Wisconsin Electric
POWER COMPANY

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10 CFR 50.55

December 20, 1990

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

Gentlemen:

DOCKET NO. 50-266
INSERVICE INSPECTION PLAN FOR THE THIRD INTERVAL
POINT BEACH NUCLEAR PLANT UNIT 1

On December 1, 1990, Point Beach Nuclear Plant Unit 1 began its third ten-year interval for inservice inspection. Pursuant to 10 CFR 50.55a(g)4(ii), a long term inspection program was developed in accordance with the 1986 edition of ASME Section XI. A summary of the program is provided as Attachment A and an explanation of abbreviation used therein is provided as Attachment C.

Pursuant to 10 CFR 50.55 a(g)5(iii), we have enclosed in Attachment B requests for relief from those inspection requirements which are not possible or impractical for the Point Beach Nuclear Plant. Your prompt review and approval of these relief requests will be appreciated.

If further details concerning this inspection plan are required, please contact Messrs. G. R. Sherwood or J. F. Kohlwey at (414) 755-2321.

Very truly yours,

James Zach for

C. W. Fay
Vice President
Nuclear Power

Attachments

Copy to: NRC Regional Administrator, Region III
NRC Resident Inspector

26012[illegible]

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1/1

Attachment A

ISI PROGRAM SUMMARY

I. INTRODUCTION

The Inservice Inspection Program at Point Beach Nuclear Plant, Unit 1 utilizes Inspection Program B, as defined by Article IWA-2432 of the 1986 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." The components listed in the ISI Long Term Plan were allocated over the third 10-year inspection interval of the 40-year plant lifetime in accordance with the provisions of Articles IWx-2500. The third 10-year interval is scheduled to begin in December, 1990 and continue through November, 2000. This inspection interval is further divided into inspection periods of 3-1/3 years or 40 months each. In keeping with this anticipated schedule, the interval is to be divided as follows:

Third Inspection Interval

December 01, 1990 - November 30, 2000

Period 1	Period 2	Period 3
December 01, 1990 March 31, 1994	April 01, 1994 - July 31, 1997	August 01, 1997 - November 30, 2000

The areas to be examined were selected in accordance with the following:

(1) Title 10 of the Code of Federal Regulations, Part 50, Section 55a, Revised as of January 1, 1990.

(2) Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1974 Edition with Addenda through and including Summer 1975, for determining the extent of examination for Category B-J piping welds.

(3) Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1986 Edition, for determining the remaining ISI requirements.

II. BASIS FOR QUALITY GROUP CLASSIFICATION FOR INSERVICE INSPEC...N

General Design Criterion 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that structures, systems and components important to safety be tested to quality standards commensurate with the importance of the safety functions to be performed. Also, in accordance with IWA-1400 of ASME Section XI of the Boiler and Pressure Vessel Code, the Owner is responsible for the determination of the appropriate Code classes for each component of the power plant, identification of the system boundaries for each class of components subject to inspection, and selection of components exempt from examination requirements.

Title 10 of the Code of Federal Regulations Part 50.55a(c), (d) and (e), provide the criteria to classify systems Code Class 1, 2, and 3. The guidance for quality group classifications of components may be found in Regulatory Guide 1.26 and in Section 3.2.2 (also referred to as Standard Review Plan 3.2.2) of NUREG-0800. These two documents were used extensively for the classification of components for ISI at Point Beach Nuclear Plant (PBNP), Unit 1.

10 CFR 50.55a requires that components of the reactor coolant pressure boundary, as defined in 10 CFR 50.2, be tested to the highest available national standards. This corresponds to the quality standard required for Quality Group A (ISI Class 1) of the NRC system described in Regulatory Guide 1.26, Article B. As such, the reactor coolant pressure boundary at PBNP Unit 1 was defined per 10 CFR 50.2.

Standard Review Plan 3.2.2, "System Quality Group Classification" contains information used by the NRR as a guideline to review applications to operate nuclear power plants. In this Standard Review Plan (SRP), there is a list of fluid systems considered to be important to safety for pressurized water reactor (PWR) plants which are reviewed with regard to quality group classification to meet the requirements of General Design Criterion 1.

Table 1 is a list of these systems and their corresponding PBNP Flow Diagrams. Table 2 is a list of other systems that perform safety related functions and as such have been considered for inclusion in the ISI program.

Regulatory Guide 1.26, "Quality Group Classification and Standards for Water -, Steam -, and Radioactive - Waste - Containing Components of Nuclear Power Plants," describes a quality classification system related to specified national standards that may be used to determine quality standards acceptable to the NRC staff for satisfying General Design Criterion 1 for Class 2 and Class 3 components.

Instrumentation impulse lines beyond the root valves have not been classified for ISI since they are not required for the system to perform its safety function nor upon a single active failure would they prevent the system from performing its safety function when required. Only if an instrument had been identified to be necessary for a system to fulfill its safety function would the piping beyond the root valve have been classified for ISI. The lines are typically 0.065-in. wall thickness and rated to withstand in excess of 5000 psig. Therefore no leakage is expected from this tubing. However, should a leak occur, there would be no danger to health and public safety since the sumps and drains are capable of handling all leakage of this magnitude. In addition, since this tubing is 3/8-in. diameter, one charging pump has the capability to provide adequate reactor water make-up if needed to maintain the appropriate reactor coolant levels.

III. LONG TERM EXAMINATION PLAN

The Long Term Plan was prepared in accordance with ASME Section XI, IWA-1400(c) to describe how Point Beach Nuclear Plant, Unit 1, will meet the requirements of ASME Section XI for the third inspection interval. Further, via the "Component Listing," the Long Term Plan documents which components will be subjected to examination and the proposed schedule for those examinations. Note that the Long Term Plan is not rigid and that it is subject to revisions and updating throughout the course of the inspection interval.

1. Class 1 Systems and Components

Class 1 systems and components have been incorporated into the Class 1 portion of the (Long Term Plan) Component Listing based upon previous inservice examination information with modifications incorporated where necessary to reflect the changes in upgrading from the 1977 Edition of Section XI with Addenda through and including Summer 1979 to the 1986 Edition of Section XI, except for Category B-J piping welds.

Category B-J piping welds have been incorporated into the Class 1 portion of the Component Listing using the 1974 Edition with Addenda through and including Summer 1975 to determine the extent of examination as allowed by 10 CFR 50.55a(b)(2)(ii). This method of selection was utilized since insufficient information is available to properly perform a weld examination selection based on the selection criteria allowed by more recent editions of the Code.

The following Class 1 components (Category B-J only) are considered to be exempt from nondestructive examinations other than pressure tests based on the exemption criteria of the 1974 Edition with Addenda through and including Summer 1975.

Component	Exemption Criteria
All 1-in. and smaller Class 1 Piping	IWB-1220(b)(3)

The following Class 1 components (other than Category B-J) are considered to be exempt from nondestructive examinations other than pressure tests based on the exemption criteria of the 1986 Edition.

Component	Exemption Criteria
Excess Letdown Heat Exchanger	IWB-1220(b)(2)

2. Class 2 Systems and Components

Class 2 systems and components have been identified and accordingly classified for ISI by Wisconsin Electric (WE). Class 2 systems and components have been incorporated into the Class 2 portion of the Component Listing based upon previous inservice examination information with modifications incorporated where necessary to reflect the changes in upgrading from the 1977 Edition of Section XI with Addenda through and including Summer 1979 to the 1986

Edition of Section XI including Categories C-F-1 and C-F-2 (formerly C-F) piping welds.

The examination category C-F has been revised from the second inspection interval to coincide with examination categories C-F-1 and C-F-2 in the 1986 Edition of the Code. This change required the addition of Class 2, high pressure safety injection piping greater than or equal to 2-in. in diameter and less than or equal to 4-in. in diameter. This piping was previously Exempt ISI Class 2.

The following Class 2 components <u>within</u> RHR, ECC, and CHR systems are considered to be exempt from nondestructive examinations other than pressure tests based on the exemption criteria of the 1986 Edition of the Code.

<u>Component</u>	<u>Exemption Criteria</u>
All 4-in. and smaller Class 2 piping (except for high pressure safety injection piping)	IWC-1221(a)
High pressure safety injection piping 1 1/2-in. and smaller	IWC-1221(b)
Piping from RWST to SI, CS, and RHR Pump Suction Shutoff Valves	IWC-1221(e)
Piping from CS shutoff valves to the Ring Header Discharge	IWC-1221(f)
Accumulator Tanks	IWC-1221(e)
Refueling Water Storage Tank	IWC-1221(e)
Boric Acid Tanks	IWC-1221(e)
Spray Additive Tank	IWC-1221(e)
Containment Spray System Components other than piping	IWC-1221(a)
Non-Regenerative Heat Exchanger Nozzles	IWC-1221(c)

The following Class 2 components within systems <u>other than</u> RHR, ECC, and CHR systems are considered to be exempt from nondestructive examinations other than pressure tests based on the exemption criteria of the 1986 Edition.

<u>Component</u>	<u>Exemption Criteria</u>
All 4-in. and smaller Class 2 piping	IWC-1222(a)
Seal Water Heat Exchanger (Tube Side)	IWC-1222(c)

Volume Control Tank	IWC-1222(c)

The following Class 2 components are considered to be exempt from examination requirements of IWC-2500 based on the exemption criteria of the 1986 Edition.

All pipe support members and support components encased in concrete	IWC-1230

3. Class 3 Systems and Components

Class 3 systems and components have been identified and accordingly classified for ISI by Wisconsin Electric. Class 3 systems and components have been incorporated into the Class 3 portion of the Component Listing based upon previous inservice examination information with modifications incorporated where necessary to reflect the changes in upgrading from the 1977 Edition of Section XI with Addenda through and including Summer 1979 to the 1986 Edition of Section XI.

The following Class 3 components are considered to be exempt from nondestructive examinations based on the exemption criteria of the 1986 Edition.

Component	Exemption Criteria
Integrally Welded Attachments of all Class 3 Supports and Restraints to components that are on 4-in. and smaller Piping (Except for Auxiliary Feedwater Systems)	IWD-1220.1
Integrally Welded Attachments to all Class 3 Hangers and Supports that are located in systems or portions of systems not required for reactor RHR, CHR, or ECCS and operates at 275 psig and 200°F or less.	IWD-1220.2

4. Component Supports

Component (NF) supports were scheduled for examination in accordance with ASME Section XI, 1986 Edition, for the third inspection interval.

IV. CODE INTERPRETATIONS

Interpretation(s) of e intent of specific ASME Section XI requirements (1986 Edition of Section XI) have been applied to the following areas of the Point Beach Nuclear Plant, Unit 1.

Regenerative Heat Exchanger

The Regenerative Heat Exchanger consists of three identical vessels interconnected by piping. In accordance with Note 3 of Table IWC-2500-1, Category C-A, these vessels including the interconnected piping are being considered as multiple vessels of similar design for purposes of examination scheduling.

