0894

Department of Energy
Clinch River Breeder Reactor
Plant Project Office
P.O. Box U
Oak Ridge, Tennessee 37830
Docket No. 50-537

File: 05.10

February 23, 1979

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Boyd:

AMENDMENT NO. 48 TO THE PRELIMINARY SAFETY ANALYSIS REPORT FOR CLINCH RIVER BREEDER REACTOR PLANT

The application for a Construction Permit and Class 104(b) Operating License for the Clinch River Breeder Reactor Plant, docketed April 10, 1975, in NRC Docket No. 50-537, is hereby amended by the submission of Amendment No. 48 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment No. 48 includes: an update to Section 9.5, "Inert Gas Receiving and Processing System"; an update to Section 9.13.1, "Conventional Fire Protection System"; and other updates and revisions, as well as responses to NRC's request for additional information contained in letters dated December 1, 1976, and March 30, 1977.

A Certificate of Service, confirming service of Amendment No. 48 to the PSAR upon designated local public officials and representatives of the EPA, will be filed with your office after service has been made. Three signed originals of this letter and 97 copies of this amendment, each with a copy of the submittal letter, are hereby submitted.

Sincerely,

Raymond L. Copeland

Raymond L. Copeland
Acting Assistant Director
for Public Safety

PS:79:034

Enclosure

cc: Service List
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SUBSCRIBED and SWORN to before me this 6th day of February, 1979.

Notary Public

My Commission Expires ... 1980

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1/23/79

PAGE REPLACEMENT GUIDE FOR

AMENDMENT 48

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 48 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket No. 50-537. Amendment 48 consists of new and replacement pages for the PSAR text and question/response supplement pages.

The following attached sheets list Amendment 48 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

Amendment 48

Clinch River Breeder Reactor Plant

Preliminary Safety Analysis Report

(Docket No. 50-537)

This forty-eighth amendment to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report includes an update to sections describing the Inert Gas Receiving and Processing System, the Conventional Fire Protection System, as well as other updates and revisions and responses to NRC's request for additional information. Vertical margin lines on the left hand side of the page are used to identify new design information while lines on the right hand side identify question/response information.

A page replacement guide appears following the list of responses to NRC questions.

Reference: NRC Letter Dated December 1, 1976

NRC
Ques. No.

020.49

Reference: NRC Letter Dated March 30, 1977

NRC
Ques. No.

001.700
001.701

Amendment 48

Page Replacement Guide

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Amendment 48

Question/Response Supplement

The Question/Response Supplement contains an Amendment 48 tab to be inserted following page Q-i (Amendment 47, November 1978). Page Q-i (Amendment 48, February 1979) is to follow the Amendment 48 tab.

The Questions/Response Supplement pages listed below should be inserted in the proper numerical order following the correct section tabs. The parenthesis beside each question indicates the number of pages in each Question/Response.

NRC
Ques. No.

001.700 (9)
001.701 (1)
020.49 (1)

Remove These Pages	Insert These Pages
Q020.47-1	Q020.47-1
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Q120.66-3, 4	Q120.66-3, 4

1.6 MATERIAL INCORPORATED BY REFERENCE

1.6.1 Introduction

This section identifies technical reports incorporated by reference into the PSAR. Some of the technical reports cited were produced for the LMFBR program under the direction of the Energy Research and Development Administration (ERDA) and, therefore, contain the disclaimer notice as required by ERDA manual Appendix 3201, Part II-D. In support of the construction permit application for the Clinch River Breeder Reactor Plant, however, any such disclaimer notice should be considered to be deleted and therefore of no effect.

1.6.2 References

4|36 1. Deleted.

42 2. WARD-D-0185, "Clinch River Breeder Reactor Plant Integrity of Primary and Intermediate Heat Transport System Piping in Containment", September 1977.

3. WARD-D-0115, "Development and Application of a Cumulative Mechanical Damage Function for Fuel Pin Failure Analysis in LMFBR Systems", April 1976.

4. WARD-D-0005, "Demo Code" LMFBR Demonstration Plant Simulation Model, Rev. 4.

5. WARD-D-0090, "CRBRP Decay Power Analysis", January 1976.

36 6. Deleted.

7. AI Report No. 99-TI-413-039, "EVTM/CLEM Full Scale Test Analysis" R.G. Hanson, issued August 15, 1975.

8. AI Report No. 99-TI-413-042, "Subscale Emissivity Test Analysis (EVTM)", D. Vanevenhoven, issued October 17, 1975.

42|16 9. "Hypothetical Turbine Missile Data and Probability of Occurrence for 3600-RPM-23-Inch LSB Unit for Use with Liquid Metal Cooled Fast Breeder Reactor", General Electric Co., August 4, 1977.

18 10. "Third Level Thermal Margins in the Clinch River Breeder Reactor Plant", April 1976.

40 11. WARD-D-0178, "CRBRP Closure Head Capability for Structural Margin Beyond Design Basis Loading", Revision 3, November 1978.

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12. WARD-D-0174, "CRBRP; Active Pump and Valve Operability Verification
44 Plan", April 1977.

47 13. WARD-D-0165, "Requirements for Environmental Qualification of CRBRP
Class 1E Equipment", August 1978.

14. WARD-D-0218, "Structural Response of CRBRP Scale Models to a Simu-
48 lated Hypothetical Core Disruptive Accident", October 1978.

33

2.4.11.5.3 Normal Plant Service Water System

The Normal Plant Service Water Pumps take suction from the main cooling tower basin and provide a supply of water to components
33 listed in Table 9.9-1 during normal operation. Heat is dissipated through evaporation cooling by the main cooling towers. Additional
43 discussion is provided in Section 9.9.1.

2.4.11.5.4 Emergency Plant Service Water System

The Emergency Plant Service Water System is designed to provide sufficient cooling water to permit the safe shutdown and the maintenance of the safe shutdown condition in the event of an accident resulting in the loss of the Normal Plant Service Water System or the loss of both plant AC power supply and all offsite AC power supplies. The Emergency Plant Service Water System is not used during normal plant operation. The system provides the Emergency Chilled Water System chiller condensers and the standby Diesel Generators with cooling water. The Emergency
43 Plant Service Water System is described in Section 9.9.2.

33

33 2.4.11.5.5 Deleted

33 48

2.4.11.6 Heat Sink Dependability Requirements

2.4.11.6.1 Circulating Water System - see 2.4.11.5.2

2.4.11.6.2 Emergency Plant Service Water System

The Emergency Plant Service Water System is a standby system and functions only following an accident occurrence. Emergency Diesel Generator and Nuclear Services heat loads will be dissipated by use of either of two cells of the Seismic Category I mechanical draft cooling tower.

The Emergency Plant Service Water System is composed of two totally redundant loops. Each loop has the capability to provide sufficient cooling water for shutting down the plant and maintaining a safe-shutdown condition for a period of 30 days.

One below grade Seismic Category I water reservoir serves both loops of the emergency water supply and houses sufficient water to assure uninterrupted operation of the water volume for 30 days. The Emergency Cooling Towers and Emergency Cooling Tower Basin have been located and constructed such that the complex will survive the site maximum flood elevation of 809 feet as described in Section
43 2.4.4. Further description of the system is provided in Section 3.4.1.

33

2.4.12 Environmental Acceptance of Effluents

All liquid effluents from CRBRP operation enter the Clinch River through the plant discharge structure located at approximately Clinch River Mile 16. Processed radioactive liquid waste and steam cycle blowdown will be discharged and mixed with mechanical draft cooling tower blowdown before being released to the river. Processed radioactive liquid wastes will be released on a batch basis with each batch being sampled prior to release to assure proper radioactive concentrations. Based upon the activity analysis, the wastes will either be released under controlled conditions or recycled for further processing. Steam cycle blowdown will be released continuously but periodic measurement

Criterion 3 FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of structures, systems, and components.

RESPONSE

The Non-Sodium Fire Protection System provides the plant with equipment, piping, valves, detectors, instrumentation and controls to prevent or mitigate the consequences of a non-sodium fire.

It consists of the following:

Water Supply System
Wet Sprinkler System
Preaction Sprinkler System
Water Spray System
Carbon Dioxide Gas Blanketing System
Halon 1301 Gas Blanketing System
Standpipe System
Portable Fire Extinguisher System
Fire Detection System

The general description of the above systems is provided in Section 9.13.1 and Table 9.13-4. The fire prevention and protection systems to be provided for all the areas associated with the safety related structures, systems and components are listed in Table 9.13-3.

In areas with safety related structures, systems and components, the Non-Sodium Fire Protection System piping and components (such as sprinkler heads) will be designed so that neither piping failures nor inadvertent operation of the system fire protection components due to a seismic event will result in the loss of function of safety related structures, systems and components. This is accomplished through the use of seismically qualified pipe supports, and dry pipe preaction
32 sprinklers within areas containing safety related equipment. Standpipes 48

serving safety-related equipment are Seismic Category I and will be 48
supplied by a Seismic Category I water supply system if necessary. Building isolation valves will be specified as Seismic Category I.

Electrical power for the Fire Protection System will be provided from the normal plant AC power distribution system. If normal AC power is unavailable, the water supply system pressure will be maintained by two diesel-driven fire pumps, and the fire detection system will be energized by a non-Class IE 4-hour DC battery/inverter system that has the capability of being connected to an emergency diesel generator through qualified isolation devices. The Non-Sodium Fire Protection System will be designed in accordance with applicable codes and standards. 48

Fire barriers will provide isolation between areas such as:

Steam Generator Building from Intermediate Bay, Maintenance Bay, Auxiliary Bay and Diesel Generator Building.

Access to all buildings, other than the Reactor Containment Building, will be designed such that there will be multiple means of access for operating personnel and there will be multiple means of access for fire fighting personnel.

The largest potential source of fire from fuel oil is in the vicinity of the standby diesel generator fuel oil storage tanks, located below grade adjacent to the Diesel Generator Building. As these tanks are located below grade, the chance of an accident is reduced. Physical separation provided between the two tanks limits the spreading of fire from one tank to the other. Since either tank is capable of fulfilling the emergency fuel oil requirements, a safe shutdown of the plant will not be jeopardized by a fire in either tank.

Charcoal filters will be bounded and separated by fire barriers, 48
and the filter units will be made redundant, so that safe shutdown of the plant will not be jeoparidized by a fire in either filter.

Table 9.13-3 lists the safety related areas of the plant containing combustible materials. The burning characteristics of these materials such as maximum fire intensity, flame spread, smoke generation and toxicity of combustion products are listed in Table 9.13-2. A detailed fire hazards analysis will be provided in the FSAR and will evaluate the potential fire hazards throughout the plant and the effect of postulated design basis fires relative to maintaining the ability to perform safety shutdown functions and minimizing radioactive releases to the environment. This analysis will serve to confirm the adequacy of
32 48

the present fire protection system which is based on a preliminary fire hazards analysis. Noncombustible and heat resistant materials will be 48 used throughout the plant wherever practical to minimize the fire intensity in any combustion zone. The integrity of vital areas, components and systems is assured through the use of redundacy, physical separation and fire barriers, and administrative controls of materials brought into 48 vital areas.

The design features of the fire detection system are provided in Table 9.13-4. The alarm system will be designed such that the failure of single fire detection devices do not affect the operation of remaining detection devices connected to the same detection zone. The interconnecting circuitry between the detection devices within a zone will be continuously supervised, and a break in the circuitry will be annunciated both locally and in the Control Room.

The entire plant will be encircled by a cement-line, coal tar enamel coated, underground ductile iron piping fire loop having a minimum 48 diameter of 12 inches. Two runouts from the fire pump discharge header will serve the fire loop.

Section 9.13 describes the Non-Sodium Fire Protection System.

The electrical design criteria for circuit integrity and fire
32 protection are described in Section 8.3.

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TABLE 3.2-2 (Continued)

PRELIMINARY LIST OF SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS AND ASSIGNED SAFETY CLASSES(3)

Components	Safety Class(1)	Location(2)	
Steam Generator System			
Evaporators	2	SGB	
Superheaters	2	SGB	
Steam Drums	3	SGB	
Sodium-Water Reaction Pressure Relief Systems	3	SGB	20
IHTS Na Dump Tank	3	SGB	
SWRP Rupture Disk Assemblies (4)	2	SGB	36
35 S.G. Water and Steam Components, Piping and Valves	3	SGB	
Steam Generator Auxiliary Heat Removal System			
Air-Cooled Condensers	3	SGB	20
Auxiliary Feedwater Pumps (w/o motor drives)	3	SGB	
Protected Water Storage Tank (PWST)	2	SGB	
Connecting Piping & Valves (Extending from PWST to and including the First Valve)	2	SGB	
Turbine Drive	3	SGB	
Connection Piping and Valves (except piping from PWST to and including the first valve)	3	SGB	20
Containment Isolation Valves (Within their associated fluid systems)	2	RCB, IB	
Containment Cleanup System	3	RSB	
Containment Annulus Air Cooling System	3	RSB	
Containment Annulus Filtration System	2	RSB	36
Refueling System			36
Ex-Vessel Storage Tank (EVST)	2	RSB	
EVST Guard Vessel	3	RSB	
EVTM Containment Pressure Boundary	3	RSB	43
4 Spent Fuel Transfer Station	NSC	RSB	

TABLE 3.2-2 (Continued)

PRELIMINARY LIST OF SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS AND ASSIGNED SAFETY CLASSES(3)

Components	Safety Class(1)	Location(2)	
Inert Gas Receiving and Processing System			
Primary Cover Gas Lines (Recycle Argon)	2	RCB	
Equalization Line Between Reactor Vessel Primary Pump and Overflow Vessel	2	RCB	
RAPS (Outside Containment)	3	RSB	
RAPS (Inside Containment)	3	RCB	36
CAPS (Outside Containment)	3	RSB	
Control Building Ventilation			
Fan	3	CB	
Filters	3	CB	
Air Conditioning Unit	3	CB	
Emergency Plant Service Water System (5)	3	SGB,DGB	20
Emergency Chilled Water System (5)	3	SGB,CB,DGB RSB,RCB	20
Auxiliary Mechanical Systems for Diesel Generators (Details to be provided)	3	DGB	
Recirculating Gas Cooling System (Portions Serving: Na makeup pump cold trap pipeways, Na makeup pump and vessels, EVS pump and cold trap, EVS pumps and pipeways and the third loop.)	3	RSB,RCB	45
Control Room Heating, Ventilating and Air Conditioning System	3	CB	32, 1
Non-Sodium Fire Protection System			15
Standpipe System (Nuclear Island) Piping and Valves	Note(10)	SGB,CB,DGB RSB,RCB	
Standpipe System Seismic Category I Pumps	Note(10)	DGB	48

TABLE 3.2-2 (Continued)

PRELIMINARY LIST OF SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS AND ASSIGNED SAFETY CLASSES(3)

Notes:

44 (1) Safety Classes are defined in Sections 3.2.2.1 through 3.2.2.4

(2) RCB - Reactor Containment Building
 IB - Intermediate Bay of the SGB
 SGB - Steam Generator Building
 RSB - Reactor Service Building
 CB - Control Building
 DGB - Diesel Generator Building 36

(3) All components will be seismically qualified by analysis unless otherwise noted; motors are included with the mechanical components they drive.

(4) The SWRPRS rupture disc assemblies will be seismically qualified by analysis based on rupture data obtained during dynamic testing. 20

(5) Control panel attached to chillers will be qualified by test.

44 (6) Out to First Isolation Valve

(7) Within Dual Isolation Valves

(8) Downstream of Isolation Valve

44 (9) Downstream of First Isolation Valve

(10) Non-Safety Class, Seismic Category I. 48

TABLE 3.2-5 (Continued)

PRELIMINARY LIST OF ASME CODE CLASSIFICATIONS
FOR SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS

Component	Code/Code Class(1)	Location(2)
Emergency Plant Service Water System	ASME-III/3	SGB,DGB
Emergency Chilled Water System	ASME-III/3	SGB,CB,DGB, RSB,RCB
15 Normal Chilled Water System	ASME-III/3	RCB,RSB
Auxiliary Mechanical Systems for Diesel Generators (Details to be provided)	ASME-III/3	DGB
Control Room Heating, Ventilating, and Air Condition System Isolation Valves 32	ASME-III/3	CB
Non-Sodium Fire Protection System	Note(9)	SGB,CB,DGB
Standpipe System (Nuclear Island) Piping and Valves 48		RSB, RCB
Standpipe System Seismic Category I Pumps	Note(9)	DGB 48

Notes:

(1) Including applicable code cases.

(2) RCB - Reactor Containment Building
IB - Intermediate Bay of the SGB
SGB - Steam Generator Building
RSB - Reactor Service Building
CB - Control Building
DGB - Diesel Generator Building
AEB - Auxiliary Equipment Building

(3) Only piping from containment isolation valves to the filter
32 intake; filters and discharge ductwork per Reg. Guide 1.52.

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TABLE 3.2-5 (Continued)

PRELIMINARY LIST OF ASME CODE CLASSIFICATIONS
FOR SEISMIC CATEGORY I MECHANICAL SYSTEM COMPONENTS

Notes (Continued):

44 | 34 (4) System will meet the requirements of Reg. Guide 1.52.
(5) Out to First Isolation Valve
(6) Within Dual Isolation Valves
(7) Downstream of Isolation Valve
44 | (8) Downstream of First Isolation Valve
(9) Non-Safety Class, Seismic Category I | 48

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3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment (Not Applicable)

3.8.2 Steel Containment System

3.8.2.1 Description of the Containment

The Containment Vessel is a low leakage, free-standing, all
welded steel vessel anchored to the base mat with a steel lined concrete
bottom in the form of a vertical right cylinder having an inside diameter
of 186 feet and with side walls extending approximately 169 feet from
the flat bottom liner at the base to the spring line of the ellipsoidal-
45 spherical dome. The cylindrical shell is embedded in concrete up to the
47 elevation of the operating floor. On the inside of the Containment
Vessel, there is the continuous reinforced concrete wall comprising the
peripheral boundary of the internal concrete structure. Butting against
the outside face of the steel shell from elevation 733 feet up to the
45 elevation of the underside of the operating floor, there is another rein-
forced concrete wall of sufficient thickness designed to prevent buckling
45 of the steel shell. Neither of the two concrete walls are considered
part of the containment vessel. Alumina-silica insulation is attached 33
to the inside surface of the Containment Vessel from elevation 816 feet
to elevation 823 feet. The insulation is 3 inches thick and has a value
of 0.0267 Btu/hr - ft ^{o}F. Its purpose is to limit the shell temperature
48 at elevation 816 feet during Design Basis Accidents to less than 130^{o}F.

