June 4, 2001

Mr. Guy G. Campbell, Vice President - Nuclear
FirstEnergy Nuclear Operating Company
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1 - RELIEF REQUEST NOS. RR-A18, RR-A19, AND RR-A20 FOR THE SECOND 10-YEAR INSERVICE INSPECTION INTERVAL (TAC NO. MA7210)

Dear Mr. Campbell:

By letter dated November 13, 1999, supplemented March 24, 2001, FirstEnergy Nuclear Operating Company (FENOC), the licensee, submitted three requests for relief from specific ASME (American Society of Mechanical Engineers) Code, Section XI requirements for inservice inspection (ISI) for the Davis-Besse Nuclear Power Station (DBNPS), Unit 1. The Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the information provided in relief request numbers RR-A18, RR-A19, and RR-A20.

Relief request RR-A18 was reviewed against the requirements of the 1986 Edition of the ASME Code, Section XI and 10 CFR 50.55a(a)(3)(ii). The staff finds the licensee's request for relief in RR-A18 acceptable for the second 10-year ISI interval at DBNPS.

Relief request RR-A19, Parts A, B, C, and D were evaluated. The staff finds the examination coverage requirements for Parts A, B, C, and D to be impractical. Therefore, relief is granted for the coverage requirements obtained in accordance with 10 CFR 50.55a(g)(6)(i) for the second 10-year ISI interval. The staff also reviewed the licensee's use of the Performance Demonstration Initiative (PDI) protocol for the reactor pressure vessel welds. The staff finds the licensee's use of the PDI protocol for the reactor vessel welds acceptable, and is hereby authorized, pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year ISI interval at DBNPS.

The staff also completed its review of relief request RR-A20. The staff finds the alternative examination proposed is acceptable and the use thereof is hereby authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the second 10-year ISI interval at DBNPS.

The staff has reviewed the information and associated proposed alternatives provided by FENOC regarding these relief requests. Enclosed is the NRC staff's safety evaluation report (SER) on its review of the DBNPS relief request submittal.

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

Enclosure: Safety Evaluation

cc w/encl: See next page

The staff has reviewed the information and associated proposed alternatives provided by FENOC regarding these relief requests. Enclosed is the NRC staff's safety evaluation report (SER) on its review of the DBNPS relief request submittal.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE SECOND INSERVICE INSPECTION INTERVAL

RELIEF REQUEST NOS. RR-A18, RR-A19, AND RR-A20

DAVIS-BESSE NUCLEAR POWER STATION

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NUMBER 50-346

1.0 INTRODUCTION

By letter dated November 13, 1999, as supplemented by letter dated March 24, 2001, First Energy Nuclear Operating Company (FENOC) submitted three requests for relief from certain examination requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI. The information provided by the licensee in support of the request for relief from Code requirements has been evaluated pursuant to the provisions of 10 CFR 50.55a(a)(3) and 10 CFR 50.55a(g)(6)(i). The basis for disposition is documented below.

2.0 BACKGROUND

Inservice inspection (ISI) of the ASME Code Class 1, 2, and 3 components shall 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 written 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 (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. For Davis-Besse Nuclear Power Station (Davis-Besse) the applicable edition of Section XI of the ASME Code for the second 10-year ISI interval is the 1986 Edition.

3.0 RELIEF REQUESTS

3.1 Relief Request Number RR-A18

The components for which relief is requested:

Reactor Coolant System small diameter (≤1 inch) vent, drain, and instrument piping ASME Section XI, Class 1 Piping

Code Requirement (as stated):

Subsection IWB-2500, Table-2500-1, Examination Category B-P, Item Nos. B15.51 and B15.71 of the 1986 Edition, No Addenda, of ASME Section XI requires a system hydrostatic test at or near the end of each inspection interval. The pressure-retaining boundary during the hydrostatic test shall include all Class 1 components within the system boundary.

Relief is requested from performing the hydrostatic test of the Class 1 piping and valves downstream of the first isolation valve of small diameter vent, drain, and instrument piping.

Licensee's Basis for 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 (RCS) by another short pipe nipple. Manually operated isolation valves provide double isolation of the RCS and are closed during normal operating conditions.

