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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Duke Energy
Oconee Nuclear Station, Unit 1
Docket Nos. 50-269
Third Ten Year Inservice Inspection Interval
Request for Relief No. 02-004

Pursuant to 10 CFR 50.55a(g)(5)(iii), attached is a Request for Relief from the requirement to examine 100% of the volume specified by the ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition with no Addenda (as modified by Code Case N-460). This request is to allow Duke Energy to take credit for twenty (20) limited ultrasonic examinations on Reactor Vessel welds described in the attached request. During examination of the subject Unit 1 welds, the ultrasonic examination coverage did not meet the 90% examination requirements of Code Case N-460. The obtainable volume coverage for each weld examination is indicated on the attached request. Achievement of greater examination coverage for these welds is impractical due to piping/valve geometry, interferences, and existing examination technology. Therefore, Duke Energy requests that the NRC grant relief as authorized under 10 CFR 50.55a(g)(6)(i).

If there are any questions or further information is needed you may contact R. P. Todd at (864) 885-3418.

Very truly yours,

W. R. McCollum, Jr.
Site Vice President

Attachment

A047

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Proposed Relief in Accordance with 10 CFR 50.55a(g)(5)(iii)
Inservice Inspection Impracticality

Duke Energy Corporation
Oconee Nuclear Station – Unit 1 (EOC-20)
Third 10-Year Interval – Inservice Inspection Plan
ASME Section XI Code – 1989 Edition with No Addenda

	I.	II. & III.	IV.	V.	VI.	VII.
Limited Area/Weld I.D. Number	**System / Component for Which Relief is Requested:** Area or Weld to be Examined	**Code Requirement from Which Relief is Requested:** 100% Exam Volume Coverage Exam Category Item No. Fig. No. Limitation Percentage	**Basis for Relief**	**Alternate Examinations or Testing**	**Justification for Granting Relief**	**Implementation Schedule**
1-RPV-WR34	NC System Reactor Vessel Lower Shell to Lower Head Ring Circumferential Weld	Exam Category B-A Item No. B01.011.005 Fig. IWB-2500-1 36% Volume Coverage	See Paragraph "A"	See Paragraph "G"	See Paragraph "H"	See Paragraph "K"
1-RPV-WR35	NC System Reactor Vessel Lower Head Cap to Lower Head Ring Circumferential Weld	Exam Category B-A Item No. B01.021.003 Fig. IWB-2500-3 42% Volume Coverage	See Paragraph "A"	See Paragraph "G"	See Paragraph "H"	See Paragraph "K"
1-RPV-WR13	NC System Reactor Vessel Outlet Nozzle-to-Vessel Weld @ 90°	Exam Category B-D Item No. B03.090.001 Fig. IWB-2500-7(a) 82% Volume Coverage (UT from vessel I.D.)	See Paragraph "B"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR13	NC System Reactor Vessel Outlet Nozzle-to-Vessel Weld @ 90°	Exam Category B-D Item No. B03.090.001A Fig. IWB-2500-7(a) 82% Volume Coverage (UT from nozzle I.D.)	See Paragraph "B"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"

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1-RPV-WR13A	NC System Reactor Vessel Outlet Nozzle-to-Vessel Weld @ 270°	Exam Category B-D Item No. B03.090.002 Fig. IWB-2500-7(a) 82% Volume Coverage (UT from vessel I.D.)	See Paragraph "B"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR13A	NC System Reactor Vessel Outlet Nozzle-to-Vessel Weld @ 270°	Exam Category B-D Item No. B03.090.002A Fig. IWB-2500-7(a) 82% Volume Coverage (UT from nozzle I.D.)	See Paragraph "B"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR54	NC System Reactor Vessel Core Flood Nozzle-to-Vessel Weld @ 0°	Exam Category B-D Fig. IWB-2500-7(a) Item No. B03.090.007 (UT from vessel I.D.) 81% Volume Coverage Item No. B03.090.007A (UT from nozzle ID) 0% Volume Coverage (not able to scan)	See Paragraph "C"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR54A	NC System Reactor Vessel Core Flood Nozzle-to-Vessel Weld @ 180°	Exam Category B-D Fig. IWB-2500-7(a) Item No. B03.090.008 (UT from vessel ID) 81% Volume Coverage Item No. B03.090.008A (UT from nozzle ID) 0% Volume Coverage (not able to scan)	See Paragraph "C"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR54	NC System Reactor Vessel Core Flood Nozzle Inside Radius Section @ 0°	Exam Category B-D Item No. B03.100.007 Fig. IWB-2500-7(a) 52% Volume Coverage	See Paragraph "D"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"