The vessel includes: its shell, a ¼" bottom liner plate, one
45 access airlock, one emergency egress airlock, vacuum relief system, one
equipment hatch, penetrations, inspection ladders, miscellaneous appur-
tenances and attachments. The configuration of the Containment Building
is shown in figures in Section 1.2. The design lifetime of the contain-
39 ment vessel shall be 30 years.

3.8.2.2 Applicable Codes, Standards and Specifications

3.8.2.2.1 Codes

The Containment Vessel will be designed, material procured,
fabricated, installed and tested in accordance with the requirements of
the ASME B&PV Code, Section III, Division 1, 1974 Edition with Addenda
through Winter 1974, and all applicable addenda and Code Cases, for Class
43 MC, and ASME-III, Division 2, 1975 Edition, Subsection cc with Addenda
through Winter 1975, for the steel lined concrete containment bottom.
The design shall also meet the requirements of the Class MC Section of
RDT Standard E15-2T, "Requirements for Nuclear Components".

The quality assurance procedures will be in accordance with RDT Standard F2-2 as well as meeting the requirements of the ASME Code, 1
45 Section III, Divisions 1 and 2.

All structural steel non-pressure parts such as ladders, walkways, handrail, etc. will be designed in accordance with the American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel Buildings (AISC, February 12, 1969).

3.8.2.2.2 <u>Design Specification Summary and Design Criteria</u>

The Containment Vessel, including all access openings and penetrations will be designed such that the leakage of radioactive materials from the Containment under conditions of temperature and pressure resulting from the extremely unlikely faults could not cause undue risk to the health and safety of the public and will not result in potential offsite exposures in excess of guideline values of 10CFR100.

The Steam Generating System is located in the Steam Generator
Bay which consists of three cells. Each cell contains one of the three
independent steam generating loops. The south and west side of the
structure are structurally connected to the Intermediate Bay and Diesel
Generator Building respectively. The structural system of this Bay is
essentially the same as the Intermediate Bay. The runway rails of a 115
ton/15 ton gantry crane at roof level are supported on the north and 44
45 south walls of this Bay. The gantry crane is capable of handling major
equipment such as intermediate pumps, evaporators and superheaters, and
transferring them to the Maintenance Bay which is located east of the
45 Steam Generator Bay.

The portion of the Steam Generating System such as Water/Steam
Circulating System and the Steam Generator Auxiliary Heat Removal System
(SGAHRS) is located in the Auxiliary Bay. The exterior walls, interior
walls and the floor system of the Auxiliary Bay are structurally the
same as the Steam Generator Bay. The south side of the Auxiliary Bay
45 is structurally connected to the Steam Generator Bay.

The Maintenance Bay is the portion of the Steam Generator Building containing a railroad siding and facilities for maintenance, cleaning,
and laydown of Steam Generator Building equipment. This bay is a Seismic
Category I structure but is not tornado-hardened. It is constructed of
metal roof decking and metal wall siding supported on structural steel
47 beams and columns.

48 The overall dimensions of the above four structures are as
follows:

			Inside Length (ft.)	Inside Width (st.)	Overall Height (ft.)
48 6	1.	Intermediate Bay	260	Varies from 17' to 162'	124
	2.	Steam Generator Bay	228	74	140
	3.	Auxiliary Bay	228	30	153
48	4.	Maintenance Bay	84	77	108

The top of the foundation mat for the Steam Generator Building,
excluding the Maintenance Bay, is at elevation 733' and grade is at
elevation 815'. The maintenance area of the Maintenance Bay is founded
on competent rock. The laydown area and railroad tracks are founded on
48 Class "A" backfill.

48 45 39 See Section 1.2 for the Steam Generator Building General Arrangements and the general layout and configuration of the structures.

3.8.4.1.4 Diesel Generator Building

The Diesel Generator Building is a Seismic Category I, tornado-hardened reinforced concrete structure extending down to the Nuclear Island
45 33 common base mat at elevation 733'-0".

The DGB houses equipment and facilities used in the production of electrical power from the Emergency Diesel Generators. In addition, it houses the PHTS and IHTS sodium pump motor generators used to supply power to the IHTS and PHTS pumps in loop #3, and the switchgear and associated breakers for all IHTS and PHTS pumps.

The building is designed to allow separation of the two safety-related Emergency Diesel Generators and associated power production and distribution equipment. In order to obtain a safe margin between the Diesel Generator Operating Frequency and the Diesel Generator Building Resonating Frequency:

a. Floors at El. 816'-0" and below are supported by (3) concrete walls.

b. Floor at El. 816'-0" is a 4'-0" thick concrete slab.

Another important function of the building is to provide an equipment removal path and routing area from the Nuclear Island Buildings to the Turbine Generator Building. To fufill this function, a corridor approximately twenty-six feet wide, containing an equipment removal hatch is provided along the eastern edge of the building at each elevation.

The two safety-related, redundant diesel generators and auxiliary equipment are located at elevation 816'-0". The floor below, elevation 794'-0", houses the emergency electrical power distribution equipment and the diesel fuel oil pumps which transfer oil from the buried storage tanks outside. Elevation 765'-0" houses the breakers for the PHTS and IHTS sodium pumps and 13.8 kv and 4.16 kv switchgear. The base elevation, 733'-0" houses the PHTS and IHTS sodium pump motor generators for loop #3. Detailed equipment arrangements are shown on the Diesel Generator
45 Building Arrangement Drawings in Section 1.2.

5. The consistent application of conservative assumptions with respect to the radioactive term, the energy release to the cell, and the mitigating contribution of other design features, insures that the functional design basis for the RAPS Surge and Delay Tank Cell provides adequate conservation. 1

3A.1.3 Design Description

There are 13 independent inner cells in the RCB having
44 inert atmospheres cooled by the Recirculating Gas Cooling System.
The Recirculating Gas Cooling System is described in detail in
47 44 Section 9.16. Table 3A.1-3 lists these inner cells indicating the
equipment contained in each and the atmosphere cooled. The inner
cell arrangements and equipment layouts are shown in the RCB General
39 Arrangement drawings in Section 1.2.

The steel-lined reinforced concrete walls completely
enclosing each inner cell insure cell structural integrity under
dynamic effects associated with pipe breaks or equipment failure
resulting in the generation of missiles. The pressure boundaries
44 of the inner cells are designed for the pressures listed in Table
3A.1-3. Should dynamic effects following component failure result
in a sodium leak or spill, the steel liners are designed to contain
the sodium and prevent degradation of the concrete, while the inert
44 atmosphere provided by the Inert Gas Receiving and Processing System 25
protects against sodium fires. Design and analysis procedures for
the inner cell concrete structures are described in Section 3.8.3.4,
37 and for the cell liners in Paragraph 3.5 of the Appendix 3.8-B.

48

9.5.5.3 Nitrogen Distribution Subsystem

The specific instrumentation requirements for the Nitrogen Distribution Subsystem are:

1) Control of the use of auxiliary nitrogen bottles to respond to low-nitrogen pressure in valve-actuation line headers

2) Pressure regulation for nitrogen-inerted cells

3) Pressure and/or flow regulation for equipment cooling circuits for the control rod drive mechanism and for the RAPS and CAPS cold boxes

4) Control of cell atmosphere purges, by automatic-sequencing cell atmosphere sampling unit, and provision for on-line analyses for oxygen, water vapor, and radiation levels

5) Controls to divert the cell purge gas exhaust to CAPS when radioactivity exceeds the low-level radiation setpoint

6) Purge controls for cell atmosphere on selected signal of high oxygen content or water vapor level

7) Alarm signal when cell atmosphere radiation exceeds the high-level radiation setpoint, with operator re-set only.

9.5.5.4 RAPS and CAPS

The RAPS and CAPS subsystems have specific control requirements for the functions listed below:

a. RAPS

   1) Pressure regulation in the vacuum vessel

   2) Pressure regulation in the surge vessel

   3) Gas flow rate regulation at the inlet to the cryogenic section (control of flow from surge vessel)

   4) Alarm on signal of high radiation in surge vessel

   5) Radiation level measurement and indication of RAPS effluent stream to recycle argon vessels, with alarm on high signal

   6) Manual flow bypass controls for cold box

   7) Manual controls for the diversion to CAPS of cold-box effluent and maintenance purge, and automatic pressure relief to CAPS
48 of overpressure in cold-box components

1) Gas pressures and temperatures

2) Gas flow rates

3) Liquid levels in supply vessels

4) Valve position status for selected valves

5) Piping and component temperatures

6) Component pressure drops.

c. Controls

The following general control functions are to be provided as required:

1) Liquefied Gas Supply Vessels: level control to automatically switch to full tanks in sequence on low-level signals from another tank or tanks, and high-flow shutdown capability

2) Supply Headers: pressure reduction and regulation

3) Vessel Cover Gases: pressure regulation and over-pressure relief

4) Containment Isolation: remote controls for valves.

9.5.5.2 Argon Distribution Subsystem

The specific instrumentation requirements for the Argon Distribution Subsystem are:

1) Control of the use of the auxiliary in-containment argon bottles to respond to low argon pressure in the normal supply headers

2) Pressure regulation for the fuel handling cell atmosphere

3) Controls for automatic regenerative operation of the FHC atmosphere purification unit

4) Temperature controls on freeze vents, vapor traps, vapor condensers, and heated argon lines

5) Controls to minimize reactor cover gas pressure oscillation during temperature transients

6) Control of flow of recycle argon cover gas to the PHTS pumps and of total flow from the reactor and PHTS cover gas spaces to RAPS.

48

The liquefied gas stations are also fitted with equipment sized to provide gas at flow rates, pressures, and durations in excess of the minimum requirements, thus providing a margin of safety.

The supply of gas for the essential function of valve operation is ensured by the installation of supplementary high-pressure gas bottles in the RCB and in the RSB. These serve as safe shutdown protections in the event of an interruption or loss of the principal gas supply.

9.5.4 Test and Inspections

The components and piping of the IGRP System meet the requirements of the applicable sections of the ASME Boiler and Pressure Vessel Code and ANSI Code B31.1.0. NEMA Standards are applied to the electrical equipment. The system design, procurement, manufacturing, construction, and installation conform with the quality assurance requirements of 10 CFR 50, Appendix B, and RDT F 2-2.

9.5.5 Instrumentation Requirements

9.5.5.1 General System Requirements

The following instrumentation requirements are common to all of the IGRP subsystems:

a. Functions

The Inert Gas Receiving and Processing Instrumentation System shall perform the following functions:

1) Monitor process parameters and positions of selected valves

2) Maintain process parameters within normal prescribed operating ranges

3) Provide for overriding the normal control loops in the event of abnormal conditions of pressure, temperature, or gas analysis, including both venting and blocking-off of subsystems

4) Provide for automatic isolation of all process gas lines entering or leaving the Reactor Containment Building

5) Provide for the remote manual operation of valves (both proportional and on-off control, as required)

6) Provide local, centralized, and main control room I&C panels to accomplish the above.

b. Indicators

The following process variables will be logged, indicated,
48 recorded, or alarmed, as is appropriate:

In case of a major leakage of cover gas of sufficient radioactivity to overload CAPS, a radiation detector located in the effluent line will cause recirculation of the effluent through CAPS until it is sufficiently decontaminated to be released to the heating and ventilating system safely. This option is time limited by the amount of injected cooling nitrogen that can be accumulated within the subsystem. When that limit is reached, CAPS compressors are stopped and the radioactive gas is held in the system. After sufficient decay time, the radioactive gas can be vented to H&V by operator action.

All penetrations of containment by IGRP System piping are protected by double isolation valves (one inside, one outside) that prevent the escape of contamination through the pipes and out of containment when high activity levels exist in containment. Specific details of containment isolation are presented in Section 6.2.4.

The effects of off-normal events that cause RAPS piping or vessel ruptures outside of containment are discussed in Chapter 15. In brief, the system design lends itself to procedural actions that will contain gaseous radwaste within leak-tightness-specified equipment cells under the worst circumstances. This delay in releasing gaseous radwaste very significantly mitigates the effect of an accident by allowing the decay of radioactivity within the cell, before initiating cleanup procedures.

The fuel handling cell will be well sealed, so that the inleakage rate of air and its moisture will be small. The presence of sodium vapor in the cell will tend to reduce the concentration of the water vapor by reaction to form NaOH. In addition, the NaOH "smoke" that settles out in the cell will also be a getter for water vapor because of the hygroscopicity of NaOH(s).

The FHC Argon Purification Unit (APU) is to be procured from a supplier as a unit having specified and demonstrated performance capabilities. The specifications for the unit are to be developed in the course of system engineering, during which period the parameters that affect the oxygen and water vapor concentrations will be established. The selected fabricator will be required to prepare and submit a design for approval prior to beginning fabrication of the FHC-APU. The fabricated unit will then, as a condition of acceptance, be required to demonstrate that it meets the specified performance requirements (75 volume ppm maximum water vapor and oxygen).

9.5.3.2 Availability of Inert Gases

The supply of inert gases to other systems in the CRBRP is ensured by the installation of a complex of liquid gas storage vessels and distribution systems. Both argon and nitrogen are available at the RSB and the SGB. In addition, there is an independent installation of
48 liquid nitrogen for emergency fire and accident control uses in the SGB.

the design discussion of the RAPS and CAPS subsystems is presented in Section 11.3. The following evaluation draws on the information in that section.

The evaluation of the design of the IGRP System is based on the degree to which the system meets its major objective, that the radioactivity release to the environment be as low as reasonably achievable, and the corollary objectives that all requirements for the use and control of inert gases, both for normal and off-normal conditions, be satisfied. The following questions are addressed.

9.5.3.1 Control of Radioactive Gases

The principal safety consideration in the design of the IGRP is that the leakage of discharge of radioactive gases to both restricted and unrestricted areas must not only be lower than the maximum permissible concentration (as given by 10 CFR 20 under normal conditions) but must also be as low as is reasonably achievable.

The RAPS subsystem (Section 11.3), by means of a cryogenic still, reduces and maintains the radioactivity in the recycle argon cover gas at a steady-state level such that the piping that distributes the gas to the reactor head and to the primary pumps does not present a radioactivity hazard to operating or maintenance personnel. RAPS reduces the cover gas activity concentrations for most of the radioisotopes by many orders of magnitude, the average decontamination factor being approximately 1,000*. This is particularly important because a large contribution to the total plant release of radioactivity is the diffusion of cover gas through the reactor head seals. A second, but very much smaller contribution to the plant release of radioactivity is the leakage of recycle argon gas from the buffered reactor head seals.

This gas, which originates as RAPS subsystem effluent, also leaks from the seals into the reactor head access area and is released to the atmosphere through the RCB heating and ventilating system. The performance of RAPS is sufficiently effective that these seal leakages result in site boundary dose rates that are a small fraction of the normal background dose rate. Site boundary doses presenting specific values are given in Section 11.3.7.

The small but finite expected leakage or diffusion of cover gas through piping and components into the primary sodium system cells is another source of potential radioactivity release to the environment. In order to prevent the direct release of this activity, purged cell atmospheres containing leaked radioactivity are processed in CAPS to remove gaseous fission product activities. The two delay beds of CAPS provide a decontamination factor of about 62, averaged for all the radioactive isotopes processed. This capability is more than adequate to handle expected normal leakage.

* Average decontamination factor is the total influent radioactivity divided by the totalled effluent radioactivity for all radioisotopes.

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provides nitrogen to ensure the uninterrupted operation of certain essential valves in the event of pressure loss in the nitrogen supply header. A control valve automatically restores pressure in the valve actuation circuit when an abnormal decrease in operating pressure is sensed. A check valve which isolates the valve circuit precludes auxiliary supply blowdown to the remainder of the failed circuit.

9.5.2.2.6 Nitrogen Supplies at SGB

The normal-use nitrogen supply for the SGB is stored as liquid nitrogen in two Dewars, with 3000 gal. capacity each, on the SGB pad. The liquid nitrogen is converted to gas by an ambient-air vaporizer (at 15,000 scfh nominal rating) for each Dewar. Normal usage is supplied from one Dewar, with a level sensor automatically switching tanks upon depletion to a pre-set level. A control override allows the option of simultaneously supplying nitrogen from both tanks, so that doubling the flow rate to meet abnormal demands is possible.

The nitrogen supply for sodium-water reaction control is stored as liquid nitrogen in one Dewar of 3000 gal. capacity, also located on the SGB pad. The liquid nitrogen from this Dewar is vaporized by 3 vaporizers (total capacity 750 scfm). Under normal conditions, the vaporizers can be maintained at temperature and also provide a small flow to maintain positive pressure in the inerted Sodium-Water Reaction Pressure Relief System (SWRPRS). Nitrogen gas at 200 psig is provided during normal use and during a sodium reaction accident.

9.5.2.2.7 Nitrogen: SGB Distribution

Headers branch from the normal-use supply to provide nitrogen for service stations in the SGB, sodium maintenance area, and hot shop. Effluent gases from these operations will be discharged to CAPS. Provision is made to supply nitrogen for inerting the ex-containment PSST cell in the IB when sodium is present.

9.5.2.2.8 Nitrogen: Sodium-Water Reaction and Fire Control

Nitrogen gas is provided to the SWRPRS to maintain a positive pressure in the inerted atmosphere. In the event of a sodium-water reaction, nitrogen purge will be initiated to prevent the establishment of explosive mixtures of hydrogen within the SWRPRS.

A nitrogen gas supply at a minimum flow rate of 150 scfm and 190 psig is provided to the water-steam side of the Steam Generator System following system blowdown. This prevents sodium from entering the water side in the event of a leak in any one of the nine sodium-water heat exchangers.

9.5.3 Safety Evaluation

An evaluation of the design of the IGRP System must include the functions and operations of the RAPS and CAPS subsystems, as well as
48 those of the argon and nitrogen subsystems. As has been noted above,

about 10,000 vppm. Reduction from this value to the 1000 vppm limit will be done by purging with nitrogen.

Nitrogen for service maintenance operations is available at service stations located within the RCB.

9.5.2.2.3 <u>Nitrogen: RCB Auxiliary Supply</u>

An auxiliary supply of nitrogen gas is stored in high-pressure standard cylinders located within a cell in the tornado-hardened RCB. This nitrogen is used to ensure the uninterrupted operability of certain essential valves in the event of pressure loss in the nitrogen supply header. A control valve automatically restores pressure in the valve actuation circuit when an abnormal decrease in operating pressure is sensed. A check valve then isolates the valve circuit from the main supply line in order to preclude auxiliary supply blowdown to the remainder of the failed supply circuit.