The system hydrostatic test is performed in Mode 3 with the RCS at full temperature and pressure. Performance of the system hydrostatic test 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 RCS. FENOC proposes to perform the system hydrostatic test of the RCS 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-4400 provides the requirements for the hydrostatic pressure testing of piping and components after repairs by welding to

the pressure boundary. IWA-4400(b)(5) exempts component connections, piping, and associated valves that are nominal pipe size (NPS) 1 inch and smaller from system hydrostatic tests following repairs by welding. The requirements of IWA-4400 also apply to replacements.

2. The non-isolable portion of the RCS drain and vent connections will be pressurized and visually examined as required. Only the isolated portion of the small diameter drain, vent, and instrument 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 Operating License Technical Specifications (TS 3.4.6.1 and 3.4.6.2) require RCS leakage monitoring during normal plant operation (Modes 1, 2, 3, and 4). Should any technical specification limits be exceeded, corrective actions, include plant shutdown, are required to identify the source of leakage and restore the RCS boundary integrity.

Personnel safety and as low as reasonably achievable (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 RCS nominal operating pressure and temperature. Opening this inboard isolation valve under RCS full temperature and 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.

2. Performing the system hydrostatic 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 hydrostatic test is performed near the end of the outage at full temperature and pressure following completion of all RCS work. The system hydrostatic test is a critical path activity. To minimize the time the RCS 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 hydrostatic test and the duration of the outage.

Licensee's Proposed Alternative:

The system hydrostatic test will be performed with all small diameter vent, drain, and instrument valves in their normal closed position. The VT-2 examination will extend to and include the outboard closed valve in the RCS boundary.

Relief is requested from performing the hydrostatic test of the Class 1 piping and valves downstream of the first isolation valve of small diameter vent, drain and instrument piping.

FENOC considers the requirement to pressurize the downstream portions of small diameter vent, drain, and instrument piping a hardship 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).

Staff Evaluation

The Code requires that all Class 1 components within the RCS boundary undergo a system hydrostatic test once per interval. The licensee has proposed an alternative to the hydrostatic test requirements for the subject line segments. The line segments, as stated by the licensee, include two manually operated valves separated by a short pipe nipple that is connected to the RCS via another short pipe nipple. The line configuration, as outlined, provides double isolation of the RCS. Under normal plant operating conditions, the subject line segments would see RCS temperatures and pressures only if leakage through the inboard valve occurs. For the licensee to perform the Code required test, it would be necessary to manually open the inboard valves to pressurize the line segments. Pressurization by this method would defeat the RCS double isolation and may cause safety concerns for the personnel performing the examination. Typical line/valve configurations are in close proximity to the primary and secondary RCS piping. Manual actuation (opening and closing) of these valves is estimated to expose plant personnel to 0.4 man-rem per test. Therefore, the Code requirement to perform the system hydrostatic test on these isolated line segments presents a hardship for the licensee. In addition, manual actuation (opening and closing) of these valves exposes plant personnel performing the test.

The licensee will visually examine the isolation valves in the normally closed position for leaks and evidence of past leakage during the system leakage test each refueling outage. Also, the RCS vent and drain connections will be visually examined with the isolation valves in the normally closed position during the 10-year ISI pressure test. The licensee's proposed alternative will provide reasonable assurance that structural integrity is maintained for the subject line segments. Imposition of the Code requirement on the Davis-Besse Nuclear Power Station (DBNPS) would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second ISI interval at Davis-Besse.

3.2 Relief Request Number RR-A19

The components for which relief is requested:

Part A: Examination Category B-A, Item Number B1.11 - reactor vessel lower shell to bottom head circumferential weld (Weld Number RC-RPV-WR-34)

Part B: Examination Category B-A, Item Number B1.21 - reactor vessel bottom head circumferential weld (Weld Number RC-RPV-35)

Part C: Examination Category B-D, Item Number B3.90 - outlet nozzle to reactor vessel shell welds (Weld Numbers RC-RPV-WR-13/14/72-X and RC-RPV-WR-13/14/72-Z)

Part D: Examination Category B-D, Item Number B3.100 - core flood nozzle inner radius sections (Weld Numbers RC-RPV-WR54/55-W-IR and RC-RPV-WR-54/55-Y-IR)

Code Requirement (as stated):

10 CFR 50.55a(g)(6)(ii)(A)(2) requires licensees to implement the examination requirements of the 1989 Edition of ASME Section XI for reactor vessel shell welds. Subsection IWB, Table IWB-2500-1, Examination Category B-A, Item B1.11 requires essentially 100 percent of the shell weld length be examined. As defined in 10 CFR 50.55a(g)(6)(ii)(A)(2), essentially 100 percent means more than 90 percent of the examination volume of each weld where the reduction in coverage is due to interference by another component, or part geometry.

