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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

NRC 2003-0060 10 CFR 50.55a

July 3, 2003

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

POINT BEACH NUCLEAR PLANT
DOCKETS 50-266 and 50-301
REQUEST FOR RELIEF FROM THE PROVISIONS OF ASME SECTION XI
IMPRACTICAL REQUIREMENTS

Point Beach Nuclear Plant (PBNP) is required to have an Inservice Inspection (ISI) Program for examination of ASME Class 1, 2, and 3 systems and their supports. During the third Inservice Inspection interval, ISI examinations were performed on the required number of components in accordance with 10 CFR 50.55a and ASME Section XI, 1986 Edition with no Addenda.

On June 30, 2002, Point Beach Nuclear Plant ended its third interval ISI Program. The required examinations were conducted in a manner that met all Section XI Code requirements to the extent practical. There were areas where the Code required coverage could not be obtained. Where possible, PBNP substituted other welds and supports to avoid those situations where complete Code compliance could not be attained. In many cases, PBNP was successful with finding an alternate area where full Code compliance could be achieved. This Request for Relief details those areas where the Code required examination criteria could not be completely met, and to what extent Code compliance was achieved. This relief does not address any limited examinations on the Reactor Pressure Vessel, as those were requested and approved at an earlier date.

PBNP requests relief in accordance with 10 CFR 50.55a(g)(5)(iii), as these areas either could not be examined in accordance with Section XI Code requirements or without significant modifications to the plant. We recognize that this request should have been submitted by June 30, 2003 and will address this issue via our corrective action process.

This letter contains no new commitments and no revision to existing commitments.

A. J. Cayia

A. J. Cayia
Site Vice-President

RLT/kmd

Attachment

A047

cc: Regional Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, NRR, USNRC
NRC Resident Inspector - Point Beach Nuclear Plant
Department of Commerce, State of Wisconsin, Mike Verhagen

Attachment
Point Beach Nuclear Plant, Units 1 and 2

Component Identification

Code Class: 1, 2

Examination Categories: See attached tables

Item Numbers: See attached tables

Description: Limited Section XI Code examinations during Code required examinations

Code Requirements

Third Inservice Inspection Interval

1986 Edition of Section XI, no Addenda

ASME Code Case N-460, Alternative Examination Coverage for Class 1 and 2 Welds

Volumetric and surface examinations of welds and base material will be examined in accordance with the applicable Examination Category and Item Number.

Basis for Relief

During the third inservice inspection interval, Point Beach Nuclear Plant (PBNP) used the requirements of the 1986 Edition of ASME Section XI for examination of components. This edition of Section XI has specific requirements for what components will be examined and to what extent they will be examined. This edition of Section XI was followed with the exception of the selection of Class 1 piping welds. In accordance with 10 CFR 50.55a(b)(2)(ii), PBNP used the 1974 Edition of Section XI for selection of Class 1 Category B-J piping welds. This selection criteria states that for each inservice inspection interval, a different 25% of the population of class 1 piping welds shall be examined. Since this was the third 10-year interval, essentially 50% of the Class 1 piping welds had already been examined, leaving only 50% of the total Class 1 welds available for selection and examination. During previous intervals, most of the areas selected for examination were in low radiation areas and were relatively easy to access. This meant that, for the third interval, a larger percentage of areas available for examination were in higher radiation areas or required additional support in order to gain access. These factors led to many of the welds being examined having a higher probability of having limited examinations. For all other Examination Categories, PBNP used the Section XI required examination selection criteria.

During the performance of scheduled examination, there were numerous instances where examiners reported some type of interference. PBNP personnel evaluated every instance where this situation was reported, and when possible, alternative examination areas were selected to avoid having to examine restricted areas. This reduced the population of welds where a limited examination would be encountered. On welds where alternatives were not available, additional techniques were performed to increase coverage where possible, such as using steeper angle beam ultrasonic techniques, approaching the examination area from a different direction, or in limited cases, radiography was employed. In each case, this increased examination area coverage to the extent practicable and reduced the number of welds with limited examinations.

In order to gain additional access to the areas where limited examinations were encountered, major modification of components would be required. These modifications could be extensive, up to and including complete replacement. In cases where minor grinding would allow additional coverage, this was performed.

Major modification of components is not a feasible approach nor is it required to obtain additional coverage. Modifying systems and components to improve examination area coverage would result in additional dose with marginal improvement in quality or safety. 10 CFR 50.55a(g)(1) states that plants of PBNP's vintage (construction permit before January 1, 1971) must meet the requirements, except design and access provisions and preservice examination requirements, set forth in ASME Section XI. Access has been improved over the years to many areas of the plants, further reducing those areas with limitations, but it is not a requirement to make every area available for examination.

Removal of structural interferences to gain additional examination area coverage is costly in time and dose with little impact to quality or safety. Structural steel and component supports, were not designed for removal. If any were removed, temporary support structures would need to be designed and installed to compensate for the reduction in load bearing capacity. Temporary modifications, such as these, could require extensive scaffold building and temporary structural components, resulting in additional accumulated radiation dose.

For volumetric examinations, PBNP examined the required areas to the extent practical using ultrasonic examination techniques. Since May 22, 2000, many of these components were examined using personnel qualified in accordance with Appendix VIII of Section XI and as implemented by the Performance Demonstration Initiative (PDI). All examinations performed prior to the implementation of the 10 CFR 50.55a(g)(6)(i)(C) Appendix VIII requirements, show the limitations of the 1986 Edition of Section XI. All examinations performed after the implementation of the Appendix VIII requirements show limitations based on component configuration, material, or limitations in the PDI procedures being used. Where possible and allowed by the applicable PDI Generic Procedure, additional angles were used to increase coverage. In many cases, no combination of ultrasonic angle beam examinations would cover the entire examination area. In each case, the maximum feasible coverage was obtained.

Radiography was performed on many welds to increase or obtain complete coverage. This reduced the number of limited examinations. Each component with limited examination coverage was evaluated to determine whether radiography would increase coverage. Use of radiography to improve examination area coverage was balanced with the risk associated with performing radiography and the personnel doses expected.

For surface examinations, the maximum surface area that was available was examined. Where possible, additional insulation was removed, including asbestos insulation. Occasionally, where possible, supports or other components were moved to provide additional coverage. Again, the removal of supports and structural steel could adversely impact the system or other components, and would increase radiation exposure and provide little or no benefit. Where combinations of insulation or component removal would not increase coverage, the maximum amount of coverage possible was obtained.

PBNP performs system leakage tests in accordance with the Pressure test requirements of Section XI, Examination Categories B-P and C-H. These pressure tests cover every component within the Code boundaries established by PBNP. There has been no through wall leakage noted of any component during the third inservice inspection interval. Where leakage was noted at mechanical connections, these were corrected in accordance with maintenance procedures.