Subsection IWE

The rules and requirements for inservice inspection, repair, and replacement of Class MC pressure retaining components and their integral attachments, and for steel portions of CC pressure retaining components not backed up by concrete and their integral attachments as described in Subsection IWE of the Code have not been imposed upon Commission licensees by the NRC. Therefore, this subsection will not be considered to be a requirement. Refer to Federal Register, Vol. 53, No. 87/Thursday, May 5, 1988, Page 16053 relative to "10 CFR 50 Codes and Standards for Nuclear Power Plants."

Limited Exams

A request for relief will not be required or submitted for Class 1 or Class 2 weld examinations in which 90% or greater coverage as required by the Code is achieved. However, all exam limitations will be documented and reviewed by the ANII.

V. RELIEF REQUESTS

During the course of Preservice and prior Inservice Inspections at Point Beach Nuclear Plant, Unit 1, certain areas were identified where total compliance with ASME Section XI is not possible. Relief Requests h[illegible]een prepared for each of these areas in accordance with 10 CFR 50.5[illegible](5). The relief requests contain specific information to support the [illegible]s for relief from the requirement. Detailed information relating to each Relief Request may be found in Attachment B.

VI. CODE CASES

The guidance of the Code Cases listed in Regulatory Guide 1.147, as well as other Code Cases may be used during the course of examinations performed in the third inspection interval. The following is a summary of those Code Cases and how they will be applied to Point Beach Nuclear Plant, Unit 1 during the third inspection interval:

Case N-460, Approved 07-27-88

The Examination Tables (IWx-2500-1) in the Code frequently use the expression "essentially 100%" when describing the extent of the Class 1 or Class 2 weld length or volume to be examined. 10 CFR 50.55a(g)(5)(iii) states that if a licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit information to support the determinations, i.e., a Relief Request.

PBNP will utilize Code Case N-460 (Approved 07-27-88) which states, when the entire examination volume or area on any Class 1 or Class 2 weld cannot be examined due to interference by another component or part geometry, a reduction in examination coverage may be accepted provided the reduction in coverage for that weld is less than 10%. The applicable examination records shall identify both the cause and the percentage of reduced examination coverage. The implementation of this Code Case means that a request for relief will not be required or submitted for examinations in which 90% or greater coverage is achieved. However, all exam limitations will be documented and reviewed by the ANII.

VII. AUGMENTED EXAMINATION PROGRAMS

Augmented examinations are those examinations that are performed above and beyond the requirements of ASME Section XI. Below is a summary of those examinations performed by Point Beach Nuclear Plant, Unit 1, but are not specifically addressed by Section XI, or the examinations will be performed in addition to the requirements of the Code on a routine basis during the third inspection interval.

A. PROGRAMS SUMMARY

Augmented examinations performed at PBNP, Unit 1 on a continuous or ongoing basis are as follows:

1. IE BULLETINS

   a. IEB 82-02 - "Threaded Fasteners Used in Reactor Coolant System," WE Letter, "Reply to IE Bulletin 82-02," dated August 11, 1982.

3. REGULATORY GUIDES

   a. RG 1.14 - Reactor Coolant Pump Flywheel

   b. RG 1.65 - Reactor Pressure Vessel Closure Studs

   c. RG 1.150 - Reactor Pressure Vessel Beltline Weld Examinations

4. NRC COMMITMENTS

   a. ANSI N14.6 - Heavy Load Lifting Devices - WE Letter to H. R. Denton of NRR, "Special Lifting Devices," Enclosure Item 4, dated September 28, 1983.

      (1) Reactor Vessel Head Lifting Rig
      (2) Reactor Vessel Internals Lifting Rig
      (3) Reactor Coolant Pump Motor Lifting Rig

   b. Main Steam Bypass Line Energy Absorbers - NRC Letter to C. W. Fay, dated April 10, 1987, and Enclosed Safety Evaluation Report, "Energy Absorbers as Replacements for Snubbers on the Unit 1 Main Steam Bypass Line," Section 3.3 and 5.2.

B. PROGRAM IMPLEMENTATION

Those augmented examination programs that are included in the Long Term Plan and their requirements are stated below.

1. IEB 82-02 - "Threaded Fasteners Used in Reactor Coolant System"

   Bolting used in the reactor coolant system, excluding the RPV studs, pressurizer heaters, valve bonnets and pump flange connections installed on lines having a nominal diameter less than

6-in., and control rod drive mechanisms, will receive a Code required VT-1 visual examination and a supplemental surface examination when it is removed. The surface examination is in addition to the Code required VT-1 examination. Closures will not be disassembled and the bolting removed expressly for the purpose of examination unless bolting degradation is suspected.

When bolting is removed for maintenance, and the Code required exam has been completed for the inspection period/interval or does not require an exam of all the bolting, a visual exam of the bolting will be performed by trained maintenance personnel. This examination may not necessarily be conducted in accordance with ASME Section XI.

2. Regulatory Guide 1.14 - "Reactor Coolant Pump Flywheel Integrity"

Regulatory Guide 1.14 requires the following reactor coolant pump flywheel examinations to be performed:

(1) An in-place volumetric examination of the bore and keyway area shall be performed once every three years.

(2) A 100 percent volumetric examination of the entire flywheel shall be performed at 10-year intervals.

Due to the construction of the flywheels, a complete examination of the bore and keyway while the flywheels are in-place, in strict accordance with the regulatory guide, cannot be performed. The flywheel is constructed of two disks that are bolted together. Only the top surface and edge of the flywheel disc are accessible while the flywheel is in place. Only the top disk can be examined volumetrically with this configuration. Therefore, meaningful examination data could only be attained by the removal of the flywheel.

Historically, there has been no evidence of flaws that may be detrimental to the flywheels since they were placed in operation. The reactor coolant pump motor stator and rotor must be inspected every five years in accordance with routine maintenance procedure RMP# 2M. During this time, the flywheels can be removed and made accessible for inspection. ALARA concepts dictate that this is the most advantageous time to perform the most meaningful examinations on the areas of interest. Therefore, the following examinations at a frequency of every five years (to coincide with RMP# 2M) will be performed to meet WE's interpretation of the intent of the Regulatory Guide:

(1) A surface examination of the reactor coolant pump flywheel bore and keyway.

(2) A 100 percent volumetric examination of the flywheel.

(3) A visual examination of the surface of the flywheel.

3. Regulatory Guide 1.65 - "Materials and Inspections for Reactor Vessel Closure Studs"

Regulatory Guide 1.65 requires the examination of pressure vessel stud bolting in accordance with Section XI of the ASME Code supplemented as follows:

(1) A surface examination shall be performed with the studs removed.

(2) Examinations shall be performed on a representative sample and on a reasonable geometric distribution.

The examinations as scheduled in the Long Term Plan will meet the intent of this Regulatory Guide.

4. Regulatory Guide 1.150 - "Reactor Pressure Vessel Beltline Weld Examinations"

Regulatory Guide 1.150 requires improved ultrasonic examination methods of reactor pressure vessel beltline region welds. Point Beach Nuclear Plant has elected to adopt the guidance provided in this regulatory guide to ensure the integrity of the RPV Beltline Region Welds.

5. Heavy Load Lifting Devices

Welds on the following lifting devices will be examined to meet WE's commitment to the Nuclear Regulatory Commission. Point Beach Nuclear Plant has committed to perform examinations on critical welds analyzed in WCAP 10082 in response to ANSI N14.6 and NUREG-0612.

(1) Reactor Vessel Head Lifting Rig

(2) Reactor Vessel Internals Lifting Rig

(3) Reactor Coolant Pump Motor Lifting Rig

These examinations are not considered as ASME, Section XI, but WE will use ASME as a guide for the examinations. A surface exam will be performed on the critical welds listed above. The personnel performing the examinations will be qualified in accordance with the applicable edition and addenda of the Code. The weld examinations will be scheduled such that 100% of the critical welds will be examined during each 10-year interval in concurrence with the ISI Program. Essentially, an equal portion will be performed during each 40-month period.

The acceptance standards of Paragraphs NF-5340 and NF-5350 of ASME, Section III, Division 1, of the corresponding edition and addenda of Section XI in effect will be used.

In addition to the surface examination of the critical welds discussed above, a visual examination of the lifting devices will be performed each outage to check for defects and deformation by experienced, competent maintenance personnel. These visual examinations of the lifting devices are the responsibility of the maintenance department and as such are not within the scope of the Long Term Plan.

5. Main Steam Bypass Line Energy Absorbers

The Main Steam Bypass Line has not been classified for ISI and as such is outside the scope of ASME Section XI. However, Energy Absorbers have been installed on the main steam bypass line and will be examined in accordance with the WE commitment to the NRC.

The Energy Absorbers will receive a VT-3 visual examination on a rotating basis such that each one will be examined within a 10-year interval. Plates that are determined to be cracked will be replaced and the occurrence will be reported to the NRC providing root cause determination, implications to piping integrity, and proposed corrective actions. In addition, scratch plates will be examined to determine if any unanticipated loadings may have occurred to the energy absorber.

TABLE 1

FLUID SYSTEMS IMPORTANT TO SAFETY FOR PWR PLANTS

(Extracted from NUREG-0800, SRP 3.2.2)

SYSTEM	PBNP FLOW DIAGRAM
Reactor Coolant System	541F091
Emergency Core Cooling System	110E017
Containment Spray System	110E017
Chemical and Volume Control System	684J741 684J972 PBM-226
Boron Thermal Regeneration System	Note 1
Boron Recycle System	684J741 Note 1
Residual Heat Removal System	110E018
Component Cooling Water System	110E018
Spent Fuel Pool Cooling & Cleanup System	110E018
Sampling System	541F092
Service Water System	M-207
Compressed Air System	M-209
Emergency Diesel Engine Fuel Oil Storage and Transfer System	M-219 Note 2
Emergency Diesel Engine Cooling Water System	Note 3
Emergency Diesel Engine Starting System	M-219 Note 4
Emergency Diesel Engine Lubrication System	Note 3
Emergency Diesel Engine Combustion Air Intake and Exhaust System	Note 3

TABLE 1
(Continued)

SYSTEM	PBNP FLOW DIAGRAM
Main Steam System	M-201
Feedwater System	M-202
Auxiliary Feedwater System	M-217
Steam Generator Blowdown System	M-201
Containment Cooling System	110E017 M-207
Containment Purge System	M-215
Ventilation Systems for Control Room and Engineered Safety Features Rooms	Note 5
Combustible Gas Control System	M-224
Condensate Storage System	M-217

NOTES FOR
TABLE 1

NOTE 1: The Boron Recycle System at PBNP, Unit 1, are used for Reactivity Control only and do not perform a safety function. However, since portions of this system are not isolated from the Chemical and Volume Control System, they have been included in the ISI Program. PBNP design does not employ a Boron Thermal Regeneration system, however, the boron temperature is maintained by the recirculation process.

