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Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

December 22, 1978

Director, Nuclear Reactor Regulation
Att: Mr Dennis L Ziemann, Chief
Operating Reactors Branch No 2
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 -
BIG ROCK POINT PLANT - DESCRIPTION OF
INSERVICE INSPECTION PROGRAM AND REQUESTS
FOR RELIEF FROM SPECIFIC ASME CODE REQUIREMENTS

The attachment to this letter comprises a description of the Big Rock Point inservice inspection program and requests for relief from specific requirements of Section XI of the ASME Boiler and Pressure Vessel Code. These requests are submitted for approval in accordance with guidance provided by NRC letter dated November 24, 1976.

Consumers Power Company letter dated April 7, 1978 specified that the current 40-month inservice inspection interval for the Big Rock Point Plant had been extended to be coincident with the scheduled refueling outage and would end January 31, 1979. Since that letter, the refueling outage schedule has been delayed. Therefore, the 40-month inservice inspection period for which the attached relief requests are applicable is now scheduled to begin April 1, 1979.

The attached description and requests for relief reference a composite version of Section XI of the ASME Code utilizing both Summer '75 and Summer '77 addenda as a basis. This code version is identical to that previously used as a basis for the inservice inspection program at our Midland Plant and is consistent with the proposed amendment to 10 CFR 50.55a submitted to the ACRS Subcommittee on Regulatory Activities for review April 11, 1978. Subsequent discussions with members of the NRC staff have identified that issuance of the proposed amendment to 10 CFR 50.55a is expected in the near future and that this amendment will likely reference the Summer '78 version of Section XI. The differences between the Summer '78 code and the composite code used in the attached are administrative in nature when applied to a plant such as Big Rock Point and will affect, at most, examination scheduling. It is, therefore, requested that these requests be reviewed on a technical basis with the understanding that appropriate revisions will be made relative to code use upon publication of the amendment to 10 CFR 50.55a.

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Final internal review of the attachment is being completed concurrently with this submittal. No substantive changes are anticipated as a result of this review. Any changes made will be submitted to you as soon as they are completed.

David A Bixel (Signed)

David A Bixel
Nuclear Licensing Administrator

CC: JGKeppler, USNRC

PROPOSED REQUEST FOR RELIEF FROM
PROVISIONS OF ASME B&PV CODE-SECTION XI*
PURSUANT TO 10 CFR 50, SECTION 50.55a(g)(6)(i)

*SPECIFIC YEARS AS DISCUSSED IN SECTION 2.0

TABLE OF CONTENTS

SECTION 1.0

Description of the Big Rock Point 40-Year Inservice Inspection Master Plan

DESCRIPTION OF THE BIG ROCK

POINT 40-YEAR INSERVICE INSPECTION MASTER PLAN

I. INTRODUCTION

This document is a plan for inservice examinations (ISI) to be performed over a 40-year period on Class 1, Class 2, and Class 3 components and systems (and their supports) of Consumers Power Company's (CPCo) Big Rock Point Nuclear Power Plant.

A. HISTORICAL BACKGROUND

The initiation of Commercial Service for Big Rock Point was on December 8, 1962. The start of the first Ten-Year Interval was on January 1, 1972. during the first two 40-month intervals, in order to comply with Section 9 of the Technical Specifications of the " Operating License (DPR-6) for Big Rock Point Nuclear Plant," which discusses ISI requirements of Class 1 components and systems, the nondestructive examinations were performed to satisfy the requirements of the ASME Section XI Code, 1971 edition including the Winter 1972 Addenda. In February 1976, the NRC ammended Paragraph 55a(g) of 10 CFR 50 to require Nuclear Power Plants to upgrade their Technical Specifications in the areas of the ISI requirements and the functional testing of pumps and valves. By amending Paragraph 55a(g) and by invoking Regulatory Guide 1.26. the NRC required Nuclear Plants to upgrade their systems to include not only Class 1 systems, but also Class 2 and Class 3 systems in their ISI programs.

B. UPGRADING CRITERIA

The construction of this Plan was based on the following documents:

(1) Big Rock Point Nuclear Plant's Piping and Instrument Diagrams, and Plant Q-List.

(2) Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components, As identified in Section 2.0 of this document.

(3) U.S. NRC "Rules and Regulations, Title 10, Chapter 1, Code of Federal Regulations-Energy," Part 50.55a;

(4) Applicable sections of Section 9.0 of the Technical Specifications of the " Operating License (DPR-6) for the Big Rock Point Nuclear Power Plant."

Components were scheduled for examination in accordance with the Plant's Technical Specifications and Section XI of the code. Examinations are conducted in accordance with Section 2.0 of this document.

C. REFERENCES

(1) 10 CFR 50.55a(g)

C. REFERENCES (CONT'D)

(2) Operating License DPR-6, Technical Specifications for the Big Rock Point Plant, Docket No 50-155, Section 9.0, as modified by Technical Specification change request dated July 27, 1978.

(3) ASME Boiler and Pressure Vessel Code, Section XI.

(4) Consumers Power Quality Assurance Program Procedure for Operations, No 10-52.

(5) Consumers Power Operating Services Department Procedure, No OSD-25.

D. GENERAL

(1) This Inservice Inspection Plan for the four 10-Year Inservice Intervals has been developed by Consumers Power Company for use at Consumers Power Company's Big Rock Point Nuclear Power Plant. This Plan incorporates all periodic surveillance requirements of references C-2 and C-3 for the 40-Year Service Lifetime. The length of the second three and one-third year period has been extended by approximately six to seven months to permit the next inspection to be coincident with the scheduled refueling outage. Therefore, the start of the third interval third which is the date the update becomes effective is approximately April 1, 1979.

(2) Responsibility for the Maintenance of this Plan, and the development of subsequent Plans rests with the Operating Services Department as defined in Reference C-4. Only copies marked "Controlled" will be issued revisions.

(3) In view of the fact that the Big Rock Point Nuclear Plant went into Commercial Operation well before the issuance of Section XI of the ASME B&PV Code, the Inspection Access requirements of IS-142, '71 Edition, were not available to impact the plant design parameters. The Technical Specification/Relief Request Section of this Plan details specific code requirements which cannot be met.

(4) Examination methods delineated in the tables are intended to be representative of past ISI practice or of preservice methods utilized. In either case, it should be recognized that either UT or RT are acceptable volumetric exams and either PT or MT are acceptable surface exams. Unique weld joint parameters may, of course, dictate more restrictive selection criteria; e.g., high background radiation will preclude RT, stainless materials will preclude MT, etc. It is intended that the process which selects exam methods for inspections under this plan treat UT and RT as interchangeable and PT and MT as interchangeable with consideration given to past practice in light of the reproductibility of results.

The following Table summarizes the inspections to be performed at the Big Rock Point Nuclear Plant per the updated ISI 40-Year Plan:

Table 1

BIG ROCK POINT INSERVICE INSPECTION

(PER ASME BVPV CODE, SECTION XI, AS IDENTIFIED IN SECTION 2.0)

CLASS 1

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
REACTOR VESSEL				
Longitudinal and circumferential shell welds; meridonal and circumferential head welds; Vessel-to-Flange and Head-To-Flange circumferential welds	B1.1	B-A	18	6
Primary Nozzle-To-Vessel welds and nozzle inside radiused section	B1.4	B-D	30	12
Vessel penetrations, including control rod drive and instrument penetrations	B1.5	B-E	37	32
Nozzle-to-Safe End Welds	B1.6	B-F	37	6
Nuts	B1.7	B-G-1	42	42
Pressure retaining bolts and studs	B1.8	B-G-1	42	42
Ligaments between threaded stud holes	B1.9	B-G-1	1	1
Closure Washers-bushings	B1.10	B-G-1	42	42
Pressure retaining bolting	B1.11	B-G-2	5	5
Integrally welded vessel supports	B1.12	B-H	20	---
Vessel interior	B1.15	B-N-1	---	---
Interior attachments and core support structures	B1.16	B-N-2	24	24
Core support structures	B1.17	B-N-3	---	---
Control Rod Drive Housing	B1.18	B-O	32	32**
Pressure retaining components; support restraints-mechanical, hydraulic	B1.19	B-P	---	---
**Only two out of the three control rod drive housing welds, are volumetrically inspectible and shall be done. Geometry precludes volumetric examination of third weld. (See Relief Requests.				
HEAT EXCHANGERS - STEAM GENERATORS				
A) Emergency Condensor				
* Based upon relief requests				

BIG ROCK POINT INSERVICE INSPECTION

CLASS 1

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
HEAT EXCHANGERS - STEAM GENERATORS (Cont'd)				
Longitudinal and circumferential welds	B3.1	B-B	4	4
Nozzle-to-Head welds and Nozzle Inside Radius	B3.2	B-D	8	4
Pressure Retaining Bolting	B3.10	B-G-2	2	2
B) Clean Up Regen Heat Exchanger (4)				
Longitudinal and Circumferential Welds	B3.1	B-B	16	16
Nozzle-to-Head Welds and Nozzle Inside Radiused Section	B3.2	B-D	32	16
Integrally Welded Vessel Supports	B3.7	B-H	16	16
C) Clean Up Non-Regen Heat Exchanger				
Longitudinal and Circumferential Welds	B3.1	B-B	2	2
Nozzle-to-Head Welds and Nozzle Inside Radiused Section	B3.2	B-D	4	0
Integrally Welded Vessel Support	B3.7	B-H	2	2
D) Steam Drum				
Longitudinal and Circumferential Welds	B3.1	B-B	16	4
Nozzle-to-Head Welds and Nozzle Inside Radiused Section	B3.2	B-D	34	17
Nozzle-to-Safe End Welds	B3.3	B-F	10	10
Pressure Retaining Bolting	B3.6	B-G-1	2	2
Integrally Welded Vessel Supports	B3.7	B-H	18	4
E) Clean Up Demin Tank				
Longitudinal and Circumferential Welds	B3.1	B-B	5	5
Nozzle-to-Head Welds and Nozzle Inside Radiused Section	B3.2	B-D	5	5
Integrally Welded Vessel Supports	B3.7	B-H	3	3
Pressure Retaining Bolting	B3.10	B-G-2	1	1

BIG ROCK POINT INSERVICE INSPECTION

CLASS 1

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
SYSTEM IDENTIFICATION:				
1) Emergency Condensor System (ECS)				
2) Liquid Poison System (LPS)				
3) Shutdown Cooling System (SCS)				
4) Reactor Clean Up System (RCS)				
5) Main Steam System (MSS)				
6) Main Recirculation System (MRS)				
7) Core Spray System (CSS)				
8) Redundant Core Spray (RDC)				
9) Control Rod Drive (CRD)				
10) Feedwater System (FWS)				
11) Reactor Depressurization (RDS)				
Safe End-to-Piping Welds and Safe-End in Branch Piping Welds	B4.1	B-F		
LPS			2	2
SCS			2	1
RCS			4	3[1]
MSS			1	1
CSS			1	1
RDS			10	10[1]
[1]One B4.1, B-F is a Socket Weld and therefore cannot be examined volumetrically a PT examined will be done.				
Pressure Retaining Bolts and Studs	B4.2	B-G-1		
RCS			3	3
Pressure Retaining Bolting	B4.3	B-G-1		
RCS			2	2
MSS			7	7
CSS			1	1
RDC			2	2
RDS			14	14
Circumferential and Longitudinal Piping Welds 4 in. and greater	B4.4	B-J		
ECS			57	57

BIG ROCK POINT INSERVICE INSPECTION

CLASS 1

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
SCS			43	32
NSS			37	37
MRS			124	116
CSS			22	19
EHC			31	31
CRD			4	4
FWS			15	15
RHS			30	30
Circumferential and Longitudinal Butt Welds Less Than 4 inch	B4.5	B-J		
LPS			54	54
RCS			175	175
NSS			69	69
Branch Pipe Connections	B4.6	B-J		
ECS			5	5
LPS			1	1
MSS			1	1
MRS			9	9
CSS			1	1
RHS			9	9
Socket Welds	B4.8	B-J		
LPS			21	15
RCS			58	58
MSS			182	177
CRD			80	79
RHS			50	50
Integrally Welded Supports	B4.9	B-K-1		
ECS			10	10
LPS			3	1
SCS			3	3
RCS			6	6
MSS			6	6
MRS			77	75
CSS			1	1
CRD			1	1[2]
RHS			13	1
Support Components - Structural Support Restraints - Mechanical, Hydraulic	B4.10	B-K-2		

