quality System Certificate.

2.5 Code Case N-528

Case N-528 applies to metallic material (meeting the definition of IWA-9000) that is purchased, exchanged, or transferred between nuclear plant sites. Case N-528 provides an alternative to the specific administrative requirements of Section III that refer to possession of a Certificate of Authorization or Quality System Certificate (Materials). The case was approved by the ASME Boiler and Pressure Vessel Committee on December 12, 1994, and reaffirmed on August 14, 1997. Code Case N-528-1 was approved on May 7, 1999, and is still active.

Code Case N-528 provides an alternative to the requirements of NCA-3700/NCA-3800 in that the responsibilities of the N Certificate Holder are, in fact, imposed on the supplying plant. All documentation required by NCA-3700/NCA-3800 are provided to the receiving plant with the material.

For material that has been fabricated in accordance with specific dimensional requirements in addition to those provided in a national standard (e.g., nonwelded valve bonnet or nonwelded pump casing), Code Case N-528 requires the licensee to include in the evaluation of suitability, required by IWA-7220 (IWA-4150 for the 1991 addenda through the 1995 edition, no addenda and IWA 4160 subsequent to the 1995 edition, no addenda), an evaluation of the material for its intended application, including any differences that might affect form, fit, or function.

The licensee shall obtain, and incorporate into its plant record system, certifying documentation that the subject material was purchased in accordance with the provisions of NCA-3700/NCA-3800 and maintained in accordance with the supplier's quality assurance program.

The licensee shall also obtain and incorporate into its plant records system, certification provided by the supplier that the material was not placed in service, nor subject to any operation that might affect the mechanical properties of the material. The licensee shall document, on the ASME Owner's Report for Inservice Inspection (Form NIS-2), each instance in which Code Case N-528 was applied.

## 3.0 TECHNICAL EVALUATION

With the exception to the ASME Section XI administrative requirements explicitly stated by Code Case N-528, the licenses makes no changes to its approved Appendix B program or regulatory guides to which it has committed. The licensee's quality assurance program conforms to the guidance provided by RG-1.33, Revision 2 and RG-1.123, Revision 1.

With respect to Appendix B criteria, Criterion VII provides the specific regulatory requirements for control of purchased material, equipment, and services. Criterion VII requires, in part, that

> "...measures be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery. Documentary evidence that material and equipment conform to the procurement requirements shall be available at the nuclear power plant or fuel reprocessing plant site prior to installation or use of such material and equipment. This documentary evidence shall be retained at the nuclear power plant or reprocessing plant site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment."

The licensee requests relief in the specific area of source evaluation. In effect, the supplying plant fulfills the regulatory requirement for source evaluation by originally procuring the material and documentation in conformance with Section III of the Code and subsequently maintaining the material in accordance with its approved Appendix B quality assurance program. In addition, Code Case N-528 stipulates that the documentary evidence required by Criterion VII be transferred to the licensee with the material and subsequently maintained by the licensee.

Other regulatory procurement requirements continue to apply. The licensee is responsible for ensuring that the material is in conformance with all other Code requirements, applicable design requirements, its Appendix B program, and other regulatory requirements and commitments. The licensee is also responsible for ensuring that the item is suitable for the intended application and documenting this evaluation. The proposed alternative is acceptable on the basis that it provides an acceptable level of quality and safety.

## 4.0 CONCLUSIONS

The staff has evaluated Code Case N-528 as an acceptable alternative to certain administrative requirements of Section III, when material is purchased, exchanged, or transferred between nuclear plant sites. The code case requires that the material was originally procured in compliance with ASME Code Section III requirements, maintained in conformance with an approved Appendix B program, and not subject to any operation that might affect the mechanical properties of the material.

The licensee is responsible for ensuring that the received documentation is complete and in compliance with Code requirements, that the material meets the design requirements for the intended application, and that the material conforms to the licensee's Appendix B program and all other regulatory requirements and commitments.

These requirements provide reasonable assurance that the proposed alternative provides an acceptable level of quality and safety in accordance with paragraph 50.55a(a)(3)(i). Therefore, the alternative provided by Code Case N-528 is acceptable for the licensee's third 10-year inspection interval, which expires September 20, 2010, or until such time Code Case N-528 is published in a future revision of RG 1.147. At that time, if the licensee intends to continue to implement Code Case N-528, it must follow all provisions in the subject code case with the limitations or conditions specified in RG 1.147, if any.

## 5.0 REFERENCES

1. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, 1978.

2. Regulatory Guide 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," Revision 1, 1977.

3. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, "1995 Code Cases," published July 1, 1995, and "1998 Code Cases," published July 1, 1998.

4. Regulatory Guide 1.147, "Inservice Code Case Acceptability, ASME Section XI, Division 1," Revision 12, May 1999.

5. Letter, FirstEnergy Nuclear Operating Company to the NRC, "Third 10-year Interval Inservice Inspection Program for Davis-Besse Nuclear Power Station Unit 1, dated September 19, 2000.

Principal Contributor: R. McIntyre, NRR

Date: September 30, 2002

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI

RELIEF REQUEST NOS. RR-A18 AND RR-A19

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION

DOCKET NO. 50-346

## 1.0 INTRODUCTION

By a letter, dated September 19, 2000, FirstEnergy Nuclear Operating Company (the licensee) requested approval to use Davis-Besse Unit No. 1 Technical Specification 3/4.7.7 as an alternative to the ASME Code Section XI, Subsection IWF-5200(a) and (b) and IWF-5300(a) and (b) for the examination and testing requirements for snubbers (RR-A18), and the use of the requirements of IWA-2317 of the 1998 Edition of ASME Section XI as an alternative to the provisions of IWA-2313 and IWA-2314 for visual examination personnel performing VT-3 snubber examination (RR-A19).

As a result of its review of the licensee's submittal, the staff identified certain areas where additional information and clarification were needed from the licensee. The licensee responded to the staff questions in a telephone conference held on January 24, 2002, and documented its responses and revised Relief Request RR-A18 in a letter dated February 6, 2002.

## 2.0 REGULATORY EVALUATION

The inservice inspection (ISI) of the ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (ASME 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). Section 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the Nuclear Regulatory Commission (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) will meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code Section XI, 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. The applicable edition of Section XI of the ASME Code for the Davis-Besse Nuclear Power Station Unit No. 1 third 10-year ISI interval is the 1995 Edition and Addenda through the 1996 Addenda.

Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility, information will be submitted to the Commission in support of that determination and a request must be 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/or may impose alternative 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.

## 3.0 TECHNICAL EVALUATION

### 3.1 Basis for Relief Requests

IWF-5200(a) of the 1995 Edition,1996 Addenda of ASME Section XI requires preservice examination be performed in accordance with ASME/ANSI OM, Part 4, using the VT-3 visual examination method described in IWA-2213. IWF-5200(b) of the 1995 Edition,1996 Addenda of ASME Section XI requires preservice tests be performed in accordance with ASME/ANSI OM, Part 4. IWF-5300(a) of the 1995 Edition,1996 Addenda of ASME Section XI requires inservice examination be performed in accordance with ASME/ANSI OM, Part 4, using the VT-3 visual examination method described in IWA-2213.

ASME section XI, Subsection IWF-5200(a) and (b) and Subsection IWF-5300(a) and (b) of 1995 Edition,1996 Addenda specify that snubber examinations and tests be performed in accordance with the 1987 Edition with OMa-1988 of ASME/ANSI OM, Part 4.