The attached figures show typical configurations for the affected welds and components, giving a visual representation of the areas where examination coverage could not be attained. Figures 3 and 4 show typical one and two directional examination coverage. The Percent Coverage column shows either an average of the examination coverage, or if more specific information was available, the coverage from each side of the weld.

Proposed Alternative

PBNP proposes to use the examination volume or surface coverage obtained during the third interval examinations on the listed components in lieu of the Code required volumes and surfaces (see attachments). The coverage obtained meets the intent of ASME Section XI and provides an acceptable level of quality and safety.

Conclusion

The examinations performed during the third inservice inspection interval were performed to the extent possible. Additional coverage was impractical, as modification of systems, structures, and components would have resulted in significant radiation exposure with minimal increase in the level of quality and safety.

Period for Which Relief Is Requested

Relief is requested for the third inservice inspection interval at PBNP, which ended on June 30, 2002.

Attachments

Attachment 1 – Listing of Limited Examinations for the Third Inservice Inspection Interval

Attachment 2 - Sketches of Areas With Limited Examination Coverage

Attachment 1
Listing of Limited Examinations for
the Third Inservice Inspection Interval

Point Beach Nuclear Plant Unit 1								
Component Identification	Catgy/ Item no.	System or Component	Configuration	Fig. No.	NDE Method	Limitation Description	Percent Full Coverage	Remarks
RPV-HFlange	B-A B1.40	Reactor Pressure Vessel	Head to Flange	15	UT	Limited by welded attachments of head lift rig and flange configuration	86%	100% Exam From Head Side, Partial exam from flange side
PZR-SprayNoz-IRS	B-D B3.120	Pressurizer Inside Radius Section	Inside Radius Section	11	UT	Examination limited due to nozzle configuration and permanent insulation support rings	73%	
RHE-N1-IRS	B-D B3.150	Regenerative Heat Exchanger	Inlet Nozzle Inner Radius	16	UT	Exam limited due to nozzle configuration.	~50%	
RHE-N4-IRS	B-D B3.150	Regenerative Heat Exchanger	Outlet Nozzle Inner Radius	16	UT	Exam limited due to nozzle configuration.	~50%	
RC-34-MRCL-AI-03	B-F B5.130	Reactor Coolant	Elbow to S/G Inlet Nozzle	7	UT	Elbow is static cast stainless steel and is highly attenuative, and has limiting ~1/2" step. Nozzle blend radius limits proper transducer contact	31%	An additional 31% partial coverage was obtained
RC-36-MRCL-AII-01	B-F B5.130	Reactor Coolant	S/G Outlet Nozzle to Elbow	7	UT	Elbow is static cast stainless steel and is highly attenuative, and has limiting ~1/2" step. Nozzle blend radius limits proper transducer contact	2%	An additional 29% partial coverage was obtained
RC-34-MRCL-BI-03	B-F B5.130	Reactor Coolant	Elbow to S/G Inlet Nozzle	7	UT	Elbow is static cast stainless steel and is highly attenuative, and has limiting ~1/2" step. Nozzle blend radius limits proper transducer contact.	31%	An additional 31% partial coverage was obtained

Point Beach Nuclear Plant Unit 1								
Component Identification	Catgy/ Item no.	System or Component	Configuration	Fig. No.	NDE Method	Limitation Description	Percent Full Coverage	Remarks
RC-36-MRCL-BII-01	B-F B5.130	Reactor Coolant	S/G Outlet Nozzle to Elbow	7	UT	Elbow is static cast stainless steel and is highly attenuative, and has limiting ~1/2" step. Nozzle blend radius limits proper transducer contact.	14%	An additional 36% partial coverage was obtained
RPV Closure Nut	B-G-1 B6.10	Reactor Pressure Vessel	Closure Nut	12	MT	Middle 1/3 of threaded area inside of nut is inaccessible	67%	Geometric limitation of nut in the threaded area
RC-34-MRCL-BI-02	B-J B9.11	Reactor Coolant	Pipe to Elbow	8	UT	Elbow is static cast stainless steel and is highly attenuative. Elbow has limiting ~1/2" step. Weld has limiting ~1/2" step.	74%	
RC-36-MRCL-BII-02	B-J B9.11	Reactor Coolant	Elbow to Pipe	8	UT	Elbow is static cast stainless steel and is highly attenuative. Elbow has limiting ~1/2" step.	79%	An additional 21% partial coverage was obtained on the remainder of the examination area.
RC-36-MRCL-BII-06	B-J B9.11	Reactor Coolant	Elbow to Pump	7	UT	Elbow is static cast stainless steel and is highly attenuative. Elbow has limiting ~1/2" step. Pump configuration does not allow any examination.	0%	An additional 59% partial coverage was obtained from the elbow side. The nozzle on Fig. 7 is similar to the configuration of the RC pump but with less area from which to examine.
RC-08-DR-1001-01	B-J B9.11	Reactor Coolant	Branch Connection to Pipe	14	UT	Limited exam due to branch connection configuration	46%	An additional 54% partial coverage was obtained on the remainder of the examination area.

Point Beach Nuclear Plant Unit 1								
Component Identification	Catgy/ Item no.	System or Component	Configuration	Fig. No.	NDE Method	Limitation Description	Percent Full Coverage	Remarks
SIS-10-SI-1003-19	B-J B9.11	Safety Injection	Elbow to Valve 1SI-867A	10	UT	Valve is cast stainless material and is highly attenuative. Limited exam due to valve configuration	9%	An additional 41% partial coverage was obtained on the remainder of the examination area. Best effort examination on 50% of examination area (valve side) see note 1.
RC-10-SI-1003-20	B-J B9.11	Reactor Coolant	Valve 1SI-867A to Elbow	10	UT	Valve is cast stainless material and is highly attenuative. Limited exam due to valve configuration	28%	An additional 27% partial coverage was obtained on the remainder of the examination area. Best effort examination on 50% of examination area (valve side) see note 1.
RC-10-SI-1003-21	B-J B9.11	Reactor Coolant	Elbow to Branch Connection	14	UT	Limited exam due to branch connection configuration	33%	An additional 18% partial coverage was obtained on the remainder of the examination area. Best effort examination on 50% of examination area (pipe side) see note 1.
AC-10-SI-1001-19	B-J B9.11	Auxiliary Cooling (SI)	Pipe to Valve 1SI-867B	10	UT	Valve is cast stainless material and is highly attenuative. Limited exam due to valve configuration	14%	An additional 36% partial coverage was obtained on the remainder of the examination area. Best effort examination on 50% of examination area (valve side) see note 1.