The 1986 Edition of ASME Section XI, Subsection IWB -2500-1, Examination Category B-A, Item B1.21 for reactor vessel circumferential head welds, requires essentially 100 percent of the weld length as defined by Figure IWB-2500-3 be examined. Code Case N-460, Alternative Examination Coverage for Class 1 and Class 2 Welds, states that a reduction in examination coverage on any Class 1 or Class 2 weld may be accepted provided the reduction in coverage for that weld is less than 10 percent.

The 1986 Edition of ASME Section XI, Subsection IWB, Table IWB-2500-1, Examination Category B-D, Items B3.90 and B3.100 for reactor vessel nozzle to vessel welds and the nozzle inside radius section, require essentially 100 percent of the weld length as defined by Figure IWB-2500-7 be examined. Code Case N-460, Alternative Examination coverage for Class 1 and Class 2 Welds, states that a reduction in examination coverage on any Class 1 or Class 2 weld may be accepted provided the reduction in coverage for that weld is less than 10 percent.

Licensee's Basis for Relief (as stated):

The reactor vessel welds are examined from the inside surface using the Framatone [sic] URSULA inspection manipulator in conjunction with the Framatome ACCUSONEX data acquisition and analysis system. URSULA is a computer controlled, remotely operated manipulator which uses a contact ultrasonic test (UT) head to obtain ultrasonic data for the detection and sizing of indications. The contact head is fitted with an array of transducers in direct contact with the reactor vessel surface. With the ACCUSONEX data acquisition system multiple channels of ultrasonic data are amplified, filtered, digitized and processed, and integrated with the transducer position to provide computer generated images of the examination volume. The URSULA and ACCUSONEX system has been successfully demonstrated and qualified in accordance with Supplements 4 and 6 of the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix VIII, using the Performance Demonstration Initiative (PDI) protocol.

Part A: Reactor Vessel Lower Shell to Bottom Head Circumferential Weld (RC-RPV-WR-34)

Licensee's Basis for Relief :

ASME Section XI, 1986 Edition, Subsection IWB, Figure IWB-2500-1 defines the required examination volume. It is not possible to obtain ultrasonic interrogation of greater than 90 percent of this volume due to interference caused by the core support lugs. The core support lugs are welded to the reactor vessel shell just above the lower shell to bottom head weld and extend approximately 2 inches below the centerline of the weld. These lugs restrict the URSULA manipulator's ability to move to areas necessary to fully examine the required volume. Access to approximately 40 percent of the examination volume is restricted. The remaining 60 percent of the examination volume will be examined by techniques which have been demonstrated and qualified in accordance with Supplements 4 and 6 of the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix VIII, using the PDI protocol. These examinations will be performed from both sides of the weld scanning both parallel and perpendicular to the weld. In addition to the required ultrasonic examination, the welds attaching the core support lugs will receive a VT-3 examination in accordance with Table IWB-2500-1, Examination Category B-N-2, Code Item B13.30, Interior Attachment Welds Beyond Beltline Region. This VT-3 examination will identify the structural condition of the core support lug welds which would indicate if the area of the lower shell to bottom head weld had been subjected to any excessive loads. The core support lugs were visually examined during the ten year reactor vessel examination in 1990. No deficiencies were noted during the examination. The subject weld RC-RPV-WR-34 was examined during the April 2000 outage. The actual examination coverage for this weld was 62 percent.

The licensee states that due to the configuration of the reactor vessel, it is impractical to meet the examination coverage requirements of the 1986 Edition of ASME Section XI. The licensee requested relief in accordance with 10 CFR 50.55a(g)(5)(iii).