10 CFR 50.55a(b)(3)(v) permits the use of the Subsection ISTD of the ASME OM Code 1995 Edition up to and including the 1996 Addenda in lieu of the 1987 Edition with OM-1988 of ASME/ANSI OM, Part 4.

Snubber examination and testing is currently performed in accordance with the Davis-Besse Unit No. 1 technical specifications. The Davis-Besse Unit No. 1 technical specifications meet the requirements of NRC Generic Letter 90-09, Alternative Requirements for Snubber Visual Inspection Intervals and Corrective actions.

The requirements for the examination and testing of snubbers in the Davis-Besse Unit No. 1 technical specifications are similar to and provide more thorough examinations and tests than required by ASME/ANSI OM Code, Subsection ISTD. For example, while the examination boundary specified in Subsection ISTD 2.1 includes only the snubber assembly from pin to pin, inclusive, the Davis-Besse Unit No. 1 technical specifications extend examination includes the attachments to the foundation or supporting structures. In addition, while the Davis-Besse Unit No. 1 technical specifications require examination and testing of snubbers which are installed on non-safety related systems when their failure or failure of the system on which they are installed would have an adverse effect on safety-related system during a dynamic event, the snubbers under IWF-5000 would not include these non-safety related snubbers because they would not be within the ASME Section XI Class 1, 2, or 3 boundaries.

The 1995 Edition, 1996 addenda of ASME Section XI requires personnel conducting VT-3 examinations be qualified and certified to comparable levels of qualification as defined in ANSI/ASNT CP-189 and the Employer's written practice. IWA-2317 of the 1998 Edition of the ASME Section XI provides alternative requirements for the qualification of VT-3 examination personnel. The requirements of IWA-2317 are less burdensome than qualifying and maintaining the VT-3 certification program required by IWA-2313. IWA-2317 makes it feasible to train and qualify experienced personnel to perform VT-3 examinations.

3.2 Staff Evaluation

The licensee states that its applicable edition of Section XI of the ASME Code for the Davis-Besse Nuclear Power Station Unit No. 1 third 10-year ISI interval is the1995 Edition and Addenda through the 1996 Addenda. ASME section XI, Subsection IWF-5200(a) and (b) and Subsection IWF-5300(a) and (b) of1995 Edition,1996 Addenda specify that snubber examinations and tests be performed in accordance with the 1987 Edition with OMa-1988 of ASME/ANSI OM, Part 4. 10 CFR 50.55a(b)(3)(v) permits the use of the Subsection ISTD of the ASME OM Code 1995 Edition up to and including the 1996 Addenda in lieu of the 1987 Edition with OM-1988 of ASME/ANSI OM, Part 4.

The licensee states that snubber examination and testing is currently performed in accordance with the Davis-Besse Unit No. 1 technical specifications, and the Davis-Besse Unit No. 1 technical specifications meet the requirements of NRC Generic Letter 90-09, Alternative Requirements for Snubber Visual Inspection Intervals and Corrective actions. Therefore, the licensee's testing frequency for snubbers is acceptable to the staff.

The Davis-Besse Unit No. 1 technical specifications require a 10 percent representative sample of the snubbers be tested each refueling outage with each snubber requiring testing at least once every ten refueling outages. This is similar to the Subsection ISTD 10 percent sampling plan, and is acceptable. Subsection ISTD 7 requires the establishment of failure mode groups when test failures occur within a Design Test Plan Group. If the cause of the failure can be determined and the failure is determined to be isolated, no further testing is required of the test group when using Subsection ISTD. The Davis-Besse Unit No. 1 technical specifications state that any snubber failure in tests, whether isolated or not, requires testing an additional 10 percent of the snubbers within the failed snubber's group to ensure the acceptability of the group. This testing continues in 10 percent increments until that additional 10 percent sample is acceptable. The staff finds the licensee's 10 percent sampling plan and the 10 percent increment testing method comparable to the ISTD's requirements and is acceptable.

Subsection ISTD 2.1 specifies the examination boundary to be the snubber assembly from pin-pin, and the Davis-Besse Unit No. 1 technical specifications specify the same examination boundary and extend it to attachments. The staff finds the examination boundary in the Davis-Besse Unit No. 1 technical specifications acceptable. The staff also finds the Davis-Besse Unit No. 1 technical specifications' requirements to examine and test snubbers which are installed on non-safety related systems when their failure or failure of the system on which they are installed would have an adverse effect on safety-related system during a dynamic event acceptable.

Subsection IWA-2317 of the 1998 Edition of ASME Section XI provides alternative requirements for the qualification of VT-3 examination personnel. The staff accepted Subsection IWA-2317 of the 1998 Edition of ASME Section XI because it provides an acceptable level of quality and safety comparable to that which would be obtained by using personnel qualified to the levels of qualification as defined in ANSI/ASNT CP-189.

## 4.0 CONCLUSION

The staff concludes that the licensee's proposed use of the Davis-Besse Unit No. 1 Technical Specification 3/4.7.7 as an alternative to the ASME Code Section XI, Subsection IWF-5200(a) and (b) and IWF-5300(a) and (b) for the examination and testing requirements for snubbers (RR-A18), and use of the requirements of IWA-2317 of the 1998 Edition of ASME Section XI as an alternative to the provisions of IWA-2313 and IWA-2314 for visual examination personnel performing VT-3 snubber examination (RR-A19) provide an acceptable level of quality and safety in regard to the examination and testing of snubbers and VT-3 examination personnel qualifications. Therefore, the licensee's proposed alternatives with regard to the examination and testing of snubbers and VT-3 examination personnel qualifications are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year interval, which expires September 20, 2010.

Principal Contributor: J. Ma, NRR

Date: September 30, 2002

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI

RELIEF REQUEST NOS. RR-E1 THROUGH RR-E8

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE NUCLEAR POWER STATION

DOCKET NO. 50-346

## 1.0 INTRODUCTION

By the letter dated September 19, 2000 (Reference 1), FirstEnergy Nuclear Operating Company (the licensee), proposed several alternatives to the requirements of Subsection IWE of Section XI of the ASME Code (Relief Requests RR-E1 through RR-E8) for its Davis-Besse Nuclear Power Station (DBNPS). The NRC's findings with respect to authorizing the alternatives or denying the proposed requests are discussed in this evaluation.

## 2.0 REGULATORY EVALUATION

In the Federal Register dated August 8, 1996 (61 FR 41303), the Nuclear Regulatory Commission (NRC) amended its regulations to incorporate by reference the 1992 edition with 1992 addenda of Subsections IWE and IWL of Section XI of the ASME Boiler and Pressure Vessel Code (Code). Subsections IWE and IWL provide the requirements for inservice inspection (ISI) of Class CC (concrete containment), and Class MC (metallic containment) of light-water cooled power plants. The effective date for the amended rule was September 9, 1996, and it requires the licensees to incorporate the new requirements into their ISI plans and to complete the first containment inspection by September 9, 2001. However, a licensee may propose alternatives to or submit a request for relief from the requirements of the regulation pursuant to 10 CFR 50.55a(a)(3) and (g)(5).

## 3.0 TECHNICAL EVALUATION

### 3.1 Relief Request RR-E1:

#### 3.1.1 Code Requirements:

IWE-2500, Table IWE-2500-1, Examination Category E-D, "Seals, Gaskets, and Moisture Barriers," requires seals and gaskets on airlocks, hatches, and other devices to be visually examined, VT-3, when disassembled.

3.1.2 Requirements from Which Relief is Requested:

Relief is requested from performing the Code-required visual examination, VT-3, on the metal containment seals and gaskets.

3.1.3 Basis for Relief:

The penetrations discussed below contain seals and gaskets.

