FirstEnergy

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Docket Number 50-346

License Number NPF-3

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: Request for Relief from Certain American Society of Mechanical Engineers (ASME) Code Requirements for Inservice Inspection for the Davis-Besse Nuclear Power Station

Ladies and Gentlemen:

The purpose of this letter is to request relief pursuant to 10 CFR 50.55a(a)(3) from certain requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for the Davis-Besse Nuclear Power Station (DBNPS). Details concerning the requested relief are provided in attached Relief Requests RR-A18, RR-A19, and RR-A20.

Relief Request RR-A18 requests relief from Section XI of the ASME Code, 1986 Edition, No Addenda, Table IWB-2500-1, Code Items B15.51 and B15.71, which require a system hydrostatic test of Class 1 piping and valves. Performance of the Reactor Coolant System hydrostatic test requires that normally closed vent, drain, and instrument valves be opened to pressurize the short piping between the inboard and outboard isolation valves. Relief from repositioning these valves while performing the system hydrostatic test is requested in accordance with 10 CFR 50.55a(a)(3)(ii).

Relief Request RR-A19 requests relief from Section XI of the ASME Code, 1986 Edition, No Addenda, Table IWB-2500-1, Examination Categories B-A and B-D. Code Case N-460 requires the examination coverage of Class 1 welds be greater than 90 percent. Due to the configuration of the DBNPS Reactor Vessel, it is not possible to obtain greater than 90 percent examination coverage on the Reactor Vessel Lower Shell to Bottom Head Circumferential Weld, the Reactor Vessel Bottom Head Circumferential Weld, the Outlet Nozzle to Reactor Vessel Shell Welds, and the Core Flood Nozzle Inner Radius Sections. Relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii).

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Relief Request RR-A20 requests relief from paragraph IWA-2232 of Section XI of the ASME Code, 1986 Edition, No Addenda for the Reactor Vessel Flange to Shell Weld. This paragraph requires ultrasonic examinations of vessel welds be performed in accordance with Article 4 of ASME Section V. FENOC proposes to examine the Reactor Vessel Flange to Shell Weld in accordance with Appendix VIII of the 1995 Edition, 1996 Addenda of ASME Section XI. Relief is requested in accordance with 10 CFR 50.55a(a)(3)(i).

NRC approval of these relief requests is requested by January 31, 2000, in order to support the Twelfth Refueling Outage which is scheduled to commence on April 1, 2000.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 321-8466.

Sincerely yours,

GMW/s

Attachments

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. N. Bailey, DB-1 NRC/NRR Project Manager
K. S. Zellers, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

RELIEF REQUEST
RR-A18

Component Description:

Reactor Coolant System small diameter (≤ 1 inch) vent, drain, and instrument piping

ASME Code Class:

ASME Section XI, Class 1 Piping

ASME Examination Requirements

Subsection IWB-2500, Table IWB-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.

Basis for Relief:

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 hydrostatic test is performed in Mode 3 with the Reactor Coolant System 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 Reactor Coolant System. FENOC proposes to perform the system hydrostatic test of the Reactor Coolant System 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 Reactor Coolant System 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 Reactor Coolant System leakage monitoring during normal plant operation (Modes 1, 2, 3, and 4). Should any 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 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 Reactor Coolant System work. The system hydrostatic 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 hydrostatic test and the duration of the outage.

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).

It should be noted that the NRC approved a similar relief request for the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TAC Nos. MA2118 and MA2119) on September 3, 1998.

Alternative Examination:

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 Reactor Coolant System boundary.

RELIEF REQUEST
RR-A19

Component Description:

- Reactor Vessel Lower Shell to Bottom Head Circumferential Weld (Weld Number RC-RPV-WR-34)
- Reactor Vessel Bottom Head Circumferential Weld (Weld Number RC-RPV-35)
- Outlet Nozzle to Reactor Vessel Shell Welds (Weld Numbers RC-RPV-WR-13/14/72-X and RC-RPV-WR-13/14/72-Z)
- Core Flood Nozzle Inner Radius Sections (Weld Numbers RC-RPV-WR-54/55-W-IR and RC-RPV-WR-54/55-Y-IR)

The location of these welds in the Davis-Besse Nuclear Power Station Reactor Vessel is shown in Figure 1.

ASME Code Class:

ASME Section XI, Class 1

ASME Examination Requirements

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, Table 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.