9.5.2.2.4 <u>Nitrogen: RSB Distribution</u>

The 150 psig RSB header branches off into several lower pressure headers that service the needs of other systems as well as those of the RAPS and CAPS subsystems within the RSB.

RSB cells and pipeways containing sodium components are inerted with nitrogen during normal operation. The cell pressures are maintained by a feed and bleed arrangement, and a purge function controls impurity levels. (See Section 9.5.2.2.2)

The RAPS and CAPS cold boxes are inerted with nitrogen at a continuous low flow rate during operation. These flows are vented directly to the respective cells, so that the cell atmospheres become nitrogen-rich. The cell pressures are maintained by back-pressure regulators that bleed the cell effluents to CAPS.

The nitrogen requirement to the cold boxes serves two purposes: to inert the cold boxes so that water condensation within the cryogenically-cooled structure is prevented and to provide gas for valve operation. The cold boxes would not be effected adversely by high purge flows nor would there be an impact on the CAPS decontamination process. The only consequence of such flows would be increased nitrogen utilization.

Nitrogen for service maintenance operations is available at service stations located within the RSB.

Nitrogen gas is provided as a cover gas for the Dowtherm tanks used in the chilled water system.

9.5.2.2.5 <u>Nitrogen: RSB Auxiliary Supply</u>

An auxiliary supply of nitrogen gas, stored in high-pressure
48 standard cylinders located within a cell in the tornado-hardened RSB,

9.5.2.2.1 Nitrogen Supply at RSB

The RSB and RCB nitrogen supply is stored as liquid nitrogen in two Dewars, each with [illegible]000 gal. capacity, on the RSB pad. An ambient air vaporizer on each Dewar can evaporate the liquid nitrogen at a nominal flow rate of 15,000 scfh. Normal nitrogen usage is supplied from one Dewar, with a level sensor automatically switching tanks upon depletion of a pre-set level. A control override, however, allows the option of simultaneously supplying nitrogen from both tanks so that doubling the flow rate to meet peak demands is possible.

9.5.2.2.2 Nitrogen: RCB Distribution

The header feeding the RCB contains one isolation valve on each side of the containment penetration, providing automatic shutoff capability on either side in the event of nitrogen pressure loss. The header inside containment branches off into (1) a low pressure header feeding all of the normally inerted cells and pipeways within containment, (2) a high pressure line for actuation of valves in cells that are normally inerted, (3) a line to the CRDM assembly recirculation cooling system, and (4) a line to provide sparging gas to the sodium component cleaning operation.

Cells and pipeways containing sodium components in the RCB are normally inerted with nitrogen atmosphere, as is the CRDM cooling system. Each inerted cell or group of cells has inlet and outlet control valves that maintain preset cell pressures, in addition to having automatic cell purging for maintaining required oxygen or water-vapor levels. Purge flow is automatically activated by a cell atmosphere sampling and analysis unit that periodically monitors the O_2 and H_2O levels in each cell atmosphere. Radioactivity is also monitored but does not activate purging.

The inerting system for RCB and RSB cells (except FHC) is designed for normally controlling the oxygen concentration within the cells to a maximum of 2 vol %. The design base for the cell gas inerting system is a net inward leakage of air of 1% of the cell volume per day. When the cell is inerted to 2% oxygen, which amounts to 10% of the oxygen content of the inleaking air, the water vapor content in the cell will also be 10% of that in the in-leaking air. The Heating and Ventilating System normally controls the humidity of the air in the building to 40% R.H. at 75°F (water partial pressure, 0.12 atm). Because the cells are to be steel-lined, dehydration of the concrete will not contribute directly to the water content of the cell gas, so that the normal partial pressure of water vapor is 0.0012 atm, or 1200 vppm water vapor.

During initial warm-up and prior to sodium loading, should the water vapor content of the cell atmosphere (which can be air) exceed the normal maximum value, this water will be removed first by cell purging with air, and then, as the Recirculating Gas Cooling System (RGCS) goes into operation, by condensation on the cooling coils. At steady-state,
48 this unit will limit the water vapor content of the cell atmosphere to

While one of the loops of this unit operates, the other loop is regenerated by flowing mixed argon-5% hydrogen gas through the copper bed to reduce copper oxide. The water produced by this reaction is removed in the dryer bed.

9.5.2 Nitrogen Distribution System

9.5.2.1 Design Basis

Nitrogen is to be supplied for (1) cooling and inerting the atmospheres of the cells and pipeways containing radioactive sodium and the Control Rod Drive Mechanism, (2) actuating pneumatically-operated valves in the inerted cells, (3) cover gas for the Dowtherm tanks in the chilled water system, (4) purging the IHTS steam generators and evaporators in the event of a sodium-water reaction, (5) a cover gas for the Sodium Water Reaction Pressure Relief System (SWRPRS), and (6) miscellaneous handling and maintenance services.

The SGB nitrogen supply for the sodium-water reaction purge is sized to provide 750 scfm of nitrogen for a maximum of 12 hours.

The nitrogen supply rate to be available for the RCB and RSB cell purge requirements is to be 250,000 scfd.

The nitrogen subsystem is to include a sampling system that periodically samples each nitrogen-inerted cell and analyzes the cell atmospheres for radioactivity, oxygen, and water vapor content. The radioactivity analysis determines the selection of exhausting cell gases to CAPS if radioactive, or to heating and ventilating if not radioactive.

Either the oxygen or the water vapor monitor reading can be selected for automatically initiating cell atmosphere purge. The oxygen content operating range is 0.5 to 2.0%; the water vapor concentration upper limit is 1000 vppm. The oxygen range is chosen to provide enough oxygen to prevent nitriding of the steel, and yet not exceed a fire-limiting concentration of oxygen. The water vapor is limited in order to assure early detection in the event of a small sodium leak.

9.5.2.2 Design Description

The nitrogen distribution subsystem consists of three sets of liquid nitrogen supply sources in Dewars (vaporized to gaseous nitrogen for usage), two sets of gaseous nitrogen supply sources in pressure bottles, and the necessary valves and piping to meet the requirements discussed in Section 9.5.2.1. The supply at the RSB provides the nitrogen to the RSB and RCB. One of the two supply sources at the SGB provides the normal needs of the SGB and shop area and the other supply source provides protection for sodium fire and sodium-water reaction accidents
48 in the SGB.

The RSB header also supplies argon to the ex-vessel storage tank (EVST), the fuel handling cell (FHC), and other Reactor Refueling System Components, such as the RSB plug storage facility, floor service stations, EVST seals, and cask corridor (Dowtherm control panel).

9.5.1.2.6 Fresh Argon Supply at the Steam Generator Building (SGB)

Argon for the Steam Generator Building (SGB) is stored as liquid in two Dewars located on the SGB pad. These Dewars have a capacity of 1500 gal. each and are equipped with fill and vent lines. Normally only one Dewar is in operation. When it is nearly empty, a low-level instrumentation signal operates automatic controls to shut off that Dewar and to open a full Dewar to the supply header. A control override allows drawing on all Dewars simultaneously.

Two ambient-air vaporizers on each Dewar can evaporate the liquid argon at a nominal maximum gas flow rate of 250 scfm each, at 200 psig. With both Dewars on-line, therefore, approximately 1000 scfm of argon gas at 200 psig can be delivered.

The argon flow from these Dewars passes through a filter and into a main header. Branch lines serve the sodium receiving station, the incoming sodium drum sodium sampling packages, and the intermediate sodium characterization package.

9.5.1.2.7 Fresh Argon: SGB Distribution

The argon flow from the main header in the SGB-Intermediate Bay (SGB-IB) is divided into three headers that serve the respective IHTS loops in the SGB. Each header services the following components: line vents (freeze vents), rupture disc spaces, leak detection service, intermediate sodium pump seal purge and oil gravity tank, sodium dump tank, and the pressure equalization line between the intermediate sodium pump and intermediate sodium expansion tank, providing cover gas for both. Purged gas from the sodium pump oil leakage collection tank and oil gravity tank passes through an oil vapor trap before release to the atmosphere outside of the SGB.

9.5.1.2.8 Vacuum Services

The argon distribution subsystem incorporates permanently installed vacuum pumps. Several locations are provided for movable pumps that may be temporarily connected to evacuation stations.

9.5.1.2.9 Fuel Handling Cell Atmosphere Purification

The FHC atmosphere purification unit continuously processes a side-stream of argon gas drawn from and returned to the cell gas cooling stream. The unit contains two parallel gas-treating trains, each basically
48 consisting of a copper bed to remove oxygen and a dryer.

Two ambient-air vaporizers on each Dewar can evaporate the liquid argon at a nominal maximum gas flow rate of 2000 scfh each, at 175 psig. With both Dewars on-line, therefore, approximately 8000 scfh of argon gas at 175 psig can be delivered.

The argon from the Dewars passes through a filter and is then divided into three main headers that supply argon to the RCB, RSB, and other ex-containment components.

9.5.1.2.3 Fresh Argon: RCB Distribution

The RCB header enters the building with isolation valves on each side of the penetration. This header supplies argon to the primary sodium storage vessel, with a feed and bleed system at a normal pressure of 1 psig, and to the recycle argon storage vessels.

The RCB header also supplies argon to the primary sodium plugging temperature indicator, the primary sodium sampling package, the floor/wall service stations, the reactor head inflatable seals, and the IVTM storage facility.

The RCB header also supplies argon to the primary sodium line freeze vents, which are furnished argon during startup, maintenance, and sodium drain and fill at a nominal pressure of 5 psig; the pressure can be increased, if needed, to 50 psig. This header also supplies cover gas argon for the NaK system and the make-up pump drain vessel.

9.5.1.2.4 Fresh Argon: RSB Ex-Containment Distribution

The RSB ex-containment header supplies make-up argon to the ex-containment primary sodium storage vessels in the Intermediate Bay. The normal pressure in the storage vessels is 1 psig, but this can be increased to 50 psig during tank drain. These vessels can be vented either through a vapor trap and a pressure control valve to the Cell Atmosphere Processing System (CAPS) or to a vacuum station and then to the CAPS.

9.5.1.2.5 Fresh Argon: RSB Distribution

The RSB header supplies argon at the required pressures to the gas chromatograph, the fission gas monitor module, and the gas sampling trap. A branch line provides argon purge to the RAPS cold box.

The RSB header supplies argon through regulators to the Auxiliary Liquid Metal System EVS Na and NaK components and to the Impurity Monitoring and Analysis System EVS sodium sampling package. The sodium lines have freeze vents that are furnished with argon during startup, maintenance, and sodium drain and fill operations at a nominal pressure
48 of 5 psig. This pressure can be increased to 50 psig.

The use rate of argon by these services is variable and is dependent on operator options. Under start-up conditions, the flow will be maximum, and a minimum supply capability of 95,000 scfd of argon is to be provided.

Argon is to be used for all services involving sodium-wetted components, such as fuel handling, sampling, and maintenance services. This gas also is ultimately exhausted through CAPS to the atmosphere.

Argon is also to be supplied for purging and inerting IHTS components and for sodium-water reaction control purposes.

9.5.1.2 Design Description

The argon distribution subsystem is composed of liquid argon Dewars with vaporizers, gaseous argon bottles, piping, valves, vapor traps, filters, vessels, relief systems, freeze vents, and oil traps as necessary to distribute the argon to meet the requirements described in Section 9.5.1.1.

9.5.1.2.1 Recycle Argon Distribution

Argon from the primary recycle cover gas storage vessels in the RCB is reduced in pressure to supply cover gas to the reactor vessel, primary sodium overflow vessel, and primary pumps cover gas spaces, which are all interconnected by a pressure equalization line. This cover gas system is maintained at a pressure of 6 in. w.g. by a feed and bleed control system.

There is a continuous transfer of argon cover gas from the reactor and the primary pumps via the equalization line to the primary sodium overflow vessel and then through a 5-scfm vapor trap that removes sodium vapor. This vapor trap consists of a vapor condenser and two parallel aerosol filters (one redundant). The gas flows back to RAPS for processing before recycling. A 1-scfm sample of cover gas is taken from the equalization line and is passed through a 1-scfm sodium vapor trap to the Impurity Monitoring and Analysis System. This gas and the cover-gas bleed from the primary pumps are also returned to RAPS.

9.5.1.2.2 Fresh Argon Supply at RSB

Argon for services in the Reactor Service Building (RSB), the Reactor Containment Building (RCB), and the Intermediate Bay (IB) is stored as liquid in two Dewars, located on the RSB pad. These Dewars have a capacity of 1500 gal. each and are equipped with fill and vent lines. Normally, only one of the Dewars is in operation. When it is nearly empty, a low-liquid-level instrumentation signal operates automatic controls that shutoff that Dewar and open the other Dewar to the supply header. A control override allows drawing on both Dewars simul-
48 taneously.

Both stainless steel and carbon steel are used in the IGRP System. All piping and components that are exposed to sodium vapor are fabricated from Type 304 or 316 stainless steel. All piping and components used in cryogenic services are made of Type 304 or other low-temperature alloy steel. The RAPS subsystem piping is made of Type 304 stainless steel. The remaining piping and components are made of carbon steel. All the gas vessels are to be made of carbon steel, as is much of the argon and nitrogen gas distribution subsystems.

After these vessels have been fabricated, their interiors are to be cleaned with abrasives and solvents to remove rust and scale. The tanks are then to be evacuated and back-filled with a dry inert gas. The tanks are to be maintained at this positive pressure until they are installed. Following installation and leak-testing of the weld joints, the tanks are again to be evacuated and filled with a dry inert gas.

The active valves are listed in Table 9.5-4. These valves must be operable during and after design events, such as Safe Shutdown Earthquake (SSE).

The following sections describe in detail the Argon Distribution Subsystem and Nitrogen Distribution Subsystem. The RAPS and CAPS subsystems are described in detail in Section 11.3.

9.5.1 Argon Distribution System

9.5.1.1 Design Basis

Argon is to be supplied for the liquid metal system cover gas spaces for purging, filling, and draining the liquid metal systems, for buffered and inflatable head seals, for the atmosphere in the Fuel Handling Cell, and for services connected with fuel handling, sampling, and maintenance operations.

The reactor closure head seals are to be supplied with argon. The seals are designed so that a small amount of this gas is expected to leak directly into the head access area (HAA). This gas may contain radioactive gases, which are expected to diffuse through the seals. The radioactivity content of the seal buffer gas must be maintained at an activity concentration, either by purification in the RAPS subsystems or through the use of fresh argon, such that seal diffusion losses of cover gas to the head access area will not exceed one tenth the MPC_{40} (maximum permissible concentration for first 40 hour work week) concentration in the HAA. The argon feed and bleed system will maintain the reactor cover gas pressure at 6 ± 2 in. w.g.

Fresh argon is to be used for the PHTS and associated equipment purging, filling, and draining services in the RCB and RSB and for the Fuel Handling Cell supply, in order to minimize the amount of radioactivity in the affected components or in cells whose atmospheres are
48 ultimately exhausted through CAPS.

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9.5 INERT GAS RECEIVING AND PROCESSING SYSTEM

The Inert Gas Receiving and Processing (IGRP) System consists of the following four subsystems: (1) Argon Distribution Subsystem, (2) Nitrogen Distribution Subsystem, (3) Radioactive Argon Processing Subsystem (RAPS), and (4) Cell Atmosphere Processing Subsystem (CAPS).

The Argon Distribution Subsystem (Figures 9.5-1 through 9.5-3) provides cover gas to all free liquid metal surfaces and to component and reactor head seals.

The Nitrogen Distribution Subsystem (Figures 9.5-4 through 9.5-10) provides inerting gas for cells containing primary sodium components, cover gas for auxiliary coolant surfaces, inert gas for maintenance operations, gas for driving pneumatic valve operators, and inert blanketing gas for fire control.

Figures 9.5-1 through 9.5-10 summarize the argon and nitrogen gas services provided by the IGRP System to interfacing systems.

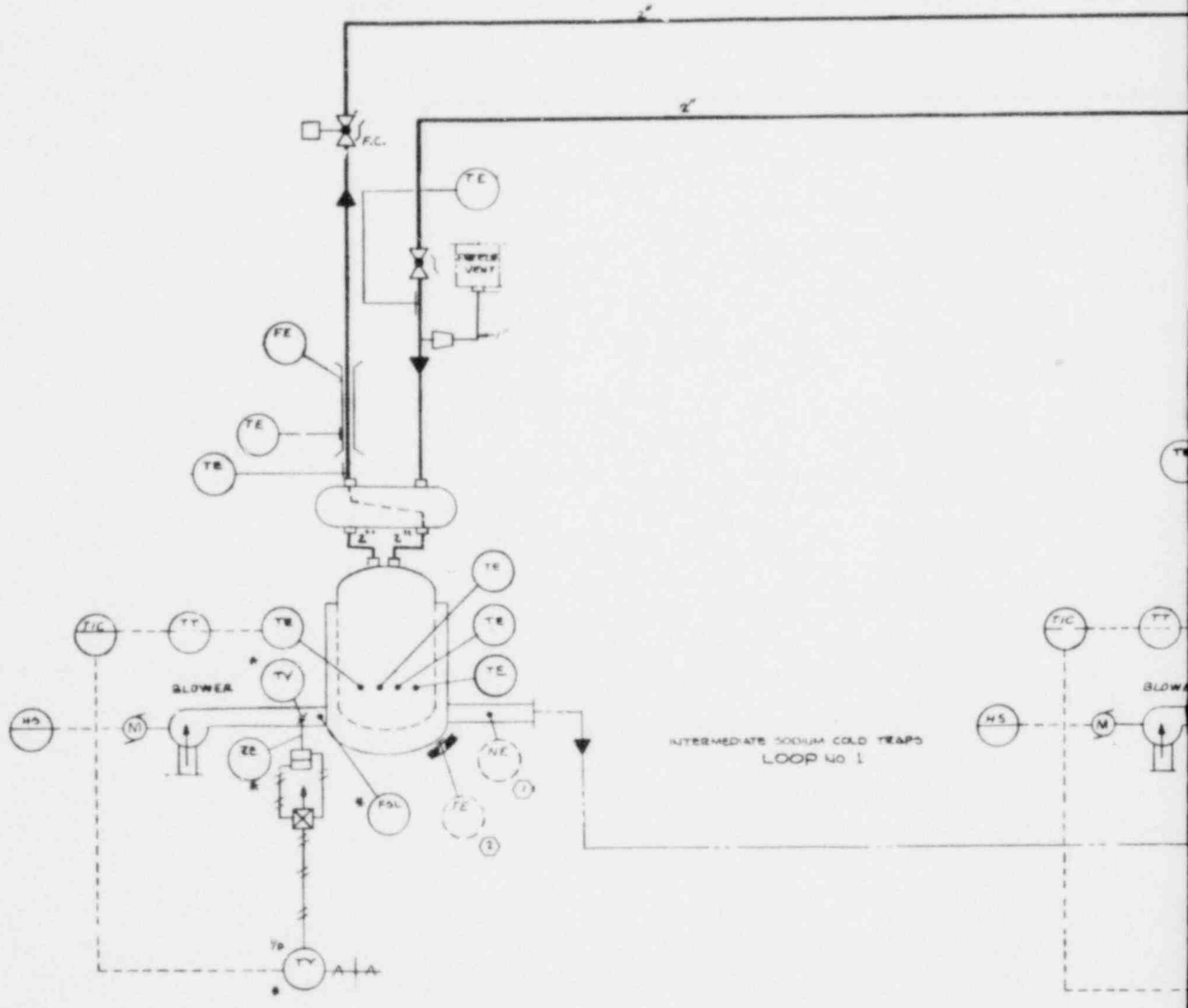
The Inert Gas Receiving and Processing System has several vessels that contain gases under pressure; these are listed in Table 9.5-2 which identifies their names, the contained gas, the design, operating, maximum pressures, the operating temperatures, the vessel volume, and the maximum available stored energy (PV product) for the maximum pressure. Table 9.5-3 is a summary of the locations of the vessels. All of these vessels are either inside a cell within a Category I building or outside such a building. The cell and building walls provide the required protection of equipment essential for a safe reactor shutdown. Figure 9.5-11 shows the locations and arrangements of the equipment items located by the item numbers in Table 9.5-3.