Electrical Penetrations:

Electrical penetrations use a header plate attached to a containment penetration nozzle flange with redundant O-rings between the header plate and flange face. Modules through which electrical conductors pass are installed in the header plate. One type, manufactured by Amphenol, uses seals and gaskets to assure leak-tight integrity. A second type, manufactured by Conax, uses a set of compression fittings. Replacement modules for the Amphenol penetrations use a combination of O-rings and compression fittings. Each penetration is pressurized by dry nitrogen to maintain and monitor integrity and to prevent the intrusion of moisture into the penetration.

These seals and gaskets cannot be inspected without disassembly of the penetration to gain access to the seals and gaskets.

Containment Personnel, Equipment, and Emergency Escape Hatches:

The personnel, equipment, and emergency escape hatches utilize an inner and outer door with gasket surfaces to ensure the leak-tight integrity. These hatches also contain other gaskets and seals such as the handwheel shaft seals, electrical penetrations, blank flanges, and equalizing pressure connections that require disassembly to gain access to the gaskets and seals.

Seals and gaskets receive a 10 CFR Part 50, Appendix J, Type B test. As noted in 10 CFR Part 50, Appendix J, the purpose of Type B tests is to measure leakage of containment or penetrations whose design incorporates resilient seals, gaskets, sealant compounds, and electrical penetrations fitted with flexible metal seal assemblies. The seals and gaskets in these joints are therefore proven adequate through Appendix J testing.

The equipment hatch is removed during maintenance outages, when necessary, and during refueling outages. Prior to final closure, the hatch gaskets and door sealing face are inspected for damage that could prevent sealing. The personnel hatch and emergency escape hatch are included in the DBNPS preventive maintenance program. This program requires that the door gaskets be inspected for damage that could prevent sealing and be coated with an approved silicon lubricant to preserve their integrity. These inspections are performed each refueling outage. Prior to establishing containment integrity, the equipment hatch, personnel hatch, and the emergency escape hatch are tested in accordance with 10 CFR Part 50, Appendix J to confirm their sealing capability.

When the electrical penetrations, airlocks and hatches containing seals and gaskets are tested in accordance with 10 CFR Part 50, Appendix J, degradation of the seal or gasket material would be revealed by an increase in leakage rate. Corrective measures would be applied and the component retested. Repair or replacement of seals and gaskets is not subject to Code (1995 Edition, 1996 Addenda) rules in accordance with Paragraph IWA-4120(b)(5) of ASME Section XI.

The visual examination of seals and gaskets in accordance with IWE-2500, Table IWE-2500-1 is a burden without any compensating increase in the level of safety or quality.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). Testing the seals and gaskets in accordance with 10 CFR Part 50, Appendix J will provide adequate assurance of the leak-tight integrity of the seals and gaskets.

Subsection IWE of the 1998 Edition of ASME Section XI no longer requires the examination of seals and gaskets.

3.1.4 Alternative Examination:

The leak-tightness of seals and gaskets will be tested in accordance with 10 CFR Part 50, Appendix J.

3.1.5 Justification for Granting Relief:

The functionality of containment seals and gaskets is verified during the Type B testing required by 10 CFR Part 50, Appendix J. This testing provides an acceptable level of quality and safety in lieu of the Code required visual examinations.

3.1.6 Staff Evaluation of RR-E1:

As an alternative to the requirements (VT-3 examinations) of the ASME Section XI, Subsection IWE, the licensee proposed to use leak-rate testing in accordance with 10 CFR Part 50, Appendix J to examine the leak-tight integrity of containment seals and gaskets.

In its relief request, the licensee stated that because the seals and gaskets associated with these penetrations are not accessible for examination when the penetration is assembled, containment penetrations seals and gaskets must be disassembled and re-assembled for the purpose of performing the VT-3 visual examination. These activities (disassembly and reassembly of seals and gaskets) associated with a VT-3 visual examination would result in hardship without a compensating increase in the level of quality and safety, and also would introduce the possibility of component damage that would not otherwise occur. The periodic test in accordance with 10 CFR Part 50, Appendix J will detect and measure local leakage of containment or penetrations whose design incorporates resilient seals, gaskets, sealant compounds, and electrical penetrations fitted with flexible metal seal assemblies. If unacceptable leakage is identified during the test, corrective measures would be taken and the components would be re-tested.

Also, the staff finds that the changes to ASME Section XI, 1992 Edition, 1993 Addenda reflect that disassembly of joints for the sole purpose of performing visual examination is unwarranted. Requiring the licensee to disassemble components for the sole purpose of inspecting seals and gaskets would place a significant hardship on the licensee without a compensating increase in the level of quality and safety.

On the basis discussed above, the staff concludes that the alternative proposed by the licensee will provide reasonable assurance of the functional capability and integrity of the containment penetration seals and gaskets during the testing required by 10 CFR Part 50, Appendix J. Therefore, on the basis that compliance with the specific requirements of the Code would result in hardship without a compensating increase in the level of quality and safety, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year inservice inspection interval, which expires September 20, 2010.

3.2 Relief Request RR-E2:

3.2.1 Code Requirements:

ASME Section XI, Subsection IWE-2200(g) requires that when paint or coatings are reapplied, the condition of the new paint or coating shall be documented in the preservice examination records.

3.2.2 Requirements from Which Relief is Requested:

Relief is requested from the requirement to perform a preservice inspection of new paint or coatings.

3.2.3 Basis for Relief:

SECY 96-080, "Issuance of Final Amendment to 10 CFR Section 50.55a to Incorporate by Reference the ASME Boiler and Pressure Code (ASME Code), Section XI, Division 1, Subsection IWE and Subsection IWL," dated April 17, 1996, response to Comment 3.2 about IWE-2200(g) states, "In the NRC's opinion, this does not mean that a visual examination must be performed with every application of paint or coating. A visual examination of the topcoat to determine the soundness and the condition of the topcoat should be sufficient." This is currently accomplished through the inspection performed by the DBNPS coating maintenance program.

The adequacy of applied coatings is verified through the inspections performed by the DBNPS coating maintenance program. The coatings on the interior surface of the containment vessel are considered nuclear safety-related. They are applied and inspected in accordance with the licensee's NRC-approved Quality Assurance Program. This program endorses NRC Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," June 1973, and ANSI Standard N101.4-1972, "Quality Assurance Protective Coatings Applied to Nuclear Facilities." The following requirements are applicable for coatings applied to the interior surface of the containment vessel:

- The quality assurance requirements of Section 3 of ANSI N101.4 applicable to the coating manufacturer are imposed on the coating manufacturer through the procurement process.

- Coating application procedures are developed based on the manufacturer's recommendations for application of the selected coating systems.

- Coating applicators are qualified to demonstrate their ability to satisfactorily apply the coatings in accordance with the manufacturer's recommendations.

- Quality Control personnel perform inspections to verify conformance of the coating application procedures. Section 6 of ANSI N101.4 is used as a guideline in the establishment of the inspection program.

- Quality Control inspection personnel are qualified to the requirements of Regulatory Guide 1.58, Revision 1.

- Documentation demonstrating conformance to the above is maintained.

The condition of the coatings are examined every four to six years in accordance with 10 CFR 50.65, "Requirements for Monitoring Effectiveness of Maintenance at Nuclear Power Plants." The general visual examination required by IWE is also performed each inspection period. These periodic examinations will identify evidence of flaking, blistering, peeling, discoloration, or other signs of coating distress that might be indicative of degradation of the containment structural integrity.