Point Beach Nuclear Plant Unit 1								
Component Identification	Catgy/ Item no.	System or Component	Configuration	Fig. No.	NDE Method	Limitation Description	Percent Full Coverage	Remarks
RC-10-SI-1001-21	B-J B9.11	Reactor Coolant	Elbow to Pipe	17	UT	Limited by three welded attachments.	89%	Best effort examination on 4% of examination area (pipe side) Welded attachments are 1" wide and cover the weld and pipe side. see note 1.
AC-06-SI-1001-21	B-J B9.11	Auxiliary Cooling (SI)	Pipe to Valve 1SI-853D	10	UT	Valve is cast stainless material and is highly attenuative. Single side exam due to valve configuration	50%	Best effort examination on 50% of examination area (valve side) see note 1.
AC-08-RHR-1006-02	C-F-1 C5.11 B	Auxiliary Cooling (RHR)	Elbow to Valve 1RH-718B	10	UT	Valve is cast stainless material and is highly attenuative. Single side exam due to valve configuration	36%	An additional 35% partial coverage was obtained on the remainder of the examination area. Note 2
SIS-04-SI-1001-11	C-F-1 C5.11 B	Safety Injection	Elbow to Flange	1	UT	Single side exam from elbow only due to flange configuration. Elbow inner radius prevents complete transducer contact	39%	Note 2
1SI-850A-Welds	C-G C6.20	Safety Injection	Valve Body Welds	See Rmks	PT	Welds partially obstructed due to permanent restraints.	Weld 1 - 51% Weld 2 - 61% Weld 3 – 100% Weld 4 – 100%	Valve has four body welds. Two welds are partially covered by pipe restraints.

NOTE 1 - Examinations performed utilizing the rules of ASME Section XI, Appendix VIII (as implemented by the Performance Demonstration Initiative), as directed by 10 CFR 50.55a.

NOTE 2 – In accordance with Wisconsin Electric letter VPNPD-91-360 from J.J. Zach to NRC, dated October 16, 1991, Wisconsin Electric agreed to extend the criteria for selection of Class 2 piping welds to a wall thickness greater than 0.312 inches. The item number C5.11B was created to differentiate these welds from those required under the regular C-F-1 examination areas.

NOTE 3 – All percentages rounded to the nearest 1%.

Point Beach Nuclear Plant Unit 2								
Component Identification	Catgy/ Item no.	System or Component	Configuration	Fig. No.	NDE Method	Limitation Description	Percent Full Coverage	Remarks
RPV-HFlange	B-A B1.40	Reactor Pressure Vessel	Head to Flange	15	UT	Limited by welded attachments of head lift rig and flange configuration	87%	100% Exam From Head Side, Partial exam from flange side
PZR-SprayNoz-IRS	B-D B3.120	Pressurizer	Inside Radius Section	11	UT	Examination limited due to nozzle configuration and permanent insulation support rings	88%	
RHE-N1-IRS	B-D B3.150	Regenerative Heat Exchanger	Inlet Nozzle Inner Radius	16	UT	Exam limited due to nozzle configuration.	~50%	
RHE-N4-IRS	B-D B3.150	Regenerative Heat Exchanger	Outlet Nozzle Inner Radius	16	UT	Exam limited due to nozzle configuration.	~50%	
PZR-SurgeNoz-SE	B-F B5.40	Pressurizer	Safe-End to Nozzle	6	UT	Single side exam due to nozzle configuration	0%	100% partial coverage was obtained from the safe-end side Nozzle blend radius prevents proper transducer contact
RC-34-MRCL-AI-05	B-F B5.70	Reactor Coolant	Safe-End to A S/G Inlet Nozzle	6	UT	Scan limited by safe-end and nozzle geometries	59%	Nozzle blend radius prevents proper transducer contact
RC-34-MRCL-BI-05	B-F B5.70	Reactor Coolant	Safe-End to B S/G Inlet Nozzle	6	UT	Scan limited by safe-end and nozzle geometries	59%	Nozzle blend radius prevents proper transducer contact
RPV Closure Nut	B-G-1 B6.10	Reactor Pressure Vessel	Closure Nut	12	MT	Middle 1/3 of threaded area inside of nut is inaccessible	67%	Geometric limitation of nut

Point Beach Nuclear Plant Unit 2								
Component Identification	Catgy/ Item no.	System or Component	Configuration	Fig. No.	NDE Method	Limitation Description	Percent Full Coverage	Remarks
RC-34-MRCL-AI-04R1	B-J B9.11	Reactor Coolant	Elbow to S/G Inlet Nozzle Safe-End	5	UT	Elbow is cast material and is highly attenuative, and both elbow and safe-end have a limiting ~1/2" step	40%	Partial coverage was obtained on the remaining 60% of the examination area.
RC-36-MRCL-AII-01R1	B-J B9.11	Reactor Coolant	S/G Outlet Nozzle Safe-End to Elbow	5	UT	Elbow is cast material and is highly attenuative, and both elbow and safe-end have a limiting ~1/2" step	36%	Partial coverage was obtained on the remaining 64% of the examination area.
RC-36-MRCL-AII-03	B-J B9.11	Reactor Coolant	Pipe to Elbow	8	UT	Elbow is cast material and is highly attenuative, and has limiting ~1/8" step	83%	Partial coverage was obtained on the remaining 17% of the examination area.
RC-34-MRCL-BI-04R1	B-J B9.11	Reactor Coolant	Elbow to S/G Inlet Nozzle to Safe End	5	UT	Elbow is cast material and is highly attenuative, and both nozzle and safe-end have a limiting ~1/2" step.	17%	Partial coverage was obtained on the remaining 83% of the examination area.
RC-36-MRCL-BII-01R1	B-J B9.11	Reactor Coolant	S/G Outlet Nozzle Safe End to Elbow	5	UT	Elbow is cast material and is highly attenuative, and both elbow and safe-end have a limiting ~1/2" step.	30%	Partial coverage was obtained on the remaining 70% of the examination area.
RC-36-MRCL-BII-04	B-J B9.11	Reactor Coolant	Elbow to Pipe	8	UT	Elbow is cast material and is highly attenuative, and has a limiting ~1/2" step. Weld configuration prevents proper transducer contact.	25%	Partial coverage was obtained on an aditional 24% of the examination area.
RC-10-AC-2001-01	B-J B9.11	Reactor Coolant	Branch Connection to Elbow	14	UT	Limited exam due to branch connection configuration	54%	Partial coverage was obtained on the remaining 46% of the examination area.