BIG ROCK POINT INSERVICE INSPECTION

CLASS 1

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
ECS			1	1
LPS			9	9
SCS			7	4
RCS			26	26
MSS			17	17
FWS			60	60
CSS			5	5
HDC			4	4
CRD			13	13
FWS			1	1
RDS			13	13
Pressure Retaining Component all of the above Class 1 Systems shall be subjected to the System Pressure Tests of IWB-5000 and the Visual Examinations (VT-2) of IWA-5240	B4.11	B-P		
[2]Due to Configuration and Purpose of Support a Visual Examination shall be made of Lugs				
PUMP PRESSURE BOUNDARY				
1) Reactor Clean Up Pump				
Support Components-Structural Support Restraints-Mechanical Hydraulic	B5.5	B-K 2	1	1
Pump Casing Welds	B5.6	B-L-1	1	1
Pump Casings	B5.7	B-L-2	1	1
Pressure Retaining Bolting	B5.9	B-G-2	1	1
2) Main Recirc Pump				
Support Components-Structural Support Restraints-Mechanical Hydraulic	B5.5	B-K 2	10	10
Pump Casing Welds	B5.6	B-L-1	2	2
Pump Casings	B5.7	B-L-2	2	2
Pressure Retaining Bolting	B5.9	B-G-2	2	2
VALVE PRESSURE BOUNDARY				
Pressure Retaining Bolting >2"	B6.3	B-G-1	34	34
Valve Bodies	B6.7	B-M-2	53	47

BIG ROCK POINT INSERVICE INSPECTION

CLASS 1

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
VALVE PRESSURE BOUNDARY (Cont'd)				
Pressure Retaining Bolting <2"	B6.9	B-G-2	36	36

BIG ROCK POINT INSERVICE INSPECTION

CLASS 2

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
PRESSURE VESSELS				
Class 2 Vessels to Be Examined are as Follows:				
1) High Pressure Heater 2) Shutdown Heat Exchangers (2) 3) Core Spray Heat Exchanger 4) Liquid Poison Tank				
Pressure Vessel Inspections Per Table IWC-2600-1				
PIPING				
Piping Systems				
1) Control Rod Drive (CRD) 2) Feedwater (FWS) 3) Emergency Condensor (ECS) 4) Liquid Poison System (LPS) 5) Main Steam System (MSS) 6) Post Incident Cooling (PIS) 7) Reactor Clean-Up (RCS) 8) Reactor Shutdown (RSS)				
Piping Welds - 1/2" or Less	C2.1	C-F		
ECS PIS RSS			5 61 71	5 61 71
Piping Welds - Over 1/2 inch	C2.2	C-F		
CRD FWS MSS			18 65 19	18 65 19
Pipe Branch Connection	C2.3	C-F		
FWS			3	3
Integrally Welded Supports	C2.4	C-E-1		
FWS MSS			6 3	6 3

BIG ROCK POINT INSERVICE INSPECTION

CLASS 2

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
Support Components - Structural Support Restraints - Mechanical Hydraulic	C2.5	C-E-2		
CRD			1	1
FWS			14	14
MSS			3	3
PIS			8	8
RSS			8	8
Pressure Retaining Components all of the above Class 2 Systems shall be subjected to the System Pressure Test of IWC-5000 and the Visual Examination (VT-2) of IWA-5240	C2.7	C-H		
PUMPS				
Class 2 Pumps to be Examined are:				
1) CRD Pumps (2)				
2) Shutdown Pumps (2)				
3) Core Spray Pumps (2)				
Pump Examination per Table IWC-2600-1 Section XI				
VALVES				
The following Class 2 Valves are subject to examination:				
1) CV-4000 10-FWS-201				
2) FW-23 10-FWS-201				
3) FWS-6 10-FWS-201				
4) CV-2075 10-FWS-201				
5) FW-300 6-FWS-204				
6) FW-6 6-FWS-204				
7) FW-301 6-FWS-205				
8) FW-2 6-FWS-205				
9) CV-7500 12-MSS-201				
10) MO-7067 10-MSS-204				
11) CV-4200 10-MSS-204				
12) FP-29 6-PIS-201				
13) VPI-302 6-PIS-201				
14) SW-3 6-PIS-203				
15) MO-7059 8-RSS-201				
16) MO-7057 8-RSS-202				
17) SC-30 6-RSS-203				
18) SC-4 6-RSS-203				

BIG ROCK POINT INSERVICE INSPECTION

CLASS 2

COMPONENTS TO BE EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
19) SC-3 6-RSS-204 20) SC-300 6-RSS-205 21) SC-2 6-RSS-205 SC-1 6-RSS-206 The Valves are scheduled for examination per ASME Section XI Table IWC-2600-1.				

BIG ROCK POINT INSERVICE INSPECTION

CLASS 3

SYSTEMS EXAMINED	COMPONENTS EXAMINED	SECTION XI ITEM NO.	SECTION XI CAT	ITEM NO TOTALS	NO. TO BE INSPECTED *
1) Condensate Pump Suction 2) Condensate Tank Pipe 3) Demin Water System 4) Reactor Cleanup System 5) Reactor Cooling Water 6) Service Water System Post Incident Cooling	Pressure retaining components within the boundaries of systems required to operate in support of Normal Plant safty functions and maintaining the Reactor in Cold Shutdown.	D1.1	D-A	Pressure Test whole systems per Table IWD-2500-1	---
Reactor Cooling Water	Component Supports and Restraints, within the boundaries of the above systems, for components exceeding 4 in.Pipe Size.	P1.2	D-A	41	41
Service Water System Post Incident Cooling	Pressure Retaining Components within the boundaries of systems or portion of systems required to to operate in support of the Post-Accident Safety Functions of Emergency Core Cooling, Containment Heat removal and atmosphere clean up, and long term residual heat removal from reactor.	D2.1	D-B	Pressure Test whole systems per Table IWD-2500-1	---
1) Reactor Cleanup System 2) Reactor Cooling Water	Pressure Retaining Piping, Pumps and valves within the boundary of systems or portions of systems required to operate in support of residual heat removal from spent fuel storage pool.	D3.1	D-C	Pressure Test whole systems per Table IWD-2500-1	---
Reactor Cooling Water	Component Supports and restraints within the boundary of the above systems for components exceeding 4 inch nominal pipe size diameter.	B3.2	D-C	3	3

SECTION 2.0
APPLICABLE EDITIONS AND ADDENDA OF ASME BOILER AND PRESSURE VESSEL CODE-SECTION XI

2.0 APPLICABLE EDITIONS & ADDENDA OF ASME BOILER & PRESSURE VESSEL CODE - SECTION XI [1]

Pursuant to the draft Paragrapn 50.55a(g) of 10 CFR Part 50, as reviewed by the ACRS on May 3, 1978, the in-service examination requirements applicable to nondestructive examination and system pressure testing at the Consumers Power Company, Big Rock Point Plant are based upon the rules set forth in the 1974 Edition of Section XI of the ASME Boiler and Pressure Vessel Code, including Addenda through Summer 1975.

As permitted by Paragrapn 50.55a(g)(2)(v) and 50.55a(g)(4), Consumers Power Company elects, for certain components, to meet supplemental requirements set forth in the edition of the Code and Addenda which have become effective subsequent to the 1974 Edition and Summer 1975 Addenda. The selected requirements which are contained in 1977 Edition including Addenda through Winter 1977 are identified in Table II-1 with respect to the specific rules that supersede those contained in 1974 Edition including Addenda through Summer 1975.

Consumers Power Company elects to adopt Subsection IWF - Requirements for Classes 1, 2, and 3 Component Supports which, although not yet included in the 1977 Edition of the Section XI ASME Code, is expected to be issued in the Winter 1978 Addenda. This Subsection IWF provides improved inspection guidance not contained in the earlier code Edition and Addenda issues.

[1] Basis for the selection of the applicable editions and addenda of ASME Boiler and Pressure Vessel Code Section XI, Is contained in Section 5.0

TABLE II-1

APPLICABLE ASME SECTION XI CODE
EDITIONS AND ADDENDA

Applicable Requirements From 1977 Ed W/Addenda Thru W/77	Superseded Requirements From 1974 Ed W/Addenda Thru S/75 [1]	Subject	Comments
IWA-1100	IWA-1100	Scope	
IWA-1400	IWA-1400	Owner's Responsibility	
IWA-2120	IWA-2120	Duties of the Inspector	
IWA-2130	IWA-2130	Qualification of Inspectors, Inspection Specialists and Inspection Agencies	
IWA-2210	IWA-2210	Visual Examinations	
IWA-2232	IWA-2232	Ultrasonic Examination	
IWA-2300	IWA-2300	Qualifications of NDE Personnel	
IWA-2400	IWA-2400	Inspection Intervals	Inspection Program B will be applied.
IWA-2500	Not in S/75	Extent of Examination	
IWA-3000	IWA-3000 & IWB-3300	Examination Evaluations	
IWA-4000	IWA-4000	Repair Procedures	
IWA-5000	IWA-5000	System Pressure Tests	IWA-5213(c) shall read "no holding time required ..."
IWA-6000	IWA-6000	Records & Reports	

[1] Items listed in this column are superseded by those respective items from the '77/W77 Section XI Edition listed in the column to the left.

TABLE II-1 (Cont'd)

Applicable Requirements From 1977 Ed W/Addenda Thru W/77	Superseded Requirements From 1974 Ed W/Addenda Thru S/75	Subject	Comments
IWA-7000	Not in S/75	Replacements	
IWB-2410 & 2412	IWB-2410, 2411 & 2412	Inspection Schedule	
The following categories in Table IWB-2500-1:	The following categories in Table IWB-2500:	Examination Categories	
B-D	B-D	Full Penetration Welds in Vessel Nozzles	The requirements for the first inspection interval will be implemented for each inspection interval.
B-F	B-F	Dissimilar Metal Welds	The requirements for the first inspection interval will be implemented for each inspection interval.
B-G-1	B-G-1	Bolting Larger Than 2"	
B-G-2	B-G-2	Bolting 2" & Smaller	
B-H	B-H	Vessel Supports	
Not in W/77 [2]	B-I-1	RV Cladding	
Not in W/77	B-I-2	Other Vessel Cladding	
B-K-1	B-K-1	Support Members for Piping, Valves & Pumps	W/77 will be applied except that the provisions of S/75 will be applied to the extent and frequency of examinations in order to comply with the proposed revision to 10CFR50:55a referenced in Section 1.0 of this plan.
B-L-1	B-L-1	Welds in Pump Casings	
B-M-1	B-M-1	Welds in Valve Bodies	

[2]Since this requirement has been deleted from the Code, components of this category (B-I-1, B-I-2) will no longer be scheduled for examination.