Licensee's Proposed Alternative:

The licensee examined the accessible areas with techniques that have been demonstrated and qualified in accordance with Supplements 4 and 6 of the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix VIII, using the PDI protocol. The examinations were performed from both sides of the weld scanning both parallel and perpendicular to the weld. The aggregate examination coverage of the weld and base metal areas was 62 percent of the required examination volume.

Staff Evaluation

The 1986 Edition of ASME Code Section XI Table IWB-2500-1, examination category B-A, item number B1.11 requires examination of all welds in the 1st inspection interval and one beltline region weld in successive inspection intervals. However, 10 CFR 50.55a(g)(6)(ii)(A)(2) requires all licensees to augment their reactor pressure vessel (RPV) examinations by implementing once, as part of the ISI interval in effect on September 8, 1992, the examination requirements for RPV shell welds specified in item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of Subsection IWB of the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(A) (3) and (4). The licensee is requesting relief from the Section XI requirement to examine essentially 100 percent (defined in 50.55a(g)(6)(ii)(A)(2) as more than 90 percent) of the volume of weld RC-RPV-WR-34 reactor vessel lower shell to bottom head circumferential weld. It is not possible for the licensee to obtain complete ultrasonic coverage of this weld due to interference caused by the core support lugs. Gaining additional access for examination of the subject weld would require design modifications. Imposition of this requirement would impose a significant burden on the licensee.

The licensee examined 62 percent of the required examination volume for this weld. Furthermore, the volumetric examination is supplemented with a visual (VT-2) examination of the welds during each refueling outage. In addition, the licensee has met the coverage requirements for the remaining RPV shell welds. Based on the volumetric examination coverage obtained on weld RC-RPV-WR-34, examinations conducted on other RPV welds, and the visual examinations, the staff finds that any significant patterns of degradation, if present, would have been detected and that the examinations performed provide reasonable assurance of structural integrity. Therefore, based on the impracticality of meeting the Code requirements, and the reasonable assurance of structural integrity provided by the UT examinations, relief is granted for the UT examination coverage obtained of weld RC-RPV-WR-34 in accordance with 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(g)(6)(i) for the second ISI interval.

In addition, the licensee stated they inspected the accessible areas using techniques that have been demonstrated and qualified in accordance with Supplements 4 and 6 of the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix VIII, using the PDI protocol. The PDI program based on the criteria of Appendix VIII, Section XI of the ASME Code, Supplements 4 and 6 of the 1995 Edition, 1996 Addenda, requires that ultrasonic equipment, procedures, and examiners perform a performance demonstration to verify the effectiveness of the examination. A comparison of the licensee's PDI protocol technique to the current Code and Regulatory Guide (RG) 1.150, indicates that the PDI protocol technique provides an equivalent or better examination of the reactor vessel. The staff finds that the use of the PDI protocol program for the reactor vessel examination provides an equivalent or better examination than that of the licensee's current Code of record and RG 1.150.

Part B: Reactor Vessel Bottom Head Circumferential Weld
(RC-RPV-WR-35)

Licensee's Basis For Relief:

ASME Section XI, 1986 Edition, Subsection IWB, Figure IWB-2500-3 defines the required examination volume. It is not possible to obtain ultrasonic interrogation of greater than 90 percent of this volume due to interference caused by the incore instrument nozzles. The incore instrument nozzles protrude through the bottom head of the reactor vessel to a height of approximately 1 foot from the inside surface of the bottom head. Access to approximately 28 percent of the examination volume is restricted. The remaining 72 percent of the examination volume will be examined with techniques which have been demonstrated and qualified in accordance with Supplements 4 and 6 of the 1995 Edition, 1996 Addenda of ASME Section XI, Appendix VIII, using the PDI protocol. These examinations will be performed from both sides of the weld scanning both parallel and perpendicular to the weld. Due to the configuration of the reactor vessel, it is impractical to meet the examination requirements of the 1986 Edition of ASME Section XI. Relief is requested in accordance with 10 CFR 50.55a(g)(5)(iii).

Licensee's Proposed Alternative Examination:

The accessible area was examined with techniques that have been demonstrated and qualified in accordance with Supplements 4 and 6 of ASME Section XI, Appendix VIII, using the PDI protocol. The examinations were performed from both sides of the weld scanning both parallel and perpendicular to the weld. The aggregate examination obtained during the second interval RPV examinations of the subject weld and base metal area was 72 percent of the required examination volume.