Recording the condition of reapplied coating in the preservice record does not substantiate the containment structural integrity. Should deterioration of the coating in the reapplied area occur, the area will require additional evaluation regardless of the preservice record. Recording the condition of new paint or coating in the preservice records does not increase the level of quality and safety of the containment.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). The DBNPS coating maintenance program currently provides an adequate level of quality and safety.

Subsection IWE of the 1998 Edition of ASME Section XI no longer requires a preservice record of reapplied coatings.

3.2.4 Alternative Examination:

Reapplied paint and coatings on the containment vessel will be examined in accordance with the DBNPS coatings program. Although repairs to paint or coatings are not subject to the repair/replacement rules of ASME Section XI (Inquiry 97-22), repairs to the primary containment boundary, if required, would be conducted in accordance with ASME Section XI Code rules.

3.2.5 Justification for Granting Relief:

The code requirement to establish a preservice record is a duplication of requirements contained in the DBNPS coatings program. The DBNPS coating program provides the inspections and quality assurance provisions for the application of coatings necessary for protecting the inside steel surfaces of the Davis-Besse containment vessel.

3.2.6 Staff Evaluation of RR-E2:

In lieu of meeting the ASME Section XI, 1992 Edition, 1992 Addenda, Subsection IWE-2200(g) requirements to perform a preservice inspection of new paint or coatings, the licensee proposed to examine the reapplied paint and coatings on the containment vessel in accordance with the DBNPS coatings program. In the “Basis for Relief” section of the request, the licensee provided a description of the requirements used for coatings applied to the interior surface of the containment vessel.

In this request, the licensee stated that the reapplied paint and coatings on the containment vessel will be examined in accordance with the DBNPS coatings program. The adequacy of paint and coatings will be verified following application through inspections performed by the DBNPS coating maintenance program. The coatings on the interior surface of the containment vessel are applied and inspected in accordance with the licensee’s NRC-approved Quality Assurance Program which meets the guidelines of Regulatory Guide 1.54, “Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants,” June 1973, and ANSI Standard N101.4-1972, “Quality Assurance Protective Coatings Applied to Nuclear Facilities.” The licensee also stated that the condition of the coatings are examined every 4 to 6 years in accordance with 10 CFR 50.65. The general visual examination required by IWE is also performed each inspection period. In addition, the licensee committed that repairs to the primary containment boundary would be conducted in accordance with the ASME Section XI Code rules.

The staff finds that SECY 96-080, response to Comment 3.2 about IWE-2200(g) states, "in the NRC’s opinion, this does not mean that visual examination must be performed with every application of paint or coating. A visual examination of the topcoat to determine the soundness and the condition of the topcoat should be sufficient." The staff also finds that the licensee used the DBNPS coatings program together with the DBNPS coating maintenance program and FENOC Quality Assurance Program for the response to NRC Generic Letter 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,” (Reference 2). Through the NRC close-out letter for Generic Letter 98-04 dated December 2, 1999 (Reference 3), this program was approved by the staff.

From the discussion above, the staff finds that the DBNPS Coatings Program is adequate for the examinations of the safety-related protective coating work and will provide an acceptable level of quality and safety for protecting containment components. On this basis, the staff concludes that the alternative proposed by the licensee to the requirements of IWE-2200(g) is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year inservice inspection interval, which expires September 20, 2010.

3.3 Relief Request RR-E3:

3.3.1 Code Requirements:

ASME Section XI, 1992 Edition, 1992 Addenda, Subarticle IWE-2500(b) requires that when paint or coatings are to be removed, the paint or coatings shall be visually examined in accordance with Table IWE-2500-1 prior to removal.

3.3.2 Requirements from Which Relief is Requested:

Subarticle IWE-2500(b) requires that when paint or coatings are to be removed, the paint or coatings shall be visually examined in accordance with Table IWE-2500-1 prior to removal.

3.3.3 Basis for Relief:

The DBNPS coating program is described in Relief Request RR-E2.

Paint and coatings are not part of the containment pressure boundary under current Code rules as they are not associated with the pressure retaining function of the component (Paragraph NE-2210(b)(5) of ASME Section III). The interiors of containments are painted to prevent rusting. Neither paint nor coatings contribute to the structural integrity or leakage tightness of the containment. Furthermore, the paint and coatings on the containment pressure boundary were not subject to Code rules when they were originally applied and are not subject to ASME Section XI rules for repair or replacement in accordance with IWA-4111(b)(5). Degradation or discoloration of the paint or coating materials on containment would be an indicator of potential degradation of the containment pressure boundary. Additional measures would have to be employed to determine the nature and extent of any degradation, if present. The application of ASME Section XI rules for removal of paint or coatings when unrelated to a Section XI repair or replacement activity, is a burden without a compensating increase in quality or safety.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). The DBNPS Coating Program currently provides an adequate level of quality and safety.

Subsection IWE of the 1998 Edition of ASME Section XI no longer requires an examination of coatings prior to removal.

3.3.4 Alternative Examination:

The condition of the containment vessel base material will be verified prior to the application of new paint or coating as required by the DBNPS Coatings Program. If degradation is identified, additional measures will be applied to determine if the containment pressure boundary is affected. Repairs to the primary containment pressure boundary, if required, will be conducted in accordance with ASME Section XI Code rules.

3.3.5 Justification for Granting Relief:

The DBNPS Coating Program is adequate to monitor the proper removal of the old paint and application of new coatings. Performing the Code required examination prior to removal of the old paint and documenting its condition in addition to performing the inspections required by the DBNPS Coatings Program would be a burden without a compensating increase in quality or safety.

3.3.6 Staff Evaluation of RR-E3:

In lieu of performing visual examination of paint or coatings in accordance with Table IWE-2500-1 prior to removal, the licensee proposed to inspect the condition of the containment base material prior to application of new paint or coatings in accordance with the DBNPS coating program. The licensee also committed that if degradation is identified, additional measures will be applied to determine if the containment pressure boundary is affected. Repairs to the primary containment pressure boundary, if required, will be conducted in accordance with ASME Section XI Code rules.

As discussed in the evaluation of Relief Request RR-E2, the staff finds that the DBNPS Coating program is adequate for monitoring the proper removal of the old paint and application of new coatings. To perform additional examinations prior to removal of the old paint and to document the condition of the old paint or coatings would result in hardship to the licensee without a compensating increase in the level of quality and safety. On this basis, the staff concludes that the alternative coating program proposed by the licensee is acceptable and the licensee's proposed alternative to the requirement of Subsection IWE-2500(b) is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year inservice inspection interval, which expires September 20, 2010.

3.4 Relief Request RR-E4:

3.4.1 Code Requirements:

Paragraph IWE-5240, "Visual Examination," of the 1995 Edition, 1996 Addenda of ASME Section XI requires that the requirements of Paragraph IWA-5240, "Visual Examination," for visual examination, VT-2, are applicable following repair, replacement, or modification.

3.4.2 Requirements from Which Relief is Requested:

Relief is requested from performing the VT-2 visual examination in connection with system pressure testing following repair, replacement, or modification under Article IWE-5000, "System Pressure Tests."

3.4.3 Basis for Relief:

Repair/replacements are performed in accordance with the DBNPS repair/replacement program which specifies the repair methods and nondestructive examinations necessary to ensure the original quality and construction requirements of the containment vessel are met. The DBNPS containment vessel is a code stamped pressure vessel which was designed and constructed in accordance with the 1968 Edition including the Summer 1969 Addenda of ASME Section III, Subsection “B” for nuclear vessels.