The Argon Distribution Subsystem also provides evacuation service to vessels and piping that are being filled with sodium or argon.

The fuel handling cell atmosphere purification unit of the Argon Distribution Subsystem removes water vapor and oxygen from the recirculated argon atmosphere of the FHC and maintains these impurities within specified levels.

The RAPS subsystem processes primary heat transport system cover gas (particularly reactor cover gas), removes radioactivity and provides a source of purified gas for recycle back to the reactor and the PHTS.

The CAPS subsystem processes gas exhausted from the cell atmospheres and from other locations within the reactor complex and ensures that effluent gases released from the CRBRP have radioactivity
48 levels that are as low as reasonably achievable.



F.C.
TE
FE
TE
TE
TIC
TT
TE
TE
TE
TE
TV
BLOWER
HS
ZE
FSL
NE
TE
TY
INTERMEDIATE SODIUM COLD TRAPS
LOOP NO 1
TIC
TT
HS

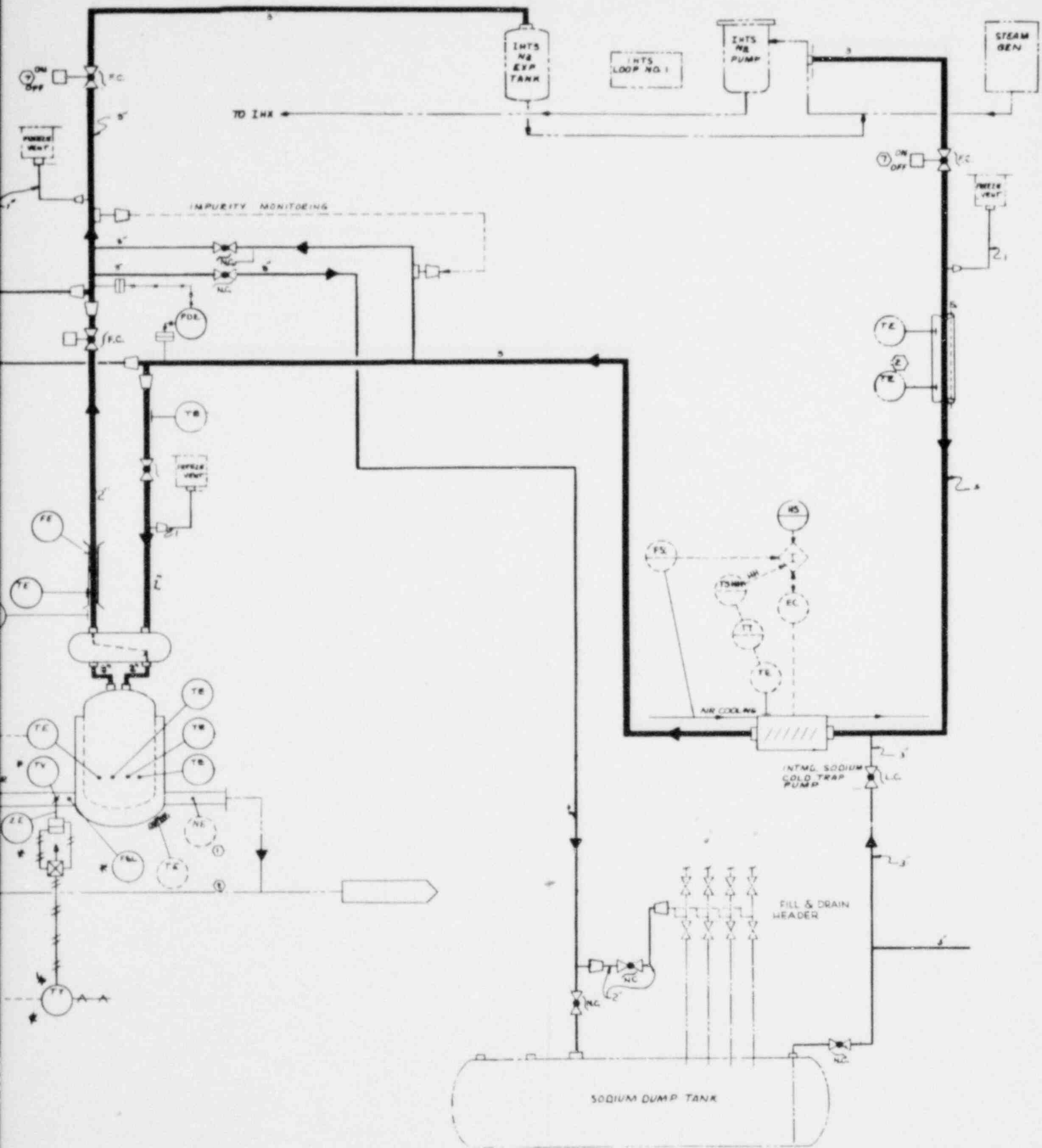


Figure 9.3-5.(Cont.) Intermediate Sodium Processing System
(Sheet 3 of 3)

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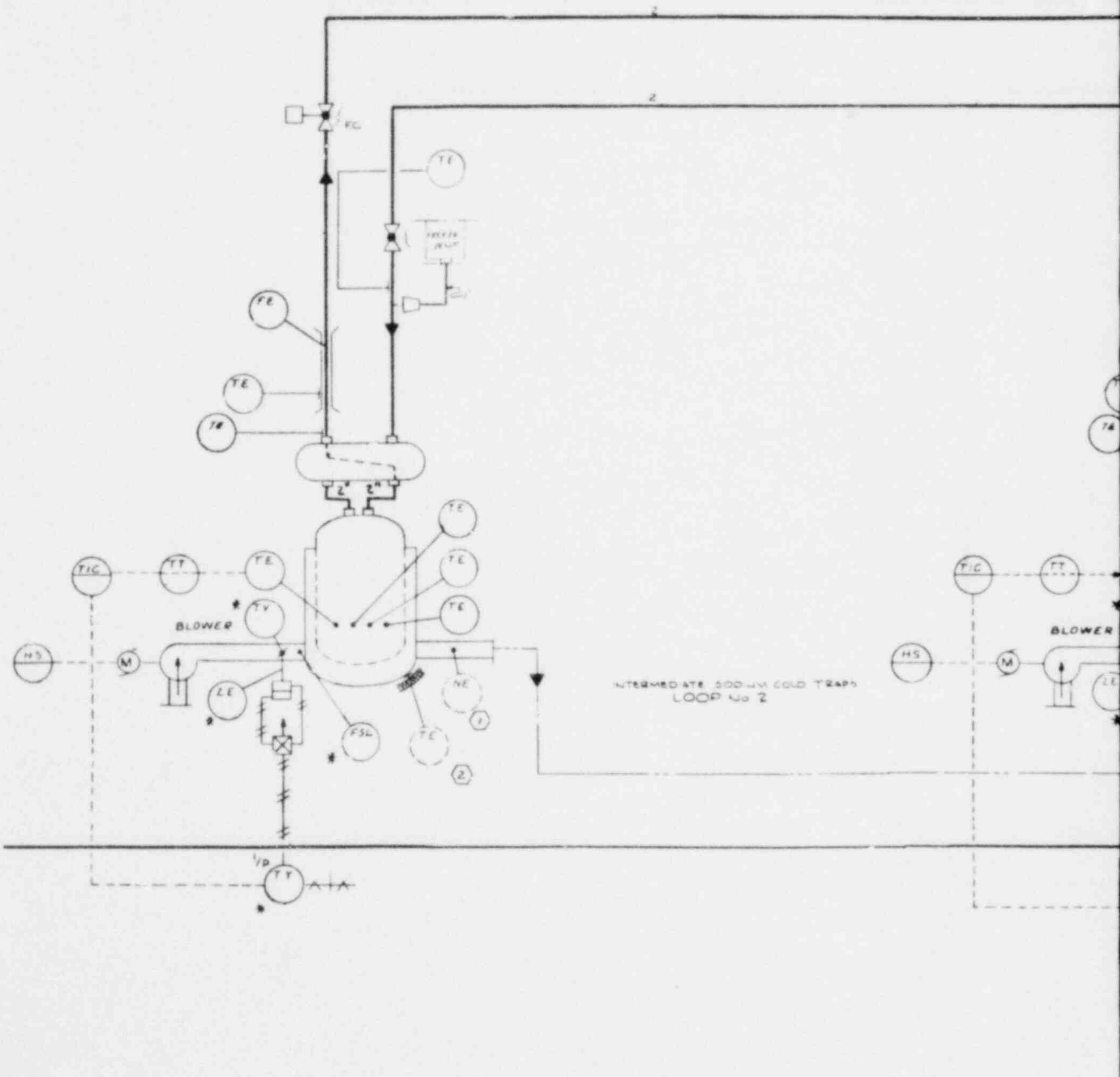

INTERMEDIATE SODIUM COLD TRAPS
LOOP No 2
BLOWER
BLOWER
TIC
TT
TE
TY
HS
M
ZE
FSL
NE
FE
TT
TE

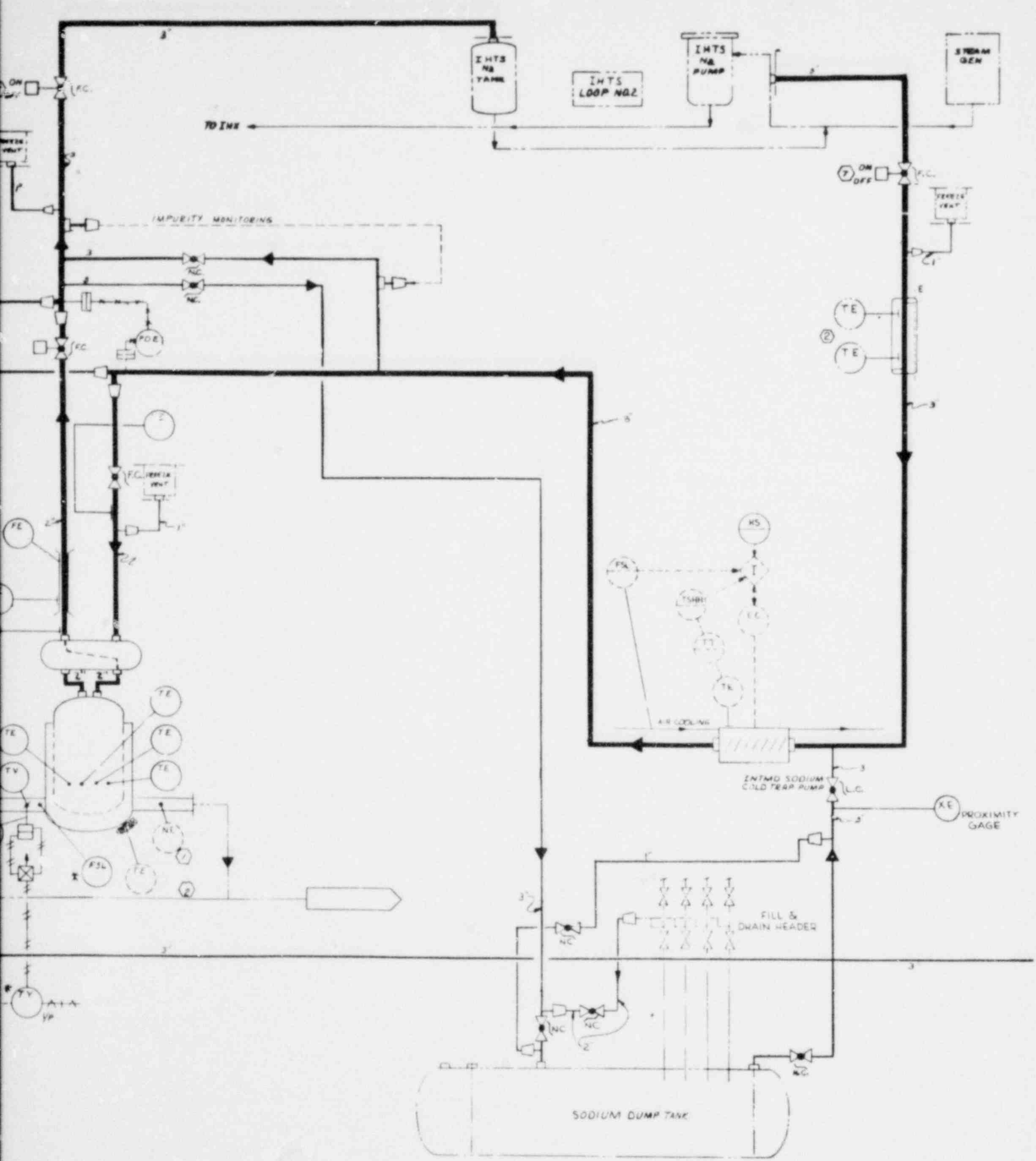

Figure 9.3-5.(Cont.) Intermediate Sodium Processing System
(Sheet 2 of 3)

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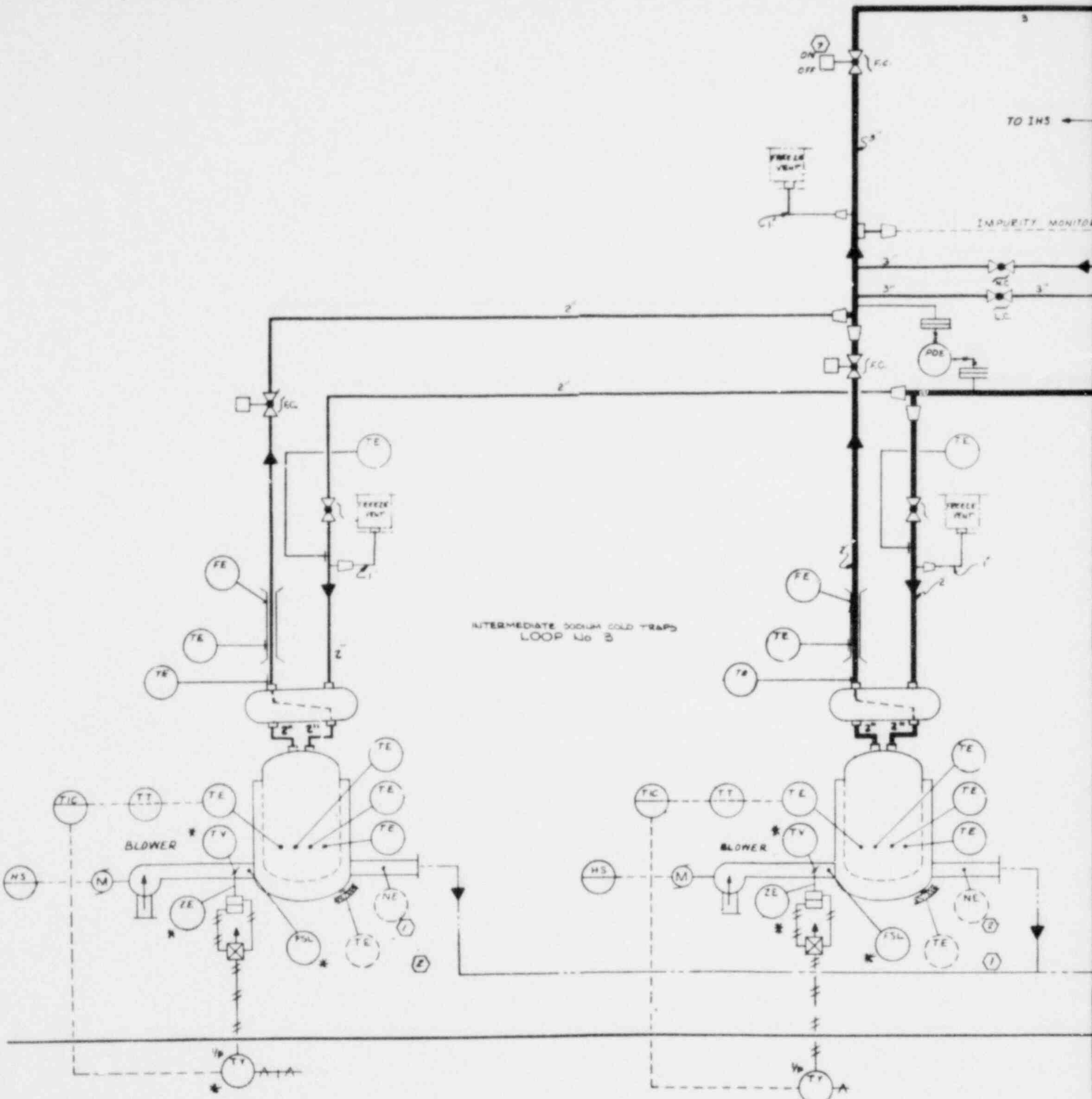