Point Beach Nuclear Plant Unit 2								
Component Identification	Catgy/ Item no.	System or Component	Configuration	Fig. No.	NDE Method	Limitation Description	Percent Full Coverage	Remarks
RC-10-AC-2001-11	B-J B9.11	Reactor Coolant	Pipe to Valve 2RH-700	10	UT	Valve is cast stainless material and is highly attenuative. Limited exam due to valve configuration and welded attachment.	77%	Support is welded to the 10" pipe and restricts 4-1/2" of weld length.
RC-06-SI-2002-26	B-J B9.11	Reactor Coolant	Elbow to Pipe	1	UT	Limited scan due to welded attachment, elbow inner radius prevents complete transducer contact.	85%	Support is welded to the 6" pipe and restricts 3-1/2" of weld length. Partial coverage was obtained on the remaining 15% of the examination area.
AC-10-RHR-2004-09	C-F-1 C5.11 B	Auxiliary Cooling (RHR)	Pipe to Tee	2	UT	Limited exam due to tee inside saddle configuration preventing complete transducer contact.	90%	Partial coverage was obtained on the remaining 10% of the examination area. see note 1.
AC-08-RHR-2006-01	C-F-1 C5.11 B	Auxiliary Cooling (RHR)	Tee to Elbow	2	UT	Limited exam due to tee inside saddle configuration preventing complete transducer contact.	73%	Partial coverage was obtained on the remaining 27% of the examination area. see note 1.
AC-08-RHR-2002-04	C-F-1 C5.11 B	Auxiliary Cooling (RHR)	Elbow to Tee	2	UT	Limited exam due to tee inside saddle configuration preventing complete transducer contact	86%	Partial coverage was obtained on the remaining 14% of the examination area. see note 1.
SIS-06-SI-2008-27	C-F-1 C5.11	Safety Injection	Pipe to Valve SI-853B	10	UT	Valve is cast stainless material and is highly attenuative. Limited scan due to pipe restraint and valve configuration	47%	Partial coverage was obtained on the remaining 53% of the examination area.

NOTE 1 – In accordance with Wisconsin Electric letter VPNPD-91-360 from J.J. Zach to NRC, dated October 16, 1991, Wisconsin Electric agreed to extend the criteria for selection of Class 2 piping welds to a wall thickness greater than 0.312 inches. The item number C5.11B was created to differentiate these welds from those required under the regular C-F-1 examination areas.

NOTE 2 – All percentages rounded to the nearest 1%.