Table IWE-2500-1, Examination Category E-P, identifies the examination method of 10 CFR Part 50, Appendix J following each repair, modification, or replacement. Paragraph IWE-5222 permits leakage tests for minor repair/replacement activities to be deferred to the next scheduled leakage test provided nondestructive examinations are performed in accordance with the repair/replacement plan. These nondestructive examinations would be required to meet the construction code requirements, which would require volumetric surface examinations based on the type of repair.

Paragraph IWE-5210 states that except as noted within Paragraph IWE-5240, “Visual Examination,” the requirements of Article IWA-5000 are not applicable to Class MC or Class CC components. Paragraph IWE-5240 states that the requirements of Paragraph IWA-5240 for visual examinations are applicable. Paragraph IWA-5240 addresses VT-2 visual examination requirements. These requirements are written to apply to systems containing fluids. The VT-2 examination requires access to the repaired area during performance of the pressure test. Access to the repaired area may not be available if the repaired area is on the interior surface of the containment vessel if a 10 CFR Part 50, Appendix J, Type A is performed. If the repaired area is subjected to a localized 10 CFR Part 50, Appendix J pressure test, the repaired area would be covered by the test fixture and not available for visual examination. Paragraph IWA-2211 defines a VT-1 visual examination as an examination conducted to detect discontinuities and imperfection on the surfaces of components, including such conditions as cracks, wear, corrosion, or erosion. Paragraph IWA-2213 defines a VT-3 visual examination as an examination conducted to determine the general mechanical and structural condition of components.

The performance of VT-1 and VT-3 examinations are more appropriate than performing a VT-2 examination during a 10 CFR Part 50, Appendix J test. Following a repair/replacement activity affecting the containment pressure boundary when a 10 CFR Part 50, Appendix J test is performed to verify the leak-tight integrity of the containment pressure boundary, a VT-3 visual examination would be appropriate. As the 10 CFR Part 50, Appendix J test will confirm the pressure boundary integrity, a VT-3 examination on the area affected by the repair/replacement activity performed either during or after the pressure test would be appropriate to verify if any conditions exist which could affect the future leak tightness of the containment vessel. If the 10 CFR Part 50, Appendix J test is deferred as permitted by IWE-5222, a more detailed VT-1 examination of the area affected by the repair/replacement would be appropriate to identify any conditions which could affect the leak tightness of the containment vessel.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). Performance of VT-1 or VT-3 examinations will provide an acceptable level of quality and safety.

3.4.4 Alternative Examination:

Following a repair/replacement activity affecting containment pressure boundary when a 10 CFR Part 50, Appendix J test is performed to verify the leak-tight integrity of the affected containment pressure boundary, a VT-3 examination will be performed during or after the pressure test on the area affected by the repair/replacement activity.

Following a repair/replacement activity affecting containment pressure boundary when a 10 CFR Part 50, Appendix J test is deferred, a VT-1 examination will be performed on the area affected by the repair/replacement activity. When the deferred pressure test is performed, a VT-3 examination will be performed as noted above.

3.4.5 Justification for Granting Relief:

The Code required VT-2 examination requires access to the repaired area during performance of the pressure test. Access to the repaired area may not be available if the repaired area is on the interior surface of the containment vessel during a full scale 10 CFR Part 50, Appendix J Type A test. If the repaired area is subjected to a localized 10 CFR Part 50, Appendix J test, access is not available as the test fixture will cover the area affected by the repair/replacement activity. The VT-2 examination is performed to confirm the leak tightness of the area of repair/replacement.

Following a repair/replacement activity affecting the containment pressure boundary when a 10 CFR Part 50, Appendix J test is performed to verify the leak-tight integrity of the containment pressure boundary, a VT-3 visual examination is proposed. In this case, the 10 CFR Part 50, Appendix J test will confirm the pressure boundary integrity. The VT-3 examination on the area affected by the repair/replacement activity will verify if any conditions exist which could affect the future leak tightness of the containment vessel. As pressure is not a factor in performing the VT-3 examination, it can be performed either during or after the 10 CFR Part 50, Appendix J test. If the 10 CFR Part 50, Appendix J test is deferred as permitted by IWE-5222, a more detailed VT-1 examination of the area affected by the repair/replacement would be appropriate to identify any conditions which could affect the leak tightness of the containment vessel prior to its testing per the requirements of 10 CFR Part 50, Appendix J.

The VT-1 or VT-3 visual examinations in conjunction with the nondestructive examinations required by the repair/replacement plan will ensure that an acceptable level of quality and safety will be attained.

3.4.6 Staff Evaluation of RR-E4:

In lieu of performing the Code required VT-2 visual examination in connection with system pressure testing following repair, replacement, or modification, the licensee proposed an alternative as follows:

- If a 10 CFR Part 50, Appendix J test is performed to verify the leak-tight integrity of the affected containment pressure boundary, a VT-3 examination will be performed during or after the pressure test on the area affected by the repair/replacement activity area after the repair or replacement is completed.

- Following a repair/replacement activity affecting containment pressure boundary when a 10 CFR Part 50, Appendix J test is deferred, a VT-1 examination will be performed on the area affected by the repair/replacement activity.

The staff finds that Table IWE-2500-1, Examination Category E-P, requires only an examination method of 10 CFR Part 50, Appendix J for the containment vessel pressure retaining boundary following each repair, replacement, or modification and does not specifically identify a VT-2 visual examination. The staff also finds that 10 CFR Part 50, Appendix J provides requirements for testing including acceptable leakage criteria to ensure the leak-tight integrity of the pressure boundary. In addition, the VT-2 visual examination based on the IWA-5240 requires access to the repaired area during performance of the pressure test. Access to the repaired area may not be available if the repaired area is on the interior surface of the containment vessel if a 10 CFR Part 50, Appendix J, Type A is performed. If the repaired area is subjected to a localized 10 CFR Part 50, Appendix J pressure test, the repaired area would be covered by the test fixture and not available for visual examination. Furthermore, the licensee committed that if a 10 CFR Part 50, Appendix J test is performed to verify the leak-tight integrity of the affected containment pressure boundary, a VT-3 examination will be performed during or after the pressure test on the area affected by the repair/replacement activity area after the repair or replacement is completed. A VT-1 examination will be performed on the area affected by the repair/replacement activity following a repair/replacement activity affecting containment pressure boundary when a 10 CFR Part 50, Appendix J test is deferred.

On the basis discussed above, the staff finds that the alternative examination proposed by the licensee will provide an acceptable level of quality and safety for protecting the containment pressure boundary integrity. Therefore, the staff concludes that the licensee's alternative coating program is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year inservice inspection interval, which expires September 20, 2010.

3.5 Relief Request RR-E5:

3.5.1 Code Requirements:

Paragraphs IWE-2420(b) and IWE-2420(c) of the 1995 Edition, 1996 Addenda of ASME Section XI, require that when a component is acceptable for continued service or when the examinations result in a repair/replacement activity, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period listed in the schedule of the inspection program of Paragraph IWE-2411, "Inspection Program A," or Paragraph IWE-2412, "Inspection Program B," in accordance with Table IWE-2500-1, Examination Category E-C.

3.5.2 Requirements from Which Relief is Requested:

Relief is requested from the requirements of Paragraphs IWE-2420(b) and IWE-2420(c) to perform successive examination of repairs/replacements.

3.5.3 Basis for Relief:

The purpose of a repair/replacement is to restore the component to an acceptable condition for continued service in accordance with acceptance standards of Article IWE-3000. Paragraph IWE-4160, "Verification of Acceptability," requires the owner to conduct an evaluation of the suitability of the repair/replacement including consideration of the cause of failure.