ON
OFF
F.C.
TO IHS
FREEZE VENT
IMPURITY MONITO
PDE
TE
FE
INTERMEDIATE SODIUM COLD TRAPS
LOOP No 3
TIC
TT
TV
BLOWER
HS
M
ZE
FSL
NE
TY

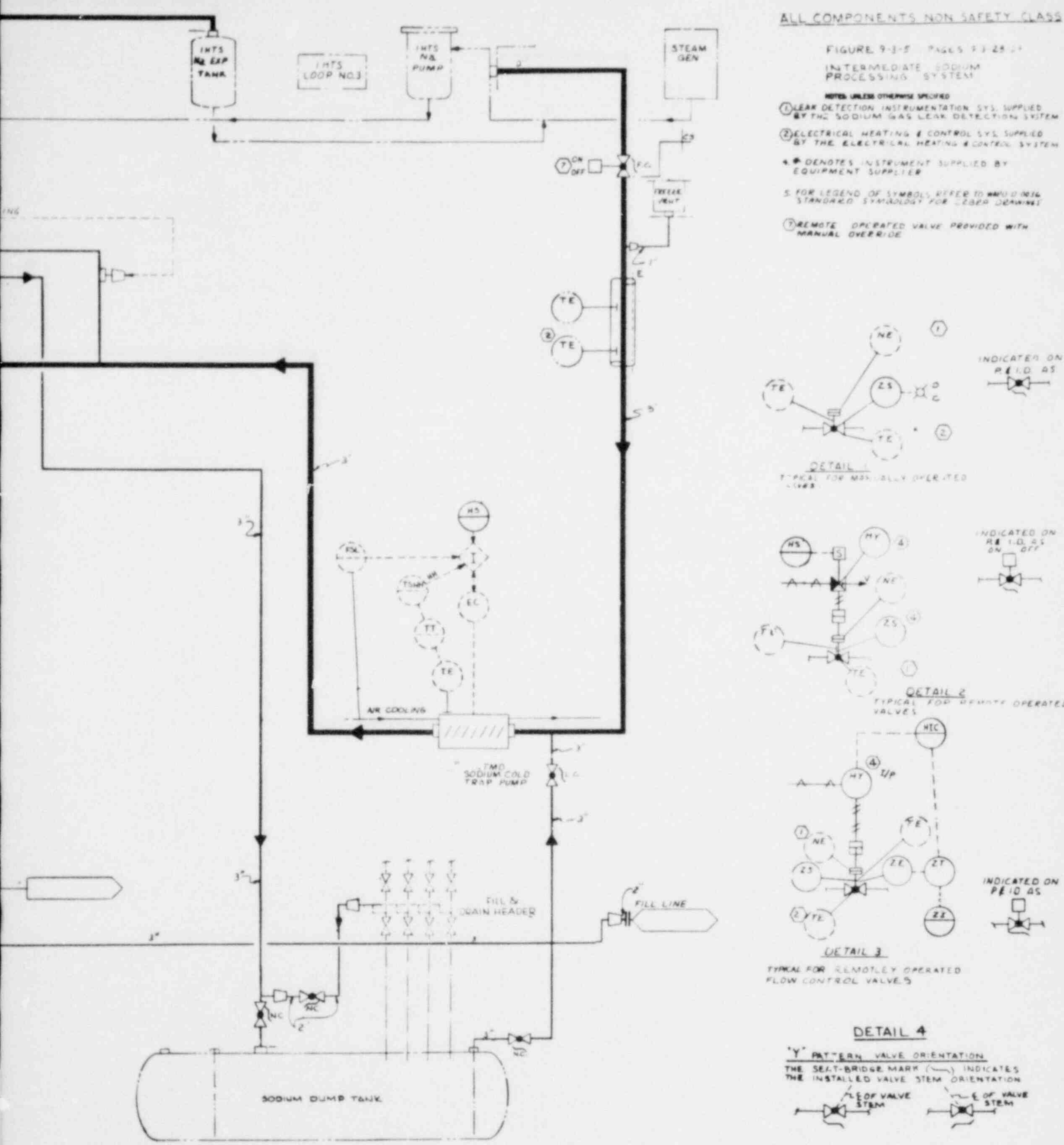
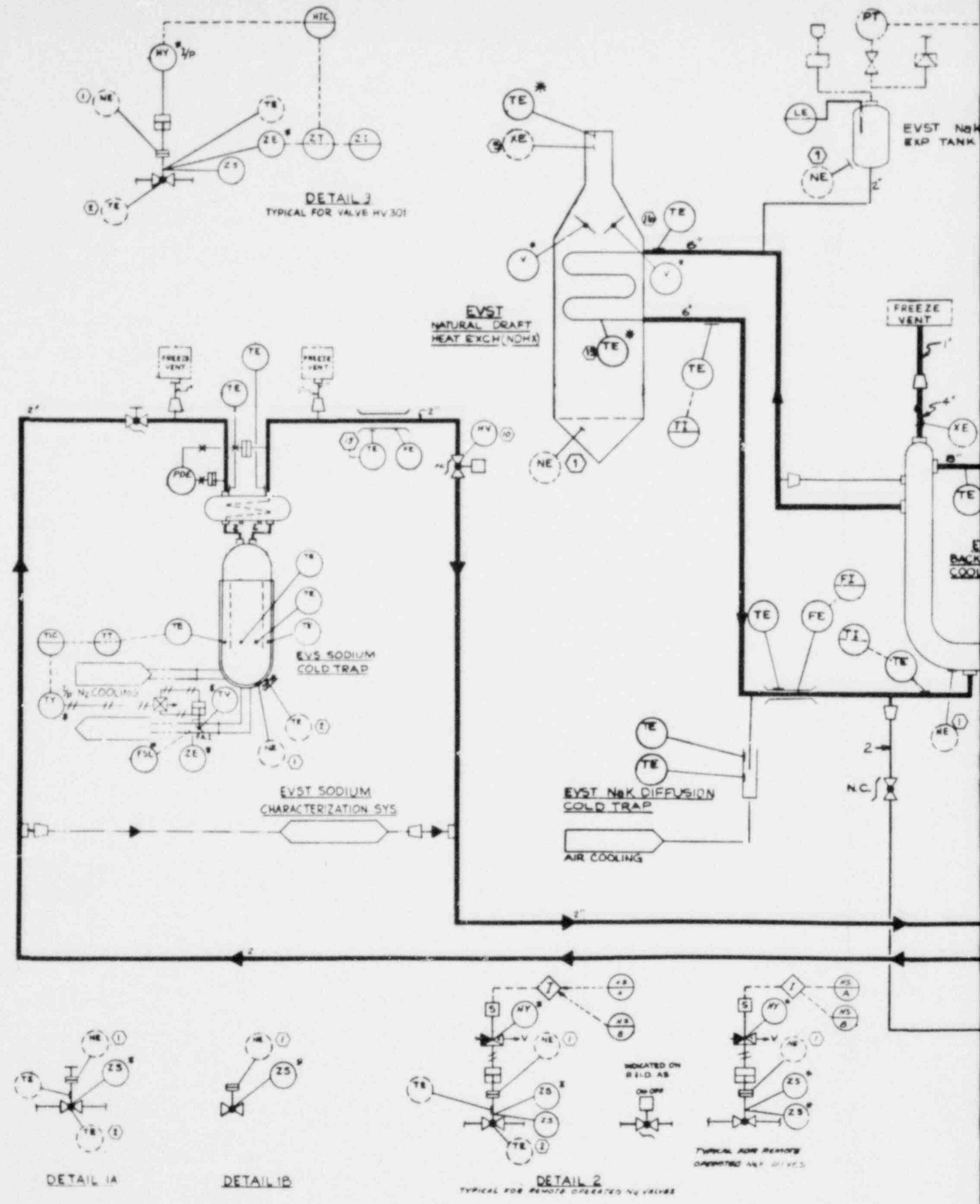


Figure 9.3-5. Intermediate Sodium (Sheet 1 of 3) Processing System



DETAIL 3
TYPICAL FOR VALVE HV 301
EVST
NATURAL DRAFT
HEAT EXCH (NDHX)
EVST NaK
EXP TANK
FREEZE
VENT
EVS SODIUM
COLD TRAP
N2 COOLING
EVST SODIUM
CHARACTERIZATION SYS
EVST NaK DIFFUSION
COLD TRAP
AIR COOLING
N.C.
DETAIL 1A
DETAIL 1B
DETAIL 2
TYPICAL FOR REMOTE OPERATED N.C. VALVES
INDICATED ON P&ID AS

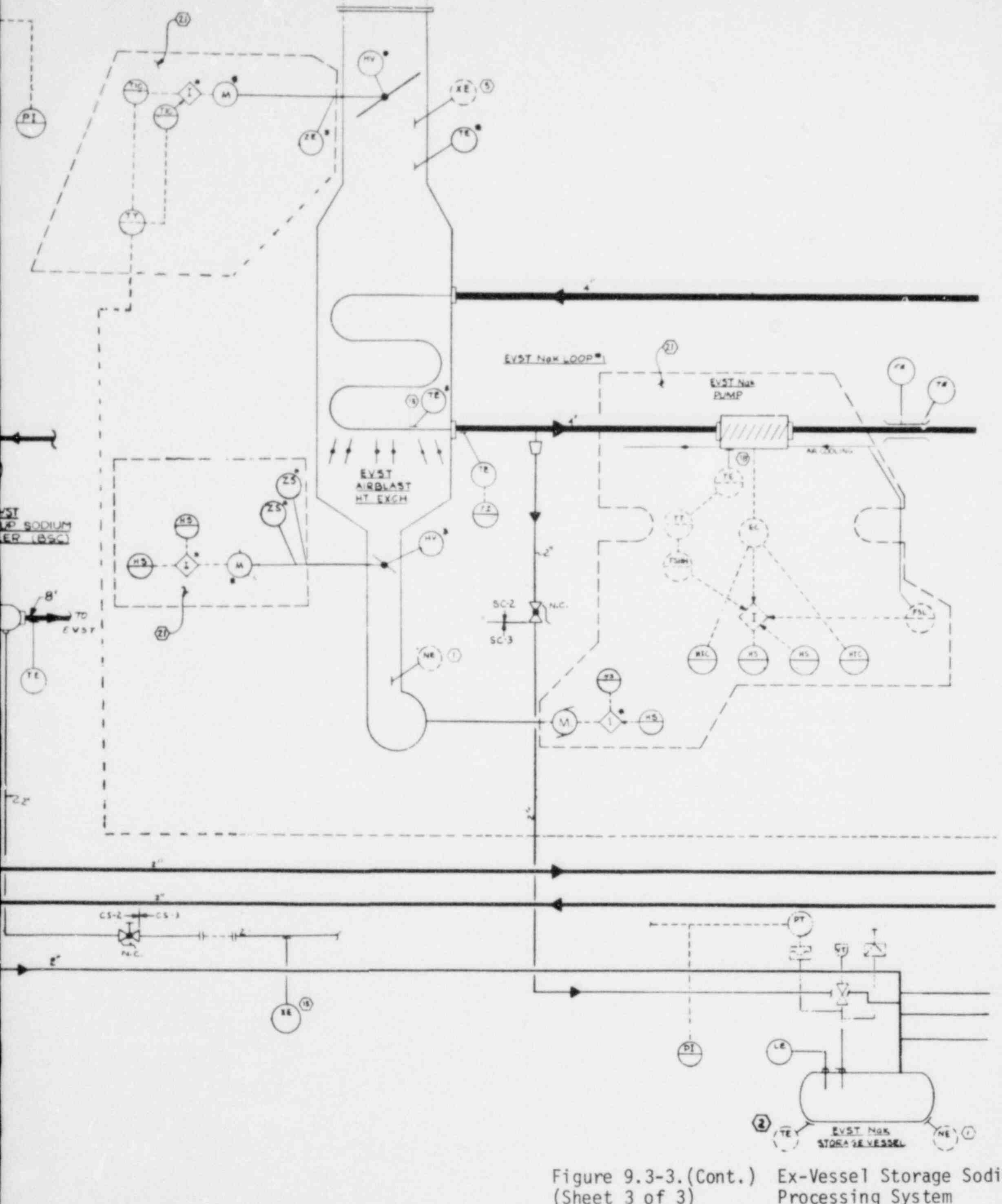
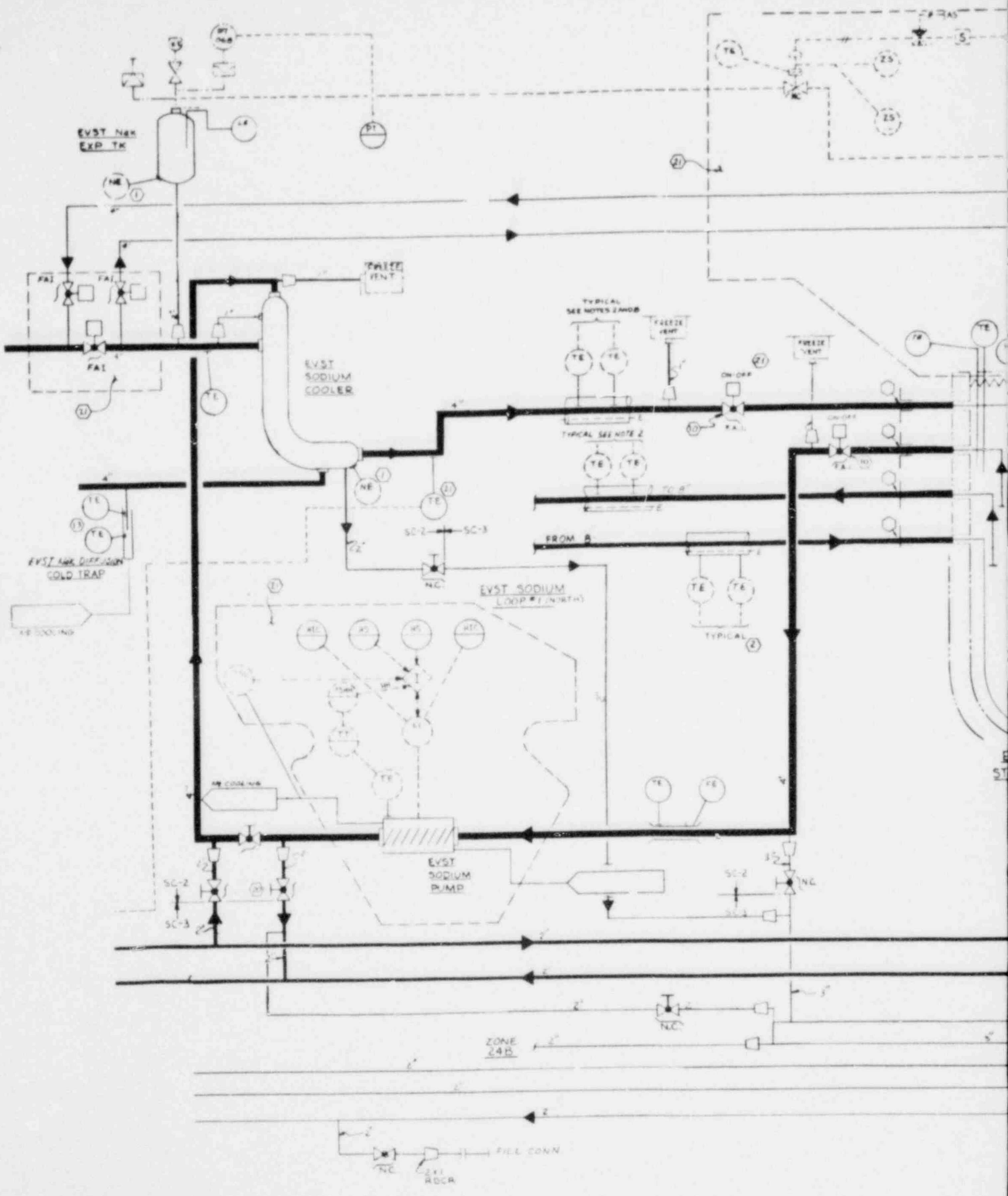


Figure 9.3-3.(Cont.) Ex-Vessel Storage Sodium Processing System
(Sheet 3 of 3)



EVST NaK EXP TK
FAI
FAI
FAI
FREEZE VENT
EVST SODIUM COOLER
TYPICAL SEE NOTES 2 AND 8
FREEZE VENT
FREEZE VENT
ON-OFF
FAI
TYPICAL SEE NOTE 2
ON-OFF
FAI
TE
TE
NE
NE
TO B
FROM B
TE
TE
TYPICAL
SC-2
SC-3
N.C.
EVST NaK DIFFUSION COLD TRAP
NaK COOLING
EVST SODIUM LOOP #1 (NORTH)
NaK COOLING
EVST SODIUM PUMP
SC-2
SC-3
SC-2
SC-3
N.C.
N.C.
ZONE 24B
FILL CONN
N.C.
2x1 REDCR