Staff Evaluation

Examination Category B-A, Item B1.21 requires 100 percent volumetric examination of all circumferential head welds each inspection interval. Code Case N-460, "Alternative Examination Coverage for Class 1 and Class 2 Welds," states that a reduction in examination coverage on any Class 1 or Class 2 weld may be accepted provided the reduction in coverage for that weld is less than 10 percent. The licensee is requesting relief from the Section XI requirement to examine essentially 100 percent of weld RC-RPV-WR-35 reactor vessel bottom head circumferential weld. Gaining additional access for examination of the subject weld would require design modifications. Imposition of this requirement would impose a significant burden on the licensee.

The licensee is able to examine 72 percent of the required examination volume for this weld. The accessible area was examined with techniques that have been demonstrated and qualified in accordance with Supplements 4 and 6 of ASME Section XI, Appendix VIII, using the PDI protocol. The staff finds that the extent of examination completed (72 percent) would detect any existing patterns of degradation and provides reasonable assurance of the continued structural integrity for the RPV. Therefore, based on the impracticality of meeting the Code requirements, and the reasonable assurance of structural integrity provided by the examinations, relief is granted for the second ISI interval in accordance with 10 CFR 50.55a(g)(6)(i).

Part C: **Outlet Nozzle to Reactor Vessel Shell Welds:**
(RC-RPV-WR-13/14/72-X and RC-RPV-WR-13/14/72-Z)

Licensee's Basis for Relief (as stated):

ASME Section XI, 1986 Edition, Subsection IWB, Figure IWB-2500-7(a) defines the required examination volume of this weld. It is not possible to obtain ultrasonic interrogation of greater than 90 percent of this volume due to the configuration of the outlet nozzle boss. The contour of the nozzle radius (See Figure 4 of licensee's submittal) restricts the URSULA manipulator's ability to scan the weld and adjacent base material from the vessel shell. Access to approximately 36 percent of the examination volume from the vessel shell is restricted. The remaining 64 percent of the examination volume will be examined from the shell side in accordance with the 1986 Edition of Section XI. As there is no limitation for the examination from the nozzle bore, 100 percent of the weld is examined by at least two angles.

This weld was examined in the first interval during the ten-year reactor vessel examination in 1990. The examination coverage during the 1990 examination is approximately the same as will be examined during the second interval ten-year reactor vessel examinations. No indications exceeding the acceptance criteria of ASME Section XI were noted during the 1990 examinations.

Due to the configuration of the reactor vessel, it is impractical to meet the examination requirements of the 1986 Edition of ASME Section XI. Relief is requested in accordance with 10 CFR 50.55a(g)(5)(iii).

Licensee's Proposed Alternative Examination:

The Outlet Nozzle to Shell welds were examined from the shell side to the extent possible. The examinations were performed from both sides of the welds scanning both parallel and perpendicular to the welds. The actual examination coverage obtained during the April 2000 outage was 64 percent for weld WR-13/14/72-X and 62 percent for weld WR-13/14/72-Z. The requirements of the 1986 Edition of the ASME Section XI were met for the examination from the nozzle bore.

Staff Evaluation

The Code requires 100 percent volumetric examination of the subject RPV outlet nozzle to shell welds. However, complete examination is restricted due to the configuration of the reactor vessel. To gain access for examination, the RPV nozzle would require design modifications. Imposition of this requirement would create an undue burden on the licensee.

The licensee examined these welds to the extent practical, obtaining significant coverage of at least 62 percent of the required examination volume for each weld. In addition, other Class 1 nozzles are being examined as required by the Code. Therefore, any existing patterns of degradation would be detected by the examinations and reasonable assurance of structural integrity is provided.

Based on the impracticality of meeting the Code coverage requirements for the subject nozzle-to-vessel welds, and the reasonable assurance of structural integrity provided by the examinations completed on these and other Class 1 nozzles, relief is granted in accordance with 10 CFR 50.55a(g)(6)(i).