If the repair/replacement has restored the component to an acceptable condition, successive examinations are not warranted. If the repair/replacement was not suitable, then the repair/replacement does not meet code requirements and the component is not acceptable for continued service. Neither Paragraph IWB-2420(b), Paragraph IWC-2420(b), nor Paragraph IWD-2420(b) requires a repair to be subject to successive examination requirements. Furthermore, if the repair area is subject to accelerated degradation, it would still require augmented examination in accordance with Table IWE-2500-1, Examination Category E-C. The successive examination of repairs in accordance with Paragraphs IWE-2420(b) and IWE-2420(c) constitutes a burden without a compensating increase in quality or safety.

In SECY 96-080, "Issuance of Final Amendment to 10 CFR Section 50.55a to Incorporate by Reference the ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, Division 1, Subsection IWE and Subsection IWL," dated April 17, 1996, the response to Comment No. 3.3, states "The purpose of IWE-2420(b) is to manage components found to be acceptable for continued service (meaning no repair or replacement at this time) as an Examination Category E-C component ... If the component had been repaired or replaced, then more frequent examination would not be needed."

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii). Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Subsection IWE of the 1998 Edition of ASME Section XI no longer requires successive examination of areas that have been repaired/replaced.

3.5.4 Alternative Examination:

Successive examination in accordance with Paragraphs IWE-2420(b) and IWE-2420(c) are not required for repairs made in accordance with Article IWA-4000.

3.5.5 Justification for Granting Relief:

Since IWA-4160 of the Code requires the suitability of the repair/replacement including its cause, successive examination of the repair/replacement does not provide an additional safety benefit. This is consistent with the requirements of ASME Class 1, 2, and 3 systems in which successive examinations are only required when an item is accepted by evaluation.

3.5.6 Staff Evaluation of RR-E5:

In lieu of meeting ASME Section XI, Subarticles IWE-2420(b) and (c) that require successive examinations of repaired areas in accordance with Table IWE-2500-1, the licensee proposes to use the process and acceptance examinations and evaluations required by the Code for repairs.

The staff finds that when repairs are complete, IWA-4160 requires licensees to evaluate the suitability of the repair. When a repair is required because of failure of an item, the evaluation shall consider the cause of failure to ensure that the repair is suitable. Considering that the failure mechanism is identified and corrected as required and the repair receives preservice examinations, as required, the proposed alternative will provide reasonable assurance of structural integrity. In doing this, the hardship associated with the requirements of successive examinations can be eliminated. Furthermore, Subparagraphs IWB-2420(b), IWC-2420(b), and IWD-2420(b) do not require the successive inspection of repairs for ASME Code Class 1, 2, and 3 components as required in Subparagraph IWE-2420(b) for ASME Code Class MC components. On the basis that compliance with the specific code requirements would result in hardship without a compensating increase in the level of quality and safety, the alternative proposed by the licensee is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year inservice inspection interval, which expires September 20, 2010.

3.6 Relief Request RR-E6:

3.6.1 Code Requirements:

ASME Section XI, 1992 Edition with 1992 Addenda, Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item E8.20 requires that Class MC bolted connections be subject to a bolt torque or tension test.

3.6.2 Requirements from Which Relief is Requested:

Relief is requested from meeting the requirements of ASME Section XI 1992 Edition, 1992 Addenda, Table IWE-2500-1, Examination Category E-G, Pressure Retaining Bolting, Item E8.20. Table IWE-2500-1 requires bolt torque or tension test on bolted connections that have not been disassembled and reassembled during the inspection interval. ASME Code Case N-604 will be used in lieu of the requirements of Table IWE-2500-1, Examination Category E-G, Item E8.20.

3.6.3 Basis for Relief:

ASME Code Case N-604 provides requirements, which may be used in lieu of the requirements of Table IWE-2500-1, Examination Category E-G, Item E8.20. Note 5 of Table IWE-2500-1, Examination Category E-G requires bolt torque or tension testing on bolted connections that have not been disassembled and reassembled during the inspection interval. Determination of the torque or tension value would require that the bolting be re-torqued and then re-torqued or re-tensioned.

ASME Code Case N-604 states that the following examinations may be performed to satisfy the inservice inspection requirements for pressure retaining bolting.

1. Exposed surfaces of bolted connections shall be visually examined in accordance with requirements of Table IWE-2500-1, Examination Category E-G, Item E8.10.

2. Bolting connections shall meet the pressure test requirements of Table IWE-2500-1, Examination Category E-P, Item E9.40.

Each containment penetration receives a 10 CFR Part 50, Appendix J, Type B test in accordance with the specified testing frequencies. As noted in 10 CFR Part 50, Appendix J, the purpose of Type B tests is to measure leakage of containment penetrations whose design incorporates resilient seals, gaskets, sealant compounds, and electrical penetrations fitted with flexible metal seal assemblies. The performance of the Type B test itself proves that the bolt torque or tension remains adequate to provide a leak rate that is within acceptable limits. The torque or tension value of bolting only becomes an issue if the leak rate is excessive. Once a bolt is torqued or tensioned, it is not subject to dynamic loading that would cause it to experience significant change. Only bolting which would be subject to pressure loading which would tend to induce tension in bolts under accident conditions changes (i.e., pressure-unseating containment penetrations) would be expected to affect the preload of bolting. Davis-Besse has no pressure-unseating bolting as blank flanges are installed on both the interior and exterior flanges on penetrations not in use during normal operation. Penetrations which are pressurized, such as electrical penetrations, are not considered to be pressure-unseating penetrations. Appendix J testing and visual inspection is adequate to demonstrate that the design function is met. Torque or tension testing is not required on any other ASME Section XI, Class 1, 2, or 3 bolted connections or their supports as part of the inservice inspection program.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii). Un-torquing and subsequent re-torquing of bolted connections which are verified not to experience unacceptable leakage through 10 CFR Part 50, Appendix J, Type B testing results in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The requirement to perform bolt torque or tension tests is not required in Subsection IWE of the 1998 Edition of ASME Section XI.

3.6.4 Alternative Examination:

The requirements of ASME Code Case N-604 will be implemented.

3.6.5 Justification for Granting Relief:

The only bolting in which the torque value of bolting would be affected is that bolting which is subjected to tension. This bolting would be that installed in penetrations which are pressure unseating. Davis-Besse has no pressure unseating bolting. The torque or tension testing of bolts required by Examination Category E-G, when the bolts are not disassembled, will result in a hardship without a commensurate increase in the level of quality or safety.

3.6.6 Staff Evaluation of RR-E6:

In lieu of meeting the requirements of Table IWE-2500-1, Examination Category E-G, Item E8.20, the licensee proposed to use the ASME Code Case N-604 requirements to ensure the tightness of the Class MC bolted connections. Code Case N-604 states that exposed surfaces of bolted connections shall be visually examined in accordance with requirements of Table

IWE-2500-1, Examination Category E-G, Item E8.10 (VT-1 visual examination). Bolting connections shall meet the pressure test requirements of Table IWE-2500-1, Examination Category E-P, Item E9.40 (10 CFR Part 50, Appendix J, Type B test).