NOTE 2: The Emergency Diesel Engine Fuel Oil Storage and Transfer System has been included in the ISI Program.

NOTE 3: The Emergency Diesel Engine Cooling Water, Lubrication, Air Intake and Exhaust Systems are not included in the ISI Program because adequate periodic testing and maintenance is being performed in accordance with Technical Specification 15.4.6, EMD Maintenance Instruction MI-1742, routine maintenance procedure RMP# 43, and Bi-Weekly periodic test procedures TS-1 and TS-2.

NOTE 4: The Emergency Diesel Air Starting System has been included in the ISI Program.

NOTE 5: The Control Room Ventilation System has not been included in the ISI Program. Technical Specification 15.3.12 states that if the system was found to be inoperable, there would be no immediate threat to the control room and operation could continue for a limited period of time. In addition, this system is adequately tested in accordance with Technical Specification 15.4.11 such that in the event of an accident, the resulting control room doses would be less than the allowable levels specified in Criterion 19 of Appendix A to 10 CFR 50. "Periodic Checks" are performed on other Ventilation Systems for Engineered Safety Features Rooms such that the systems are tested to quality standards commensurate with the safety function to be performed. (Reference Periodic Test Procedure, PC-1). Therefore, these Ventilation Systems have not been included in the ISI Program.

TABLE 2

OTHER SYSTEMS IMPORTANT TO SA[illegible]OR PWR PLANTS

SYSTEM	PBNP FLOW DIAGRAM
Fire Protection System	M-208 Note 1
Liquid Radwaste	Note 2
Gaseous Radwaste	684J972 PBM-229 Note 3
Containment Penetration Piping	684J971 M-214 M-215

NOTES FOR
TABLE 2

NOTE 1: The Fire Protection System has not been included in the ISI Program because this system is already tested in accordance with Technical Specification 15.4.15. The testing required by the Technical Specification exceeds the requirements that would be imposed by ASME Section XI and the additional examination would not contribute to ensuring the systems' operability or readiness to perform its safety function.

NOTE 2: The Liquid Radwaste System has not been included in the ISI Program. According to PBNP FSAR Section 14.2.2, calculations have been performed showing that upon postulated failure of the Liquid Radwaste System, the contents would remain in the auxiliary building and any subsequent discharge of radioactive liquid to the environment would be under administrative controls and would not exceed the limits specified in the plant Technical Specifications. Therefore, this system need not be classified for ISI.

NOTE 3: The Gaseous Radwaste System has been included in the ISI Program. According to PBNP FSAR Section 14.2.3, calculations have been performed showing the off-site doses following a "worst case" postulated failure of the Gaseous Radwaste System would be considerably below the guidelines limit specified in 10 CFR 100 and there would be no undue hazard to public health and safety. However, the calculated dose was 0.8 rem which exceeds the limit of 0.5 rem specified by Regulatory Guide 1.26.

<u>Attachment B</u>

RELIEF REQUESTS

PR-1-01

COMPONENT

Reactor Pressure Vessel Interior Surfaces

EXAM AREA

1. RPV-INTERIOR - Vessel Interior Surfaces

ISOMETRIC or COMPONENT DRAWING

None

ASME SECTION XI CATEGORY

B-N-1

ASME SECTION XI ITEM NUMBER

B13.10

ASME SECTION XI EXAMINATION REQUIREMENT

A visual examination (VT-3) is required every 3 years of the accessible areas of the vessel interior surfaces during a normal refueling outage.

ALTERNATIVE EXAMINATION

A visual examination (VT-3) of the reactor vessel interior will be performed when the core barrel is removed but not at a frequency greater than that specified in the Code (i.e., Once each inspection period). The core barrel will not be removed specifically for this examination.

REASON FOR LIMITATION

Only a small portion of the reactor vessel interior surfaces are accessible with the core barrel in place. There is approximately 10" of the vessel interior surface accessible from the top of the core barrel flange to the reactor vessel flange during a normal refueling outage. A meaningful examination cannot be performed unless the core barrel is removed. Removal of the core barrel requires a complete defueling of the reactor and significant ALARA impacts including exposure and contamination problems. This process would also significantly increase the risk of damage and loose parts.

RR-1-02

COMPONENT

Reactor Pressure Vessel Safety Injection Safe-End to Nozzle Welds (2 nozzles)

EXAM AREA

1. RC-04-SI-1001-33 - Safety Injection Safe-End to Nozzle at 288.5 Deg.

2. RC-04-SI-1001-19 - Safety Injection Safe-End to Nozzle at 108.5 Deg.

ISOMETRIC or COMPONENT DRAWING

Figure 1 - ISI-PRI-1127

Figure 2 - ISI-PRI-1129

ASME SECTION XI CATEGORY

B-F

ASME SECTION XI ITEM NUMBER

B5.10

ASME SECTION XI EXAMINATION REQUIREMENT

A surface and volumetric examination of each Safe-End to Nozzle weld every 10 years.

ALTERNATIVE EXAMINATION

No surface exam will be performed. A volumetric examination of each Safe-End to Nozzle weld every 10 years. The volumetric examination will be performed from the ID of the nozzle using mechanized equipment.

REASON FOR LIMITATION

A surface examination is not possible due to the inaccessibility of this area. These welds are located between the vessel and biological shield wall.

Figure 1

ISI-PRI-1127

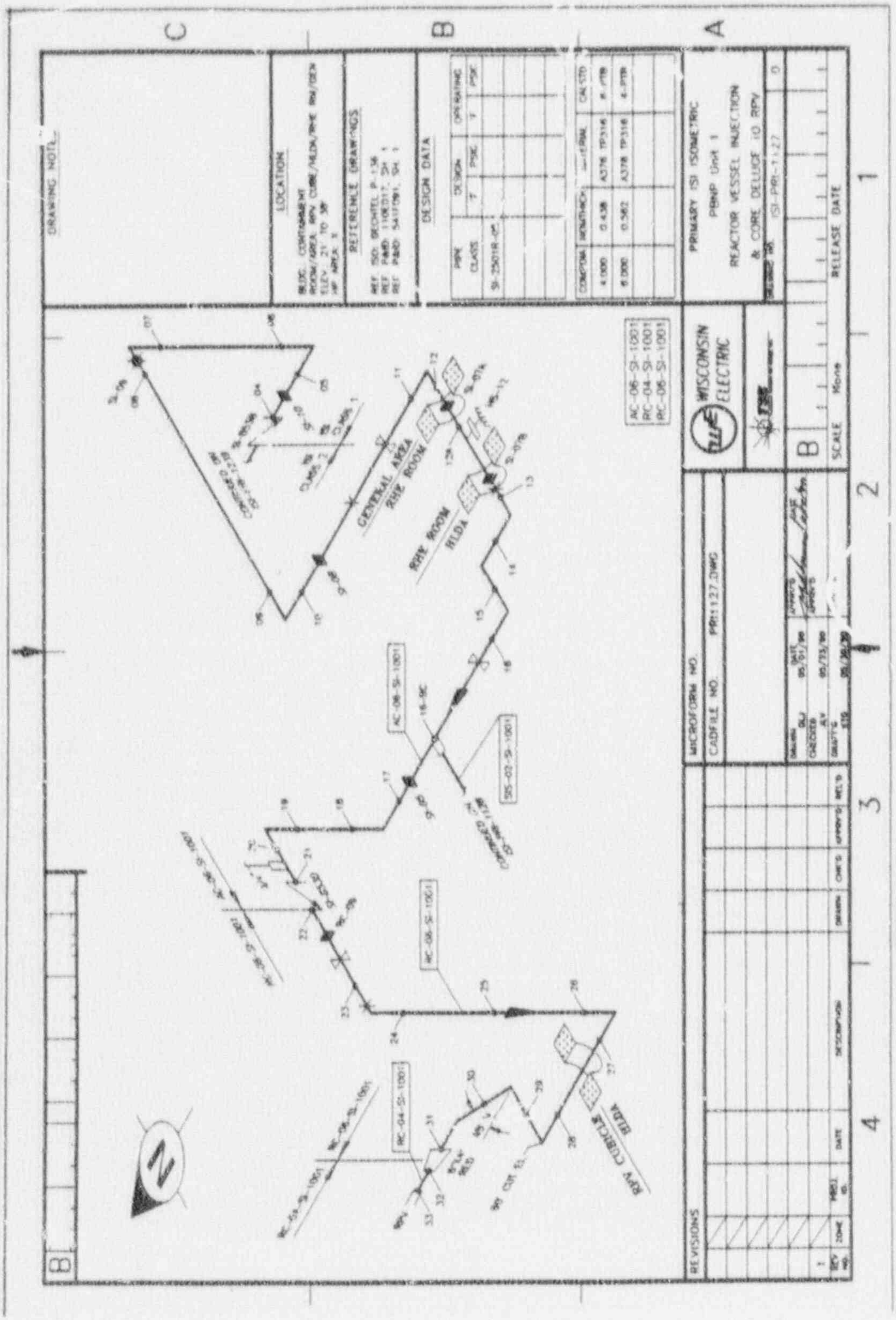
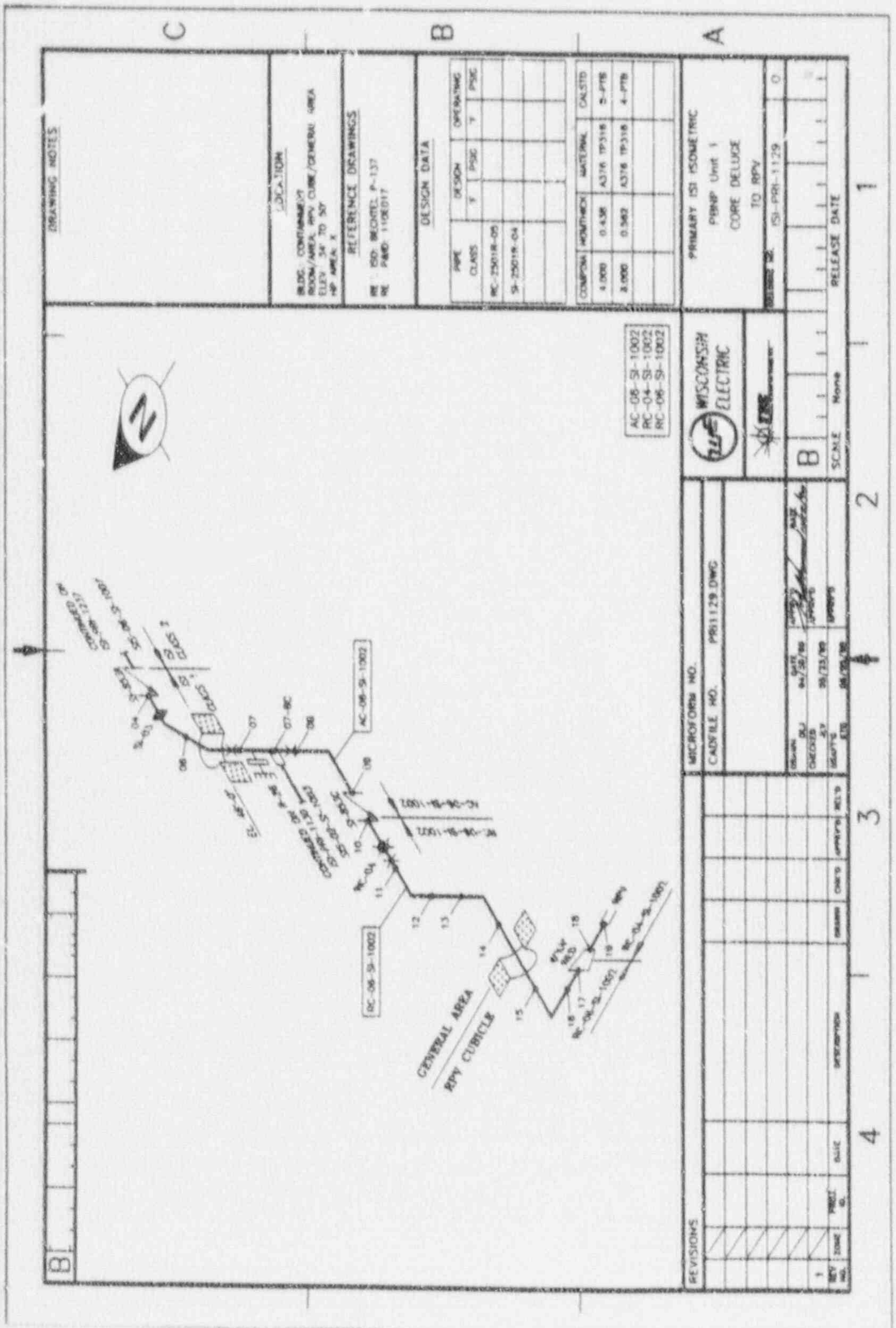