TABLE II-1 (Cont'd)

Applicable Requirements From 1977 Ed W/Addenda Thru W/77	Superseded Requirements From 1974 Ed W/Addenda Thru S/75	Subject	Comments
B-O	B-O	Welds in Control Rod Drive Housings	
B-P	B-P	All Pressure Retaining Components	
The following figures will be applied and, as such, may change the "Areas Subject to Examinations" in S/75:	The "Areas Subject to Examinations" or figures are modified for the following examination categories:		
IWB-2500-1	B-A & B-B	Vessel Circum	
IWB-2500-2	B-B	Vessel Long	
IWB-2500-3	B-B	Head Circum	
IWB-2500-4	B-C	Shell to Flange	For stud hole ligaments the area subject to examination will be limited to 1" around the stud hole in accordance with Figure IWB-2500-4 in the S/78 Addenda.
IWB-2500-5	B-C	Head to Flange	
IWB-2500-6	B-B	Tube Sheet to Head	
IWB-2500-8	B-J	Piping Welds	
IWB-2500-9	B-J	Pipe Branch	
IWB-2500-10	B-J	Pipe Branch	
IWB-2500-11	B-J	Pipe Branch	
IWB-2500-12	Figure IWB-3515.1	Closure Studs	
IWB-2500-13	B-K-1	Support Circumferential Welds	
IWB-2500-14	B-K-1	Support Circumferential Welds	
IWB-2500-15	B-K-1	Support Lug Attachment	
IWB-2500-16	Not in S/75	Pump Welds	
IWB-2500-17	Not in S/75	Valve Welds	

TABLE II-1 (Cont'd)

Applicable Requirements From 1977 Ed W/Addenda Thru W/77	Superseded Requirements From 1974 Ed W/Addenda Thru S/75	Subject	Comments
IWB-2500-18	Not in S/75	Control Rod Drive Welds	
Table IWB-2600-1	Table IWB-2600	Methods of Examination	Item B1.1 will be applied to Categories B-A, B-B & B-C. Items B5.6 and B6.7 will also be subjected to surface examination in accordance with the errata contained in S/78.
IWB-3000	IWB-3000	Flaw Indication Standards	In accordance with the errata contained in S/78 Footnote #1 to Table IWB-3515-1 is modified to delete the abbreviation "Fig." Also, Table IWB-3518-2 is modified in accordance with S/78 errata.
IWB-4000	IWB-4000	Repair Procedures	
IWB-5000	IWB-5000	Pressure Tests	
IWB-7000	Not in S/75	Replacements	
IWC-1230	Not in S/75	Concrete Encased Components	
IWC-2200	IWC-2100	Preservice Inspection	
Table IWC-2500-1			
Category C-D	Category C-D	Pressure Retaining Bolting	
Category C-H	Not in S/75	All Pressure Retaining Components	

TABLE II-1 (Cont'd)

Applicable Requirements From 1977 Ed W/Addenda Thru W/77	Superseded Requirements From 1974 Ed W/Addenda Thru S/75	Subject	Comments
Figures IWC-2520-1 Thru IWC-2520-4 and IWC-2520-6 thru IWC-2520-9	Not in S/75		
The following figures change the "Areas Subject to Examinations" in S/75:	The "Areas Subject to Examinations" are modified for the following examination categories:		
IWC-2520-1	C-A	Vessel Circum	
IWC-2520-2	C-A	Tube Sheet to Shell	
IWC-2520-3	C-B	Nozzle Welds	
IWC-2520-4	C-B	Nozzle Welds	
IWC-2520-5	C-C & C-E-1	Supports	
IWC-2520-7	C-F & C-G	Piping	
IWC-2520-8	C-F & C-G	Pumps & Valves	
IWC-2520-9	C-F & C-G	Piping	
Table IWC-2600-1	Table IWC-2600	Methods of Examination	Application of Item C1.6 expands the scope of Category C-E-2 in S/75 to include Vessels Items #C2.1, C2.2 and C2.3 will be applied to Categories C-F(a-d) and C-G(a-d) in S/75. Items #C3.1 and C4.1 will be applied to Categories C-F(e) and C-G(e).

TABLE II-1 (Cont'd)

Applicable Requirements From 1977 Ed W/Addenda Thru W/77	Superseded Requirements From 1974 Ed W/Addenda Thru S/75	Subject	Comments
IWC-3000	IWC-3000	Flaw Indication Standards	
IWC-4000	IWC-4000	Repair Procedures	
IWC-5000	IWC-5000	Pressure Tests	
IWC-7000	Not in S/75	Replacements	
IWD-2000	IWD-2000	Examination Requirements	Footnote #1 will be applied to all 3 categories in Table IWD-2500-1.
IWD-3000	IWD-3000	Flaw Indication Standards	
IWD-4000	IWD-4000	Repair Procedures	
IWD-5000	IWD-5000	Pressure Tests	
IWD-7000	Not in S/75	Replacements	
Not in W/77	Appendix I	Ultrasonic Examination	Deleted by W/76. See IWA-2232.
Appendix II	Appendix II	Owner's Data Report	
Appendix III	Not in S/75	Pipe Ultrasonic Examination	Use of Appendix III shall be modified by the attached Consumers Power Company Code Applicability Statement No 1, Revision 1.

BIG ROCK POINT

PLANT CODE USE FOR INTERVAL

THIRD 3/1/79 to 4/31/82

CONSUMERS POWER COMPANY CODE APPLICABILITY STATEMENT NO 1

SUBJECT: PIPING AND VESSEL EXAMINATIONS

I. The basic requirements for piping and vessel examinations shall be in accordance with Section V, 1974 Edition of the ASME Boiler and Pressure Vessel Code with addenda through Summer 1975 as referenced in Section XI, with Editions and Addenda as described previously. Guidance was taken from Section XI, Appendix III, 1974 Edition of the ASME Code with addenda through Winter 1975 for piping examinations.

II. The Section V requirements should be modified by the application of:

A. Code Case 1698 - Waiver of Ultrasonic Transfer Method

B. Code Case 1705 - Ultrasonic Examination - Calibration Block Tolerance.

III. The requirements of Paragraph T-511 of Article 5 of Section V will be modified as follows:

A. A pulse-echo type ultrasonic flaw indication instrument shall be used.

B. The instrument shall be equipped with a gain control calibrated in units of 2 dB or less.

IV. The ultrasonic examinations of piping components shall have the additional requirements as follows:

A. Ultrasonic examinations shall be performed in accordance with a written procedure which shall include the following information:

1. Weld type and configuration including thickness dimensions, material, and product form (casting, forging, etc).

2. Surface or surfaces from which the examinations shall be performed.

3. Surface condition.

4. The couplant used in the examination shall be the same as used in the calibration.

5. The examination technique (straight beam, angle, beam, contact, and/or immersion) shall be specified.

6. The angles and mode or modes of wave propagation in the material shall be specified.

7. The type, frequency, and transducer sizes shall be specified.

8. Special search units, wedges, shoes, or saddles if used shall be specified.

9. The type and length of search unit cables shall be specified.

10. The ultrasonic instrument type shall be specified.

11. A description of the method of calibration shall be specified.

12. The direction and extent of the scanning shall be specified.

13. Data to be recorded and the method of recording in either automatic or manual mode shall be specified.

14. The use of the automatic alarm and recording equipment, or both, if used shall be specified.

15. Any rotating, revolving, or scanning mechanisms if used shall be specified.

16. Personnel qualification requirements shall be specified.

17. The procedure shall be approved as required by Paragraph IWA-2120 of Section XI.

B. Instrument screen height linearity and amplitude control linearity shall be in accordance with established procedures and quality assurance requirements.

C. Pipe marking identification shall be in accordance with CP Co's Master Plan.

D. Calibration shall be checked by verifying the instrument sensitivity and sweep range calibration at the start and finish of each examination, with any change in examination personnel, and at least every 4 hours during the examination.

E. Calibration data shall include the following information:

1. Calibration sheet identification and date

2. Examination personnel

3. Examination procedure number and revision.

4. Basic calibration block identity.

5. Ultrasonic instrument identity and serial number.

6. Beam angle, couplant used, mode of wave propagation, and material examined.

7. Orientation of search unit with respect to the pipe longitudinal or circumferential direction.

8. Search unit identity, frequency, size, and manufacturer's serial number.

9. Special search units, wedges, wedge type, and saddle identity if used.

10. Search unit cable type and length.

11. Clock time of the initial calibration and all subsequent calibration checks.

12. Amplitude and sweep readings obtained from calibration reflectors.

F. Examination data sheets shall include at least the following information:

1. Data sheet identity number and date.

2. Examination personnel.

3. Examination procedure and revision.

4. Applicable calibration sheet identity number.

5. Identification and location of the weld or volume scanned.

6. Surface or position from which the examination is conducted.

7. Record of indications or statement that no recordable indications were observed.

G. All ultrasonic reflectors producing a response 50% or greater of the reference level shall be recorded on the appropriate Examination Record.

All ultrasonic reflectors 100% or greater of the reference level in piping, shall be investigated to determine the shape, size, identity, and location of the reflector and reported to Consumers Power.

All ultrasonic reflectors 50% or greater of the reference level in vessels shall be investigated to determine the shape, size, identity, and location of the reflector and reported to Consumers Power.

REQUESTS FOR RELIEF

REACTOR VESSEL

Item B1.1, B-A: LONGITUDINAL AND CIRCUMFERENTIAL SHELL WELDS, MERIDIONAL AND CIRCUMFERENTIAL HEAD WELDS

R-1 A. VESSEL CORE REGION

BASIS STATEMENT

Longitudinal and circumferential welds in the core region are not accessible for volumetric examination.

REMARKS

The reactor vessel is closely surrounded by concrete so examination from the outside is not possible. Inner wall of reactor vessel is inaccessible in the core region due to the presence of a thermal shield, which is immovable. Design clearance between the thermal shield and reactor vessel wall is 1.65 inches. The thermal shield extends 66.50 inches above the beltline circumferential weld.

ALTERNATIVE INSPECTION OR TEST

Each refueling outage a hydrostatic test (pre start up hydro) is performed at 1.1 times the operating pressure. This is more conservative than the prestart-up leak test at operating pressure required by the code (1WB-5221). Inspection of the six riser nozzles and vessel-to-flange weld will provide some indication of reactor vessel integrity, along with results of vessel coupon testing. During operation a failure of any weld in the reactor vessel would be readily detectable by level indication, dew cell indication, makeup water flow, and/or temperature indication in adequate time for safe shutdown.

REFERENCES

CE Drawings E-201-801-5 and E-201-794-8.
GE Drawing 141F797.
Bechtel Drawing 0740G20128 Rev C (C-128).

IMPLEMENTATION

Relief requested. Relief was previously approved May 4, 1972 pursuant to April 21, 1972 request.

R-2 B. VESSEL SHELL (Above Beltline Circ Weld)

BASIS STATEMENT

Longitudinal welds in vessel shell are generally inaccessible for volumetric examination.

REMARKS

Essentially 70-75% of the length of the longitudinal welds are inaccessible for volumetric inspection because of the thermal shield and core spray sparger and baffle supports. No technique is available for examination of the longitudinal welds. Access from the outside wall is not possible due to concrete wall. Vessel shell-to-flange weld is accessible for inspection and has been ultrasonically examined.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

CE Drawings E-201-794 and E-201-793.
Bechtel Drawing 0740G20128 Rev C.

IMPLEMENTATION

Relief requested on longitudinal welds as before.

R-3 C. VESSEL BOTTOM HEAD

BASIS STATEMENT

Meridional and circumferential welds in the bottom head are inaccessible for volumetric inspection.

REMARKS

Meridional welds are inaccessible from inside reactor because of the core support plate. Poor geometry due to the 40 penetrations on the bottom head makes the meridional welds inaccessible from the outside. The circumferential shell-to-bottom head weld is not accessible with existing equipment from the inside and the concrete wall prevents inspection from the outside.

ALTERNATIVE TEST OR INSPECTION

Hydro (see R-1).

REFERENCES

CE Drawings E-201-794 and E-201-793.
Bechtel Drawing 0740G20128 Rev C (C-128).

IMPLEMENTATION

Relief requested as before.

Item B1.4, B-D: PRIMARY NOZZLE-TO-VESSEL WELDS AND NOZZLE INSIDE RADIUS SECTION

R-4 A. 20" RECIRCULATION (RECIRC) NOZZLES (796-1A,B)

BASIS STATEMENT

These nozzles are inaccessible for volumetric examination.

REMARKS

Concrete shield wall prevents access to these nozzles from the outside. Diffuser plates prevent volumetric inspection of the nozzles from the inside.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

CE Drawing F-230-791-2.
Bechtel Drawing 0740G20128 Rev C (C-128).

IMPLEMENTATION

Relief requested as before.

R-5 B. 3" NOZZLES 1) CORE SPRAY 795-6
2) POISON INLET 796-6

BASIS STATEMENT

Volumetric techniques are not available to inspect these nozzles.

REMARKS

Access from outside of reactor vessel is prevented by concrete wall. A mechanized volumetric inspection is not possible from the inside because of the thermal sleeves present.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

CE Drawings E-201-795-5 and F-230-791-2.
Bechtel Drawing 0740G20128 Rev C (C-128).

IMPLEMENTATION

Relief requested as before.

R-6 C. 8" SHUTDOWN UNLOADING NOZZLE (795-15)

BASIS STATEMENT

This nozzle is not accessible for mechanized ultrasonic inspection.

REMARKS

A direct manual volumetric examination is not possible due to inaccessibility of nozzle. A interference with the core spray sparger prevents use of the mechanized ultrasonic device.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

CE Drawing F-230-791-2.
GE Drawing 212E456, 104 R 175, Sheet 2.
Bechtel Drawing 0740G20128 Rev C (C-128).