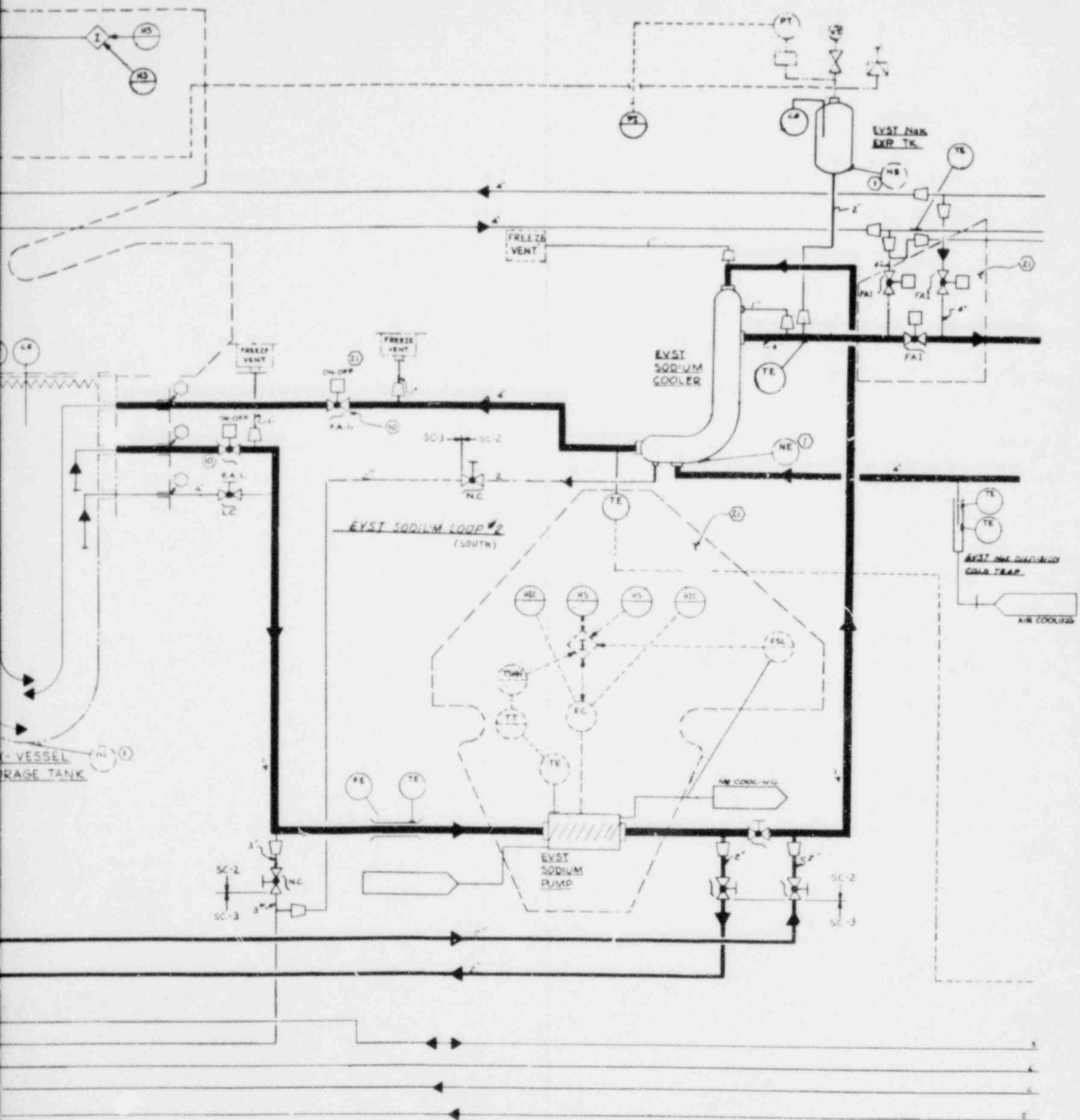
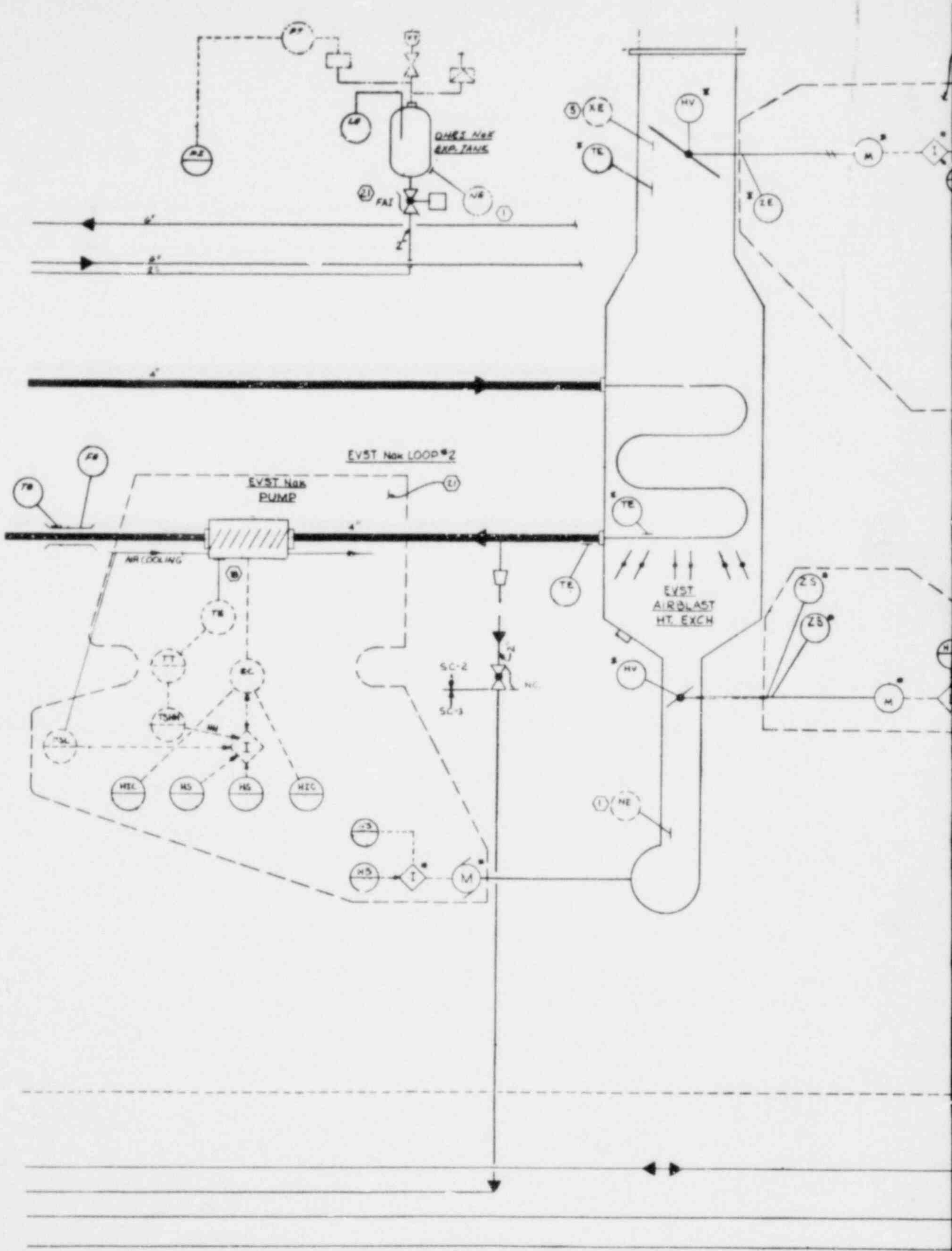
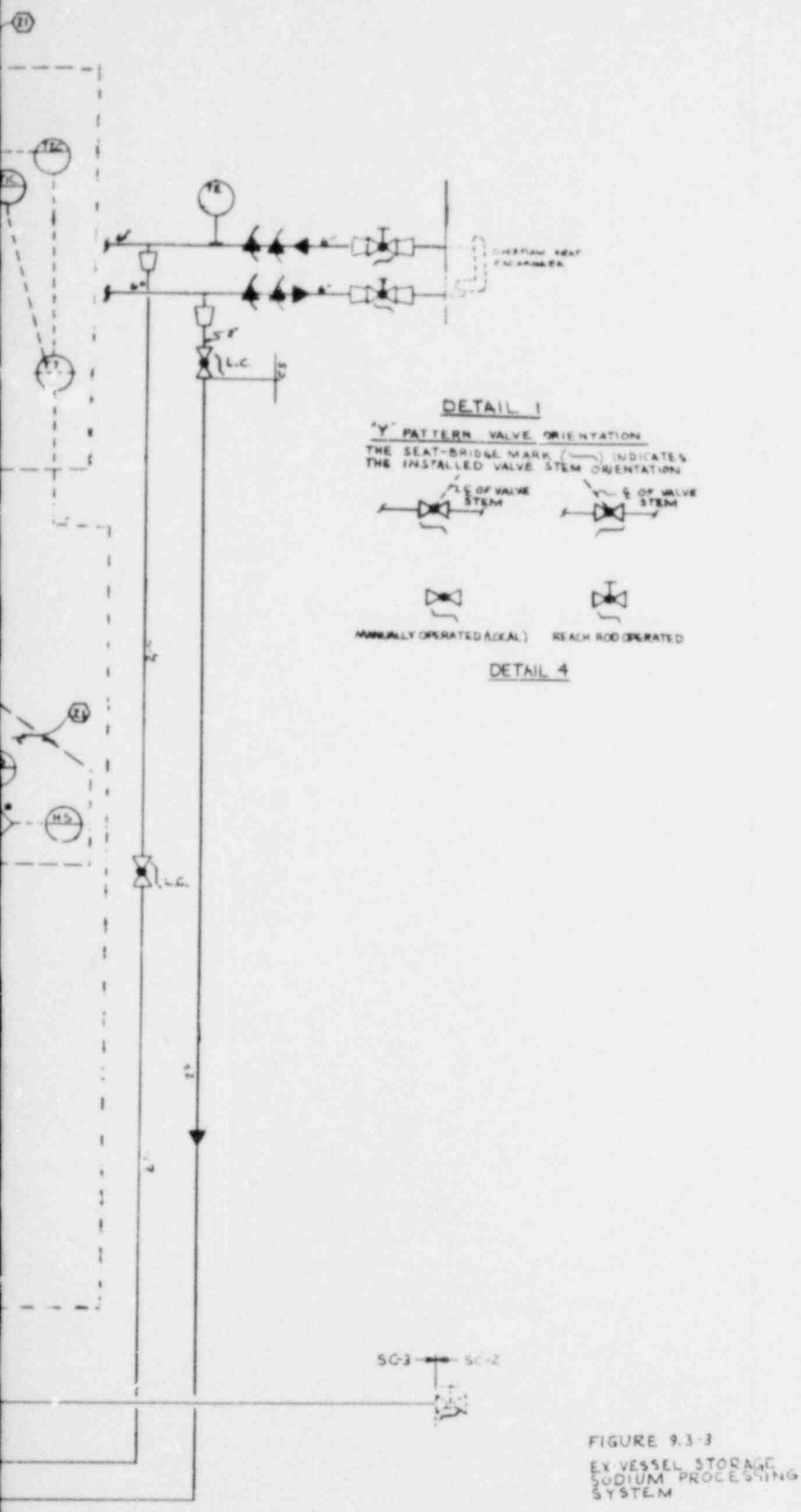


Figure 9.3-3.(Cont.) (Sheet 2 of 3) Ex-Vessel Storage Sodium Processing System


EVST NaK LOOP #2
EVST NaK PUMP
EVST AIRBLAST HT. EXCH
AIR COOLING
SC-2
SC-3
NC
HIC
HS
HS
HIC
TE
TE
TE
HV
HV
XE
ZS
ZS
M
M
M



NOTES:

(1) LEAK DETECTION INSTRUMENTATION SYS
SUPPLIED BY THE SODIUM TO GAS LEAK DETECTION SYS.

(2) ELECTRICAL HEATING AND CONTROL SYS
SUPPLIED BY THE ELECTRICAL HEATING & CONTROL SYS.

4. & DENOTES INSTRUMENT SUPPLIED BY
EQUIPMENT SUPPLIER

(5) SMOKE DETECTOR INSTRUMENTATION SYS
SUPPLIED BY THE FIRE PROTECTION SYS

6. FOR LEGEND OF SYMBOLS REFER TO WARD-D-0036,
"STANDARD SYMBOLOGY FOR CRBRP DRAWINGS"

(10) REMOTE OPERATED VALVE PROVIDED WITH
MANUAL OVERRIDE AND REACH ROD.

(13) THERMOCOUPLE WIRED TO FIELD TERMINAL.

(21) DESIGNATES IE ELECTRICAL EQUIPTMENT

22. S SAFETY CLASS

(21) NUCLEAR SAFETY RELATED
(IE ELECTRICAL EQUIPMENT)

FIGURE 9.3-3
EX VESSEL STORAGE
SODIUM PROCESSING
SYSTEM

Figure 9.3-3. Ex-Vessel Storage Sodium
(Sheet 1 of 3) Processing System

48

The circuits associated with the Standby AC Power System (Class IE electric systems) shall be separated into two or more redundant divisions. The circuits between the diesel generators and the 4160 Volt Class IE Switchgear shall be designed for a two-divisional separation (Division I and Division 2). 48

The cables for the Diverse Power Supply are separated from cables of the remaining AC power supply by routing them independent of the other safety division. 48

37

J. Administrative Responsibilities and Controls for Assuring Separation Criteria

The scheme and raceway channel identification facilitates and ensures the maintenance of separation in the routing of cables and the connection of control boards and panels. At the time of the cable routing design, the routing designer checks to ensure that the separation group scheme is compatible with a single-line diagram division designation 12 and with other schemes previously routed. Class IE cables are inspected upon installation by construction personnel for consistency with the design documents. Color identification of equipment and cabling will be used in this effort.

8.3.1.5 Physical Identification of Class IE Equipment

Each circuit and raceway is given a unique identification. This identification provides a means for distinguishing a circuit or raceway associated with a particular voltage or class as well as with a particular channel or division. The channel or division is provided by a coded digit of identification and is assigned on the basis of the 12 following criteria:

Division 1 12

A Class IE instrumentation, control, or power scheme/raceway associated with Load Group 1. 48

Division 2 12

A Class IE instrumentation, control, or power scheme/raceway associated with Load Group 2. 48

Division 3 12

A Class IE instrumentation, control, or power scheme/raceway associated with Load Group 3 (the Diverse Power Supply System).

48

48

flexible) conduits or enclosed wireways to a point where 1 foot of separation exists. A minimum horizontal separation of 1 foot shall be required between trays carrying cables of different load divisions if no physical barrier exists between them. If a horizontal separation of 1 foot minimum does not exist, a fire-resistant barrier shall be provided. 12

The fire barriers utilized will be qualified for the appropriate fire hazard as determined by the fire hazards analysis. 48

If the minimum horizontal or vertical separation does not 48
exist, a fire resistant barrier, or covered cable trays without barriers shall be provided.

I. Sharing of Cable Trays and Routing of Non-Class IE Cables

All 480 volt power cables, lighting cabinet feeders, and DC power cables carrying 30 amperes or more are run in 480 Volt trays. Medium-level signal trays carry the following cables: input and output cables for the computer other than thermocouples, instrument transmitters, recorders, tachometers, indicators, eccentricity and roto detectors, and shielded annunicator cables used with solid-state equipment. Signal cables for thermocouples, strain gauges, vibration and radiation detectors, thermal converters and RTD's are run in low-level signal trays. All other cables are run in control trays. Within a load group division, the minimum spacing between trays stacked vertically is 9 inches, tray bottom to tray bottom. The minimum spacing between trays installed side by side, within a load group division shall be 6 inches. The trays shall be constructed of steel of varying widths. All cable tray systems located in Category I structures shall have supports designed to withstand 12
seismic disturbances.

All PPS cables are run in conduit or enclosed raceways. PPS
analog circuits may be routed together in the same raceways, provided 48
the circuits have the same characteristics, such as power supply, shutdown 12
system (primary or secondary), and channel identity (A, B or C).

Vital instrument cables for the PPS may be routed together in the same raceways, provided the circuits have the same characteristics, such as power supply, shutdown system (primary or secondary), and channel- 48
identity (A, B or C).

Automatic actuation and control power circuits for the PPS may be routed in the same raceways, provided the circuits have the same 48
characteristics, such as power supply, shutdown system (primary or secondary), and train identity (PPS logic train I, II or III).

Otherwise barriers are to be installed. Metal conduit, fire barriers,
or steel wire ducts are acceptable barriers to maintain independence
without additional spatial separation over that required by Regulatory
Guide 1.75. Non-Class IE wiring is not harnessed together with Class IE
cable and is not permitted to provide a combustion path between harnesses
of different divisions. Penetrations through barriers are permitted if 12
fire stops are provided.

G. Penetration Areas

Separate penetration areas are provided for all cables that
25 must pass through the containment wall. Where possible, redundant PPS
cables shall utilize electrical penetrations spaced horizontally rather
than vertically. Cables through penetrations of the primary containment
shall be grouped such that failure of all cables in a single penetration
cannot prevent a protective action. Separation of Class IE circuits
will be maintained through penetrations. The PPS will not share penetra-
tions with non-Class IE systems. 48

The cable length for hydrogen concentration, pressure and
temperature monitors within the RCB will be minimized due to the possi-
bility of subjecting the cable to extremely high temperatures in the
TMBDB scenario described in Reference 10 of Section 1.6. To accomplish 48
this, penetrations will be provided into the annulus near the instruments.
The cable will be sized to allow for the increased ambient temperature,
25 which might occur under accident conditions.

H. Cable Spreading Room

The cable spreading rooms are the area provided above and
below the main control room where cables leaving the various control 48
board panels are dispersed into cable trays or conduits for routing to
all parts of the plant. The cable spreading rooms are arranged such 48
that Division 1 and III (which is routed in conduit) and primary PPS
cables are routed through the upper cable spreading room, and Division
II and secondary PPS cables are routed through the lower cable spreading
room. Since the cable spreading rooms are missile protected by their
seismic Category I walls and there are no internal sources of missiles,
such as high-pressure piping or rotating heavy machinery, the only
potential source of damage to redundant cables would be from fire.
Deluge spray systems are installed along cable trays as described in 48
Section 9.13.1 to ensure that potential for fire damage to cables is
minimized in the cable spreading room as described in Section 9.13.1.
Where cables of different divisions approach the same control panel with 12
separation of less than 1 foot, cables shall be run in metal (rigid or

Cable trays of different divisions which cross are to be separated vertically by a minimum clearance of 15 inches and a fire barrier, which extends for a distance of 5 feet on both sides of the crossing.

If a non-Class 1E tray crosses over or under cable trays of two redundant divisions, a fire barrier is to be installed to protect one of the redundant divisions for a distance of 5 feet on both sides of the crossing. The fire barrier is described in paragraph H following.

In any non-hazard zone, rigid steel conduit, flexible steel conduit, or steel wire duct are acceptable barriers between the two divisions. The minimum distance between these redundant enclosed raceways and between barriers and raceways shall be 1 inch, or as required by the Fire Hazard Analysis. Generally, sprinkler systems will be provided in areas of the plant containing cabling and conduit, except where the Fire Hazards Analysis determines that sprinklers are not required. In areas where redundant safety divisions are not affected by the same fire, area sprinklers will be added for economic not safety reasons. 48

Fire Hazard Zones

When routing of Class IE cable through a Fire Hazard Zone is unavoidable, only cables of one redundant division are permitted and these cables shall be protected, where necessary, by fire barriers or fire protection systems.