Figure 2

ISI-PRI-1129



RR-1-03

REQUEST FOR RELIEF WITHDRAWN

(EXAM NOT APPLICABLE TO THIRD INSPECTION INTERVAL)

RR-1-04

REQUEST FOR RELIEF WITHDRAWN

(EXAM NOT APPLICABLE TO THIRD INSPECTION INTERVAL)

<u>RR-1-05</u>

REQUEST FOR RELIEF WITHDRAWN

(RELIEF NOT REQUIRED FOR THIRD INSPECTION INTERVAL)

RR-1-06

REQUEST FOR RELIEF WITHDRAWN

(EXAM NOT APPLICABLE TO THIRD INSPECTION INTERVAL)

RR-1-07

REQUEST FOR RELIEF WITHDRAWN

(RELIEF NOT REQUIRED FOR THIRD INSPECTION INTERVAL)

RR-1-08

COMPONENT

Reactor Coolant System Safety Injection Piping Welds

EXAM AREA

1. RC-06-SI-1001-29 - Elbow to Pipe
2. RC-06-SI-1001-30 - Pipe to Elbow
3. RC-06-SI-1002-16 - Elbow to Pipe

ISOMETRIC or COMPONENT DRAWING

Figure 1 - ISI-PRI-1127

Figure 2 - ISI-PRI-1129

ASME SECTION XI CATEGORY

B-J

ASME SECTION XI ITEM NUMBER

B9.11

ASME SECTION XI EXAMINATION REQUIREMENT

A surface and volumetric exam of 25% of the circumferential joints each inspection interval (10 years) to accomplish 100% of the circumferential joints over the service life of the plant.

ALTERNATIVE EXAMINATION

No alternate examination of the above examination areas will be performed. These areas are not accessible and no examination is possible.

REASON FOR LIMITATION

A surface and volumetric exam from the exterior of the piping is not possible due to the inaccessibility of this area. These welds are located between the reactor pressure vessel and biological shield wall. A volumetric examination from the interior of the piping by mechanized equipment is also not possible due to the pipe size and configuration.

Figure 1

ISI-PRI-1127

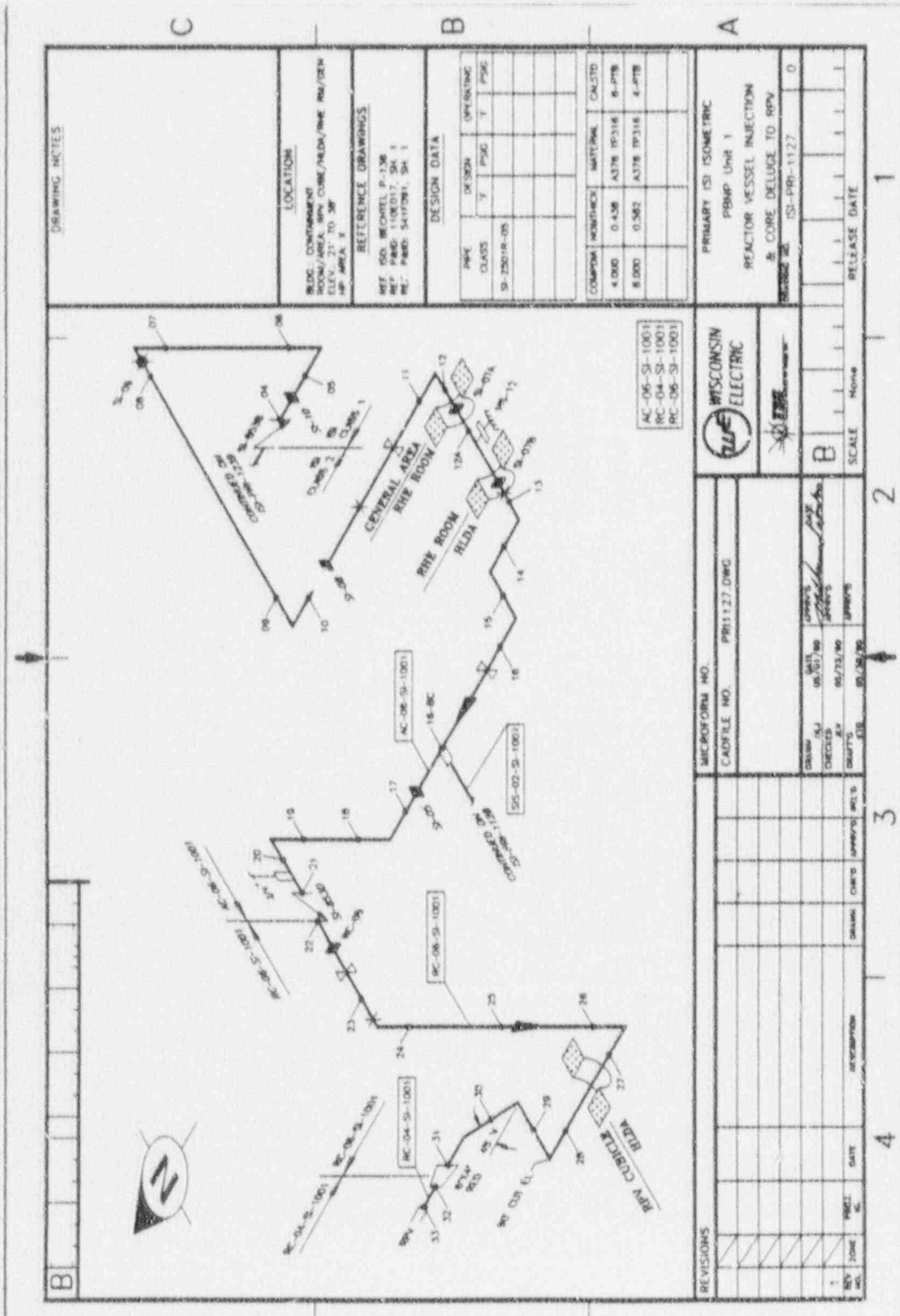
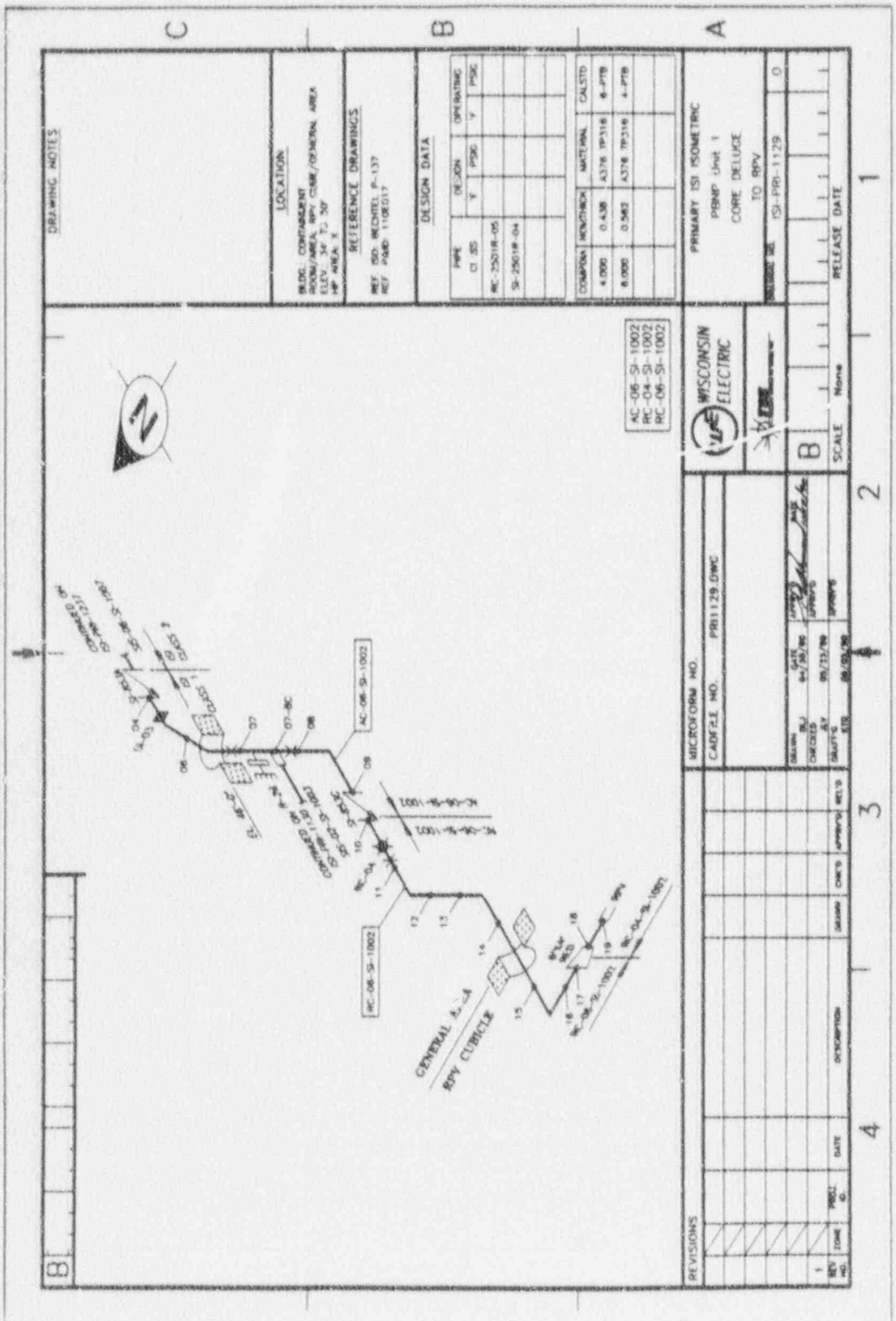