IMPLEMENTATION

Relief requested as before.

R-7 D. 3" INSTRUMENT NOZZLES (955-1A-D)

BASIS STATEMENT

Access to outside of these nozzles is not possible due to concrete wall. Volumetric examination is not possible with existing equipment.

REMARKS

Mechanized ultrasonic device developed for Big Rock's reactor vessel is not capable of inspecting these nozzles because they are on a different plane. The Big Rock device is only capable of inspecting on the same plane as the 14" risers.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

CE Drawing E-201-794-8.

IMPLEMENTATION

Relief requested as before.

Item B1.6, B-F: NOZZLE-TO-SAFE END WELDS

R-3 A. 20" RECIRC NOZZLES (796-1A,B)

BASIS STATEMENT

See B1.4, R-4.

REMARKS

See B1.4, R-4.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

See B1.4, R-4.

IMPLEMENTATION

See B1.4, R-4.

R-9 B. 8" SHUTDOWN UNLOADING (795-15)

BASIS STATEMENT

See B1.4, R-6.

REMARKS

See B1.4, R-6.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

See B1.4, R-6.

IMPLEMENTATION

See B1.4, R-6.

R-10 C. 3" NOZZLES 1) CORE SPRAY 795-6
2) POISON INLET 796-6

BASIS STATEMENT

See B1.4, R-5.

REMARKS

See B1.4, R-5.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

See B1.4, R-5.

IMPLEMENTATION

See B1.4, R-5.

R-11 D. 3" INSTRUMENT NOZZLES (955-1A-D)

BASIS STATEMENT

See B1.4, R-7.

REMARKS

See B1.4, R-7.

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

See B1.4, R-7.

IMPLEMENTATION

See B1.4, R-7.

Item B1.12, B-H: INTEGRALLY WELDED VESSEL SUPPORTS

R-12 A. SUSPENSION ROD BRACKETS (802-6A-L)

BASIS STATEMENT

Plant design does not allow access to the 24 suspension rod brackets.

REMARKS

The integrally welded support is located nine feet below the flange and at this location there is one foot between the reactor vessel wall and shield cooling jacket.

ALTERNATIVE INSPECTION OR TEST

For welds in which defects would affect the integrity of the reactor vessel boundary, the prestart-up hydro serves as an alternative inspection (see R-1). During the next outage, access for remote visual examination shall be explored.

REFERENCES

Bechtel Drawing 0740G620128 Rev C (C-128).
CE Drawing E-201-801-5.

IMPLEMENTATION

Relief requested as before. During the next outage, access for remote visual examination shall be explored.

R-13 B. VESSEL HANGER LUG (802-13 A-D)

BASIS STATEMENT

Volumetric examination of the hanger lug is not possible due to accessibility and configuration.

REMARKS

The hanger lugs, whose purpose is to prevent torsional movement of the reactor vessel, are approximately three feet above the suspension rod brackets or about six feet below the flange. Direct access to this lug is impossible due to concrete shield wall. Remote visual access is also limited due to lug support structure whose disassembly is impossible for safety considerations.

ALTERNATIVE INSPECTION OR TEST

For welds in which defects would affect the integrity of the reactor vessel boundary, the prestart-up hydro serves as an alternative inspection (see R-1). The lug's purpose is to restrict movement in a torsional direction. During normal reactor operation these lugs are essentially nonload bearing. If sufficient torsional movement is encountered during operation, a remote visual exam should identify any operations-induced defect.

REFERENCES

Bechtel Drawing 0740G620128 Rev C (C-128).
CE Drawing E-201-802-9.

IMPLEMENTATION

Relief requested as before. During the next outage, access for a remote visual inspection shall be explored.

R-14 C. STABILIZER BRACKETS (802-12 A-D)

BASIS STATEMENT

Volumetric examination of the stabilizer brackets is not possible due to accessibility.

REMARKS

These brackets are approximately 5.5 feet above the benchmark of the vessel and are inaccessible for direct inspections due to concrete shield wall. Forces caused by the stabilizers are compressive in nature, thereby eliminating shear forces. Since there are no shear forces, the weld is essentially nonload bearing. Defects in the weld would therefore have no effect on the intended use of the lug.

ALTERNATIVE INSPECTION OR TEST

For welds in which defects would affect the integrity of the reactor vessel boundary, the prestart-up hydro serves as an alternative inspection (see R-1).

REFERENCES

Bechtel Drawing 0740G620128 Rev C (C-128).
CE Drawing E-201-802-9.

IMPLEMENTATION

Relief requested as before for volumetric examination. During the next outage, access for remote visual inspection shall be explored.

Item B1.16, B-N-2: INTERIOR ATTACHMENTS AND CORE SUPPORT STRUCTURES

R-15 A. SPARGER AND CORE SPRAY SUPPORTS (801-6A-H)

BASIS STATEMENT

Design of baffle and core spray creates limited visual access to supports.

REMARKS

The steam baffle and emergency core spray sparger are supported by eight brackets welded directly to the vessel cladding. During a normal refueling outage the baffle and sparger are not removed and access to the supports is limited.

ALTERNATIVE INSPECTION OR TEST

See R-12.

REFERENCES

Big Rock FHSR, Fig 4.1.
CE Drawing E-201-794-8.

IMPLEMENTATION

A remote visual will be attempted during the next refueling outage to determine to what extent these supports can be examined.

R-16 B. THERMAL SHIELD SUPPORTS (801-4A-F) AND INLET DIFFUSER BRACKETS (801-7A-F)

BASIS STATEMENT

The six thermal shield brackets are located below the active fuel section of the core. The inlet diffuser brackets are welded directly to the bottom head. Inspection of these supports would require removal of the fuel and core support structures.

REMARKS

Removal of all the fuel and removable internals will be scheduled once every ten years, to provide access to bottom head. At this time a remote visual inspection will be attempted.

ALTERNATIVE INSPECTION OR TEST

A remote visual shall be attempted plus a prestart-up hydro (see R-12).

REFERENCES

CE Drawing E-201-794-8.
Big Rock FHSR, Fig 4.1.

IMPLEMENTATION

Remote visual inspection of these welds shall be attempted when fuel and the removable internals are removed.

R-17 C. CORE SUPPORT BRACKETS (801-5A-D)

BASIS STATEMENT

The core support brackets are located on the bottom head and are welded to the base metal. There is no access to the welds for a visual inspection.

REMARKS

Access to the supports requires removal of the core support plate which requires removal of the fuel and the removable internals. The function of the core support plate is non-load bearing so that a defective weld not affecting the vessel's pressure boundary is not critical unless the support bracket separates from the assembly and becomes loose in the vessel.

ALTERNATIVE INSPECTION OR TEST

See implementation and R-12.

REFERENCES

Big Rock FSHR, Fig 4.1.
GE Drawing 104 R 175, Sheet 2.

IMPLEMENTATION

A remote visual examination shall be attempted when fuel and removable internals are removed to ensure core support plate bracket is in place.

Item B1.18, B-O: CONTROL ROD DRIVE HOUSING

R-18 A. CRD PENETRATION WELD (799-1)

BASIS STATEMENT

No technique is available for volumetric inspection of this weld.

REMARKS

Ultrasonic examination problems are as follows (see Fig 1):

1) Ultrasonic testing from Direction A is not possible because of shrink fit of housing to head and extra cladding next to weld.

2) Ultrasonic testing from Direction B is not possible due to AIR gap present between CRD housing and backing tube.

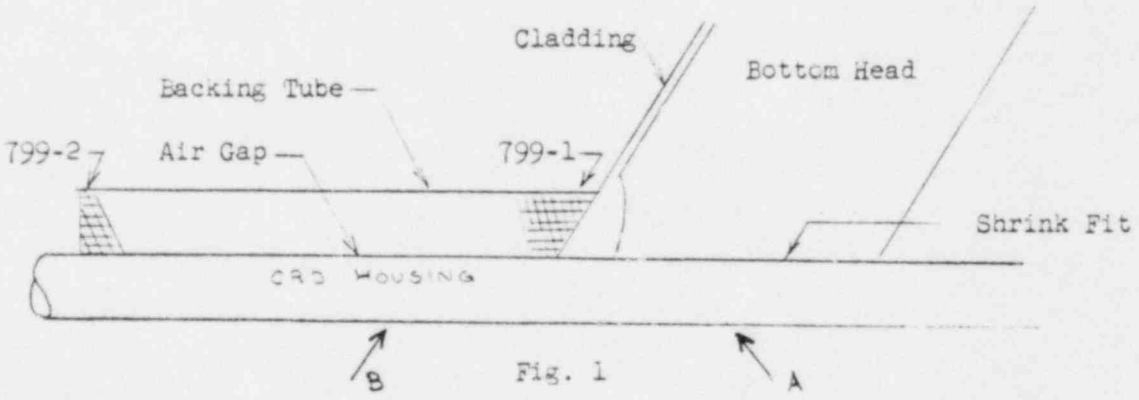


Fig. 1

ALTERNATIVE INSPECTION OR TEST

Hydro (see R-1).

REFERENCES

CE Drawing E-201-799-5.

IMPLEMENTATION

Relief requested.

R-19 B. CONTROL ROD DRIVE HOUSING-TO-FLANGE WELD (798-1; 1-32)

BASIS STATEMENT

This weld is now considered part of B-0 category and must be inspected.

REMARKS

Since this is a pressure retaining weld, it is felt it be included in the plan as Item B1.18, Category B-0.

ALTERNATIVE INSPECTION OR TEST

Not applicable.

REFERENCES

CE Drawing E-201-799-5.

IMPLEMENTATION

Include in In-Service Inspection Master Plan and meet requirement of B-0 category.

HEAT EXCHANGERS

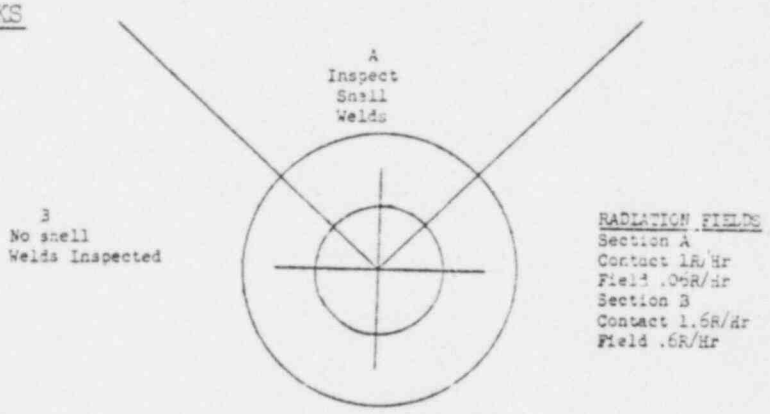
Item B3.1, B-B: LONGITUDINAL AND CIRCUMFERENTIAL STEAM DRUM WELDS

R-20

BASIS STATEMENT

The 170-240 man Rem additional exposure required during the interval to conduct a full Category B-B examination of the steam drum will not provide sufficient additional information over that provided by the partial Category B-B exam proposed to warrant the excess exposure.

REMARKS



Steam Drum

Fig 2

Access to the upper areas of the steam drum, within approximately 30° of arc either side of the top longitudinal center line, is good without extensive scaffolding. The general field is relatively low, approximately 0.06 R/h. The combination of good access and low general field will permit shell weld exams without excessive exposure. Access to the lower portions of the steam drum, within approximately 150° of arc either side of the bottom longitudinal center line, is poor without extensive scaffolding. The general field in this area is approximately ten times (~0.6R/h) that of the upper portion. No shell weld exams will be conducted in the lower portions due to the combination of high fields and poor access; ie, approximately 20 to 25 man-hours will be required (scaffolding, insulation, weld preparation, exam) to perform each exam in the lower areas; with six of the ten longitudinal and circumferential welds being located entirely in this region an additional 170 to 240 man Rem exposure will be required (28-40 man Rem per weld) to perform a complete Category B-B exam over the interval. However, by restricting inspection

to the upper portions of the steam drum, the code exam requirements can be fulfilled on four of the ten welds without accumulating the excess exposure.