Attachment 2

Sketches of Areas With Limited Examination Coverage

NRC 2003-0060

12 Pages

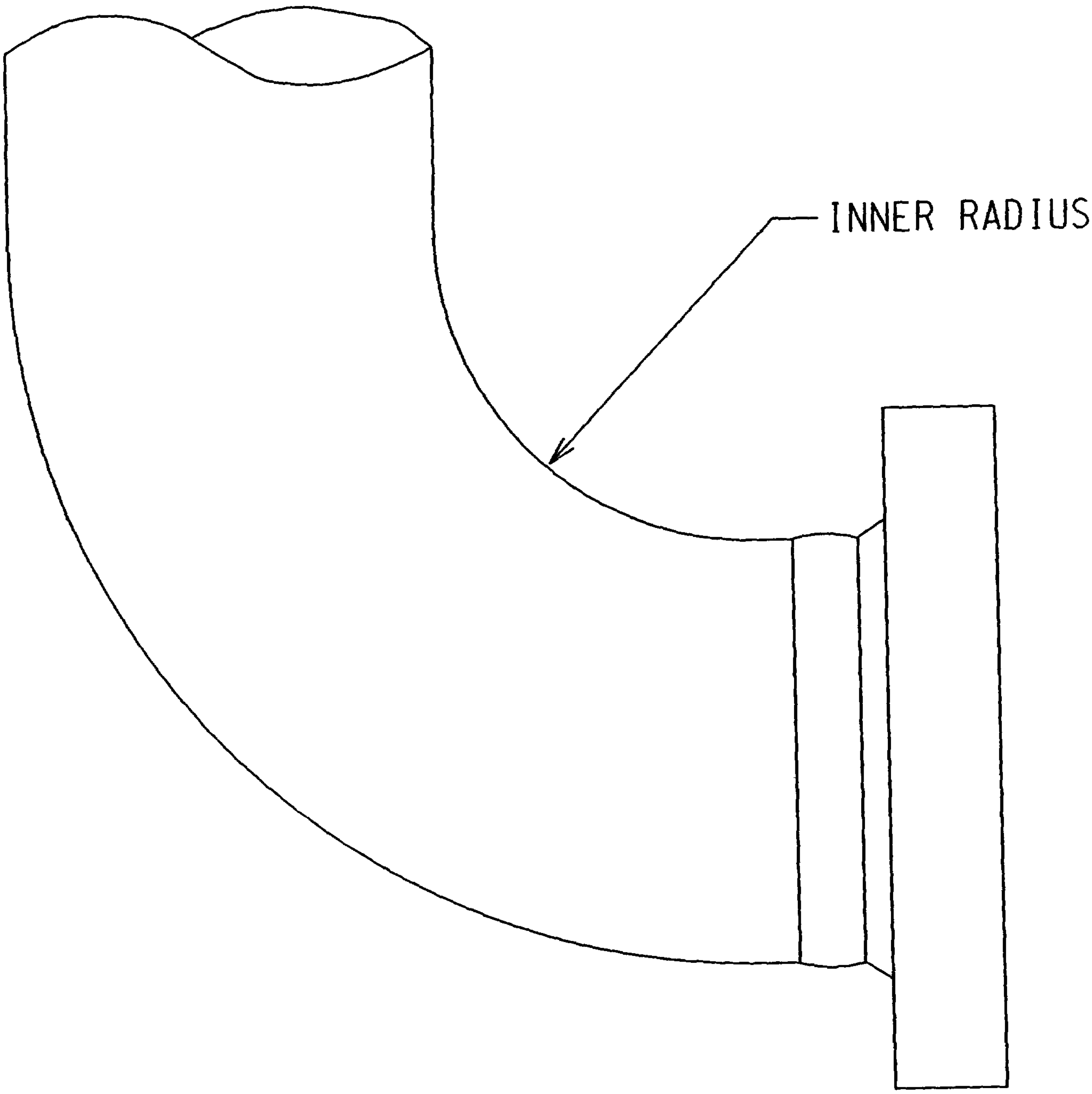


FIGURE 1

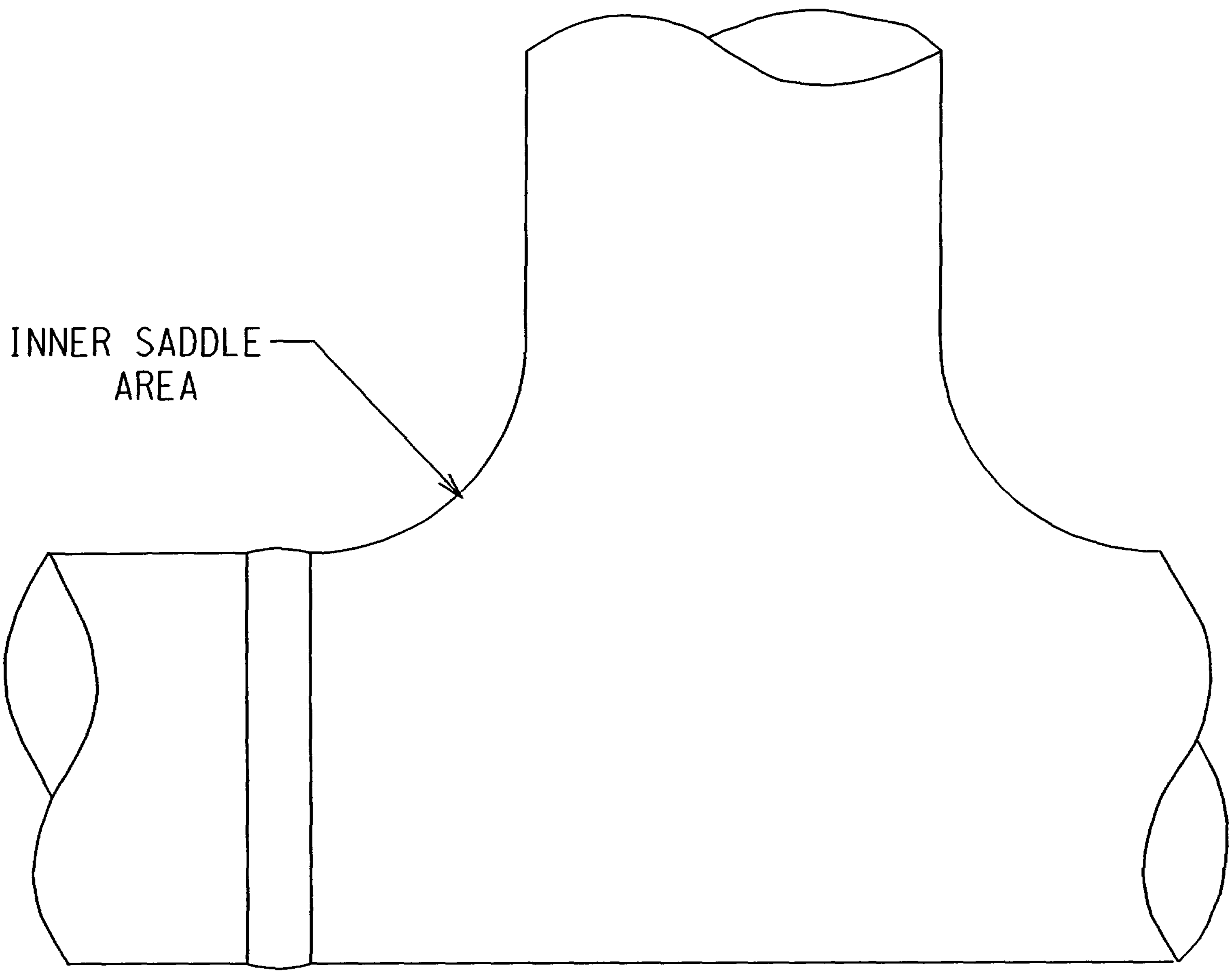


FIGURE 2

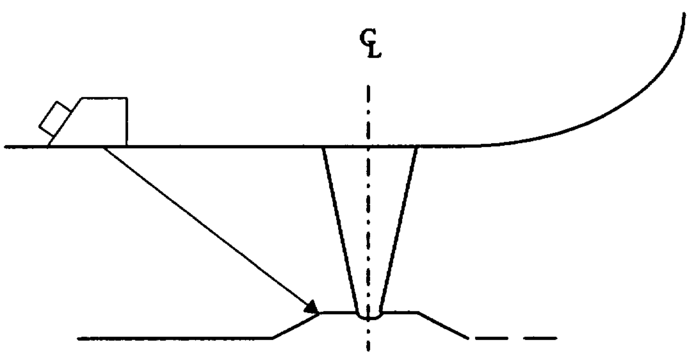


Figure 3
One Directional Coverage
(Typical)

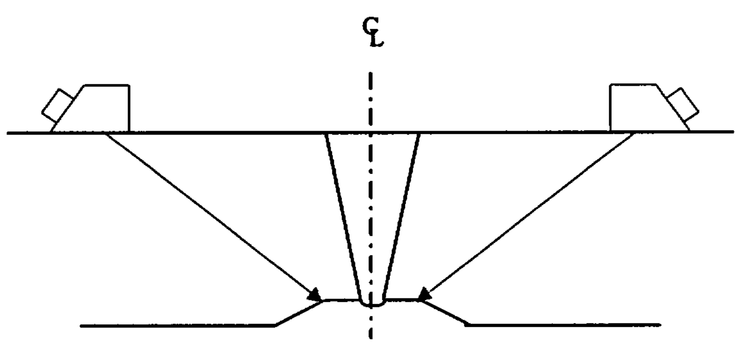


Figure 4
Two Directional Coverage
(Typical)

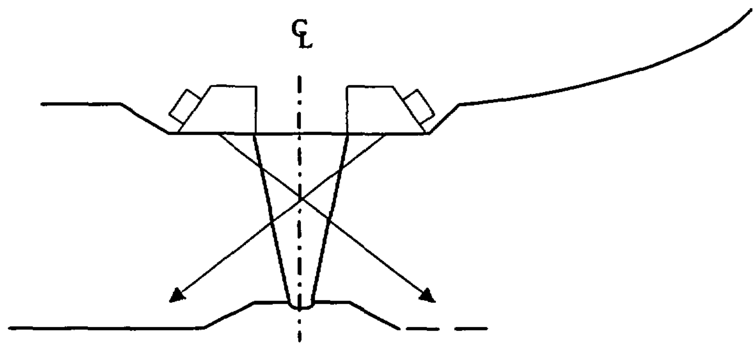


Figure 5
Elbow to Safe-End/Safe-End to Elbow
(Typical)

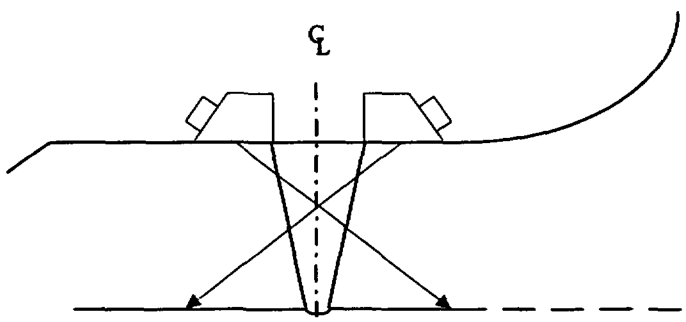


Figure 6
Safe-End to Nozzle
(Typical)

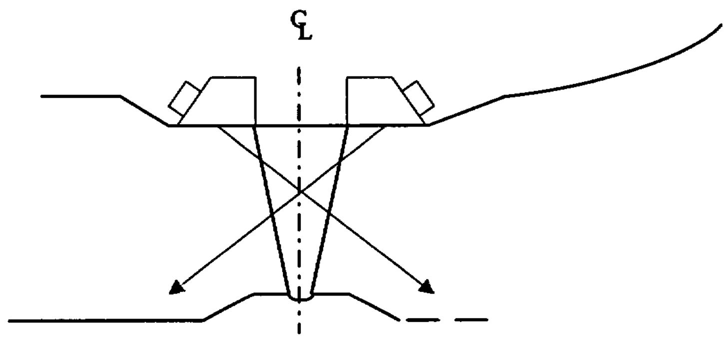


Figure 7
Elbow to Nozzle/Nozzle to Elbow
Reactor Coolant Main Loop
(Typical)