The staff finds that bolt torque or tension testing on bolted connections that have not been disassembled and reassembled during the inspection interval would require the bolting be un-torqued and then re-torqued or re-tensioned, whereas the leak testing as required by 10 CFR Part 50, Appendix J would adequately verify the leak-tight integrity of the containment. The staff also finds that torque or tension testing is not required on any other ASME Section XI, Class 1, 2, or 3 bolted connections or their supports as part of the inservice inspection program. In addition, compliance with ASME Code requirements will cause a hardship or unusual difficulty because un-torquing and subsequent re-torquing bolted connections involve unnecessary radiation exposure and costs to perform the work without a compensating increase in the level of quality and safety. Furthermore, the staff finds that the alternative approach proposed by the licensee (the test required by 10 CFR Part 50, Appendix J together with VT-1 visual examination to verify the leak-tight integrity of bolted connections for containment vessel leak-tight integrity) will provide reasonable assurance of the containment pressure boundary integrity. On this basis, the staff concludes that the alternative proposed by the licensee is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year inservice inspection interval, which expires September 20, 2010.

### 3.7 Relief Request RR-E7:

#### 3.7.1 Code Requirements:

Paragraph IWA-2210, Table IWA-2210-1 provides requirements for distance and illumination requirements for performing VT-3 visual examinations.

#### 3.7.2 Requirements from Which Relief is Requested:

Relief is requested from the provisions of Table IWA-2210-1, “Visual Examinations,” when performing VT-3 examinations required by IWE. Table IWA-2210-1 requires direct visual VT-3 examinations be performed with a minimum illumination of 50 foot-candles, and a maximum direct examination distance of 4 feet. The procedure must be demonstrated to resolve a lower case character height of 0.105 inches.

#### 3.7.3 Basis for Relief:

IWA-2210 requires visual examinations be performed in accordance with Article 9 of ASME Section V. Direct visual examination is defined in Article 9 of the 1995 Edition, 1996 Addenda of ASME Section V as a visual examination technique performed by eye and without any visual aids (excluding light source, mirrors, and/or corrective lenses). Table IWA-2210-1 requires the VT-3 examination be performed with a minimum illumination of 50 fc and a maximum direct examination distance of 4 feet.

IWA-2216 states that when remote visual examination is substituted for direct visual examination, the remote visual examination system shall have the capability of distinguishing and differentiating between colors in addition to the requirements of ASME Section V, Article 9. Remote visual examination is defined in Article 9 of the 1995 Edition, 1996 Addenda of ASME Section V as a visual examination technique used with visual aids for conditions where the area is inaccessible for direct visual examination. Remote visual examination may use visual aids such as mirrors, telescopes, borescopes, fiber optics, cameras, or other suitable instruments. Article 9 requires remote visual examination systems have a resolution capability at least equivalent to that obtainable by direct visual observation.

Considering the size of the containment structures (as compared to Class 1, 2, and 3 components), and recognizing the varied lighting conditions, the NRC provided latitude from the requirement of IWA-2216 for VT-3 remote visual examination in 10 CFR 50.55a(b)(2)(ix)(B). 10 CFR 50.55a(b)(2)(ix)(B) states that when performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which visual examination is

performed can be detected at the chosen distance and illumination. 10 CFR 50.55a(b)(2)(ix)(B) applies to remote visual examinations, but does not apply to direct visual examinations.

The Davis-Besse containment vessel is a free standing, large volume steel vessel. Access to the surfaces of the containment vessel is provided at the Elevations 565 feet, 585 feet, 603 feet, and 653 feet. Approximately 15 percent of the containment surface is within the maximum 4-foot examination distance necessary to perform a direct visual VT-3 examination. The remainder must be examined either from scaffold or by remote visual examination.

Installation of scaffold is not considered practical. Installation of scaffold would require nearly 1,600 linear feet of scaffold ranging in height of 10 feet to 40 feet. Many areas of containment do not contain sufficient room to erect scaffold within 4 feet of the containment vessel. In other areas, scaffold would restrict normal access and egress routes for personnel working in containment.

Remote visual examination may be used in lieu of building scaffold. When using remote visual examination, the maximum direct examination distance and the minimum illumination requirements of Table IWA-2210-1 may be extended or decreased respectively in accordance with 10 CFR 50.55a(b)(2)(ix)(B) provided conditions for which the visual examination is being performed can be identified. However, this relaxation in Table IWA-2210-1 applies only to remote visual examinations and does not apply to direct visual examinations. The direct visual examination distance and illumination requirements of Table IWA-2210-1 are impractical when performing containment examinations. Conditions for which the containment visual examinations are being performed can be seen at distances much greater than the maximum direct visual VT-3 examination distance specified in Table IWA-2210-1.

Remote visual examinations are qualified at a specific distance and illumination in accordance with 10 CFR 50.55a(b)(2)(ix)(B) using a chipped paint specimen or a 18 percent neutral gray card. This same chipped paint specimen or 18 percent neutral gray card will be used to qualify the maximum examination distance and minimum illumination for performing direct visual VT-3 examination. This qualification process will ensure that the direct visual and remote visual examination processes are equivalent.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i). Performance of direct visual VT-3 examinations qualified to the same standards as remote visual VT-3 examinations will provide an acceptable level of quality and safety.

Subsection IWE of the 1998 Edition of ASME Section XI no longer requires a VT-3 examination of the containment surfaces. Therefore, the requirements of Table IWA-2210-1 are no longer applicable to IWE containment examinations.

3.7.4 Alternative Examination:

Direct visual VT-3 examinations will be qualified at distances exceeding the requirements of Table IWA-2210-1 and illumination less than Table IWA-2210-1 requirements. The direct visual VT-3 examinations will be qualified on the same specimen as used to qualify the remote visual examinations.

3.7.5 Justification for Granting Relief:

Considering the size of the containment structures and recognizing the varied lighting conditions in containments, the NRC provided latitude in 10 CFR 50.55a(b)(2)(ix)(B) from the requirement of IWA-2216 for VT-3 remote visual examination. The IWA-2216 requirements are contained in Table IWA-2210-1. However, this relaxation in Table IWA-2210-1 applies only to remote visual examinations and does not apply to direct visual examinations. Conditions for which the containment visual examinations are being performed can be seen at distances much greater than the maximum direct visual VT-3 examination distance specified in Table IWA-2210-1. Qualification of direct visual VT-3 examinations to the same specimens used to qualify remote examinations will ensure that the examinations throughout the containment are consistent and will identify any conditions which may be detrimental to the leak-tight integrity of the containment vessel. Performance of direct visual VT-3 examinations qualified to the same standards as remote visual VT-3 examinations will provide an acceptable level of quality and safety.

3.7.6 Staff Evaluation of RR-E7:

The licensee described, in the “Basis for Relief” and “Justification for Granting Relief” sections, that the Davis-Besse containment vessel is a free standing, large volume steel containment. Only 15 percent of the containment surface is within the maximum 4-foot examination distance necessary to perform a direct visual VT-3 examination. It would be necessary to install and use extensive temporary scaffold systems to access the remaining portions of the containment. Even though these scaffolds can only provide limited access due to containment geometry restrictions as well as structural and equipment interferences. Because the accessibility to the major portions of the containment vessel will make it a hardship to obtain the maximum direct examination distance and minimum illumination requirements, the licensee proposed an alternative to the requirements for the measurement of illumination and direct examination distance for visual examinations specified in ASME Section XI, 1992 Edition, 1992 Addendum, Table IWA-2210-1. The licensee quoted the 10 CFR 50.55a(b)(2)(ix)(B) requirement that the code required maximum direct examination distance may be increased and the minimum illumination may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

The staff finds that visual examinations on the containment are performed to determine if damage or degradation warrant additional evaluation or repair of the structure. In order for the visual examinations to be performed in such a way as to detect damage or degradation, proper lighting is essential. Also, the installation and removal of these scaffolds would increase both worker radiation exposure and challenge personnel safety in order to meet Paragraph Table IWA-2210-1 requirements. IWA-2210 allows for remote examination as long as the remote examination procedure is demonstrated to resolve the selected test chart characters. When the proposed alternative examination is performed, the licensee also committed, in Reference 4, that the criteria to be used to qualify the direct visual VT-3 examination procedure will be established by a Responsible Professional Engineer or other responsible individual, knowledgeable in the requirements for design, inservice inspection, and testing of Class MC components.