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1-RPV-WR54A	NC System Reactor Vessel Core Flood Nozzle Inside Radius Section @ 180°	Exam Category B-D Item No. B03.100.008 Fig. IWB-2500-7(a) 52% Volume Coverage	See Paragraph "D"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR53	NC System Reactor Vessel Core Flood Nozzle-to-Safe-End Butt Weld @ 0°	Exam Category B-F Item No. B05.010.001 Fig. IWB-2500-8(c) 86% Volume Coverage (UT from nozzle I.D. in lieu of PT from O.D.)	See Paragraph "E"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR53	NC System Reactor Vessel Core Flood Nozzle-to-Safe-End Butt Weld @ 0°	Exam Category B-F Item No. B05.010.001A Fig. IWB-2500-8(c) 86% Volume Coverage (UT from nozzle side)	See Paragraph "E"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR53	NC System Reactor Vessel Core Flood Nozzle-to-Safe-End Butt Weld @ 0°	Exam Category B-F Item No. B05.010.001B Fig. IWB-2500-8(c) 86% Volume Coverage (UT from safe-end side)	See Paragraph "E"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR53A	NC System Reactor Vessel Core Flood Nozzle-to-Safe-End Butt Weld @ 180°	Exam Category B-F Item No. B05.010.002 Fig. IWB-2500-8(c) 81% Volume Coverage (UT from nozzle I.D. in lieu of PT from O.D.)	See Paragraph "E"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-RPV-WR53A	NC System Reactor Vessel Core Flood Nozzle-to-Safe-End Butt Weld @ 180°	Exam Category B-F Item No. B05.010.002A Fig. IWB-2500-8(c) 81% Volume Coverage (UT from nozzle side)	See Paragraph "E"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"

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1-RPV-WR53A	NC System Reactor Vessel Core Flood Nozzle-to-Safe-End Butt Weld @ 180°	Exam Category B-F Item No. B05.010.002B Fig. IWB-2500-8(c) 81% Volume Coverage (UT from safe-end side)	See Paragraph "E"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-53A-02-43L	NC System Reactor Vessel Core Flood Safe-End to Pipe Circumferential Weld @ 0°	Exam Category B-J Item No. B09.011.090 Fig. IWB-2500-8(c) 76% Volume Coverage	See Paragraph "F"	See Paragraph "G"	See Paragraph "I	See Paragraph "K"
1-53A-02-43L	NC System Reactor Vessel Core Flood Safe-End to Pipe Circumferential Weld @ 0°	Exam Category B-J Item No. B09.011.090A Fig. IWB-2500-8(c) 76% Volume Coverage (UT from nozzle I.D. in lieu of PT from O.D.)	See Paragraph "F"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-53A-01-1L	NC System Reactor Vessel Core Flood Safe-End to Pipe Circumferential Weld @ 180°	Exam Category B-J Item No. B09.011.100 Fig. IWB-2500-8(c) 83% Volume Coverage	See Paragraph "F"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"
1-53A-01-1L	NC System Reactor Vessel Core Flood Safe-End to Pipe Circumferential Weld @ 180°	Exam Category B-J Item No. B09.011.100A Fig. IWB-2500-8(c) 83% Volume Coverage (UT from nozzle I.D. in lieu of PT from O.D.)	See Paragraph "F"	See Paragraph "G"	See Paragraph "I"	See Paragraph "K"

Note: See Attachment A for a drawing on all the welds listed above.