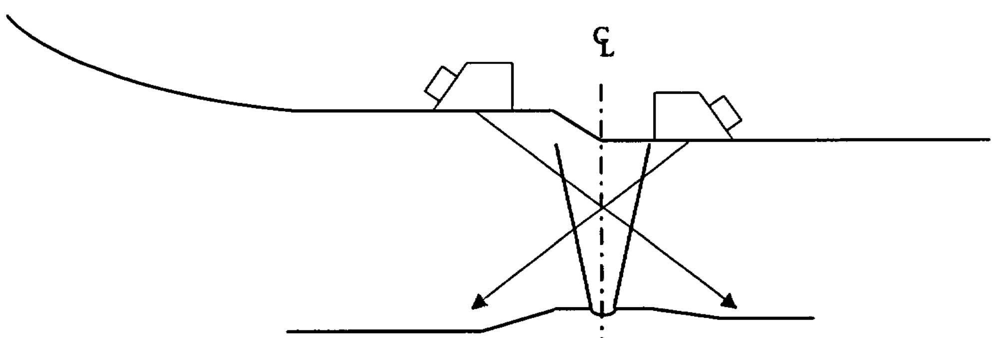


Figure 8
Elbow to Pipe/Pipe to Elbow
Reactor Coolant Main Loop
(Typical)

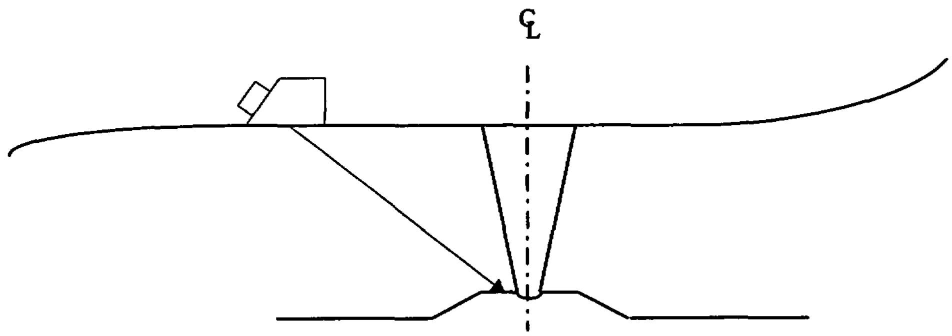


Figure 9
Elbow to Branch Connection/
Branch Connection to Elbow
(Typical)

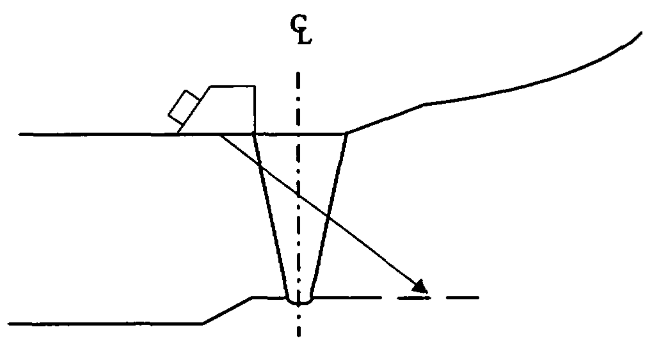


Figure 10
Pipe to Valve/Valve to Pipe
(Typical)

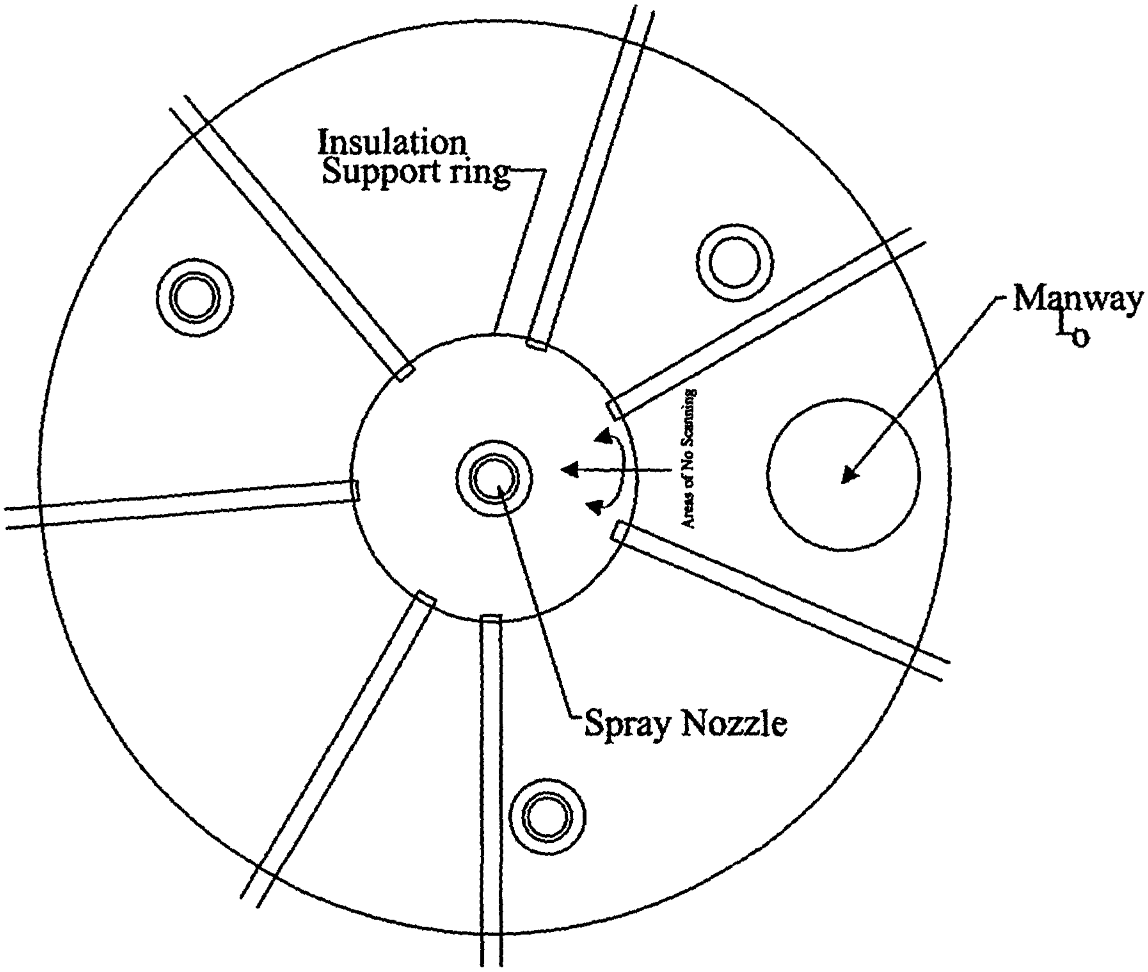


Figure 11
Pressurizer Spray Nozzle Limitations
(Typical)