Figure 2

ISI-PRI-1129



RR-1-09

REQUEST FOR RELIEF WITHDRAWN

(RELIEF COMBINED WITH RR-1-08)

RR-1-10

COMPONENT

Residual Heat Removal Heat Exchangers - Primary Side Nozzle-to-Shell Welds

EXAM AREA

1. RHR-A-N1 - Nozzle to Shell
2. RHR-A-N2 - Shell to Nozzle
3. RHR-B-N1 - Nozzle to Shell
4. RHR-B-N2 - Shell to Nozzle

ISOMETRIC or COMPONENT DRAWING

Figure 1 - RHR Heat Exchanger Nozzle Area

Figure 2 - ISI-PRI-1204

ASME SECTION XI CATEGORY

C-B

ASME SECTION XI ITEM NUMBER

C2.11

ASME SECTION XI EXAMINATION REQUIREMENT

A surface examination of each nozzle-to-shell weld every 10 years. The required examinations may be limited to one vessel or distributed among the vessels of similar design, size and service.

ALTERNATIVE EXAMINATION

The shells of these heat exchangers are 0.500-in. thick. The heat exchangers would have to be disassembled in order to perform the Code required examinations since the required exam area is concealed by the nozzle reinforcing plate. Neither of these heat exchangers will be disassembled for the sole purpose of examination. However, if a heat exchanger is disassembled, a surface examination will be performed to the extent practical and a VT-1 visual examination of the inside surface of the shell to nozzle weld area. In addition, a VT-2 visual examination for leakage of both heat exchangers will be conducted each inspection period of the areas during system leakage tests and hydrostatic pressure tests in accordance with IWA-5000 and Table IWC-2500-1.

REASON FOR LIMITATION

The surface examination from the exterior is not possible due to the reinforcing plate configuration and inaccessibility to the vessel interior. The reinforcing plate is welded to the nozzle and shell and completely covers the nozzle-to-shell weld. Figure 1 shows the details of the nozzle-to-shell weld and welded reinforcing plate.

The welded reinforcing plate, by its size and space next to the nozzle outside diameter, prevents adequate UT coverage of the nozzle-to-shell weld such that a UT examination would yield meaningless results. An ultrasonic signal transmitted at the plate surface would simply be reflected by the plate's back wall. The diameter of the reinforcing plate is such that an ultrasonic wave propagated from the nearest shell or nozzle surface would not provide adequate coverage of the nozzle-to-shell weld. Based upon these configurations, the examination of the nozzle-to-shell weld by ultrasonics is impractical.

A more meaningful examination of the nozzle-to-shell weld would be to perform an examination from the inside surface of nozzle weld area. This area is only accessible with the disassembly of the heat exchanger. The disassembly of a heat exchanger will require approximately 40 man-hours of effort in a general area radiation field of 50-100 mR per hour. The actual examination of the nozzle-to-shell weld from the inside surface of the RHR heat exchanger will require approximately 1 man-hour of effort in a radiation field of 35 Rem per hour for the "A" RHR heat exchanger and 12 REM per hour for the "B" RHR heat exchanger. Therefore, the alternate surface and VT-1 visual examinations will be performed to the extent practical only if the heat exchanger is disassembled.