IV. Basis for Relief (See Attachment A for area/weld locations.)

Paragraph A:
During the ultrasonic examination of welds 1-RPV-WR34 and 1-RPV-WR35, 100% coverage of the required examination volume could not be obtained. The examination coverage was limited to 36% and 42% respectively. Limitations were caused by the core guide lugs & flow stabilizers for WR34 and incore nozzles & flow stabilizers for WR35 that restrict the scanning surface. The percentage of coverage reported represents the aggregate coverage. In order to achieve more coverage the core guide lugs, incore nozzles and flow stabilizers would have to be moved to allow greater access for scanning, which is impractical.

Paragraph B:
During the ultrasonic examination of welds 1-RPV-WR13 and 1-RPV-WR13A, 100% coverage of the required examination volume could not be obtained. The examination coverage was limited to 82%. Limitations were caused by the outlet nozzle boss that restricts the scanning surface. The percentage of coverage reported represents the aggregate coverage. In order to achieve more coverage, the outlet nozzle boss would have to be moved to allow greater access for scanning, which is impractical.

Paragraph C:
During the ultrasonic examination of welds 1-RPV-WR54 and 1-RPV-WR54A, 100% coverage of the required examination volume could not be obtained. The examination coverage was limited to 81% of the required volume from one side of the weld. Limitations were caused by the flange taper and inlet nozzles that restrict the scanning surface. The percentage of coverage reported represents the aggregate coverage. In order to achieve more coverage, the inlet nozzles would have to be moved and the taper on the flange would have to be redesigned to allow greater access for scanning, which is impractical. In addition, because of the proximity of the flow restrictors no scanning was performed from the nozzle I.D. (0% examination coverage). In order to achieve more coverage, the flow restrictor would have to be moved to allow access for scanning, which is impractical.

Paragraph D:
During the ultrasonic examination of inside radius sections 1-RPV-WR54 and 1-RPV-WR54A, 100% coverage of the required examination volume could not be obtained. The examination coverage was limited to 52%. Limitations were caused by the flow restrictor that prevents scanning the surface. The percentage of coverage reported represents the aggregate coverage. In order to achieve more coverage, the flow restrictor would have to be moved to allow greater access for scanning, which is impractical.

Paragraph E:
During the ultrasonic examination of welds 1-RPV-WR53 and 1-RPV-WR53A, 100% coverage of the required examination volume could not be obtained. The examination coverage was limited to 86% and 81%, respectively. Limitations were caused by air at the top of nozzle that prevents the transducer from making contact for scanning the surface. The percentage of coverage reported represents the aggregate coverage. In order to achieve more coverage, the reactor coolant pumps would have to be in operation to permit reactor coolant flow which would remove the air at the top of the nozzle, which is impractical.

Paragraph F:
During the ultrasonic examination of welds 1-53A-02-43L and 1-53A-01-1L, 100% coverage of the required examination volume could not be obtained. The examination coverage was limited to 76% and 83%, respectively. Limitations were caused by air at the top of nozzle that prevents the transducer from making contact for scanning the surface. The percentage of coverage reported represents the aggregate coverage. In order to achieve more coverage, the reactor coolant pumps would have to be in operation to permit reactor coolant flow which would remove the air at the top of the nozzle, which is impractical.

V. Alternate Examinations or Testing

Paragraph G:
The scheduled 10-year code examination was performed on the referenced area/weld and it resulted in the noted limited coverage of the required ultrasonic volume. No additional examinations are planned for the area/weld during the current inspection interval.