ALTERNATIVE INSPECTION OR TEST

The exams in the upper portions will disclose the condition of those areas representative of the most severe service conditions to which the steam drum is exposed and therefore will disclose any incipient general degradation. The hydrostatic examinations performed each refueling outage (pre startup hydro) and the nozzle-to-safe end exams of the risers & downcomers will provide additional indication of the steam drum structural integrity. Failure of any weld in the steam drum shell would be ductile in nature most likely induced by a stress-corrosion or fatigue mechanism and therefore not catastrophic. Such a failure would be readily detected during operations by level indication, dew cell indication, makeup water flow, and/or temperature indication in adequate time for safe shutdown.

REFERENCES

See Master Plan Isometrics A-14 and A-15 and CE Drawing E-230-101-9.

IMPLEMENTATION

Relief is requested to perform a partial Category B-B inspection with exams limited to those welds in the upper areas of the steam drum within 30° of arc either side of the top longitudinal center line.

Item B3.2, B-D: NOZZLE-TO-SHELL WELDS AND NOZZLE INSIDE RADIUS SECTIONS

R-21 A. STEAM DRUM

BASIS STATEMENT

Nozzle-to-shell weld exams and nozzle inside radius section exams will be performed as follows:

Nozzle	Nozzle-to-Shell Weld	Nozzle Inside Radius Section
Manway A-1, A-2	Examine	Examine
Steam Outlet D-1,2,3,4	Examine	Examine
Safety Relief F-1,2,3,4,5,6	Examine	Examine
Decontaminating H	Examine	Examine
Gauge Glass J-1	Examine	Thermal Sleeve Precludes Exam
Reactor Vent L	Examine	Examine
Level M-1,3	Examine	Examine

The Downcomer (B-1,2,3,4), Riser (C-1,2,3,4,5,6), Feedwater (E-1,2), Condensate (G-1,2), Gauge Glass (J-2), and Level (M-2,4) nozzle-to-shell welds and nozzle inside radius sections will not be inspected due to the 595 to 850 man rem additional exposure required during the interval to perform these inspections. The additional information to be gained by performing a full Category B-D examination over that provided by the partial inspection described is not considered sufficient to warrant the additional exposure.

REMARKS

Examinations of nozzle-to-shell welds and nozzle inside radius sections will be performed on only those nozzles located within approximately 30° of arc either side of the top longitudinal center line of the steam drum for the reasons described in Relief Request 18above. Performing these exams on nozzles in the lower portions of the steam drum will require 595 to 850 man rem exposure over the interval (35 to 50 man rem per nozzle for 17 nozzles). These figures are based on the general field in the area of the nozzles. Contact readings on the nozzle-to-shell weld area are on the order of three times that of the general field.

ALTERNATIVE INSPECTION OR TEST

The exams in the upper portions will disclose the conditions of those areas representative of the most severe service conditions to which the steam drum is exposed and therefore will disclose any incipient general degradation. The hydrostatic examinations (pre startup hydro) performed each refueling outage and the nozzle-to-safe end exams of the risers & downcomers will provide additional indication of the steam drum structural integrity. In addition, in the case of the risers and downcomers, the nozzle-to-safe end exams scheduled to be performed represent investigation of an area of the nozzle generally subjected to higher stress levels than those encountered in the nozzle-to-shell welds and inside radius sections. Any failure encountered in the nozzle-to-shell welds or inside radius sections would be ductile in nature most likely induced by a stress-corrosion or fatigue mechanism and therefore not catastrophic. Such a failure would be readily detected during operation, by level indication, dew cell indication, makeup water flow, and/or temperature indication in adequate time for safe shutdown.

REFERENCES

See Master Plan Isometrics A-14 and A-15 and CE Drawing E-230-101-9.

IMPLEMENTATION

Relief is requested to perform a partial Category B-D inspection with exams limited to those nozzle-to-shell welds and inside radius sections of nozzles in the upper areas of the steam drum within 30° of arc either side of the top longitudinal center line.

R-22 B. CLEAN-UP REGENERATIVE HEAT EXCHANGER UNITS A, B, C AND D

BASIS STATEMENT

The channel inlets (A,B,C,D-5) and outlets (A,B,C,D-6) and the shell inlets (A,B,C,D-7) and outlets (A,B,C,D-8) inside radius sections cannot be UT inspected due to geometry.

REMARKS

The nozzle-to-shell welds on the channel inlets and outlets and shell inlets and outlets will be inspected in accordance with the code requirements. These examinations will provide adequate information concerning the general internal condition of the nozzle and shell. The inside radius section exams have been unsuccessfully attempted and will therefore not be scheduled.

ALTERNATIVE INSPECTION OR TEST

See Remarks section.

REFERENCES

See Master Plan Isometrics A-9, 11, 12 and Southwestern Engineering Drawings EM-65149, EM-86924 and DM-86366.

IMPLEMENTATION

Relief is requested to perform a partial Category B-D inspection with exams limited to the nozzle-to-shell welds.

R-23 C. CLEAN-UP NONREGENERATIVE HEAT EXCHANGER UNIT E

BASIS STATEMENT

The channel inlet (E-5) and outlet (E-6) nozzles nozzle-to-shell welds and inside radius sections cannot be UT inspected due to the close proximity of shell circumferential welds and internal baffle arrangements.

REMARKS

The channel inlet and outlet nozzle-to-shell welds will be subjected to visual and surface exams in lieu of the code required volumetric. In addition, the prestart-up hydro will provide a routine monitoring of the condition of these welds.

ALTERNATIVE INSPECTION OR TEST

See Remarks section.

REFERENCES

See Master Plan Isometrics A-10, 11, 13 and Southwestern Engineering Drawings EM-65161, BM-86365.

IMPLEMENTATION

Relief is requested to schedule a partial Category B-D inspection by performing visual and surface exams in lieu of volumetric exam on the channel inlet and outlet nozzle-to-shell welds.

R-24 D. CLEAN-UP DEMIN TANK NOZZLE INSIDE RADIUS SECTIONS

BASIS STATEMENT

The clean-up demin tank nozzles are fabricated by welding squared-ended pipe nipples into the tank shell. The portion of the pipe nipple corresponding to a nozzle inside radius section has a radius of essentially zero and therefore cannot be UT inspected.

REMARKS

The examination of the nozzle inside radius section was incorporated into the code primarily to detect thermal cycling stress degradation. The primary system water flowing into the clean-up demin tank first passes through

the clean-up regenerative and nonregenerative heat exchangers to reduce and stabilize the water temperature to below ~110°F to prevent decomposition of the demineralizer resins. This tank is therefore not subject to significant thermal cycling and inspection of the nozzle inside radius sections is therefore not required.

ALTERNATIVE INSPECTION OR TEST

The prestart-up hydro and the remainder of the Category B-D testing will provide adequate indication of tank structural integrity.

REFERENCES

See Master Plan Isometrics A-18, 19 and Infilco Drawings Y-30-4858-3, M28-4-2.

IMPLEMENTATION

Relief is requested to exempt tank nozzles from inside radius section examination.

E. EMERGENCY CONDENSER NOZZLE INSIDE RADIUS

BASIS STATEMENT

The nozzles are of an "insert type saddle" arrangement. This configuration may not be readily inspectable by ultrasonic methods.

IMPLEMENTATION

Sufficient reference drawings are not available for the inner radius section. Possibilities for inspecting the inner radius will be studied at the next outage. No relief requested.

Item B3.3, B-F: NOZZLE-TO-SAFE END WELDS

R-25 A. STEAM DRUM

BASIS STATEMENT

All steam drum dissimilar metal nozzle-to-safe end welds have been classed as B3.3, B-F welds; these include:

Downcomers	17-MRS-111,112,113,114-1
Risers	14-MRS-101,103,104,105,106-6 and 14-MRS-102-7

These welds are scheduled for examination in accordance with code requirements.

All steam drum similar metal nozzle-to-safe end welds have been classed as B4.4, B4.5 or B4.8, B-J welds since there is no safe end or dissimilar metal weld; these include:

Steam Outlet	8-MSS-101,102,103,104-1*
Feedwater	8-FWS-102,103-3,5*
Safety Relief	3-MSS-107,1,2,3,4,5,6
Condensate	4-BSS-103,104-15
Gauge Glass	1.5-MSS-110-1,22
Rx Vent	1.5-MSS-117-41**
Level	1.5-MSS-111-17
	1.5-MSS-112-17
	4 MSS-111,112-1

*These are nozzle-to-pipe welds.
**The Rx vent line has a dissimilar metal weld 1.5-MSS-117-40, elbow to safe end.

The similar metal nozzle-to-safe end welds are scheduled for examination within the code requirement to exam 25% of the weld joints so as to exclude those similar metal nozzle-to-safe end or pipe welds associated with nozzles within approximately 150° of arc either side of the bottom longitudinal center line of the steam drum as defined in Relief Requests R-1 and R-2.

REMARKS

See remarks for Relief Requests R-20 and R-21

ALTERNATIVE INSPECTION OR TEST

Not applicable.

REFERENCES

BRP ISI Master Plan.

IMPLEMENTATION

No relief is required.

R-26 B. CLEAN-UP REGENERATIVE AND NONREGENERATIVE HEAT EXCHANGER

BASIS STATEMENT

Each clean-up regenerative heat exchanger is fitted with four forged carbon steel nozzles. There are no safe ends on the clean-up regenerative heat exchangers. There are four carbon steel nozzles on the clean-up nonregenerative heat exchangers (two of the four nozzles are unclassed). There are no safe ends on the clean-up nonregenerative heat exchanger.

REMARKS

All similar metal nozzle-to-pipe welds will be examined as B4.5, B-J welds. The carbon steel nozzles to carbon steel piping welds are as follows:

A. Clean-Up Regenerative Hx
3-RCS-101-23,24,25,26,29,30,31 and 32

B. Clean-Up Nonregenerative Hx
3-RCS-101-37,38

ALTERNATIVE INSPECTION OR TEST

Not applicable.

REFERENCES

See Master Plan ISOs A-9,10,11,12,13,32,37 and Southwestern Engineering Co Drawings EM-65149, EM-86924, DM-86366, EM-65151 and BM-85365.

IMPLEMENTATION

No relief required.

R-27 C. CLEAN-UP DEMINERALIZER TANK

BASIS STATEMENT

The clean-up demineralizer tank is fitted with four carbon steel nozzles (A-234 WPB). There are no safe ends on the clean-up demineralizer tank nozzles.

REMARKS

All similar metal nozzle-to-pipe welds will be examined as B4.5, B-J welds. The carbon steel nozzle-to-pipe welds are:

3-RCS-101-89
3-RCS-110-5
2-RCS-110-5
2-RCS-111-1

There are no dissimilar metal nozzle-to-pipe welds adjacent to the clean-up demineralizer tank.

ALTERNATIVE INSPECTION OR TEST

Not applicable.

REFERENCES

See Master Plan ISOs A-18, A-19, A-35, A-36 and A-43 and Infilco Drawing Y-30-4853-3.

IMPLEMENTATION

No relief required.

Item B3.6, B-H: INTEGRALLY WELDED STEAM DRUM SUPPORTS

R-28

BASIS STATEMENT

The 392-560 man Rem additional exposure required, during the interval, to conduct a full Category B-H examination of the steam drum will not provide sufficient additional information over that provided by the partial Category B-H exam proposed to warrant the excess exposure.

REMARKS

Access to the upper areas of the steam drum, within approximately 30° of arc either side of the top longitudinal center line, is good and supports T2-1,2,3 and 4 are inspectable.

Access to the lower portions of the steam drum coupled with high radiation fields, as stated previously for the steam drum, precludes examination of the following steam drum supports:

T1-1,2,3,4
V1-1,2,3,4
V2-1,2,3,4
H1,2

ALTERNATIVE INSPECTION OR TEST

See Remarks above.

REFERENCES

See Master Plan ISO A-16 and CE Drawings E-230-101-9 and D-230-213.

IMPLEMENTATION

Relief requested to perform a partial Category B-H inspection with exams limited to those welds in the upper areas of the steam drum within 30° arc either side of the top longitudinal center line weld.

PIPING PRESSURE BOUNDARY

Item B4.1, B-F: SAFE-END TO PIPING AND SAFE-END IN BRANCH PIPING

R-29 A. 6-SCS-101-7

BASIS STATEMENT

Pipe weld is not physically accessible for nondestructive testing due to plant design.