Pipe Break Hazard Zones

To the extent practical, Class IE cables are to be routed in areas remote from high energy piping. When routing of Class IE cable in the vicinity of high energy piping is unavoidable, the following criteria are applied to determine the separation or protection requirements:

1. Cables in systems which provide reactor protective actions or safe shutdown capability, but which are not required to function as a consequence of a pipe break in the hazard zone, may experience a loss of redundancy as a result of the event, provided that they retain their functional capability. A controlled shutdown shall be initiated if a loss of redundancy occurs which degrades the ability of the reactor protective system to initiate action by engineered safety features, or which degrades the safe shutdown capability.

F. Cables Within Control Boards and Other Panels

Within control boards and other panels, harnesses of different divisions are provided with a minimum of 6 inches free air separation. 12

The quality assurance program to ensure proper installation of firestops and seals will be in accordance with Sections 1, 2, 6, and 8 of RDT Standard F2-2, included in Appendix G of Chapter 17. 12 48

E. Physical Separation Criteria for Cables of Class IE Systems

Cables of Class IE systems which are run in trays are separated 12
37 into three redundant divisions; Division 1, Division 2 and Division 3. Cables designated in each redundant division shall be run in raceways separated from cables designated in the other division and from non-Class IE cables in accordance with the physical separation criteria listed below.

The minimum physical separation maintained between cables of two redundant divisions shall vary according to cable location with respect to potential hazards. Three general classifications of hazard 12
zones are defined for physical wiring separation considerations:

1. Areas in which the only potential hazard is a fire of an electrical nature. (Non-hazard zones)

2. Areas in which a potential fire hazard could exist as a consequence of the credible accumulation of a significant quantity of flammable material. (Fire Hazard Zones)

3. Areas in which a potential hazard could exist as a consequence of postulated pipe break events in high energy lines. (Pipe Break Hazard Zones)

The design intent is to provide greater physical separation than the minimum listed where consistent with practical plant layout.

Non-Hazard Zones

In Non-Hazard Zones, a minimum horizontal clear space of 3 feet will be maintained between cable trays of the two redundant divisions. If a horizontal clearance of less than 3 feet is unavoidable, a fire 12
barrier will be provided between the two divisions.

Vertical stacking of cable trays of the two redundant divisions is to be avoided wherever possible. Where cable trays of the two redundant divisions are stacked vertically, a minimum space of 5 feet will be provided between the divisions.

Fills greater than 40% in cable tray section containing power cables require review by the design engineering group. Cable tray fills for large diameter power cables are permitted to exceed 40% without a design review, if cables are installed not more than one layer deep in the tray.

Conduit shall be sized for a maximum percent fill of the inside area of the conduit in accordance with "National Electrical Code" Art. 346.

D. Sealing Raceway Blockouts and Wall and Floor Penetrations

Fire stops will be installed for cable trays wherever the cabling system passes through fire walls and floors other than the Reactor Containment outside walls. Cable and cable tray penetrations of 12 fire barriers will be sealed to give protection at least equivalent to that required of the Fire Barrier. Penetrations will be qualified to meet the requirements of ASTM E-119. The actual fire ratings of stops and penetrations will be determined by the Fire Hazards Analysis which will be provided in the FSAR. 48

For walls and floors, other than those associated with the cable spreading room, sealing of the penetration will be accomplished by the use of a non-flowing flame resistant compound such as "Flammastic". The material used will be equivalent to "Flammastic", which is manufactured by Flamemaster Corporation. The cables will be coated with this material for a minimum distance of twelve (12) inches on each side of the opening.

The vertical walls of the cable spreading room where the cables enter will be provided with openings for the passage of cable trays. After cable installation, the wall penetrations will be sealed to prevent propagation of fire. The openings will be first filled with packed "KAO WOOL" or equal. "KAO WOOL" is a fireproof aluminum silicate manufactured by Babcock and Wilcox. The "KAO WOOL" will be held in place by a steel plate fastened to each side of the wall and covering the opening around the tray or trays. Finally, the voids between cables tray, and steel plate will be filled with a flame resistant compound such as "Flammastic", or equal. At no time during the sealing will any installation material be flammable.

For sealing the cable passage openings between the cable spreading room and the Control Room after cable installation, Multi Cable Transit (MCT) manufactured by Nelson Electric, or its equal, will be used. MCT is composed of Tectron Modules that are fitted to the cables and opening to form a fire-proof barrier. 12

During the test, the Class IE AC and DC power distribution buses not under test are continuously monitored to verify absence of voltage at these buses.

The tests are repeated for the redundant load group and source.

Functional performance of loads is verified as follows:

During preoperational testing, functional performance of auxiliaries is verified by tests.

37 NRC Regulatory Guide 1.75 48

The system design includes only associated circuits that conform to the description in paragraph 4.5 (2) of IEEE 384-1974.

The system is being designed so that no associated circuits which do not conform with paragraph 4.5 (2) of IEEE 384-1974 will become part of the design. However, if such circuits do become necessary, they will be identified as such in the FSAR and an analysis or test will be conducted in accordance with paragraph 4.5 (3) of IEEE 384-1974.

The D. C. system non-Class IE loads will be supplied from two non-Class IE batteries. All Class IE DC loads will be supplied from two Class IE 125 Volt DC and one Class IE 250 volt DC batteries. The single line diagram for Class IE DC systems are indicated in Figure
37 8.3-2.

The AC loads which are not Class IE but are required for plant availability will be connected to the redundant non-Class IE motor control centers. These motor control centers will be provided with an incoming breaker in series with the Class IE breaker feeding these motor control centers from the 480 Volt load centers and will have capability to receive power from the Diesel Generator.

In the event of loss of all off-site AC power sources, and when the Diesel Generators are required to supply power, these motor control centers will be tripped-off by the undervoltage signal which starts the Diesel Generator and will be connected manually with proper administrative control from the Control Room or from the motor control center after being tripped. The trip can be initiated by the PPS signal, by loss of bus voltage or by the overcurrent relays and after an intentional time delay to permit completion of the automatic sequential loading of all Class IE loads as depicted in Table 8.3-1a and 8.3-1b. 16

Regulatory Position C.8 places additional restrictions on adequacy of separation of redundant circuits per Section 5.1.1.1 of IEEE Std 384-1974. The Project will comply fully with the requirements as set forth in IEEE Std 384-1974 as modified by Regulatory Guide 1.75.

26 48

such motors. Where test programs are required in LMFBR design to verify the adequacy of specific design features in lieu of other verifying processes, these test programs will include qualification tests on prototype continuous-duty Class I motors installed inside the containment.

The procedure for conducting these qualification tests are those specified by IEEE Std 334-1971, "IEEE Trial-Use Guide for Type Tests of Continuous-duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations".

If and when these qualification tests are used, they will be in agreement with Regulatory Guide 1.40, as follows:

1. To the extent applicable to an LMFBR auxiliary equipment that will be part of the installed motor assembly will be qualified in accordance with IEEE Std 334-1971.

2. The qualification tests will simulate as closely as practicable those design basis events which:

    a. Require the motor to either drive equipment which mitigates the consequences of the event or provide auxiliary support to such equipment, and

    b. Affect operation of the motor's auxiliary equipment.

NRC Regulatory Guide 1.41

Preoperational tests are performed to verify the independence of the redundant Standby AC Power Sources and between the redundant load groups described in Section 8.3.1.1.

The tests are performed as follows:

a. The power sources to the Class IE 4.16 KV AC, 250V DC distribution bus of the Diverse Power Supply, and 125 Volt DC power distribution
37 buses of the redundant load group not under tests are disconnected.

b. The load group of the Safety-Related AC Distribution System under test is isolated from the Plant Power Supply and the offsite AC power supplies, simulating an actuation signal which starts the diesel generator under test.

c. The actuation signal causes sequential starting of Class IE loads as described in Section 8.3.1.1.

The tests are of sufficient duration to attain steady-state operation of the Standby AC Power System as well as steady-state operation of the loads under test.

6683-10

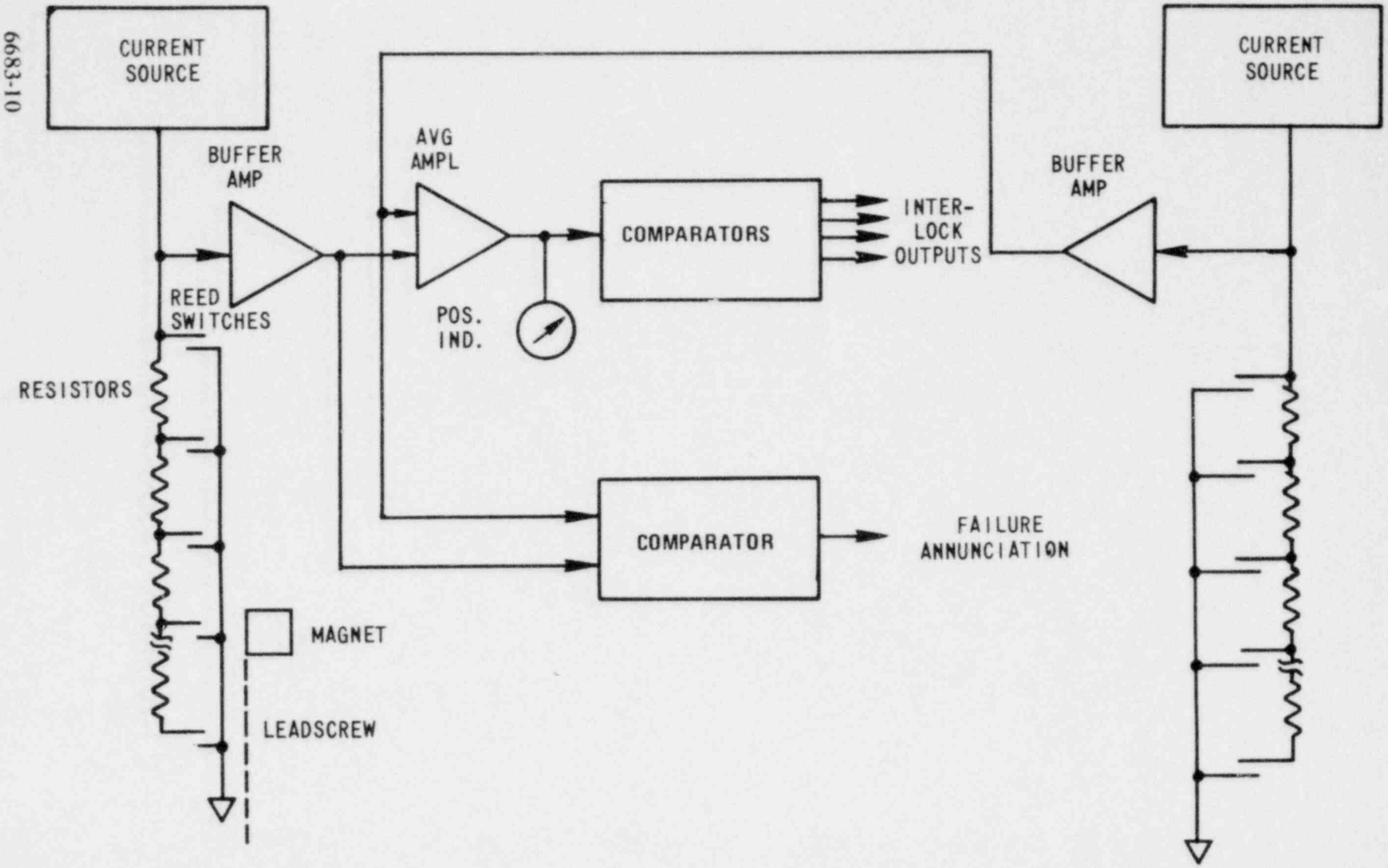


Figure 7.7-6 Absolute Position Indication System

NOTE: RELAYS ARE SHOWN FOR CONVENIENCE IN UNDERTAKING THE FUNCTION. HARDWARE IMPLEMENTATION MAY BE RELAY OR SOLID STATE.

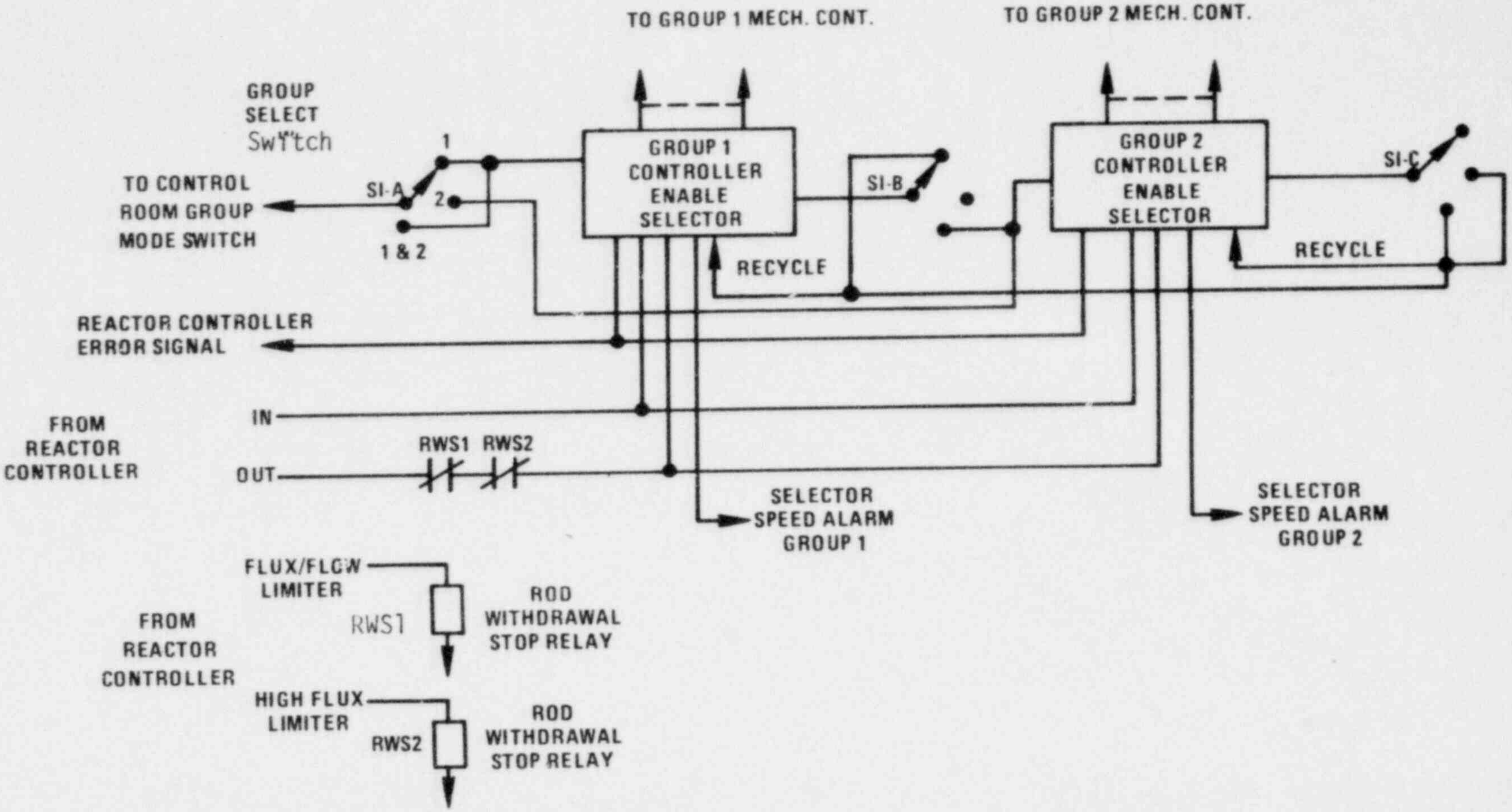


FIGURE 7.7-5 BLOCK DIAGRAM OF PRIMARY ROD GROUP CONTROL

Leak Damage Studies

Experimental studies have been conducted in the United States and in Europe over the past ten years which have given a broad background to the understanding of the behavior of leakage damage effect. Most of the experimental data taken with 2-1/4 Cr-1 Mo material have been obtained by injecting water or steam through hole type geometries at selected target configurations (jet leaks). In general, results of these studies have indicated that both adjacent tube wastage and self wastage
46 13 are possible damage mechanisms, as described in Reference 1, 3 and 4.

Adjacent tube wastage will occur with the proper leak size and orientation. For very specific conditions and geometries, some experiments have been performed where adjacent tube wastage occurred very rapidly in a localized area. Relating these specific conditions to CRBRP, adjacent tube wastage could occur which would result in tube failure in a very short period of time, less than one minute. However, it is not likely that leaks would be optimized as to leak geometry, location and orientation, as those utilized for the experiments. In the event that this did occur, the steam generator rupture discs provide necessary protection.

13

The second class of damage is self wastage around the leak site. Some experiments have noted that some very small leaks have experienced a sudden enlargement after a period or relatively steady operation as reported in Reference 1. The effect of this type of characteristic has been studied
13 by the GE/ANL Steam Generator Systems Development Program and is reported in
46 References 3 and 4.

Design Requirements

The design requirements for the Steam Generator Leak Detectors
13 have been selected as described below.

SGS Leak Detection Requirements

47

	Hydrogen Detectors	Oxygen Detectors	
Sensitivity	3 ppb	24 ppb	
Range	0.04-2 ppm	0.1-10 ppm	23
47 Response Time	≤30 sec.	≤30 sec.	

47

47 2. detection of a gradual concentration increase or decrease through several passes through the sodium.

Figure 7.5-4 illustrates typical first pass hydrogen concentration change as a function of water leak rate. As illustrated, a change in hydrogen concentration of a few ppb would be indicated at the detector for leak rates in the range of 10^{-4} lb/sec. Approximately one minute is required for the hydrogen to reach the detector and signal a leak. Detection capability can be extended to smaller leak sizes through the use of a rate of rise detection system. Several passes of sodium through the system would be required to allow the hydrogen concentration to build up. The sensitivity of this system will allow detection of leaks in the range of 10^{-5} lb/sec. Similarly, Figure 7.5-4A indicates the first pass oxygen
48 47 concentration as a function of water leak rate. Figure 7.5-5 illustrates the hydrogen concentration change with time for various sizes of leaks.

7.5.5.3.2 <u>Design Analysis</u>

A Steam Generator Leak Detection System is provided to comply
3 with CRBRP General Design Criteria 4 which calls for provision of leak detection in the Steam Generators. In order to show how the criterion will be satisfied, a review of leak damage studies is presented with the
13 resulting instrumentation requirements.