VI. Justification for Granting Relief

Paragraph H:
Ultrasonic examination of welds for item numbers B01.011 and B01.021 were conducted using personnel, equipment and procedures qualified in accordance with ASME Section XI, Appendix VIII, 1995 Edition with the 1996 Addenda as administered through the Performance Demonstration Initiative (PDI) Program. Although 100% coverage of the examination volume could not be achieved, the amount of coverage obtained for this examination provides an acceptable level of quality and integrity. (See Paragraph J for additional justification.)

Paragraph I:
Ultrasonic examination of areas/welds for item numbers B03.090, B03.100, B05.010 and B09.011 were conducted using personnel, equipment and procedures qualified in accordance with ASME Section XI, Appendix I, 1989 Edition with no Addenda. Although 100% coverage of the examination volume could not be achieved, the amount of coverage obtained for this examination provides an acceptable level of quality and integrity. (See Paragraph J for additional justification.)

Paragraph J:
Duke Energy will use pressure testing and VT-2 visual examination to compliment the limited examination coverage. The Code requires (reference Table IWB-2500-1, item numbers B15.010 and B15.050) that a system leakage test be performed after each refueling outage for Class 1. Additionally a system hydrostatic test (reference Table IWB-2500-1, item numbers B15.011 and B15.051) is required once during each 10-year inspection interval. These tests require a VT-2 visual examination for evidence of leakage. This testing provides adequate assurance of pressure boundary integrity.

Duke Energy will use VT-3 visual examination to compliment the limited examination coverage. The Code requires (reference Table IWB-2500-1, item number B13.010) that a VT-3 examination be performed after the first refueling outage and subsequent refueling outages at approximately 3 year periods. During the first and second periods of an interval a VT-3 examination is performed on areas above and below the reactor core that are made accessible for examination by removal of components during normal refueling outages. During the third period of an interval the VT-3 examination is performed on all of the reactor vessel interior surfaces at the same time that the automated UT exams are performed on the reactor vessel welds. This examination provides adequate assurance of pressure boundary integrity.

In addition to the above Code required examinations (volumetric, pressure test, and VT-3), there are other activities which provide a high level of confidence that, in the unlikely case that leakage did occur through these welds, it would be detected and isolated. Specifically, leakage from these welds would be detected by monitoring of the Reactor Coolant System (RCS), which is performed once each shift under procedure PT/1,2,3/A/0600/10, "RCS Leakage". This RCS leakage monitoring is a requirement of Technical Specification 3.4.13, "Reactor Coolant System Leakage". Leakage is also evaluated in accordance with this Technical Specification. The leakage could also be detected through several other methods. One is the RCS mass balance calculation. A second is the Reactor Building air particulate monitor. This monitor is sensitive to low leak rates; the iodine monitor, gaseous monitor and area monitor are capable of detecting any fission products in the coolant and will make these monitors sensitive to

coolant leakage. A third is the level indicator in the Reactor Building normal sump. A fourth is a loss of level in the Letdown Storage Tank.

Due to the design of the reactor vessel and location of the core guide lugs, flow stabilizers, outlet nozzle boss, flow restrictors and inlet nozzles and air in the top of some of the nozzles; it is not feasible to obtain the examination coverage required for all of the welds listed in this request for relief. Duke Energy has examined the welds/components referenced in this request to the maximum extent possible utilizing the latest in examination techniques and equipment. These welds were rigorously inspected by volumetric NDE methods during construction and verified to be free from unacceptable fabrication defects. Based on the portions and results of the required volumetric and visual examinations performed during this outage, it's our opinion that this combination of examinations provides a reasonable assurance of component integrity. Thus, an acceptable level of quality and safety will have been achieved and allowing relief from the aforementioned Code requirements will not endanger public health and safety.

VII. Implementation Schedule

Paragraph K
The scheduled third 10-year interval plan code examination was performed on the referenced area/weld resulting in limited volumetric coverage. No additional examinations are planned for the area/weld during the current inspection interval. The same area/weld may be examined again as part of the next (fourth) 10-year interval plan, depending on the applicable code year edition and addenda requirements adopted in the future.