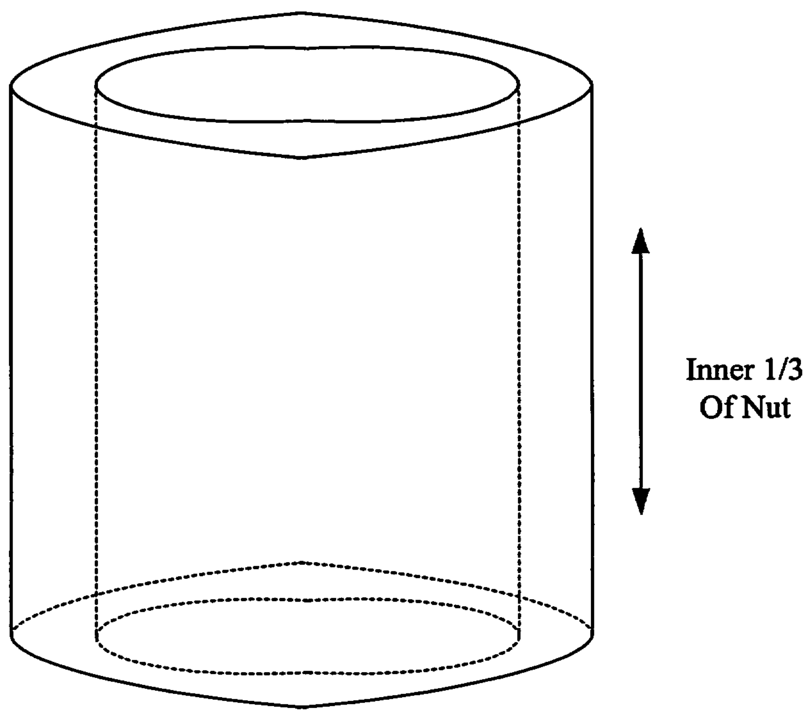


Figure 12
Reactor Pressure Vessel
Closure Nut
(Typical)

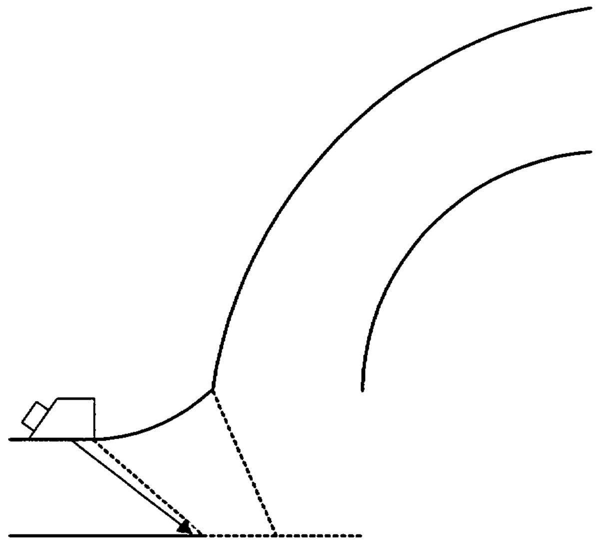


Figure 13
Reactor Coolant Main Loop
Branch Connection
(Typical)

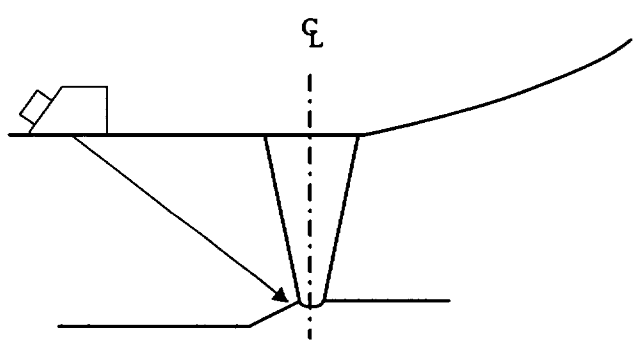


Figure 14
Elbow/Pipe to Branch Connection
Branch Connection to Pipe/Elbow
(Typical)

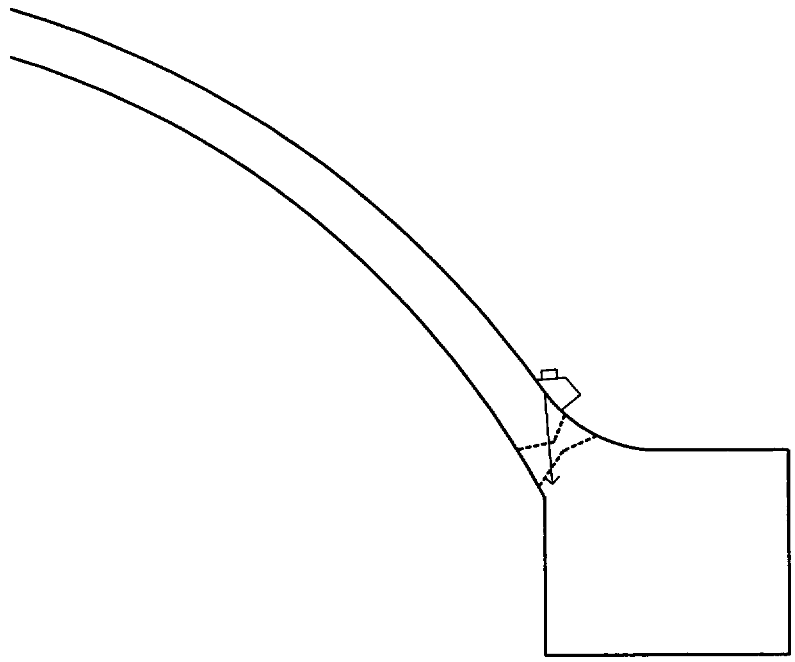
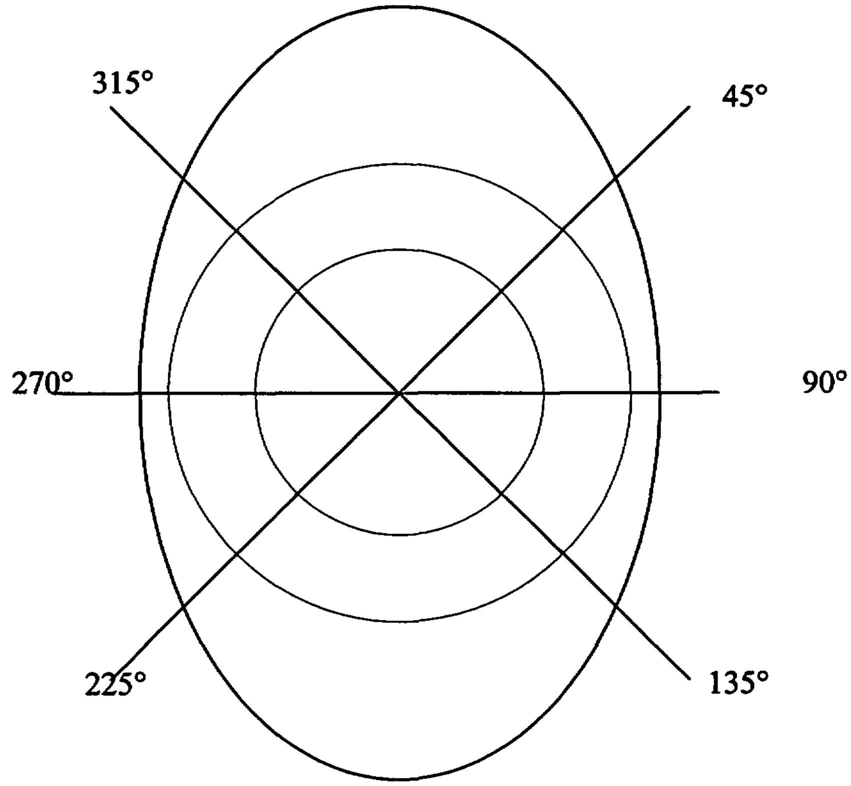