Figure 1

RHR Heat Exchanger Nozzle Area Detail

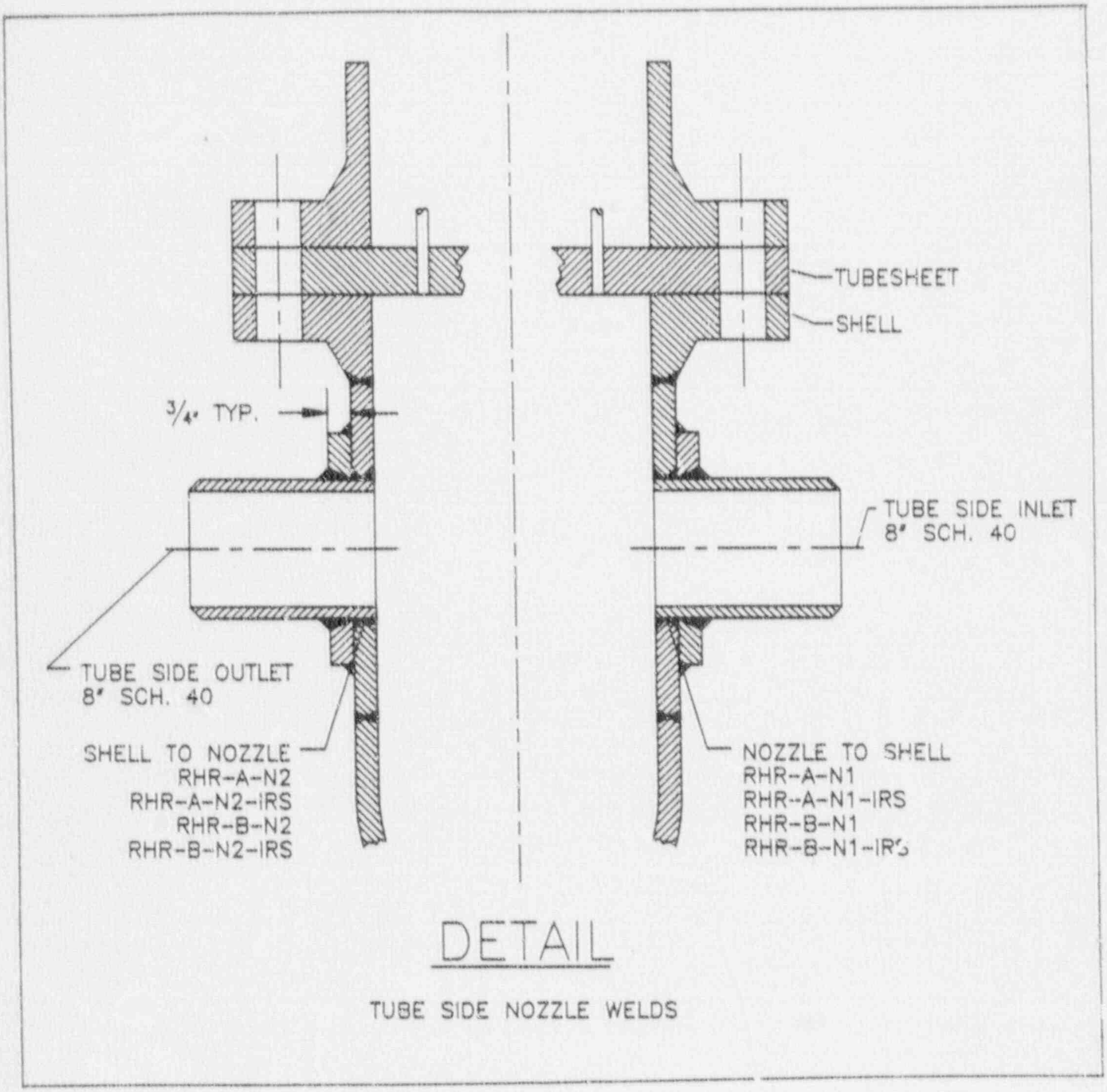


Figure 2

ISI-PRI-1204

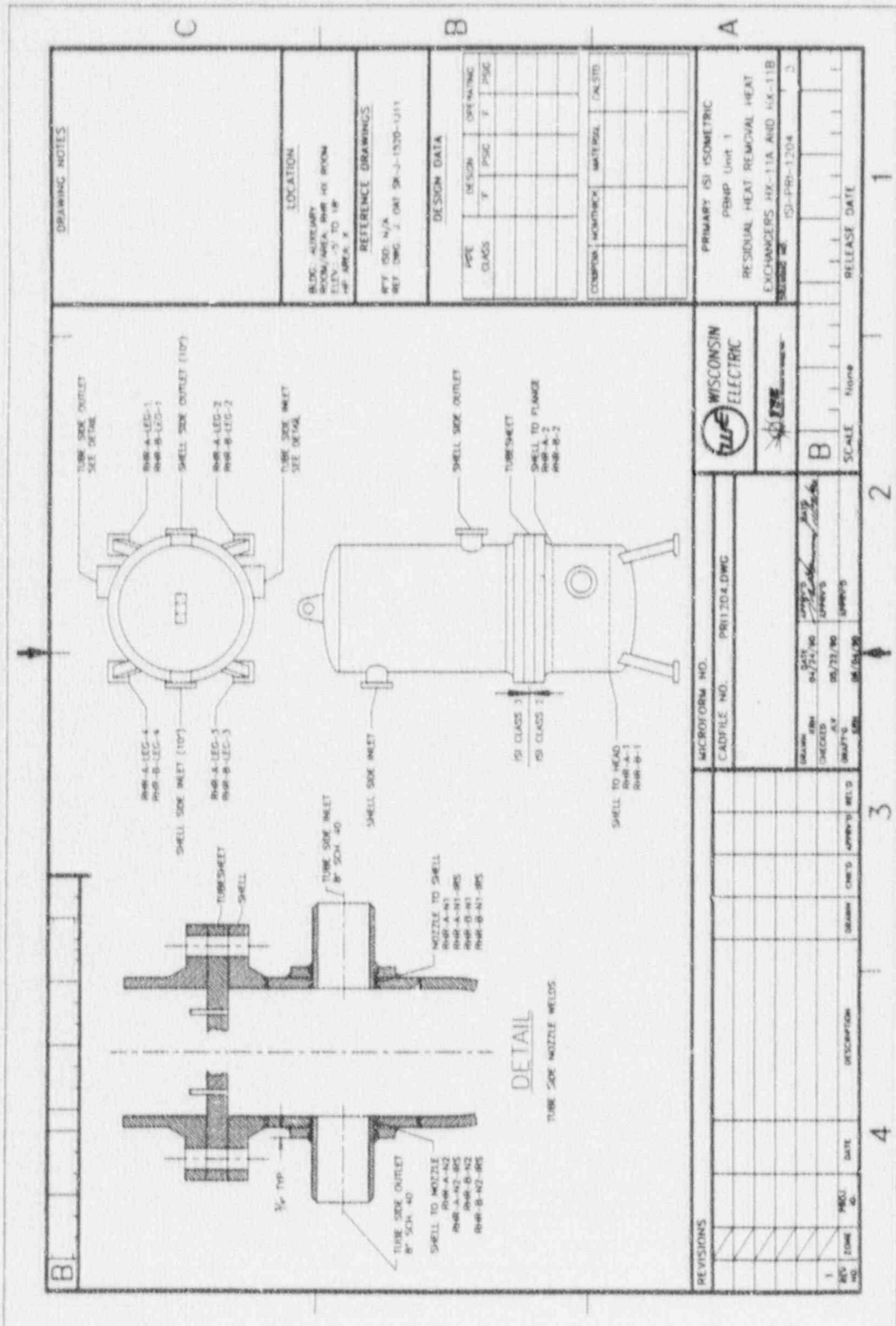
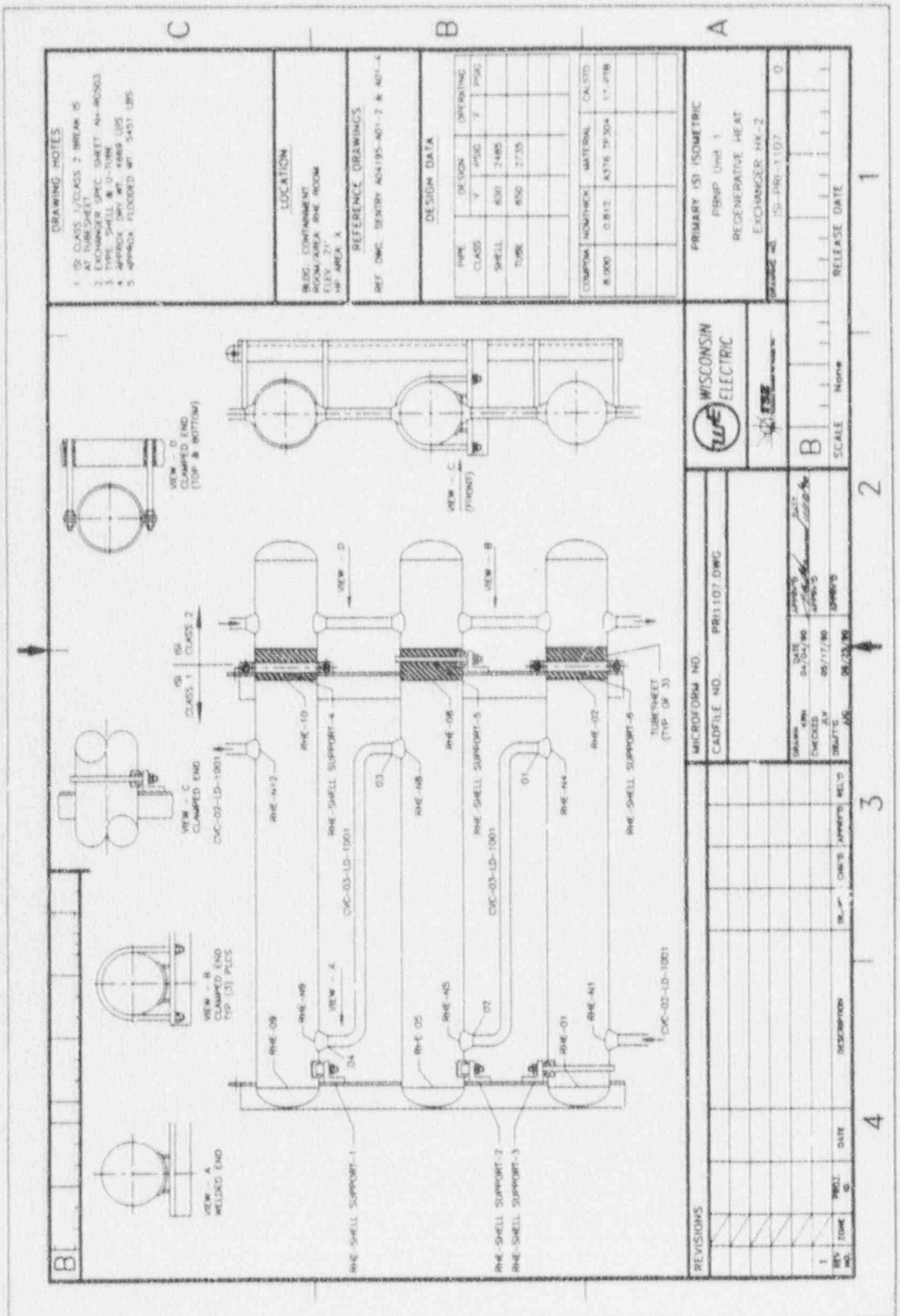


Figure 1

ISI-PRI-1107



COMPONENT

Reactor Coolant Pump Casing Welds

EXAM AREA

1. RCP-A-WELD-A - Pump Casing Weld
2. RCP-A-WELD-B - Pump Casing Weld
3. RCP-A-WELD-C - Pump Casing Weld
4. RCP-B-WELD-A - Pump Casing Weld
5. RCP-B-WELD-B - Pump Casing Weld
6. RCP-B-WELD-C - Pump Casing Weld

ISOMETRIC or COMPONENT DRAWING

Figure 1 - ISI-PRI-1109

ASME SECTION XI CATEGORY

B-L-1
B-L-2

ASME SECTION XI ITEM NUMBER

B12.10
B12.20

ASME SECTION XI EXAMINATION REQUIREMENT

B-L-1: "The examinations performed during each inspection interval shall include 100% of the pressure-retaining welds in at least one pump in each group of pumps performing similar functions in the system (e.g. recirculating coolant pumps)." The exam method shall be volumetric.

B-L-2: "One pump in each group of pumps performing similar functions in the system shall be examined during each inspection interval. This examination may be performed on the same pump selected for the category B-L-1 examination." The exam method shall be a visual examination of the pump internal surfaces.

ALTERNATIVE EXAMINATION

Each reactor coolant pump casing consists of three circumferentially oriented welds (see Figure 1). PBNP proposes to perform an exterior surface examination on one of these three welds, on one pump, once each inspection interval. The examination will include the accessible pump casing weld surface. A 100% exterior surface VT-1 visual examination will be conducted during the same outage.

In addition, PBNP proposes to perform a VT-2 visual examination of the reactor coolant pump casing during system leakage tests and hydrostatic pressure tests in accordance with IWA-5000 and Table IWB-2500-1.

If maintenance or operational problems are encountered which necessitate disassembly of the pump casing internals of either pump, a VT-3 visual examination of the interior pump casing surface or, if available, a MINAC examination of the casing welds will be performed prior to reassembly of the pump.

R ASON FOR LIMITATION

The two reactor coolant pumps (RCP) for PBNP Unit 1 are Westinghouse Model 93 pumps. Each pump casing is fabricated by welding four stainless steel (SA 351 CF8) castings together. Thus, there are three circumferential pressure retaining welds that are to be volumetrically examined in accordance with Category B-L-1. Because the physical properties of the stainless steel castings and weld material prevent meaningful ultrasonic examination, the casing welds must be examined using the miniature linear accelerator (MINAC).

This radiographic examination is performed by placing the MINAC inside the pump casing and placing the film on the outside of the pump. To perform the examination, the pump must be completely disassembled. Disassembly to this extent is far beyond any disassembly expected except for this examination. Also, insulation on the casing exterior must be removed for the placement of film. Additionally, the pump bowl must be dry for the installation of the MINAC. Therefore, all fuel assemblies must be removed from the reactor vessel and the vessel water level lowered to below the nozzles. Complete disassembly of the pump is also required to conduct the VT-1 visual examination in accordance with Category B-L-2.

This radiographic examination using the MINAC was performed on PBNP Unit 1 "B" RCP during the Fall 1981 refueling outage. In addition, the same examination has been performed at several other sites. No problems have been found with the welds at any site. Additionally, no problems have been found during the Category B-L-2 visual examination. This visual examination was conducted at PBNP by using the video camera on the MINAC.

We believe that performing a volumetric examination of the PBNP Unit 1 reactor coolant pump casing welds and a visual examination of the interior pressure retaining surface of one pump during the third inspection interval does not provide an increase in safety commensurate with the associated cost potential for inadvertent pump damage, and expected radiation exposure. The following items have been considered:

Radiation Levels

Currently the average dose rates at the RCP are:

8' elevation general area	≤ 1 to 25 mR/Hr
Below the RCP	10 to 800 mR/Hr
Inside the RCP	700 to 10,000 mR/Hr

Total Estimated Exposure

The whole body doses received during the Fall 1981 examination of Unit 1, RCP "B" are listed below. The list does not include the additional dose received while getting the plant to a condition where RCP disassembly could be performed (e.g., complete core unload).

PBNP maintenance personnel - disassembly	5,237 mR
Contractor personnel - diffuser adapter removal	3,890 mR
Contractor examination personnel	12,626 mR
Contractor personnel - insulation removal/replacement	4,490 mR
Contractor personnel - diffuser adapter replacement	1,833 mR
PBNP maintenance personnel - reassembly	6,017 mR
Total	34,093 mR

The cost estimate for this amount of exposure based on $5,000 per Man-Rem is $170,500.

Pump Disassembly

The Category B-L-1 and B-L-2 examinations require complete disassembly of the pump. The pump manufacturer (Westinghouse) does not require or recommend pump disassembly to perform normal maintenance or inspections. The only time disassembly to this degree has ever occurred was to perform this examination during the Fall 1981. Therefore, very limited experience in this area may result in significant damage or degradation to the pump. Additionally, complete pump disassembly is not anticipated for any other reason in the foreseeable future.

Pump Performance

The type of material used in these pumps is widely used throughout the industry and has performed very well. There have been no reported problems or failures with the casing welds of these model pumps. Additionally, the licensee has had no operational problems with the RCPs which could indicate potential degradation of the casing welds.