REMARKS

See Remarks section for 6-SCS-101-1 through 12 (R-32).

ALTERNATIVE INSPECTION OR TEST

A hydrostatic test (pre start up hydro) is performed before each start-up at 1.1 times the operating pressure. This is more conservative than the prestart-up leak test at operating pressure as required by the code (1WB-5221). Inspection of the remaining accessible portion of this line will provide some indication of the piping integrity.

REFERENCES

See Master Plan ISO A-28.

IMPLEMENTATION

Relief requested to examine 6-SCS-101-7 solely hydrostatically during prestart hydro.

R-30 B. DISSIMILAR METAL SOCKET WELDS

BASIS STATEMENT

Section XI does not address the subject of dissimilar metal socket welds. Plant piping contains 11 dissimilar metal socket welds. Meaningful volumetric examination of socket welds is not possible.

REMARKS

The following dissimilar metal socket welds have been designated B4.1/B-F and will be inspected by surface examination:

2-RCS-101-64
2-RDS-111-8,9
1.5-RDS-112-5,6
1.5-RDS-113-5,6
1.5-RDS-114-5,6
1.5-RDS-115-5,6
1.5-MSS-117-41

ALTERNATIVE INSPECTION OR TEST

See Implementation.

REFERENCES

See ISI Master Plan.

IMPLEMENTATION

Relief requested to examine dissimilar metal socket welds by surface examination.

R-31 C. SIMILAR METAL SAFE-END TO PIPE WELDS

BASIS STATEMENT

The following safe-end to pipe welds join two stainless steel components and are therefore similar metal welds. These safe-end to pipe welds are therefore classed as B4.4 or B4.5, B-J welds rather than B4.1, B-F welds:

20-MRS-121-20	B4.4	B-J
20-MRS-122-20	B4.4	B-J
14-MRS-101-2	B4.4	B-J
14-MRS-102-2	B4.4	B-J
14-MRS-103-2	B4.4	B-J
14-MRS-104-2	B4.4	B-J
14-MRS-105-2	B4.4	B-J
14-MRS-106-2	B4.4	B-J
8-SCS-101-1	B4.4	B-J
3-LPS-102-47	B4.5	B-J
2-MSS-121-1	B4.8	B-J
2-MSS-131-1	B4.8	B-J
2-MSS-134-1	B4.8	B-J

REMARKS

All of these welds with the exception of the 14" main recirc lines are inaccessible for examination. See the following relief requests for the basis of this statement:

1) 20" Recirc Welds R-4

2) 3" 1) Core Spray R-5
2) Poison Inlet

3) 8" Shutdown Unloading Weld R-6

4) 3" Instrument Welds R-7

ALTERNATIVE INSPECTION OR TEST

Prestart-up hydro (see R-1).

REFERENCES

CE Drawings E-201-795-5 and E-201-796-5.
Big Rock ISI Master Plan.

IMPLEMENTATION

Reclass the welds discussed above from B4.1, B-F to B4.4 or B4.5, B-J and relief requested on the safe-end to pipe welds as discussed in the Remarks section.

Item B4.4, B-J: CIRC AND LONG PIPE WELDS IN PIPING FOUR INCHES AND GREATER

R-32 A. 6-SCS-101-1 THROUGH 12; 3-CSS-101-17 THROUGH 19

BASIS STATEMENT

Physical plant layout precludes nondestructive testing of 6-SCS-101-1 through 12 and 3-CSS-101-17 through 19.

REMARKS

Welds 1 through 6 of the 6" shutdown cooling system are imbedded in the concrete reactor shield wall and are therefore not accessible for nondestructive testing. Welds 7 through 12 are inaccessible for nondestructive testing due to plant layout. The portion of Line 6-SCS-101 containing Welds 7 through 12 is greater than 35 feet above the floor. This line is at least 15 feet from all walls except the concrete reactor shield. It is not possible to erect scaffolding or to utilize a ladder to access this line due to the sloping floor at the base of the containment sphere. Access from above is precluded due to a straight, uninterrupted drop of 35 feet from the steam drum environs.

Access to 3-CSS-101, Welds 17 through 19 is possible only through small crawl spaces made available by removing blocks of shielding material. An additional 99-117 man Rem exposure is required to prepare for conduct and clean-up after the examinations.

ALTERNATIVE TEST OR INSPECTION

See R-29.

REFERENCES

BRP ISI Master Plan.
ISOs A-28 and A-71.

IMPLEMENTATION

Relief requested to examine this portion of 6-SCS-101 and 3-CSS-101 solely by hydrostatic testing during the prestart hydro.

Item B4.8, B-J: SOCKET WELDS

R-33 A. MAIN STEAM

BASIS STATEMENT

Small diameter piping directly adjacent to the reactor vessel is not inspectable due to a 2 R/h radiation field.

REMARKS

99-117 man Rem additional exposure are required to prepare for conduct and clean-up after the following examinations:

2-MSS-121-2,3
2-MSS-131-2,3
2-MSS-134-7,8

Access to the welds is possible only through small crawl spaces made available by removing blocks of shielding material.

ALTERNATIVE INSPECTION OR TEST

See R-29.

REFERENCES

See Master Plan ISOs A-56, A-59, A-61 and A-71.

IMPLEMENTATION

Relief requested to examine the above-listed welds solely by hydrostatic testing.

R-34 B. 3-LPS-102-42 THROUGH 47

BASIS STATEMENT

Welds are behind reactor shield wall and are inaccessible for examination.

REMARKS

The portion of Line 3-LPS-102 from Welds 42 to 47 located behind the reactor shield wall is not accessible. Plant design precludes access to this area.

ALTERNATIVE INSPECTION OR TEST

See R-29.

REFERENCES

See Master Plan ISO A-26.

IMPLEMENTATION

Relief requested to examine this portion of Line 3-LPS-102 solely by hydrostatic testing.

R-35 C. 2-CRD-101-22

BASIS STATEMENT

Socket weld is partially imbedded in concrete shield wall and is inaccessible for examination.

REMARKS

Weld 2-CRD-101-22 is half imbedded in concrete and is inaccessible for inspection.

ALTERNATIVE INSPECTION OR TEST

See R-29.

REFERENCES

See Master Plan ISO A-74.

IMPLEMENTATION

Relief requested to examine this weld solely by hydrostatic testing.

Item B4.9, B-K-1: INTEGRAL PIPE SUPPORTS

R-36 A. MAIN RECIRC

BASIS STATEMENT

Integrally welded pipe supports are not accessible for volumetric and surface examination due to plant design and high radiation.

REMARKS

The following supports are inaccessible for examination:

14-MRS-103-3PR-1
14-MRS-105-3PR-1

These supports are located 30 feet above the lower deck area of containment. The floor in this area is curved due to containment configuration and the use of scaffolding or ladders is not possible. Access from above is also not possible due to plant layout.

The general radiation field in this area is 0.6 R/h and 11-13 man Rem exposure per support per interval would be absorbed solely in preparation for the exam, ie, weld prep and insulation removal, performing the exam and post exam clean-up, provided access to the supports were possible, which is not the case.

ALTERNATIVE INSPECTION OR TEST

For the case in which a weld defect would affect the integrity of the piping pressure boundary, the prestart-up hydro serves as an alternative inspection. A remote visual examination will be made of the support to try and determine its integrity as a support.

REFERENCES

See Master Plan ISO A-64.

IMPLEMENTATION

Relief requested to examine these supports solely by hydrostatic testing.

R-37 B. LIQUID POISON SYSTEM

BASIS STATEMENT

Integrally welded pipe supports are not accessible for volumetric and surface examination due to plant design and high radiation.

REMARKS

The following supports are not accessible for examination:

3-LPS-102-21PR-1,2

The supports are located 50 feet above the lower deck area of containment. The floor in this area is curved due to containment configuration and the use of scaffolding is not possible. Access from above is also not possible.

The general radiation field is 0.6 R/h and 11-13 man Rem exposure per support per interval would be absorbed before, during and after examining these welds, provided access to the support were possible, which is not the case.

ALTERNATIVE INSPECTION OR TEST

See R-36.

REFERENCES

See Master Plan ISO A-26.

IMPLEMENTATION

Relief requested to examine these supports solely by hydrostatic testing.

R-38 C. RDS

BASIS STATEMENT

Pipe lugs are inaccessible for volumetric and surface examination due to configuration of pipe support; pipe lugs are nonload bearing.

REMARKS

Pipe lugs are covered by pipe support and are not accessible for volumetric and surface examination as shown below. Support is accessible for VT.

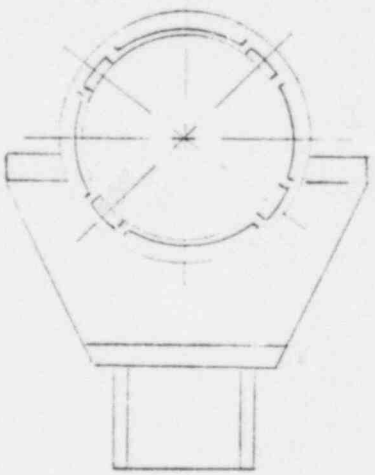


FIG 3

Support has lugs welded on pipe side of support. Lugs are welded to pipe. Pipe lugs do not contact support and serve solely to resist torsional movement. Pipe lugs are nonload bearing.

This form of support configuration is applicable to the following components:

12-RDS-101-3-PL-1,2,3,4,5,6,7,8
12-RDS-101-13-PL-1,2,3,4

ALTERNATIVE INSPECTION OR TEST

Visual inspection and hydro (see R-36).

REFERENCES

See Master Plan ISO A-82.

IMPLEMENTATION

Relief requested to examine the above welds by VT examination.

Item B4.10, B-K-2: SUPPORT COMPONENTS - STRUCTURAL

R-39

BASIS STATEMENT

The following support components are inaccessible for visual examination:

6-SCS-101-6-PS
8-SCS-101-8PR, 10PR

REMARKS

The access problem to this line is discussed in R-32.

ALTERNATIVE INSPECTION OR TEST

Refer to Implementation.

REFERENCES

See Master Plan ISO A-28.

IMPLEMENTATION

Relief requested for direct visual examination. A remote visual inspection shall be performed to determine support capabilities.

PUMP PRESSURE BOUNDARY

Item B5.6, B-L-1: PUMP CASING WELDS

BASIS STATEMENT

The location of pump casing welds for the clean-up pump and main recirculation pump cannot be determined from drawings located.

IMPLEMENTATION

Until manufacturer's drawing can be obtained that show locations of the casing welds or the welds are located by observation, no relief is requested. However, it is considered probable that volumetric examination of the casing welds will not be possible.

VALVE PRESSURE BOUNDARY

Item B6.7, B-M-2: VALVE INTERNAL EXAMINATION

R-40

BASIS STATEMENT

Main recirculation pump discharge and butterfly valve are not isolable from reactor and are therefore not inspectable.

REMARKS

The following valves are not isolable from the reactor:

Line	Valve Number
5-MRS-131	MO-N002A
5-MRS-132	MO-N002B
20-MRS-121	MO-N001A
20-MRS-122	MO-N001B
20-MRS-121	MO-N006A
20-MRS-122	MO-N006B

Examination of the above valves requires complete draining of the reactor vessel which is not possible.

ALTERNATIVE INSPECTION OR TEST

A hydrostatic test (prestart-up hydro) conducted at 1.1 times the operating pressure will serve as an alternate test. The inspection of other valves and the results will also give an indication of the condition of the internals of these valves.

REFERENCES

See Master Plan ISOs A-66, A-67, A-68 and A-69.

IMPLEMENTATION

Relief requested to examine these valves by hydrostatic testing and to exempt these valves from the VT requirements of B-M-2.