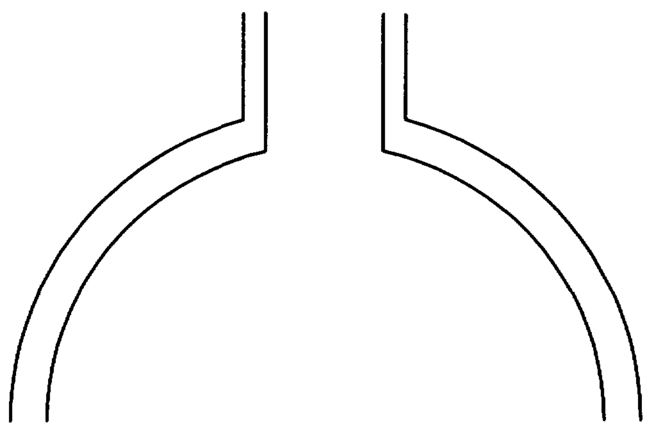
Figure 15
Reactor Pressure Vessel
Head to Flange Weld
(Typical)

Regenerative Heat Exchanger
N-1 and N-4 Nozzle Inside Radius Section
(Typical)



Top View

Examination Limited from 45 to 135 and from 225 to 315 due to nozzle configuration. 50% of total area examined, 100% of accessible area examined.



End View
(Typical)
Figure 16

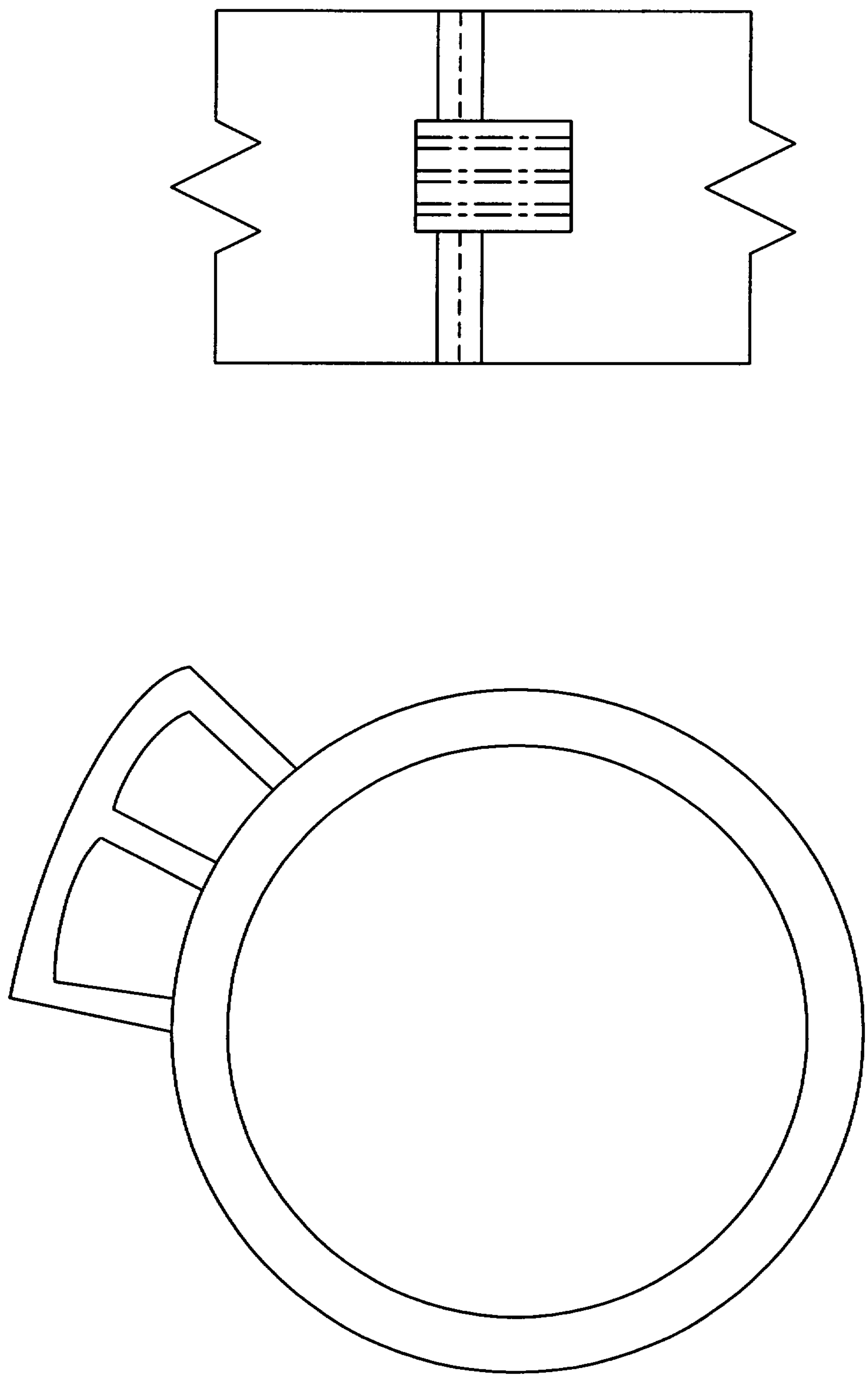


Figure 17
Welded Support Limitations
(Typical)