On the basis discussed above, the staff concludes that the examination requirements proposed by the licensee will provide reasonable assurance of the functionality and integrity of the concrete containment. Therefore, on the basis that compliance with the specific requirements of the Code would result in hardship without a compensating increase in the level of quality and safety, the licensee’s proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year inservice inspection interval, which expires September 20, 2010.

### 3.8 Relief Request RR-E8:

#### 3.8.1 Code Requirements:

ASME Section XI, 1995 Edition, 1996 Addenda, Subarticle IWE-2500(c)(3) requires one foot square grids be used when ultrasonic thickness measurements are performed on augmented examination surface areas. The number and location of the grids are determined by the owner. Subarticle IWE-2500(c)(4) requires the minimum wall thickness within each grid be determined.

#### 3.8.2 Requirements from Which Relief is Requested:

Relief is requested from using one foot square grids for augmented examination areas and the requirement to determine the minimum wall thickness within each grid. Code Case N-605 will be used as an alternative to the requirements of IWE-2500(c).

#### 3.8.3 Basis for Relief:

Subarticles IWE-2500(c)(3) and IWE-2500(c)(4) of the 1995 Edition, 1996 Addenda of ASME Section XI, require that the minimum thickness within each one foot square grid of surface areas requiring augmented examination be marked such that periodic reexamination of that location can be performed. Thickness readings are point readings. Numerous readings are necessary to identify the minimum thickness within each grid. This only identifies the thinnest area. Periodic examination of the minimum thickness point only monitors that point. It may not be the area that is the most susceptible to accelerated degradation.

Code Case N-605 provides an alternative to the one foot square grid area required by IWE-2500(c)(3). Code Case N-605 requires examination at the grid line intersections. The grid intersections may not exceed 12 inches and may be as small as 2 inches.

For a sample area of 50 square feet, Code Case N-605 requires a minimum 100 locations be monitored. For a sample area of 50 square feet, IWE-2500(c)(3) would require only 50 locations be monitored. In this instance, utilizing Code Case N-605 monitors more locations than required by IWE-2500(c)(3).

For sample areas greater than 100 square feet, Code Case N-605 requires that sufficient points be monitored to ensure at least a 95 percent confidence level that the thickness of the base metal is reduced by more than 10 percent of the normal plate thickness at 95 percent of the grid line intersections. Code Case N-605 also requires additional examinations when any measurements reveal wall thickness reduced by more than 10 percent of the nominal plate thickness.

For all examination areas, should the measurements at a grid line intersection reveal that the base material is reduced by more than 10 percent of the nominal plate thickness, Code Case N-605 requires the minimum wall thickness within each adjoining grid be determined. This is similar to the examination requirements of IWE-2500(c)(4) except that Code Case N-605 focuses resources on areas that have exhibited degradation rather than areas that have not exhibited degradation.

The Flow Accelerated Corrosion programs presently in place have proven that taking thickness readings taken at grid intersections are effective in monitoring wall thinning of piping.

Relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii). Taking numerous ultrasonic thickness readings within a grid that had not exhibited degradation results in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.8.4 Alternative Examination:

Code Case N-605 will be used to perform augmented examination of containment surface areas.

3.8.5 Justification for Granting Relief:

Code Case N-605 requires examinations be conducted at grid line intersections and only within grid sections when the grid intersection examinations reveal evidence that the base material wall thickness is being reduced. Code requirements would require numerous readings be taken within a grid that had not exhibited degradation. Compliance with the Code requirements in which examinations will be taken in areas not exhibiting degradation would result in undue hardship and unusual difficulty without a compensating increase in the level of quality and safety.

3.8.6 Staff Evaluation of RR-E8:

In lieu of meeting ASME Section XI, 1992 Edition through 1992 Addenda, Subarticles IWE-2500(c)(3) and (4) that require one-foot square grids be used when ultrasonic thickness measurements are performed on augmented examination surface areas, and the minimum wall thickness within each grid be determined, the licensee proposed to use Code Case N-605 to determine examination requirements for ultrasonic thickness measurements on areas requiring augmented examination.

Under the application of Code Case N-605 rules (as described in the request), Table IWE-2500-2 requires a minimum 100 locations be monitored for a sample area of 50 square feet. According to the licensee, utilizing Table IWE-2500-2 monitors more locations than that determined by the owner (required by the IWE-2500(c)(3) rule). For sample areas greater than 100 square feet, Table IWE-2500-2 requires: (a) sufficient locations be monitored to ensure at least a 95 percent confidence level that the thickness of the base material is reduced no more than 10 percent of the nominal plate thickness at 95 percent of the grid line intersections, and (b) additional examinations be taken when any measurement reveals that the wall thickness is reduced by more than 10 percent of the nominal plate thickness. For all examination areas, Table IWE-2500-2 requires that the minimum wall thickness within each adjoining grid be determined, if the measurements at a grid line intersection reveal that the base material is reduced by more than 10 percent of the nominal plate thickness.

On the basis discussed above, the staff finds that the alternative proposed by the licensee will provide reasonable assurance of the containment (plate) integrity. Therefore, on the basis that the alternative provides an acceptable level of quality and safety, the request for relief is granted pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year inservice inspection interval, which expires September 20, 2010.

## 4.0 CONCLUSION

Based on our review of the information provided in the requests for relief (Relief Requests RR-E1 through RR-E8), the staff concludes that for Relief Requests RR-E2, RR-E4, and RR-E8, the licensee's proposed alternatives will provide an acceptable level of quality and safety. On this basis, the proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for Davis-Besse's first 10-year containment ISI interval. For Relief Requests RR-E1, RR-E3, RR-E5, RR-E6, and RR-E7, the staff concludes that compliance with the code requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that licensee's proposed alternatives will provide reasonable assurance of containment pressure integrity. Therefore, these proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for Davis-Besse's first 10-year ISI interval which expires September 20, 2010.

RR-E6 implements Code Case N-604 and RR-E8 implements Code Case N-605. Therefore, the licensee's proposed alternatives to use Code Cases N-604 and N-605 are authorized for the third 10-year interval or until such time Code Cases N-604 and N-605 are published in a future revision of Regulatory Guide (RG) 1.147. At that time, if the licensee intends to continue to implement either Code Case N-604 or N-605, it must follow all provisions in the subject code case with the limitations or conditions specified in RG 1.147, if any.

5.0 REFERENCES

1. Letter from Guy G. Campbell, FENOC to NRC, "Third 10-Year Interval Inservice Inspection Program for Davis-Besse Nuclear Power Station, Unit 1," dated September 19, 2000.

2. Letter from J. K. Wood, FENOC to NRC, "Response to NRC Generic Letter 98-04: Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant-Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," dated November 11, 1998.

3. Letter from NRC to G. G. Campbell, FENOC, "Completion of Licensing Action for Generic Letter 98-04: Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant-Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment - Davis-Besse Nuclear Power Station, Unit 1," dated December 2, 1999.

4. Letter from Guy G. Campbell, FENOC to NRC, "Revision to Request for Relief from an American Society of Mechanical Engineers Boiler and Pressure Vessel Code Inservice Inspection Requirement at Davis-Besse Nuclear Power Station," dated September 7, 2001.

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