TABLE III-1
RELIEF REQUESTS
BIG ROCK POINT NUCLEAR PLANT
CLASS 1 COMPONENTS

Area of Relief	Sect XI Category	Component	Function	Relief Requested	Alternative Test and Examination	Implementation	Relief Request
Bl.1	B-A	Vessel Core Region	Primary Pressure Boundary	Longitudinal and Circumferential Seam Welds Inaccessible for Volumetric Exams	a) Pre-Startup Hydro[1] b) Remainder of Bl.1 Testing	a) Tech Spec b) Master Plan	R-1
Bl.1	B-A	Vessel Shell Region	Primary Pressure Boundary	Longitudinal and Circumferential Seam Welds Inaccessible for Volumetric Exams	a) Pre-Startup Hydro[1] b) Remainder of Bl.1 Testing	a) Tech Spec b) Master Plan	R-2
Bl.1	B-A	Vessel Bottom Head	Primary Pressure Boundary	Longitudinal Seam Welds Generally Inaccessible for Volumetric Exams	a) Pre-Startup Hydro[1] b) Remainder of Bl.1 Testing	a) Tech Spec b) Master Plan	R-3
Bl.4	B-D	20" Recirc Nozzles (796-1A-B)	Primary Pressure Boundary	Nozzle-to-Vessel Weld and Nozzle Inside Radius Are Inaccessible for Volumetric Examination	a) Pre-Startup Hydro[1] b) Remainder of Bl.4 Testing	a) Tech Spec b) Master Plan	R-4
Bl.4	B-D	3" Nozzles 1) Core Spray (795-6) 2) Poison Inlet (796-6)	Primary Pressure Boundary	Nozzle-to-Vessel Weld and Nozzle Inside Radius Are Inaccessible for Volumetric Examination	a) Pre-Startup Hydro[1] b) Remainder of Bl.4 Testing	a) Tech Spec b) Master Plan	R-5
Bl.4	B-D	8" Shutdown Unloading Nozzle (795-15)	Primary Pressure Boundary	Nozzle-to-Vessel Weld and Nozzle Inside Radius Are Inaccessible for Volumetric Examination	a) Pre-Startup Hydro[1] b) Remainder of Bl.4 Testing	a) Tech Spec b) Master Plan	R-6
Bl.4	B-D	3" Instrument Nozzles (955-1A-D)	Primary Pressure Boundary	Nozzle-to-Vessel Weld and Nozzle Inside Radius Are Inaccessible for Volumetric Examination	a) Pre-Startup Hydro[1] b) Remainder of Bl.4 Testing	a) Tech Spec b) Master Plan	R-7

Note: All hydrostatic tests referenced in the body of this report are the pre-startup hydros listed in the alternative test and examination column.

TABLE III-1
RELIEF REQUESTS
BIG ROCK POINT NUCLEAR PLANT
CLASS 1 COMPONENTS

Area of Relief	Sect XI Category	Component	Function	Relief Requested	Alternative Test and Examination	Implementation	Relief Request
B1.6	B-F	20" Recirc Nozzles (796-1A,B)	Primary Pressure Boundary	Nozzle-to-Safe Ends Inaccessible for UT	a) Prestart-Up Hydro[1] b) Remainder of B1.6 Testing	a) Tech Spec b) Master Plan	R-8
B1.6	B-F	8" Shutdown Unloading Nozzle (795-15)	Primary Pressure Boundary	Nozzle-to-Safe Ends Inaccessible for UT	a) Prestart-Up Hydro[1] b) Remainder of B1.6 Testing	a) Tech Spec b) Master Plan	R-9
B1.6	B-F	3" Nozzles 1) Core Spray (795-6) 2) Poison Inlet (796-6)	Primary Pressure Boundary	Nozzle-to-Safe Ends Inaccessible for UT	a) Prestart-Up Hydro[1] b) Remainder of B1.6 Testing	a) Tech Spec b) Master Plan	R-10
B1.6	B-F	3" Instrument Nozzles (955-1A-D)	Primary Pressure Boundary	Nozzle-to-Safe Ends Inaccessible for UT	a) Prestart-Up Hydro[1] b) Remainder of B1.6 Testing	a) Tech Spec b) Master Plan	R-11
B1.12	B-H	Integrally Welded Vessel Supports (802-6A-L)	Primary Pressure Boundary	Supports Are Inaccessible for Visual Examination	a) Prestart-Up Hydro[1]	a) Tech Spec	R-12
B1.12	B-H	Vessel Hanger Lugs (807-13 A-D)	Primary Pressure Boundary	Lugs Inaccessible for UT	a) Prestart-Up Hydro b) Remote Visual	a) Tech Spec b) Master Plan	R-13
B1.12	B-H	Stabilizer Brackets (802-12 A-D)	Primary Pressure Boundary	Brackets Inaccessible for UT	a) Prestart-Up Hydro b) Remote Visual	a) Tech Spec b) Master Plan	R-14
B1.16	B-N-2	Sparger and Core Spray Supports (801-6A-H)	Primary Pressure Boundary	Remote Visual as Possible	None - Inspected as Possible	a) 40-Year Plan	R-15
B1.16	B-N-2	Thermal Shield Supports and Inlet Diffuser Brackets (801-4A-F) (811-7A-F)	Primary Pressure Boundary	Remote Visual as Possible When Core Support Structure is removed.	None - Inspected as Possible	a) 40-Year Plan	R-16

TABLE III-1
RELIEF REQUESTS
BIG ROCK POINT NUCLEAR PLANT
CLASS 1 COMPONENTS

Area of Relief	Sect XI Category	Component	Function	Relief Requested	Alternative Test and Examination	Implementation	Relief Request
B1.16	B-N-2	Core Support Brackets (801-5A-D)	Primary Pressure Boundary	Remote Visual as Possible When Core Structure is removed.	None - Inspect as Possible	a) 40-Year Plan	R-17
B1.18	B-O	CRD Housing Weld 799-1	Primary Pressure Boundary	No Volumetric Technique Available for Volumetric Inspection of Weld 799-1	a) Prestart-Up Hydro[1]	a) Tech Spec	R-18
B1.18	B-O	CRD Housing-to-Flange Weld 798-1	Primary Pressure Boundary	Include in Inspection Plan	None - Examine as Required	a) 40-Year Plan	R-19
B3.1	B-B	Steam Drum Longitudinal Welds 101-2,3,5,6,8,9	Primary System Pressure Boundary	Exempt From Volumetric Exam Due To Accessibility and Exposure	a) Prestart-Up Hydro[1] b) Remainder of B3.1 Testing	a) Tech Spec b) Master Plan	R-20
B3.1	B-B	Steam Drum Circumferential Welds 101-1,4,7,10	Primary System Pressure Boundary	Perform Volumetric Only in Upper 60° Arc of Drum Circumference	a) Prestart-Up Hydro[1] b) Remainder of B3.1 Testing	a) Tech Spec b) Master Plan	R-20
B3.2	B-D	Steam Drum and Inside Radius Sections Downcomers B-1,2,3,4 Risers C-1,2,3,4,5,6 Feedwater E-1,2 Condensate G-1,2 Gauge Glass J-2 Rx Vent L Level M-1,3	Primary System Pressure Boundary	Exempt From Volumetric Exam Due To Accessibility and Exposure	a) Prestart-Up Hydro[1] b) Remainder of B3.2 Testing c) Remainder of B3.3 Testing	a) Tech Spec b) Master Plan c) Master Plan	R-21
B3.2	B-D	Clean Up Regenerative Heat Exchanger Units A, B, C, D Nozzle Inside Radius Sections on Nozzles A,B,C,D-5; A,B,C,D-6; A,B,C,D-7; A,B,C,D-8	Primary System Flow Regulation	Exempt From Volumetric Exam Due To Geometry	a) Prestart-Up Hydro[1] b) Remainder of B3.2 Testing	a) Tech Spec b) Master Plan	R-22

TABLE TII-1
RELIEF REQUESTS
BIG ROCK POINT NUCLEAR PLANT
CLASS 1 COMPONENTS

Area of Relief	Sect XI Category	Component	Function	Relief Requested	Alternative Test and Examination	Implementation	Relief Request
B3.2	B-D	Clean Up Nonregenerative Heat Exchanger Unit E Nozzle-to-Shell Welds E-5 and E-5	Auxiliary System Temperature Control	Perform Visual and Surface in Lieu of Volumetric	a) Visual and Surface[1] b) Prestart-Up Hydro	a) Master Plan b) Tech Spec	R-23
B3.2	B-D	Clean Up Nonregenerative Heat Exchanger Unit E Nozzle Inside Radius Section	Auxiliary System Temperature Control	Exempt From Examination	a) Prestart-Up Hydro[1] b) Remainder of B-D Testing	a) Tech Spec b) Master Plan	R-23
B3.2	B-D	Clean Up Demin Tank Nozzle Inside Radius Sections	Auxiliary System Water Purification	Exempt From Examination	a) Prestart-Up Hydro[1] b) Remainder of B-D Testing	a) Tech Spec b) Master Plan	R-24
B3.2	B-D	Emergency Condenser Nozzle Inside Radius Sections	Primary System Emergency Heat Removal	Investigating Feasibility of Exams	a) None - Examine as Scheduled	a) 40-Year Plan	
B3.3	B-F	Steam Drum Similar Metal Nozzle-to-Safe End Welds	Primary Pressure Boundary	B3.3, B-F Dissimilar Metal Welds Will Be Examined as Scheduled. B3.3, B-F Similar Metal Welds Shall Be Reclassified as B4.4, B4.5 or B4.8, B-J Welds. These Similar Metal Weld Inspections Shall Be Chosen on the Basis of Accessibility and Low Radiation Fields.	a) None - Examine as Required	a) 40-Year Plan	R-25
B3.3	B-F	Clean Up Regen and Nonregen Heat Exchanger Nozzle-to-Safe End Welds	Primary Pressure Boundary	None - No B3.3, B-F Welds	a) None - Examine To Requirements of B3.3, B-F Welds	a) 40-Year Plan	R-26
B3.3	B-F	Clean Up Demin Tank Nozzle-to-Safe End Welds	Primary Pressure Boundary	None - No B3.3, B-F Welds	a) None - Examine To Requirements of B3.3, B-F Welds	a) 40-Year Plan	R-27

TABLE III-1
RELIEF REQUESTS
BIG ROCK POINT NUCLEAR PLANT
CLASS 1 COMPONENTS

Area of Relief	Sect XI Category	Component	Function	Relief Requested	Alternative Test and Examination	Implementation	Relief Request
B3.6	B-H	Integrally Welded Steam Drum Supports	Primary Pressure Boundary	Partial Category B-H Inspection	a) None - Examine as Required	a) 40-Year Plan	R-28
B4.1	B-F	Socket Welds 6-SCS-101-7 Dissimilar Metal	Primary Pressure Boundary	No Access for Inspection	a) Prestart-Up Hydro[1]	a) Tech Spec	R-29
B4.1	B-F	Socket Weld Dissimilar Metal	Primary Pressure Boundary	Inspect With Surface Exams Only	a) Prestart-Up Hydro[1]	a) Tech Spec	R-30
B4.1	B-F	Reactor Vessel Safe End to Piping Welds Dissimilar Metal	Primary Pressure Boundary	Not Accessible for Volumetric Inspection. Reclass as B4.4 or B4.5 B-J Welds.	a) Prestart-Up Hydro[1] b) Remainder of B4.1 Testing on Reactor Vessel	a) Tech Spec b) 40-Year Plan	R-31
B4.4	B-J	Butt Welds on 6-SCS-101 Line (Welds 1-12)	Primary Pressure Boundary	Inaccessible for Inspection	a) Prestart-Up Hydro[1] b) Remaining Butt Welds on SCS Line	a) Tech Spec b) 40-Year Plan	R-32
B4.5	B-J	Socket Welds 2-MSS-121-2,3 2-MSS-131-2,3 2-MSS-134-7,8 3-CSS-101-17,18,19	Primary Pressure Boundary	Welds Inaccessible for Volumetric Inspection	a) Prestart-Up Hydro[1] b) Remainder of B-J Welds on Same Lines	a) Tech Spec b) 40-Year Plan	R-33
B4.5	B-J	Socket Welds 3-LPS-102 Welds 42 to 47	Primary Pressure Boundary	Welds Inaccessible for Volumetric Inspection	a) Prestart-Up Hydro[1] b) Remainder of B-J Welds on Same Lines	a) Tech Spec b) 40-Year Plan	R-34
B4.5	B-J	Socket Welds 2-CRD-101-22	Primary Pressure Boundary	Welds Inaccessible for Volumetric Inspection	a) Prestart-Up Hydro[1] b) Remainder of B-J Welds on Same Lines	a) Tech Spec b) 40-Year Plan	R-35
B4.9	B-K-1	Integral Pipe Supports 14-MRS-103-3PR-1 14-MRS-105-3PR-1	Primary Pressure Boundary	Supports Are Inaccessible for Inspection	a) Prestart-Up Hydro[1] b) Other Supports on Same Lines	a) Tech Spec b) 40-Year Plan	R-36