Basis for Relief:

The reactor vessel welds are examined from the inside surface using the Framatone URSULA inspection manipulator in conjunction with the Framatone ACCUSONEX data acquisition and analysis system. URSULA is a computer controlled, remotely operated manipulator which uses a contact 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.

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

  Basis for Relief:

  Figure 2 provides a drawing of the Reactor Vessel Lower Shell to Bottom Head Circumferential Weld (Weld Number RC-RPV-WR-34).

  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.

  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 examination. No reportable indications were noted during the 1990 examinations.

The Core Support Lugs were also visually, examined during the ten year reactor vessel examination in 1990. No deficiencies were noted during this examination.

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).

Alternative Examination:

The accessible area will be examined 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. These examinations will be 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 will be approximately 60 percent of the required examination volume.

- **Reactor Vessel Bottom Head Circumferential Weld (RC-RPV-WR-35)**

Basis for Relief:

Figure 3 provides a drawing of the Reactor Vessel Bottom Head Circumferential Weld (Weld Number RC-RPV-WR-35).

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.

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).

Alternative Examination:

The accessible area will be 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. These examinations will be 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 will be approximately 72 percent of the required examination volume.

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

Basis for Relief:

Figure 4 provides a drawing of the Outlet Nozzle to Reactor Vessel Shell Welds (Weld Numbers RC-RPV-WR-13/14/72-X and RC-RPV-WR-13/14/72-Z).

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) 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).

Alternative Examination:

The Outlet Nozzle to Shell weld will be examined from the shell side to the extent possible. These examinations will be 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 will be approximately 64 percent of the required examination volume. The requirements of the 1986 Edition of ASME Section XI will be met for the examination from the nozzle bore.

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

Basis for Relief:

Figure 5 provides a drawing of the Core Flood Nozzle that includes the Inside Radius Sections (Weld Numbers RC-RPV-WR-54/55-W-IR and RC-RPV-WR-54/55-Y-IR).

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 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).

Alternative Examination:

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

Reactor Vessel Weld Layout

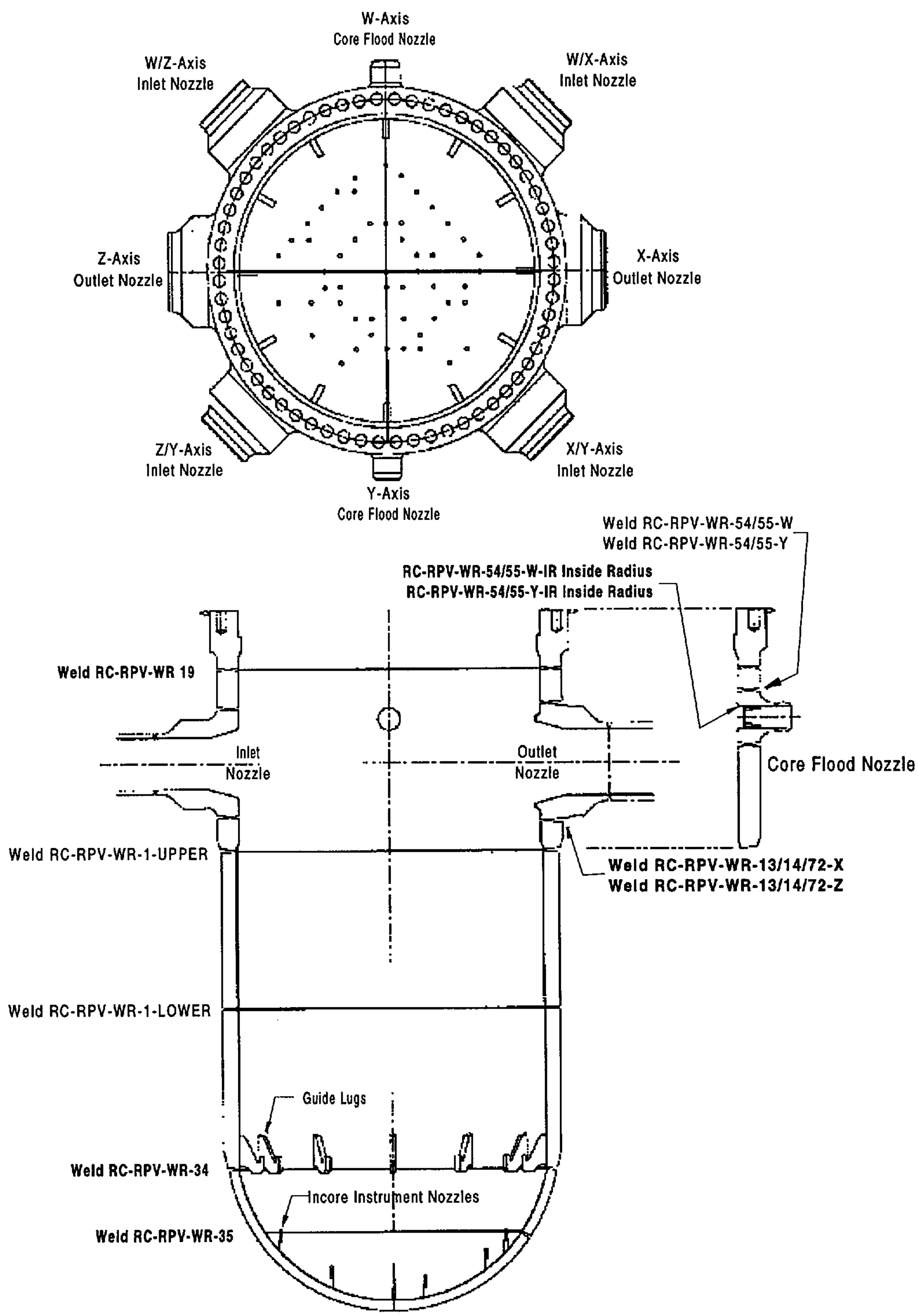
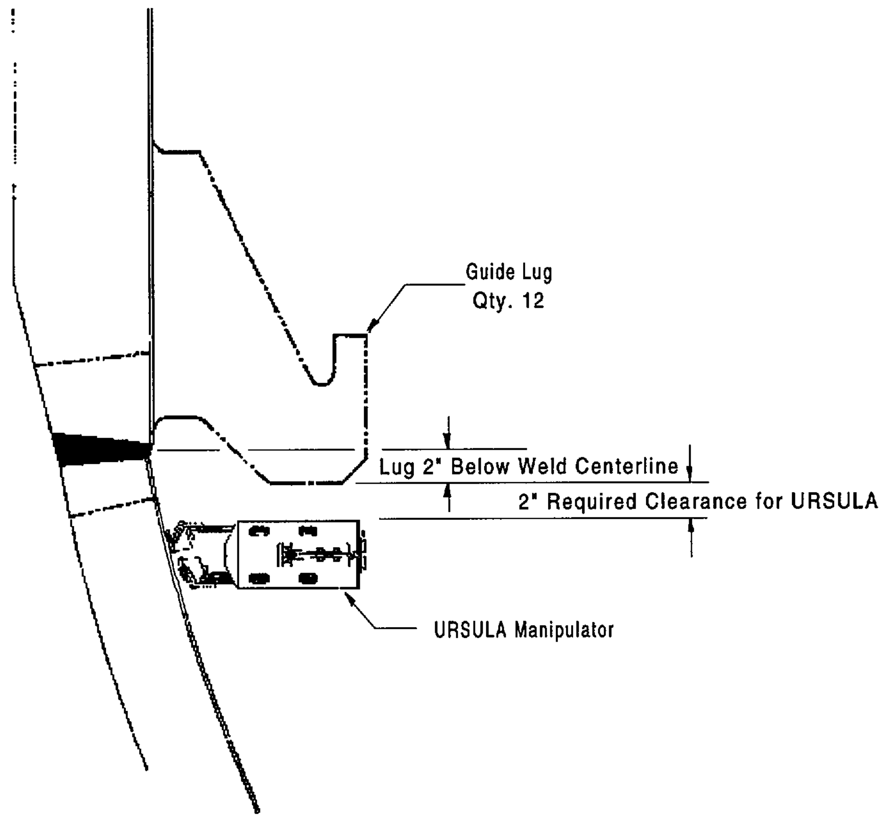