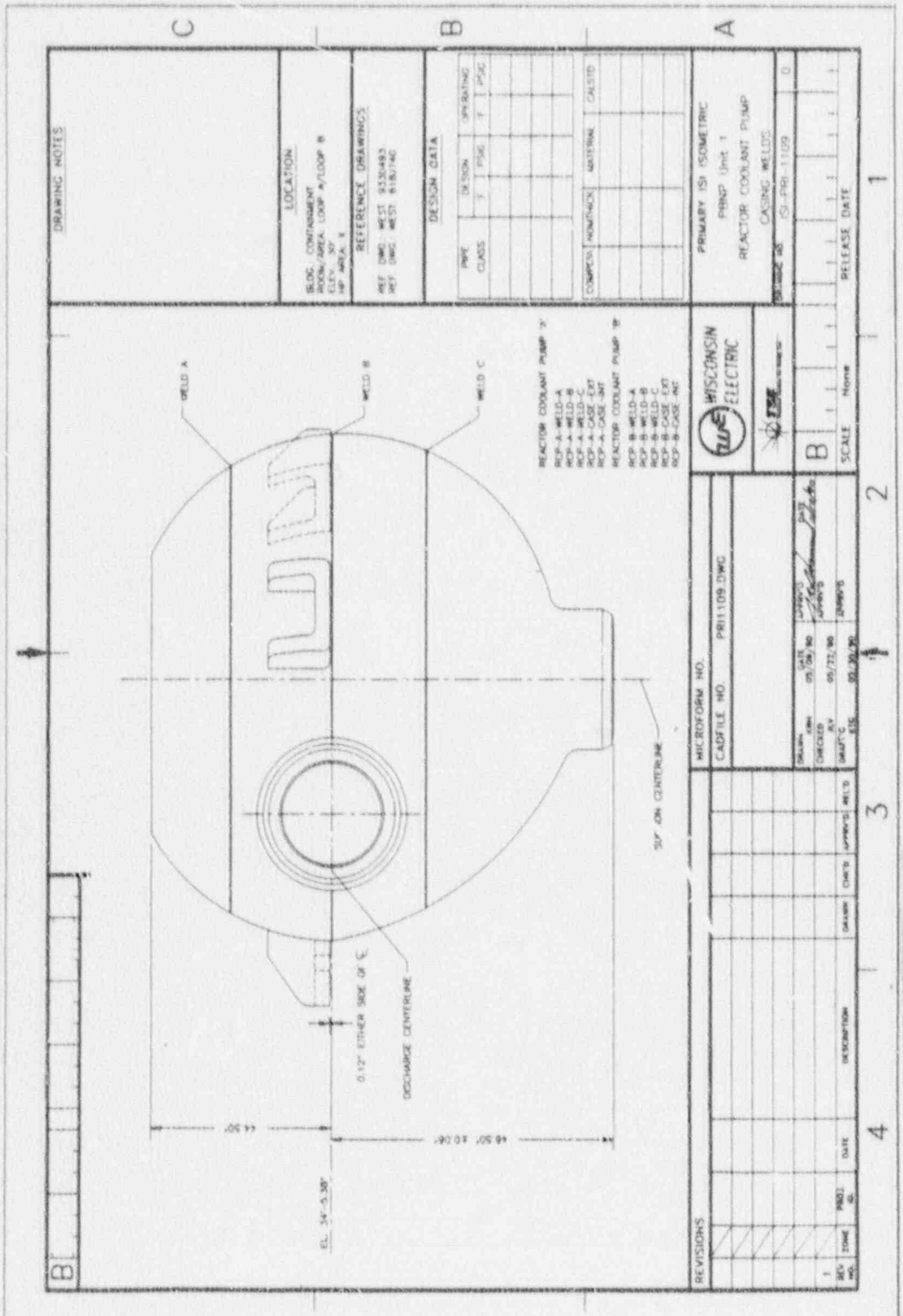
Vibration monitors are currently installed on the motor frame near the lower radial bearing and on the motor shaft above the pump casing. These monitors will alarm on panel C04 in the control room if either detects high vibration. The PBNP Operating Procedures Manual provides RCP vibration limits. If the limits are exceeded, we would expect to shut down and determine the cause. These monitors would most likely detect any problem which could lead to pump casing, weld, or rotating element failure.

Excessive Cost

The estimated cost to disassemble/reassemble the reactor coolant pump, remove and reinstall insulation and to perform the MINAC examination in 1989 was approximately $810,000. Additionally, this examination is expected to extend a refueling outage 5-7 days. The replacement power costs for this amount of time and based on current power costs of $187,000 per day would be approximately $935,000. Any minor problems which might occur could significantly increase the cost of the examination.

Figure 1

<u>ISI-PRI-1109</u>



COMPONENT

Containment Sump Valves

EXAM AREA

1. SI-850A-WLD - Valve Body Welds

2. SI-850B-WLD - Valve Body Welds

ISOMETRIC or COMPONENT DRAWING

Figure 1 - Stearns-Roger Drawing 8551/4, Sh. 1

Figure 2 - Stearns-Roger Drawing 8551/4, Sh. 2

Figure 3 - ISI-PRI-1222

Figure 4 - ISI-PRI-1223

ASME SECTION XI CATEGORY

C-G

ASME SECTION XI ITEM NUMBER

C6.20

ASME SECTION XI EXAMINATION REQUIREMENT

Perform a surface examination during each inspection interval of the valve body welds of at least one valve within each group of valves that are of the same size, design, function and service in a system. The examination may be performed from the inside or the outside surface of the valve.

ALTERNATIVE EXAMINATION

The valve body consists of two 10-in. diameter straight sections of seam welded Schedule 40 Type 304 stainless steel pipe and one elbow welded together circumferentially. The lower portion of the valve includes a longitudinal seam weld with a circumferential weld at each end. These welds will receive a surface examination from the exterior in accordance with the Code.

The upper portion of the valve has a longitudinal seam weld (approximately 11 feet in length) that, except for approximately 4 inches, is inaccessible from the exterior since it is embedded in concrete. If the valve is disassembled for maintenance, this weld will be examined from the interior to the extent practical using a remote VT-1 visual examination method.

In addition, these valves are subjected to periodic system and hydrostatic pressure tests in accordance with IWA-5000 and IWC-2500-1.

REASON FOR LIMITATION

The upper portion that is embedded in concrete can not be meaningfully examined from the inside using a surface examination method due to the length and the diameter (10-in.) of the valve. The valve will have to be disassembled and the plug and stem removed in order to complete the examination. The disassembly of these valves for the sole purpose of examination would result in unnecessary exposure to radiation and contamination and is contrary to ALARA guidelines. The examination would do little to add to the assurance of the structural integrity of these valves since they are subjected to system and hydrostatic pressure tests on a regular basis in accordance with IWA-5000 and IWC-2500-1. The valve, therefore, should not be disassembled for the sole purpose of this examination.