TABLE III-1
RELIEF REQUESTS
BIG ROCK POINT NUCLEAR PLANT
CLASS 1 COMPONENTS

Area of Relief	Sect XI Category	Component	Function	Relief Requested	Alternative Test and Examination	Implementation	Relief Request
B4.9	B-K-1	Integral Pipe Supports	Primary Pressure Boundary	Supports Are Inaccessible for Inspection	a) Prestart-Up Hydro[1] b) Other Supports on Same Lines	a) Tech Spec b) 40-Year Plan	R-37
B4.9	B-K-1	Integrally Welded Supports 12-RDS-101-3-PL-1 Through 8 and 13-PL-1 Through 4	Primary Pressure Boundary	Volumetric Exam Not Possible. Substitute a Visual.	a) Prestart-Up Hydro[1] b) Visual Inspection	a) Tech Spec b) 40-Year Plan	R-38
B4.11	B-K-2	Support Components - Structural 6-SCS-101-6-PS 8-SCS-101-8PR,10PR	Primary Pressure Boundary	Not Accessible for Direct Visual Exam	a) Remote Visual Inspection	a) 40-Year Plan	R-39
B6.7	B-M-2	Valve Internals MO-N002A, MO-N002B MO-N001A, MO-N001B MO-N006A, MO-N006B	Primary Pressure Boundary	Valves Cannot Be Isolated From Reactor so Inspection Not Possible	a) Prestart-Up Hydro[1]	a) Tech Spec	R-40
B5.6	B-L-1	Pump casing welds	Primary Pressure Boundary	Feasibility of volumetric inspection is being studied	None/Inspect as Possible	40-Year Plan	

[1]The conduct of the prestart-up hydrostatic test at a pressure of 1500 $^{+50}_{-0}$ psia (>1.1 operating pressure) will be accompanied by a visual Examination, VT-to detect the presence of any through-wall leakage, if any, of those pressure-retaining components where access for volumetric examination is impractical This test and examination provides assurance that any structural degradation that may have occurred in service will be detected and corrective actions taken prior to plant start-up. Any structural degradation that may develop while the component is in service is further monitored by fluid-level indications, dew-cell indications, makeup water flow and/or temperature indications as provided by instrumentation in the Big Rock Point Nuclear Plant for timely detection and safe shutdown of the plant.

SECTION 4.0

CLASS 2 AND 3

RELIEF REQUESTS

CLASS 2

Paragraph IWC-1210: The Examination requirements of IWC small apply to Class 2 Pressure Retaining Components (and their components).

R-41 A. CLASS 2 PIPING AT CONTAINMENT PENETRATIONS BETWEEN CLASS 3 OR NON-CLASSED (NONNUCLEAR) COMPONENTS.

BASIS STATEMENT

Implementation of this requirement is not appropriate at certain containment penetrations.

REMARKS

The following penetrations are affected by this statement:

Penetration No.	Piping on Either Side of Penetration
H-12	Class 3
H-13	Class 3
H-14	Nonnuclear
H-19	Nonnuclear
H-20	Nonnuclear
H-25	Nonnuclear
H-31	Nonnuclear

The classification of the Class 2 portion of the piping is governed by the leak-tightness requirement associated with the containment barrier, while the Class 3 or nonnuclear classification is dictated by system function under pressure and temperature operating conditions. In all cases, the system operating conditions impose loading on the components that are more severe than the external loadings imposed when the containment functions are tested.

The system pressure tests and visual examinations of Subsection IWD of the ASME Code are intended to detect any service-induced degradation resulting from the more severe operational loads that the components are expected to sustain over its service lifetime. Accordingly, the system function governs the classification under which the applicable rules of ASME Section XI Code must be applied. This position is consistent with the revision of IWA-1300 (f) approved by ASME Subcommittee on Nuclear Inservice Inspection (Agenda No ISI-77-14) which states: "The portion of piping penetrating a containment vessel required by Section III to be constructed to Class 1 or 2 rules, and which may differ from the classification of the balance of the system, need not affect the overall system

REMARKS (Cont'd)

classification which determines the applicable rules of this Section." Accordingly, where the components beyond the containment penetration areas delineated on the referenced P and I diagrams are classified as ASME Class 3, the Class 2 process piping and associated pressurization pipe will be subjected to the requirements of Subsection IWD of ASME Section XI Code. Where the components beyond the containment penetration area are classified nonnuclear class, the Class 2 components and associated pressurization pipe will not be subject to any Section XI requirements.

ALTERNATIVE INSPECTION OR TEST

Since the proposed examinations or the Class 2 portions of the above referenced systems are in accordance with code intent, no alternate inspections or tests are considered necessary. This request for relief is submitted primarily to obtain confirmation of IWA-1300 (f), which is expected to be issued in a forthcoming addenda to the ASME Section XI Code.

REFERENCES

Penetration	GE Drawing	Coordinates
H-12	0740G40111	O-11
H-13	0740G40111	H-11
H-14	0740G40125	----
H-19	0740G40125	----
H-20	0740G40133	M-20
H-25	0740G40133	C-20
H-31	0740G40121	G-21

IMPLEMENTATION

Inspection as required by code.

R-42 B. CLASSIFICATION OF CLASS 2 VENT AND DRAIN LINES BEYOND THE NORMALLY CLOSED VALVE

BASIS STATEMENT

Implementation of this requirement is not appropriate for certain piping.

REMARKS

The portion of the non-regen and regen heat exchanger channel vents and drains and shell vents and drains beyond the second normally closed valve, would generally be classified as Class 3. Since these vents and drains could have been classified as Class 3 and since the venting or draining does not relate to any of the functions listed in Table

REMARKS (Cont'd)

IWD-2500-1, These portions of the vents and drains will be classified as nonnuclear for the purposes of In-Service Inspection, And will not be subjected to examination under Section XI of the ASME Code.

ALTERNATIVE INSPECTION OR TEST

None.

REFERENCE

Bechtel Drawing 0740G40107 Revision M.

IMPLEMENTATION

Not applicable.

R-43 C. CLASSIFICATION OF LINES NOT CONTAINING WATER STEAM OR RADIOACTIVE MATERIALS AS NONNUCLEAR.

BASIS STATEMENT

These systems are not within the scope of IWA-1100 (ie, contain other than water, steam or radioactive materials). Hence, all these components are excluded from any In-Service Inspection requirements.

REMARKS

The two system under consideration for this exemption are:

1) Nitrogen charging lines for the liquid poison system.

2) The control air lines for control valves CV 4096 and CV 4097 in the heating and ventilation system.

IWA-1100 of the Section XI ASME Code defines the scope of the code rules with respect to those Class 1, 2 and 3 components (and their supports) in nuclear power plant systems that are subject to In-Service Inspections and In-Service Testing.

This code paragraph has been clarified by ASME subcommittee on Nuclear In-Service Inspection (Section XI), as approved by committee action on December 5, 1977, to read:

"IWA-1100 scope."

"This division provides rules and requirements for In-Service Inspection of Class 1, 2 and 3 components (and their supports) and In-Service Testing of pumps and valves.

REMARKS (Cont'd)

"These rules and requirements are presently only applicable to systems containing water, steam, or radioactive materials in light-water cooled nuclear plants."

ALTERNATE INSPECTION OR TEST

None.

REFERENCES

Bechtel Drawings: 1) 0740G40125 Revision N
2) 0740G40107 Revision M

IMPLEMENTATION

Not applicable.

CLASS 3

Table IWD, Items D1.2, 2.2, 3.2,: INTEGRALLY WELDED SUPPORTS

R-44 FREQUENCY OF EXAMINATION

BASIS STATEMENT

Table IWD requires visual examinations (VT-3) to be performed during each inspection period. Implementation of this requirement is impractical in view of the examination requirements for Class 1 and Class 2 components.

REMARKS

The Section XI Code requirements for Class 1 component supports (Table IWB-2500, Categories B-H and B-K-1) and Class 2 component supports (Table IWC-2520, Categories C-C and C-E-1) require less frequent examinations of component supports than the frequency required for supports for Class 3 components. The requirements can be summarized as follows:

Class	Table	Category	Examination Frequency
1	IWB-2500	B-H	100% for the first 2 intervals
		B-K-1	25% in each interval
2	IWC-2520	C-C	100% in each interval
		C-E-1	100% in each interval
3	IWD-2500-1	D-A, D-B & D-C	100% in each interval

REMARKS (Cont'd)

In view, then, of the less frequent examination requirements for Class 1 and 2 component supports and nonintegrally welded supports, examination of 100% of the integrally welded supports during each interval in lieu of examination of 100% during each period should be adequate.

ALTERNATE INSPECTIONS OR TESTS

None.

REFERENCES

Big Rock Point ISI 40-Year Master Plan, Class 3 examinations.

IMPLEMENTATION

Not applicable.

CLASS 1, 2 AND 3

TABLE IWB-2500 (CATEGORY B-P), TABLE IWC-2520 (CATEGORY C-H), AND TABLE IWC-2500-1 (ITEMS D1.2, D2.1 and D3.1)

R-45 SUBSTITUTION OF SYSTEM HYDROSTATIC TESTING FOR SYSTEM PRESSURE TESTING.

BASIS STATEMENT

Table IWB-2500 (Category B-P), Table IWC-2520 (Category C-H) and Table IWD-2500-1 (Items D1.2, D2.1 and D3.1) require the performance of both system pressure tests and system hydrostatic tests during the third inspection period. Implementation of the requirement for system pressure tests during the third inspection period is inappropriate in view of the performance of a system hydrostatic test.

REMARKS

In view of the higher pressures applied during the system hydrostatic test, it represents a more severe test than the system pressure tests which are required by the code; therefore, performance of the system hydrostatic test would make performance of the system pressure test redundant and unnecessary.

ALTERNATIVE INSPECTION OR TEST

None.

REFERENCES

Big Rock Point 40-Year ISI Master Plan.

IMPLEMENTATION

Not applicable.

TABLE IV-1

RELIEF REQUESTS
BIG ROCK POINT NUCLEAR PLANT
CLASS 2 AND 3 COMPONENTS

Area of Relief	Sect XI Category	Component	Function	Relief Requested	Alternative Test and Examination	Implementation	Relief Request
IWC-1210	---	Containment Penetrations H-12, H-13, H-14, H-19, H-20, H-25, H-31	Containment Barrier	Reclass to match piping on either side of penetration	Inspect as required by code	40-Year Plan	R-41
IWC-1210 C2.7	C-H	Heat exchanger vents and drains	Secondary pressure boundary	Piping becomes nonnuclear after second normally closed value.	None	40-Year Plan	R-42
IWC-1210 C2.7	C-H	1) Nitrogen charging line 2) Control air lines	Secondary pressure boundary	Classify as nonnuclear Not part of Section XI	None	Not applicable	R-43
Table IWD- D1.2 D2.2 D3.2	 D-A D-B D-C	Component supports and Restraints	Class 3 pipe support	Change frequency of examination	Inspect once every interval	40-Year Plan	R-44
IWB-2500 IWC-2520 IWD-2500-1	B-P C-H Items D1.1, D2.1, D3.1	Class 1, 2, and 3	Pressure Retaining Components	Performance of hydrostatic test during third period eliminates performing the system pressure test	None	40-Year Plan	R-45

SECTION 5.0

GENERIC RELIEF REQUESTS

1 ASME B&PV CODE, SECTION XI (As Identified in Section 2.0)

BASIS STATEMENT

Editions and addenda of Section XI not yet approved have been selected for use.

REMARKS

Applicable editions and addenda of the code were selected to be compatible with Consumers Power Midland Nuclear Plant's code use. Selection was based upon information received pertaining to what was expected to be approved by the NRC in 1979, Midland intends to update it's code use to include the summer 1978 addenda when adopted in 50.55a(g). To remain compatible, Big Rock Point Nuclear Plant shall update to the same code at the same time.

IMPLEMENTATION

Applicable editions and addenda of Section XI, as described in Section 2.0 of this document, will be used for updating Big Rock Point's Inservice Inspection Plan.