Part D: Core Flood Nozzle Inner Radius Sections
(Weld Numbers RC-RPV-WR-54/55-W-IR and RC-RPV-WR-54/55-Y-IR)

Licensee's Basis for Relief (as stated):

ASME Section XI, 1986 Edition, Subsection IWB, Figure IWB-2500-7 defines the required examination volume. It is not possible to obtain ultrasonic interrogation of greater than 90 percent of the inside radius section volume due to scan limitations caused by the flow restrictor located in the bore of the core flood nozzles and the radius blend between the reactor vessel shell and the bore of the core flood nozzle. The flow restrictor is welded to the bore of the core flood nozzle and therefore is not removable. This restriction prohibits any examination of the inside radius from the bore of the nozzle. The entire volume can be scanned from the vessel shell using 45 ° and 70 ° transducers. This results in 100 percent of the inside radius being scanned for circumferential flaws and 5 percent of the volume being scanned for axial flaws. This results in a composite examination of 52 percent of the inner radius volume.

This area was examined in the first interval during the ten-year reactor vessel examination requirements in 1990. The examination coverage during the 1990 examination consisted of circumferential scans. No indications were noted during the 1990 examinations.

Due to the configuration of the reactor vessel, it is impractical to meet the examination requirements of the 1986 Edition of ASME Section XI. Relief is requested in accordance with 10 CFR 50.55a(g)(5)(iii).

Licensee's Proposed Alternative Examination:

The inside radius section were examined from the vessel shell using 45 ° and 70 ° transducers. No scan was performed from the nozzle bore. Approximately 52 percent of the required examination volume of the inner radius was examined.

Staff Evaluation

ASME Section XI requires 100 percent volumetric examination of all RPV nozzle inside radius sections. However, component geometry and flow restrictors in the bore of each nozzle obstruct access and preclude performance of the 100 percent volumetric examination for the nozzle inside radius sections. These physical restrictions make the Code coverage requirements impractical for the core flood nozzles at Davis-Besse. To meet the Code requirements, design modifications would be needed to provide access for examination. Imposition of this requirement would result in a burden on the licensee.

The licensee has performed the Code-required volumetric examinations to the extent practical, obtaining approximately 52 percent coverage for the nozzle inside radius sections. In addition, all other RPV nozzle inside radius sections are required to be examined each 10-year ISI interval. The staff finds that the extent of examination completed (52 percent) for the inside radius sections, and the examination of other RPV nozzles, would detect any existing patterns of degradation and provides that reasonable assurance of the continued structural integrity has been provided. Therefore, based on the impracticality of meeting the Code requirements, and the reasonable assurance of structural integrity provided by the examinations, relief is granted for the second ISI interval in accordance with 10 CFR 50.55a(g)(6)(i).

3.3 Relief Request Number RR-A20

The components for which relief is requested:

Reactor Vessel Flange to Shell Circumferential Weld (Weld Number RC-RPV-WR-19)

Code Requirement (as stated):

The 1986 Edition of ASME Section XI requires the examination of vessel welds to comply with Article 4 of ASME Section V as amended by IWA-2232 of Section XI. Relief to use the requirements of Appendix VIII of the 1995 Edition, 1996 Addenda of ASME Section XI using the PDI protocol is requested.

Licensee's Basis for Relief (as stated):

10 CFR 50.55a as amended by the Federal Register (FR-99-24256) requires the implementation of Appendix VIII, Supplements 4 and 6 of the 1995 Edition, 1996 Addenda of ASME Section XI prior to (on) November 20, 2000. 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 DBNPS 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 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.

Licensee's Alternative Examination:

The licensee 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."

The licensee obtained full examination coverage of the subject weld with all required directions and angles. The weld was examined from both sides using examination techniques qualified in accordance with the PDI protocol. Table 1 provides a comparison of the reactor vessel shell weld examination techniques.

Staff Evaluation

The 1986 Edition of Section XI requires the examination of vessel welds to comply with Article 4 of Section V as amended by IWA-2232 of Section XI. The licensee proposes the use of ultrasonic examination procedures and techniques that have been developed to meet Appendix VIII, Supplements 4 and 6, of the 1995 Edition, 1996 Addenda for the examination of RPV shell-to-flange welds.