VIII. Other Information

The following individuals contributed to the development of this relief request:

James J. McArdle (NDE Level III Inspector) provided Sections II through V and part of Section VI.

B. W. Carney, Jr. (Oconee Engineering) provided part of Section VI.

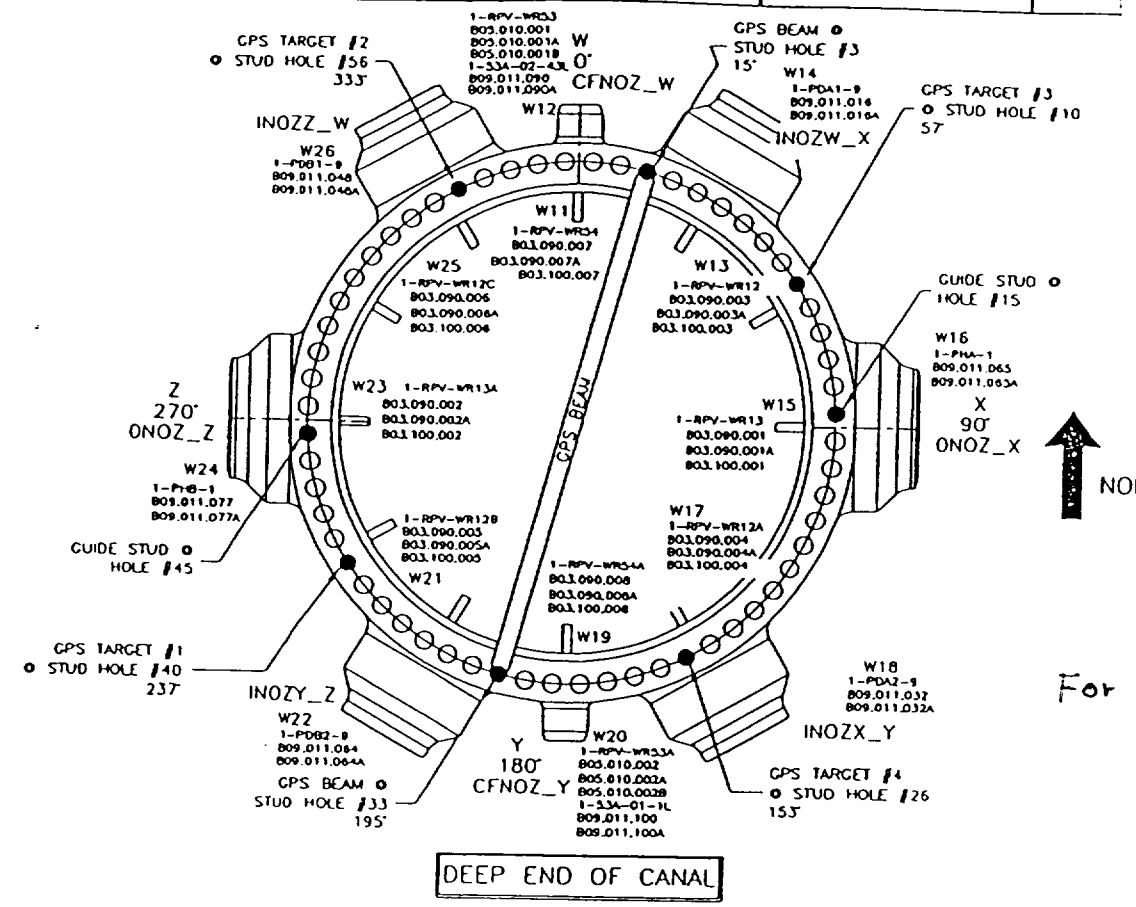
Larry C. Keith (Oconee ISI Plan Manager) compiled the remaining sections.