Figure 1

Stearns-Roger
Drawing Number 88551/4, Sh. 1

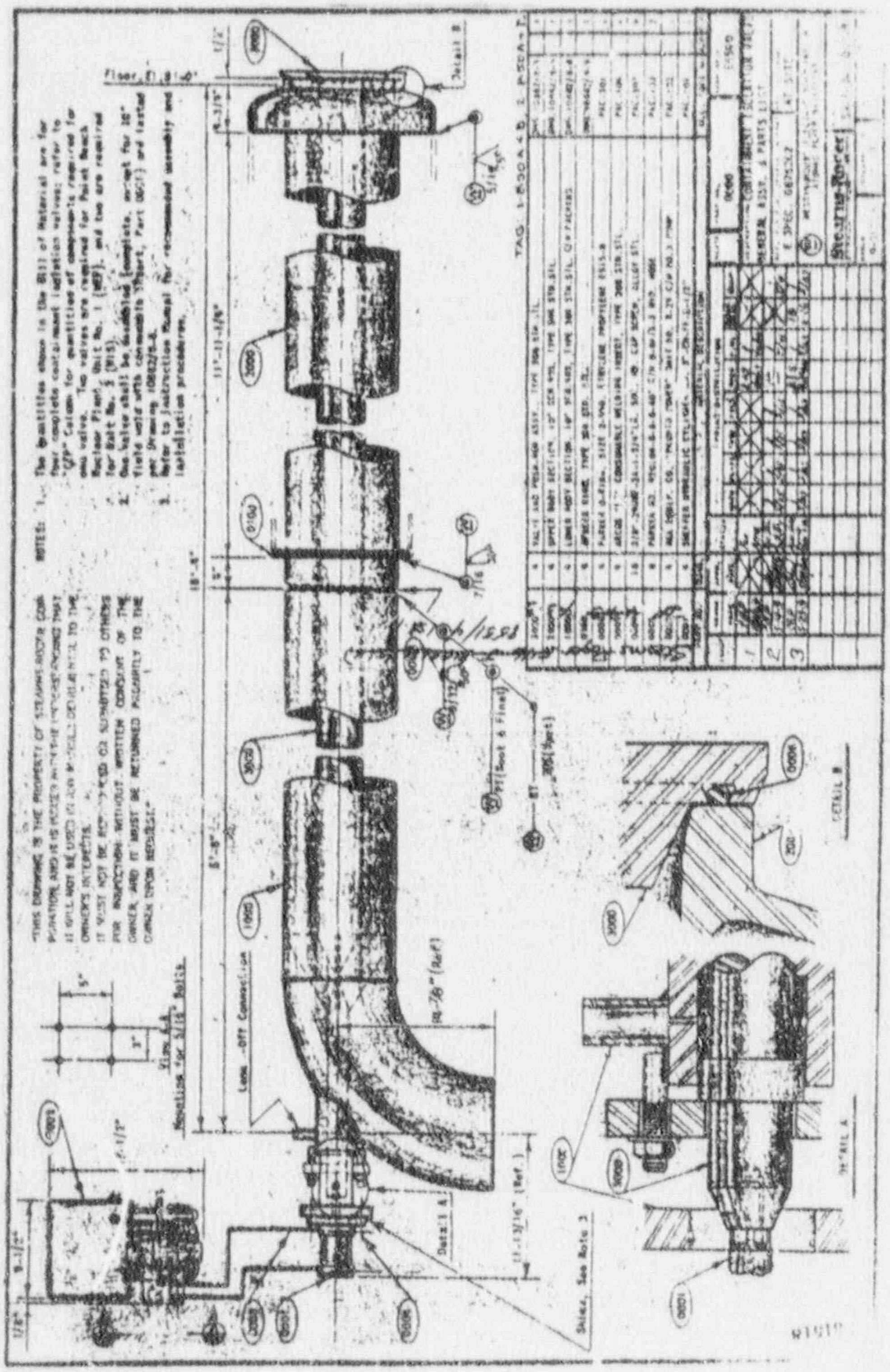


Figure 2

<u>Stearns-Roger</u>
<u>Drawing Number 88551/4, Sh. 2</u>

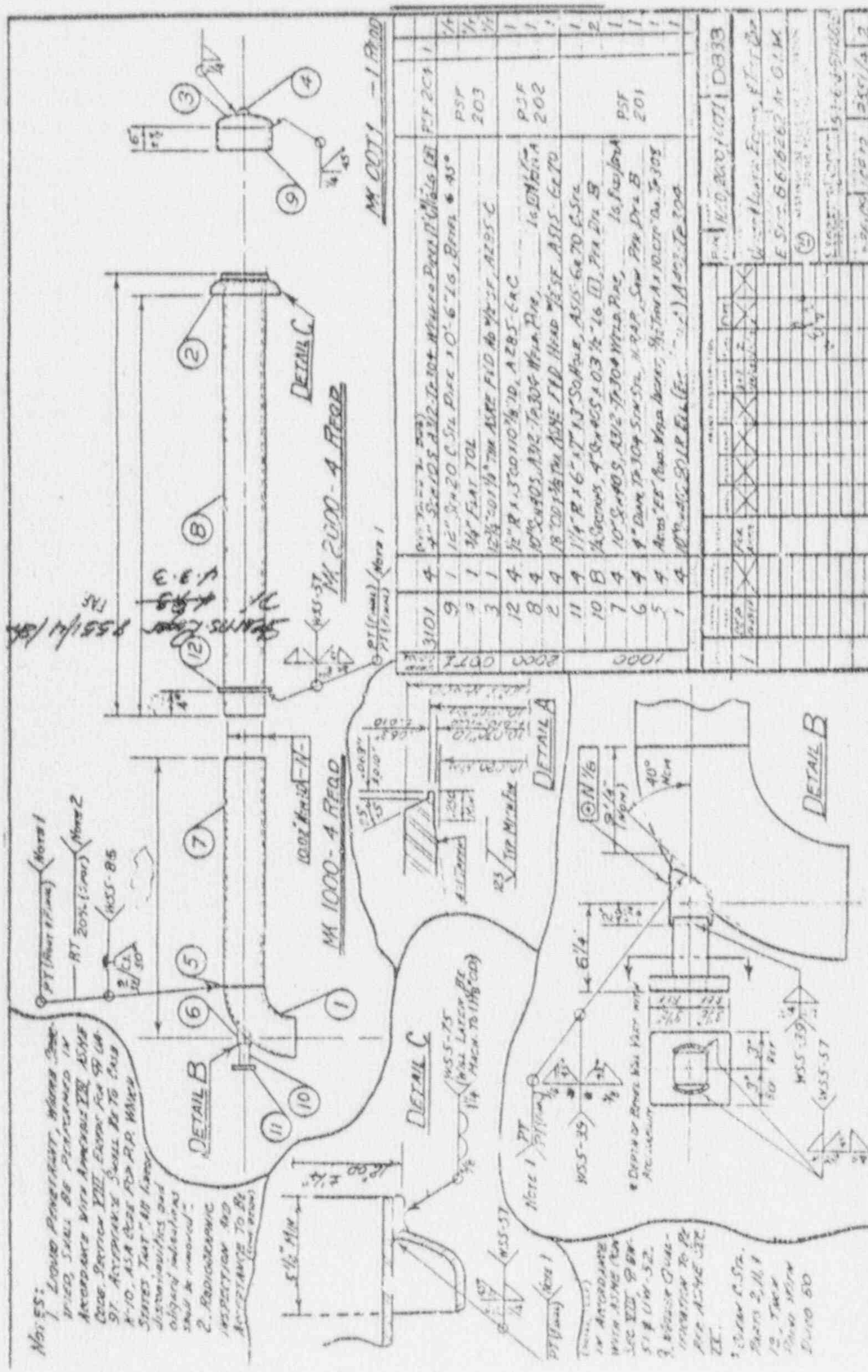


Figure 3

ISI-PRI-1222

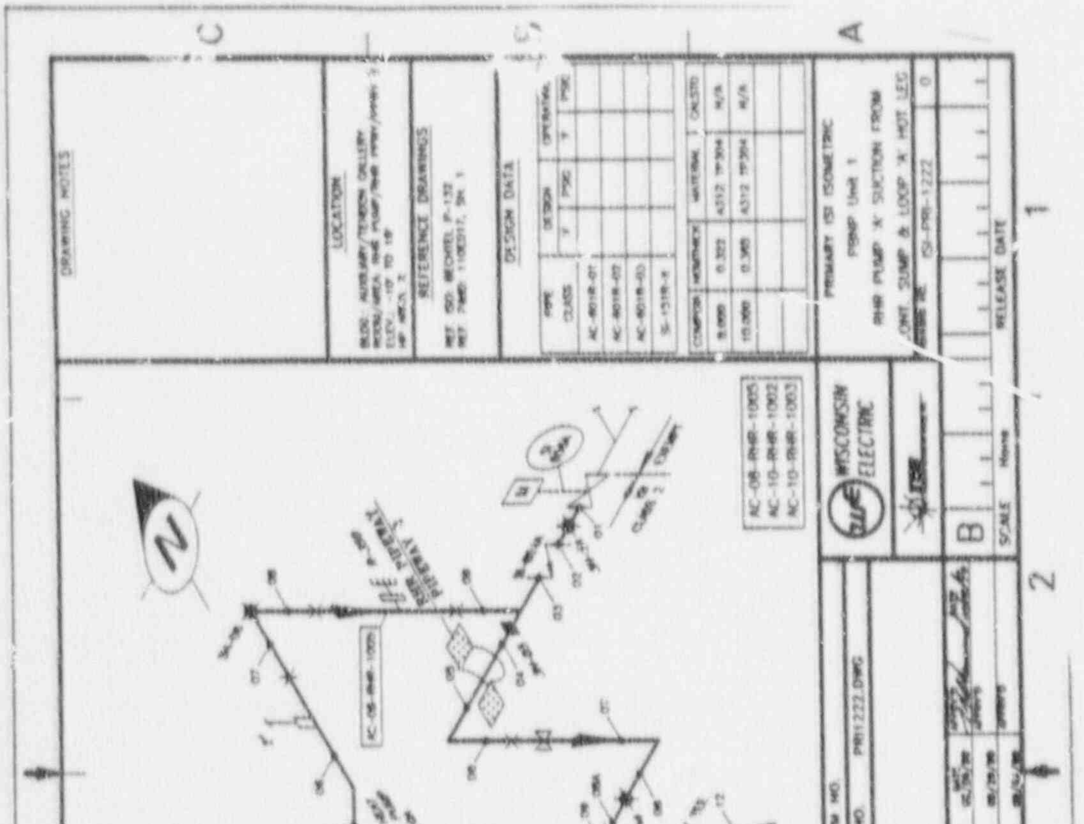
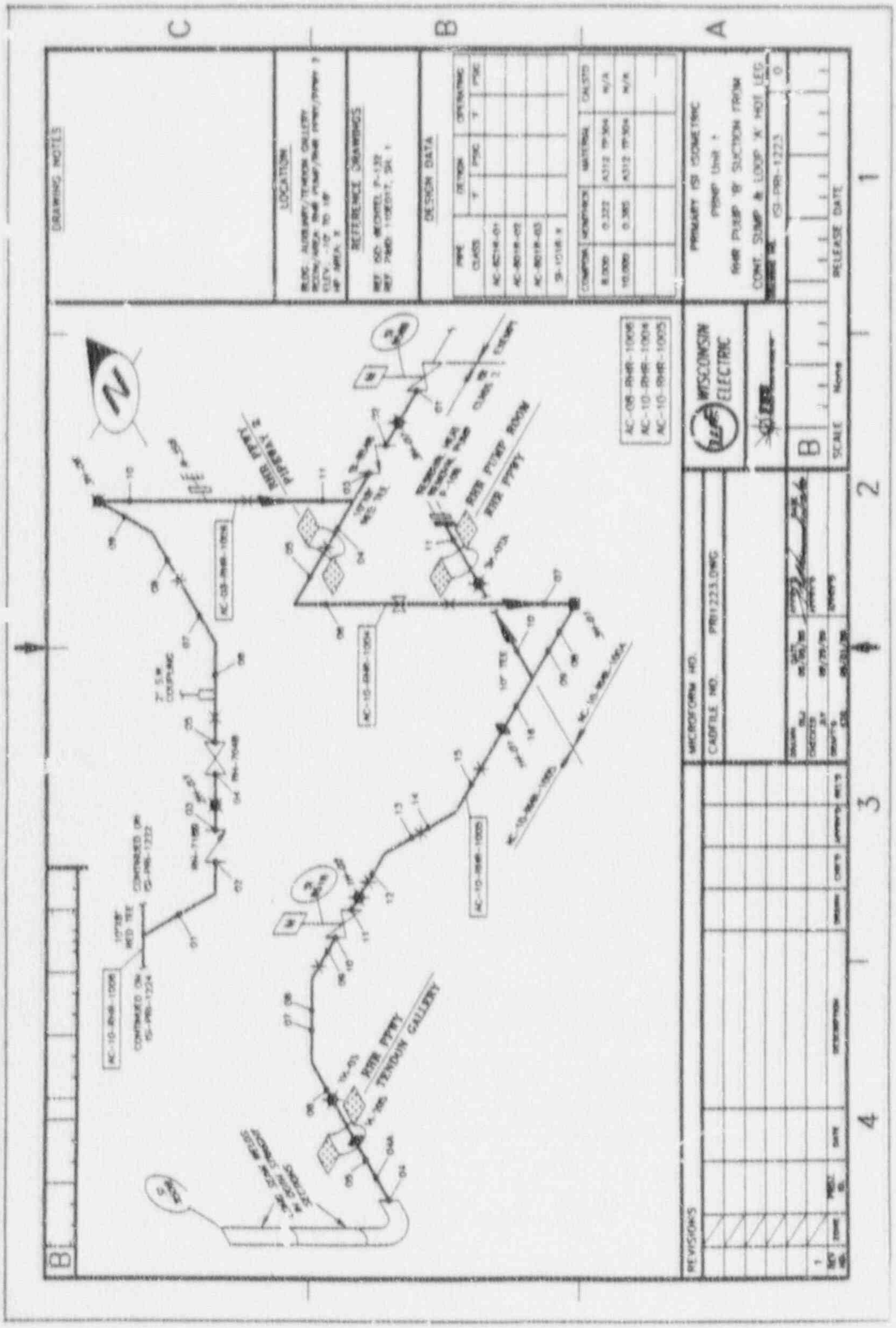


Figure 4

ISI-PRI-1223



Attachment C

LIST OF ACRONYMS

A - As Low As Reasonably Achievable

- Authorized Nuclear Inservice Inspector

ANSI - American National Standards Institute

ASME - American Society of Mechanical Engineers

CFR - Code of Federal Regulations

CHR - Containment Heat Removal

CS - Containment Spray

ECC - Emergency Core Cooling

EMD - Electro-Motive Division of General Motors

FSAR - Final Safety Analysis Report

IEB - Inspection and Enforcement Bulletin

ISI - Inservice Inspection

NRC - Nuclear Regulatory Commission

NRR - Nuclear Reactor Regulation

NUREG - Nuclear Regulation

PBNP - Point Beach Nuclear Plant

RG - Regulatory Guide

RHR - Residual Heat Removal

RMP - Routine Maintenance Procedure

RPV - Reactor Pressure Vessel

RWST - Refueling Water Storage Tank

SI - Safety Injection

SRP - Standard Review Plan

TS - Plant Technical Specification

WCAP - Westinghouse Commercial Atomic Power

WE - Wisconsin Electric Power Company