The staff has reviewed and evaluated the licensee's alternative to use UT techniques (personnel, equipment, and procedures) qualified to Appendix VIII Supplements 4 and 6. Based on the licensee's ability to obtain full coverage on the subject weld and the staff's review of the PDI protocol, the staff concludes that the proposed alternative examination of the shell-to-flange weld would provide an equivalent or better examination than the current Code requirements and RG 1.150 recommendations and thus, would provide assurance that flaws that could be detrimental to the integrity of the RPV would be detected. Therefore, the staff has determined that the licensee's proposed alternative provides an acceptable level of quality and safety. Pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized for the RPV shell-to-flange weld examination for the second 10-year ISI interval at Davis-Besse.

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TABLE - COMPARISON OF REACTOR PRESSURE VESSEL SHELL WELD EXAMINATION TECHNIQUES

Description (Code Reference)	**ASME Section V, Article 4, 1986 Edition ASME Section XI, 1986 Edition NRC Regulatory Guide 1.150, Revision 1, 1983**	**FTI Examination Procedure 54-ISI-800-03 Requirements**
Examination Angles	Section V, Article 4, T-441 requires the volume of weld and adjacent base material be scanned by straight and angle beam techniques. Two angle beams, having nominal angles of 45 and 60 ° with respect to a perpendicular to the examination surface, shall generally be used. Other pairs of angle beams are permitted provided the measured difference between the angles is at least 10 degrees.	Examination was conducted with angles of 45 ° shear wave and 45 ° longitudinal wave transducers. Additionally, a 70 ° longitudinal wave transducer was used for examining the near surface region. These examination angles were successfully qualified under PDI protocol using the PDI program test blocks.
Instrument Calibrations	Section V, Article 4, T-441 requires that instrument screen height and amplitude linearity by evaluated at least every three months. Section XI, IWA-2232 requires that these screen height and linearity checks be performed at the beginning and end of the weld examination performed on a vessel during one outage.	Instrument screen height and amplitude linearity were checked prior to and following completion of the examinations of the DBNPS reactor vessel.
System Calibrations	Section V, Article 4, T-432 requires that the original system calibration be performed on the Code basic calibration block. T-432 allows the use of different types of reference blocks and electronic simulators to perform system calibration verifications.	Initial calibration of the data acquisition system was performed on the Code basic calibration block. Periodic system checks and final calibration check was performed using simulator blocks as permitted by Section V, Article 4, T-432.
Scanning Sensitivity	Section V, Article 4, T-424 permits scanning be performed at the reference level when electronic distance-amplitude correction (DAC) is used with automated recording.	Scanning was performed at 10% of DAC.
Recording Level	Section V, Article 4, T-441 requires recording and evaluation of reflectors that produce a response equal to or greater than 50% DAC. Regulatory Guide 1.150 requires recording and evaluation at 20% DAC for the inner 25% of material thickness.	In the near surface region, non-geometric indications with a maximum amplitude greater than or equal to 20% DAC were recorded. In the subsurface region, non-geometric indications which have a maximum amplitude greater than 10% DAC for the 45 ° longitudinal scan and 20% DAC for the 45 ° shear wave were recorded.

Description (Code Reference)	ASME Section V, Article 4, 1986 Edition ASME Section XI, 1986 Edition NRC Regulatory Guide 1.150, Revision 1, 1983	FTI Examination Procedure 54-ISI-800-03 Requirements
Scan Index and Pulse Repetition Rate	Section V, Article 4, T-424 requires each pass of the search unit overlap a minimum of 10% of the transducer piezoelectric element dimension perpendicular to the direction of the scan. Section XI, IWA-2232 requires each pass of the search unit overlap at least 50% of the transducer piezoelectric element dimension perpendicular to the direction of the scan. NRC Regulatory Guide 1.150 requires a 25% maximum overlap for detection and 0.25 inch maximum increments for sizing.	A scan index of 0.50" was used for detection. A scan index of 0.20" was used for sizing. This scan index meets the requirements of T-424, IWA-2232 and Regulatory Guide 1.150.
Flaw Sizing and Evaluation	Section V, Article 4, T-441 requires amplitude based sizing at 50% DAC. Section V, Article 4, T-451 permits evaluation to alternative standards.	All recorded indications are evaluated and categorized as either geometric or non-geometric indication. Tip diffraction or satellite signals are used for measuring flaw through wall dimension. If the flaw image cannot identify evidence of flaw tips or satellite signals, amplitude based sizing techniques are used. Length sizing is performed using amplitude based techniques.