Figure 1

Weld RC-RPV-WP-34



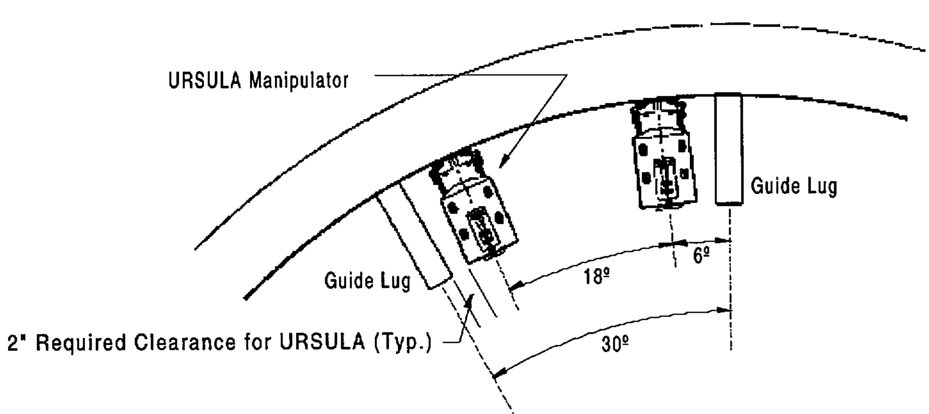


Figure 2

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Weld RC-RPV-WP-35

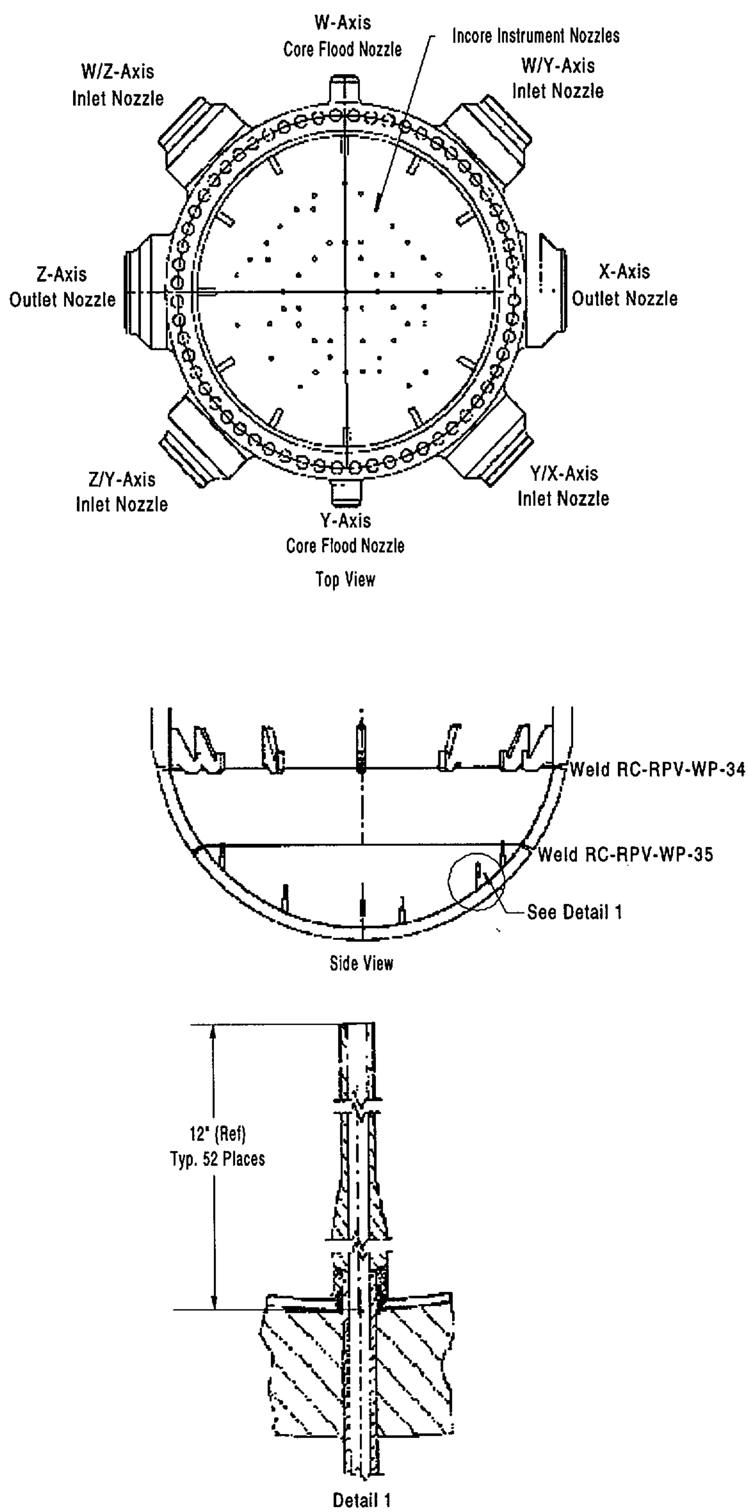


Figure 3

Welds RC-RPV-WR-13/14/72-X and -Z

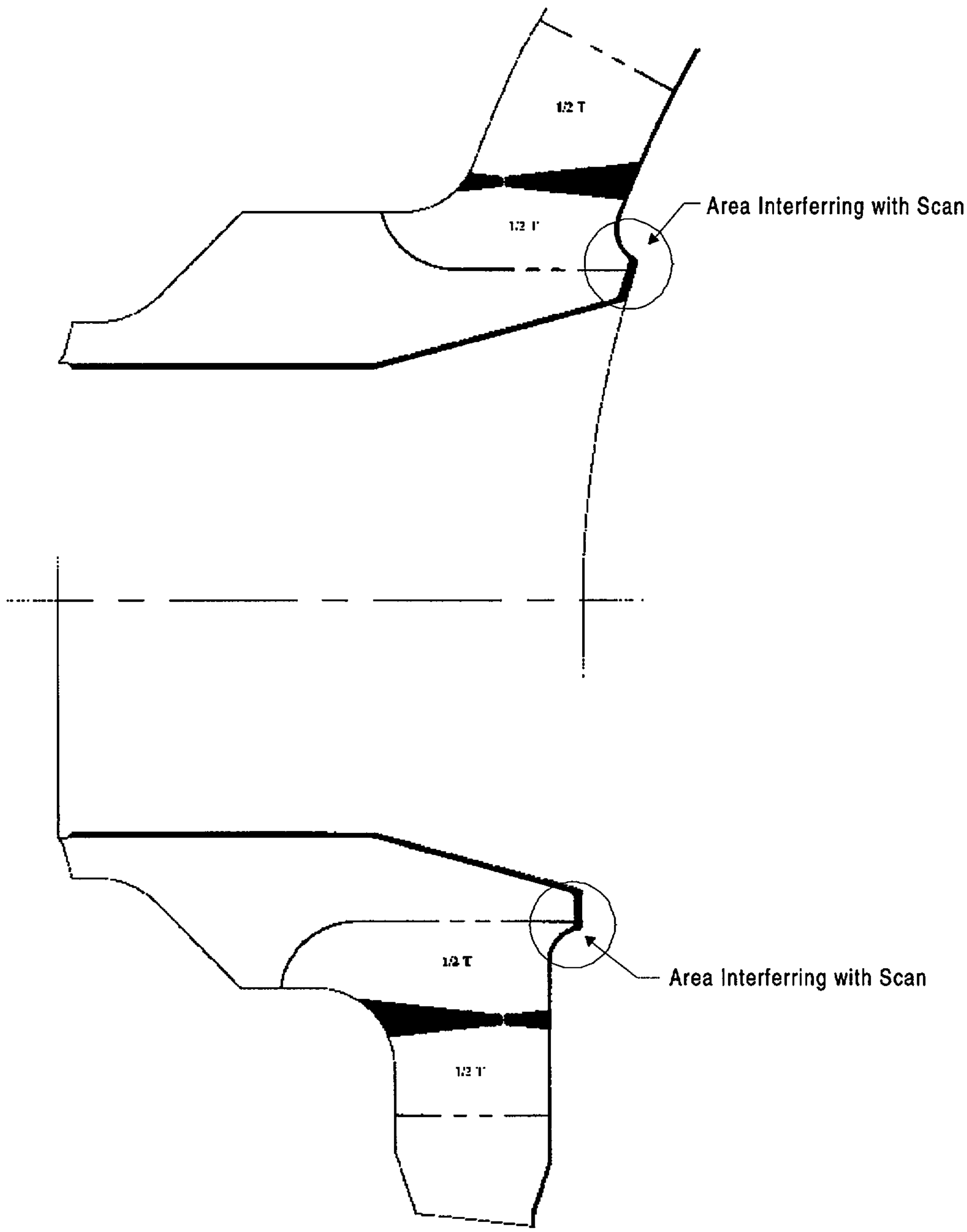
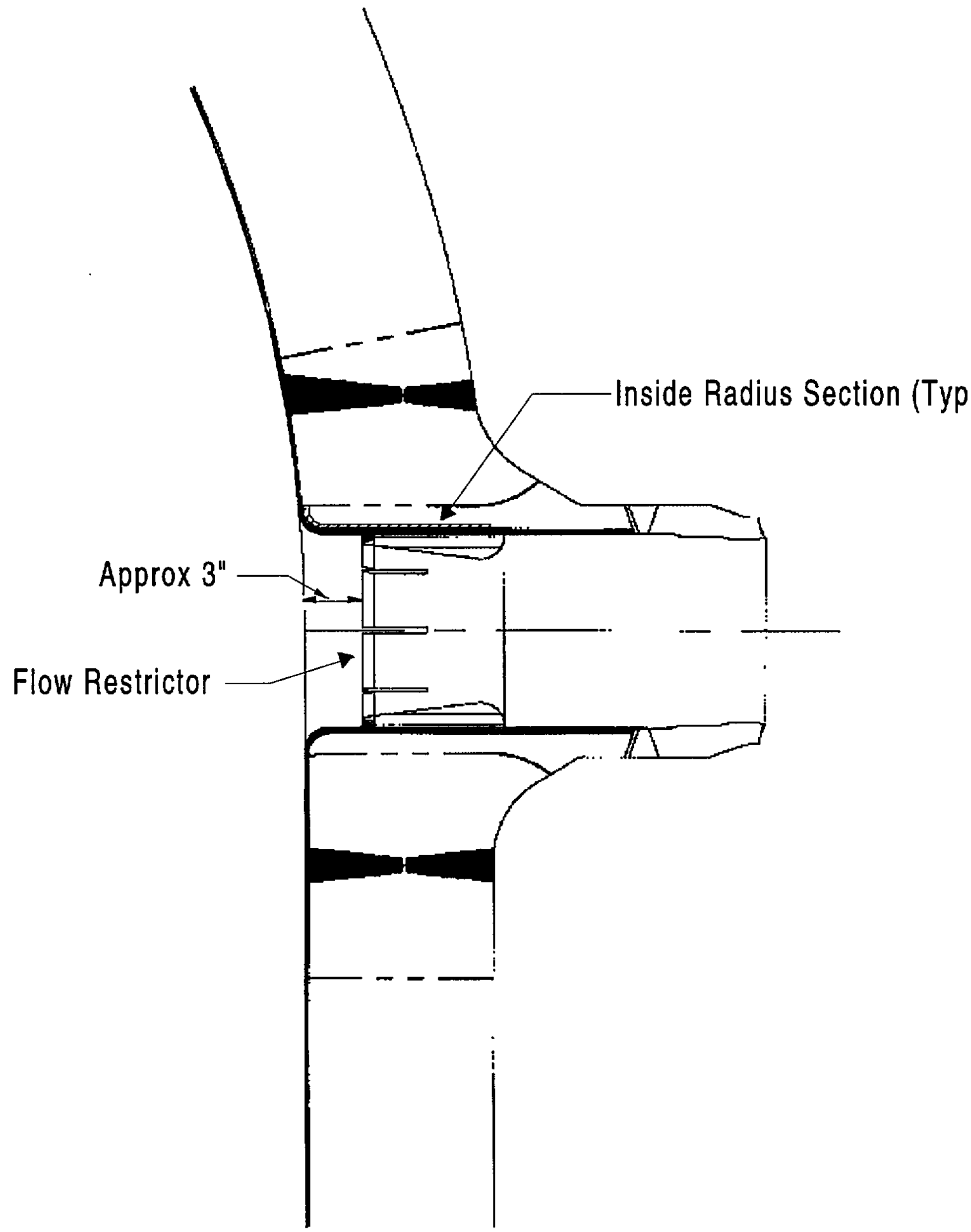


Figure 4

Inside Radius RC-RPV-WR-54/55-W-IR and -Y-IR



Top View

Figure 5

RELIEF REQUEST
RR-A20

Component Description:

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

ASME Code Class:

ASME Section XI, Class 1

ASME Examination Requirements

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 Performance Demonstration Initiative (PDI) protocol is requested.

Basis for Relief:

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 November 22, 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 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 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.

Alternative Examination:

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 Performance Demonstration Initiative (PDI) protocol.

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station in this document. Any other actions discussed in the submittal represent intended or planned actions by Davis-Besse. They are described only as information and are not regulatory commitments. Please notify the Manager - Regulatory Affairs (419-321-8466) at Davis-Besse of any questions regarding this document or associated regulatory commitments.

COMMITMENT	DUE DATE
Implement Relief Requests RR-A18, RR-A19, and RR-A20	Within 120 days after NRC approval of the specified Relief Requests.