Sponsored By: Larry C. Keith **Date** 7-23-02

Approved By: R. Kevin Rhyne **Date** 7/23/02

Attachment A Drawing on Reactor Vessel Welds

Attachment A



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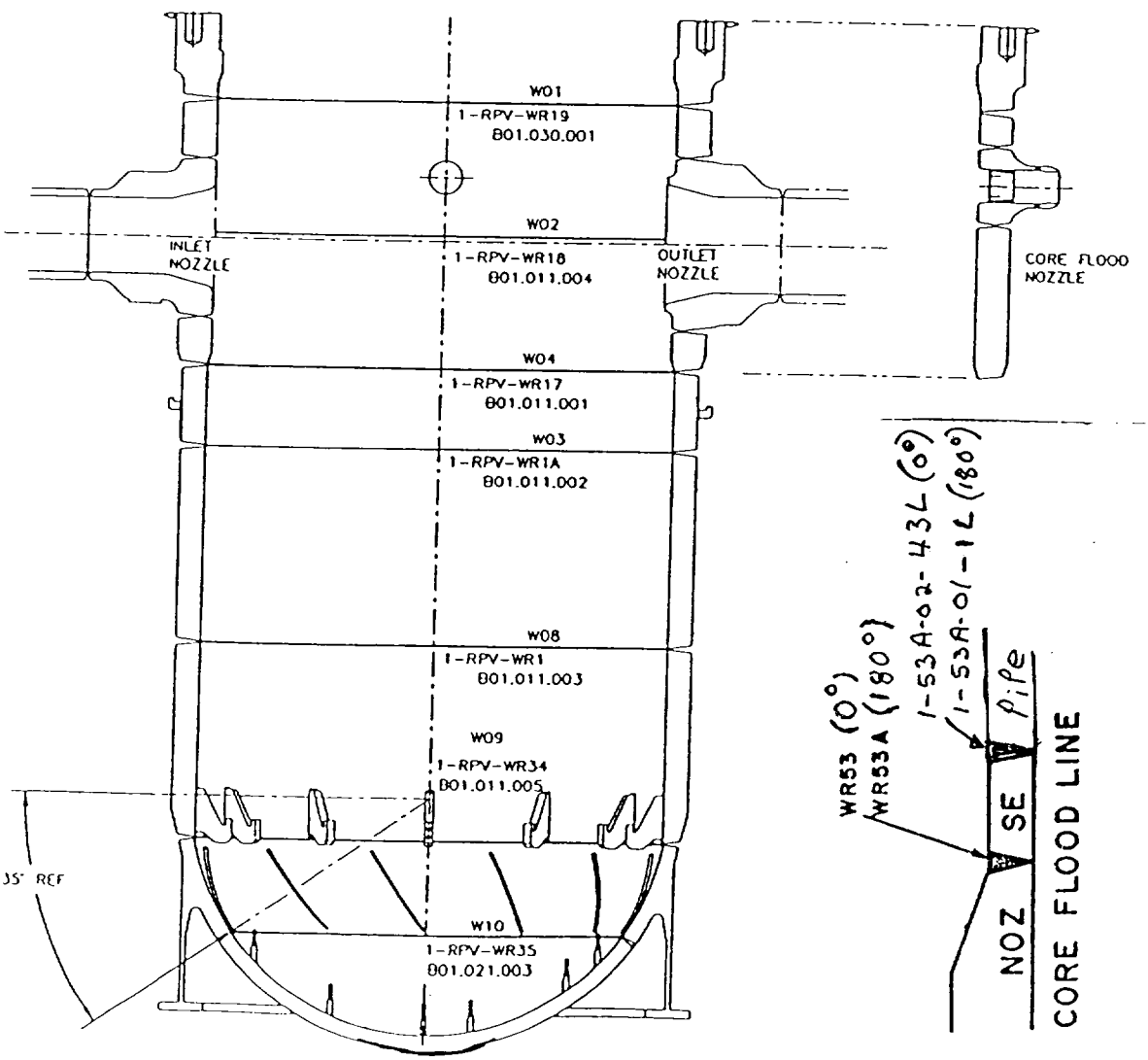
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