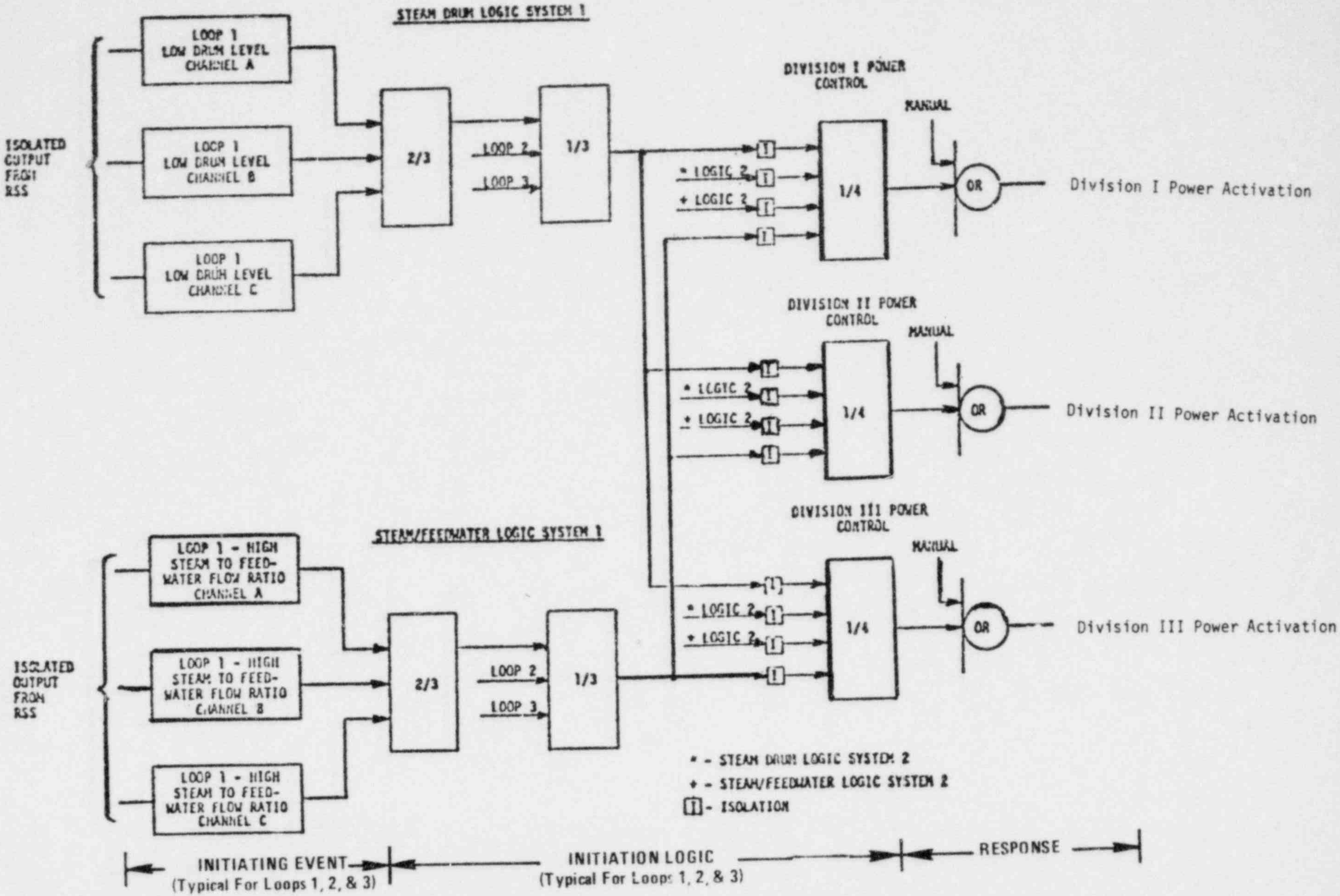


Figure 7.4-1 Steam Generator Auxiliary Heat Removal System Initiation Logic

SGS loop. The low steam drum level sensed by two of three redundant channels in any one loop provides a backup trip function. Additional redundancy is provided by three independent SGS steam supply loops serving one common trubine header. Any major break in the high pressure steam system external from the individual loop check valves will be sensed as a steam feedwater flow ratio trip signal in all three loops.

7.4.2.1.6 Actuated Device

The superheater outlet isolation valves utilize a high reali- ability air operated actuator. The superheater outlet isolation valves are designed to fail closed upon loss of electrical or air supply to the control solenoid.

7.4.2.1.7 Separation

The OSIS Instrumentation and Control System, as part of the PPS, is designed to maintain required isolation and separation between redundant channels (see Section 7.1.2).

7.4.21.8 Operator Information

Indication of the superheater outlet isolation valve position is supplied to the control room. Indicator lamps are used for open-close position indication to the plant operator.

7.4.2.2 Design Analysis

To provide a high degree of assurance that the OSIS will operate when necessary, and in time to provide adequate isolation, the power for the system is taken from energy sources of high reliability which are readily available. As a safety related system, the instrumentation and controls critical to OSIS operation are subject to the safety criteria identified in Section 7.1.2.

Redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the OSIS Instrumentation and Control System to perform its intended safety function. The system will be designed for fail safe operation and control equipment, where practical, will assure a failed position consistent with its intended safety function.

affecting the three steam supply systems and is provided if needed on a per loop basis. By definition, this zone of protection will include the high pressure steam supply system downstream from the individual loop check valves.

7.4.2.1.2 Equipment Design

A high steam flow-to-feedwater flow ratio is indicative of a main steam supply leak down stream from the flow meter or insufficient feedwater flow. The superheater steam outlet valves shall be closed with the appropriate signal supplied by the heat transport instrumenta-
48 tion system (Section 7.5). This action will assure the isolation of any steam system leak common to all three loops and also provide protection against a major steam condenser leak during a steam bypass heat removal operation.

7.4.2.1.3 Initiating Circuits

The OSIS is initiated by the SGAHRS initiation signal coincident with either a low superheater steam pressure signal or a high feedwater header pressure signal. The SGAHRS initiation signal is described in 7.4.1.1.3. This initiation signal closes the superheater outlet isola-
48 tion valves in all 3 loops when a high steam-to-feedwater flow ratio or a low steam drum level occurs in any loop. In each Steam Generator System loop, the three trip signals for high steam-to-feedwater flow ratio and the low steam drum level are input to a two of three logic network. If two of three trip signals occur in any of the 3 loops, the OSIS is nitiated, and all 3 loops are isolated from the main superheated steam system by closure of the superheater outlet isolation valves.

7.4.2.1.4 Bypasses and Interlocks

Control interlocks and operator overrides associated with the operation of the superheater outlet isolation valves have not been completely defined.

Bypass of OSIS may be required to allow use of the main steam bypass and condenser for reactor heat removal. In case the OSIS is initiated by a leak in the feedwater supply system, the operator may decide to override the closure of certain superheater outlet isolation valves.

7.4.2.1.5 Redundancy and Diversity

Redundancy is provided within the initiating circuits of OSIS. The primary trip function takes place when a high steam-to-feedwater flow ratio is sensed by two of three redundant subsystems or any one

- High Auxiliary Feedwater Temperature
- Low Pump Suction Pressure
- Low Pump Discharge Pressure
- Low Steam Turbine Inlet Pressure
- 48 | SGAHARS Initiation Logic Tripped

Additional indicators and alarms are provided at the local instrumentation and control panels. Most information is also available to the operator via the Plant Data Handling and Display System (DH&DS). All measured parameters providing either indication, alarm or input to the DH&DS are shown on the SGAHRS P&ID, Figure 5.1-5.

7.4.1.2 Design Analysis

To provide a high degree of assurance that the SGAHRS will operate when necessary, and in time to provide adequate decay heat removal, the power for the system is taken from energy sources of high reliability which are readily available. As a safety related system, the instrumentation and controls critical to SGAHRS operation are subject to the safety criteria identified in Section 7.1.2.

Redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the SGAHRS Instrumentation and Control System to perform its intended safety function. The system will be designed for fail safe operation and control equipment where practical and will, in the event of a failure, assume a failed position consistent with its intended safety function.

Because there are three redundant decay heat removal loops, the instrumentation and controls associated with each individual loop (e.g., auxiliary feedwater flow and air cooled condenser control systems) do not independently meet single failure criteria. However, when taken collectively as a system, they provide the single failure capability required.

7.4.2 Outlet Steam Isolation Instrumentation and Control System

7.4.2.1 Design Description

7.4.2.1.1 Function

The Outlet Steam Isolation Subsystem (OSIS) provides isolation of steam system pipe breaks. Steam system isolation is a necessary function for safe shutdown in those pipe break conditions

7.4.1.1.9 Operator Information

Indicators and alarms are provided to keep the plant operator informed of the status of the SGAHRS. The following items are located on the Main Control Board for operator information.

Analog Indication

- Protected Water Storage Tank Level
- Auxiliary Feedwater Flow (each loop)
- Pump Suction Header Pressure
- Pump Discharge Header Pressure
- Steam Turbine Inlet Pressure
- Feedwater Heater Inlet and Outlet Temperatures
- Air Cooled Condenser Inlet and Outlet Average Air Temperatures
- Air Cooled Condenser Return Water Flow and Temperature
- Individual Auxiliary Feedwater Pump Discharge Pressure

Indicating Lights

- Position of All Air Cooled Condenser Doors
- Position of All Isolation and Control Valves
- Operating Status of All Motors
- 48 SGAHRS Initiation Logic Reset

Annunciators

- Start-up of Air Cooled Condenser
- Start-up of Auxiliary Feedwater Pump
- Low Protected Water Storage Tank Level
- High Feedwater Pump Discharge Temperature
- Closure of Auxiliary Feedwater Pump Isolation Valves

7.4.1.1.5 Redundancy/Diversity

The SGAHRS (fluid system and mechanical components) is designed with suitable redundancy and diversity so that it can perform its safety functions following a single failure of an active component for anticipated, unlikely and extremely unlikely plant conditions. The design of SGAHRS relating to these objectives is discussed in Section 5.6.1.

Redundancy and diversity are also provided within the initiating circuitry of the SGAHRS control system. As shown in Figure 7.4-1, the system is actuated on two-out-of-three signal from either low steam drum level, or high steam-to-feedwater flow ratio.

7.4.1.1.6 Actuated Devices

All automatic valves and motors in the SGAHRS are provided with remote manual control capability, so that the entire system can be operated from the control room. Valves or motors that are automatically actuated are equipped with devices requiring manual reset by operator.

All isolation valves within the SGAHRS utilize an air operated actuator. Air is either supplied to or vented from the actuator via a three-way solenoid valve. All isolation valves are designed to fail to the position of greater safety upon loss of electrical or air supply to the control solenoid.

All required components of the SGAHRS instrumentation and control system operate on a vital electrical bus.

7.4.1.1.7 Testability

Instrumentation and controls for the SGAHRS are designed and arranged to allow for complete testability during reactor power operation. Bypassing of the actuated components (i.e., isolation valves and motors) is not required during testing as operation of these components during power operation poses no penalty on plant operation.

7.4.1.1.8 Separation

The SGAHRS instrumentation and control system, as part of the PPS, is designed to maintain required isolation and separation between redundant channels (see Section 7.1.2.2).

- **Auxiliary Feedwater Discharge Header and Pump Suction Isolation Valves.** The valves in the Auxiliary Feedwater Discharge Header and the Auxiliary Feedpump Suction Lines are provided to assure that pipe break or leakage in the Auxiliary Feedwater System Pressure Lines can be isolated. This prevents complete drainage of the Protected Water Storage Tank through the postulated break. Based on the Auxiliary Feedwater header pressure input signal and the Auxiliary Feedwater Pump speed input signal, control logic determines whether a feedwater leakage situation exists and closes the appropriate isolation valves.

7.4.1.1.3 Initiating Circuits

48 The Reactor Shutdown System (see Section 7.2) provides initiation signals to the SGAHRS Power Divisional Control System to sequentially start the three Auxiliary Feedwater Pumps and the three Protected Air Cooled Condensers when either a low steam drum level or
47 high steam-to-feedwater flow ratio occurs in any one of the three Steam Generator System (SGS) loop subsystems. In each subsystem, the three trip signals for low steam drum level and the three trip signals for high steam-to-feedwater flow ratio are each isolated and input to redundant two out of three logic networks. The outputs from the redundant logic networks are each isolated within the SGAHRS
48 divisional control system and combined in a one-of-four logic to initiate SGAHRS. If two of three trip signals occur in any subsystem, the SGAHRS is initiated. The sequence of decay heat removal events is shown in Table 7.4-1. The scheme used for initiating the SGAHRS is shown in Figure 7.4-1.

Since the automatic activation and control of auxiliary feedwater flow is necessary to assure decay heat removal, provisions are included in the design to assure that the automatic initiation takes precedence. A startup signal to the feedwater pumps overrides a manual control signal. Similarly, a signal to open the isolation valves overrides a manual closure signal.

7.4.1.1.4 Bypasses and Interlocks

Bypasses are required on the steam to feedwater flow mismatch and steam drum level subsystems to allow system reset and reactor startup without initiating SGAHRS. These bypasses will be
48 implemented as described in the Reactor Shutdown System (Section 7.2). Control interlocks associated with the operation of active components have not been completely defined.

Table 6.2-5 Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Sodium Drain Line(Spare for Future Use)	9.3	1	Globe	3"	N/A	N/A	Closed	Manual	Closed	Manual	N/A	<30	TBD
Sodium Transfer Line (In-Cont. to Ex-Cont. Stor. Tank)	9.3	1	Globe	4"	N/A	N/A	Closed	Manual	Closed	Manual	N/A	<30	C
Sodium Transfer Line (EVS Fill & Drain)	9.3	1	Globe	3"	N/A	N/A	Closed	Manual	Closed	Manual	N/A	<30	C
48 NaK DHRS From Containment	9.3	1	Globe	6"	N/A	Fail in Place	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	H
48 NaK DHRS To Containment	9.3	1	Globe	6"	N/A	Fail in Place	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	H
RAPS to Cold Box	9.5	2	Globe	1½"	CIS	Closed	Closed	Remote Manual	Open	Automatic	Remote Manual	<10	E
CAPS Inlet Header	9.5	2	Globe	3"	CIS	Closed	Closed	Remote Manual	Open	Automatic	Remote Manual	<10	E
RAPS to Recycle Argon Vessel	9.5	2	Globe	1½"	CIS	Closed	Closed	Remote Manual	Open	Automatic	Remote Manual	<10	E

47

Table 6.2-5 Lines Penetrating Containment

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Decontamination Waste Water Return	9.2	2	Gate*	3"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote Manual	<4	B
IHTS Piping Loop No. 2 Inlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping Loop No. 2 Outlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping Loop No. 3 Inlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping L Loop No. 3 Outlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping Loop No. 1 Inlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping Loop No. 1 Outlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

47

Table 5.6-13

DHRS EQUIPMENT LIST AND MATERIAL SPECIFICATIONS

	Component	ASME Section III Class	Material	Design Temperature (°F)	Design Pressure (psig)
	Overflow Vessel	1	SS	900	15
	Makeup Pump	1	SS	900	100
48	Overflow Heat Exchanger	1	SS	650	100
46	EVST Airblast Heat Exchanger	2	SS	650	100
	EVST Nak Exp Tanks	2	SS	650	100
	DHRS Nak Exp. Tank	2	SS	650	100
	EVST Nak Pumps	2	SS	650	100
	EVST Nak Diff. Cold Traps	2	SS	650	100
	Sodium Piping:				
	Overflow Line	1	SS	900	15
	Makeup Pump Suction	1	SS	900	15
46	Makeup Pump Discharge to Reactor	1	SS	900	100
26	Nak Piping	2	SS	650	100

TABLE 5.6-12

46 PACC SUBSYSTEM DYNAMIC (DESIGN FLOW) PRESSURE DROP AND HEAD LOSSES

Steam Inlet Piping	ΔP*	H2**
(1) Gate Valve (fully open)	0.05	0.17
(2) Piping and Elbows	0.7	2.46
(3) Steam drum to pipe	0.05	0.2
(4) Pipe to inlet header	0.25	0.86
	1.05	3.69
PACC Tube Bundle		
(1) Inlet header to tubes	0.005	0.015
(2) Tubes	0.32	1.13
(3) Tubes to exit header	0.002	0.007
	0.327	1.152
Condensate Return Piping		
(1) Exit header to pipe	0.016	0.06
(2) Gate Valve (fully open)	0.014	0.05
(3) Orifice flow meter	5.9	21
(4) Piping and Elbows	0.28	1.04
	6.21	25.15
Loop (Total)	7.59	29.99

*ΔP = Dynamic pressure drop (frictional losses), psi
**H_2 = Head loss, ft. water 25

5.5.3.11.5 Compatibility with External Insulation and Environmental Atmosphere

Compatibility of austenitic stainless steel with external insulation is assured as set forth in 5.3.3.10.4. Strict control of halide contents in inulation materials is required. Carbon steels and 2-1/4 CR-1 Mo are compatible with external insulation during normal operation in the absence of excessive moisture. Excessive moisture is prevented by quality controlled installation and operating procedures.

5.5.3.12 Protection Against Environmental Factors

Protection for the principal components of the SGS against environmental factors is provided by the structural integrity of the Steam Generator Building. Environmental factors to be considered include the following:

Fire Protection - See Section 9.13.
Flooding Protection - See Section 3.4.
Missile Protection - See Section 3.5.
Seismic Protection - See Section 3.7 and 3.8.
Accidents - See Section 15.6

Corrosion allowances for both steam and sodium side 2-1/4 Cr-1 Mo steel will be included in the design. These corrosion allowances are based on recommendations from the steam generator module designer (Ref. 3).

No specific protection is required for protecting Type 304 SS or 2-1/4 CR-1 Mo steels against intergranular attack, stress-corrosion or general corrosion, provided that specified sodium purity is maintained.

In water or steam, carbon steel and 2-1/4 Cr-1 Mo steel are susceptible to caustic gouging and possibly caustic stress corrosion cracking. Maintaining the feedwater and steam drum purity levels as stated below will prevent these forms of localized attack. For normal operation other than start-up conditions, the feedwater purity at the drum inlet will be specified as follows:

	Feedwater Impurities	Steady State
	Suspended solids ppm max.	0.016
	Dissolved oxygen ppm max.	0.007
	Silica, ppm max.	0.02
	Iron as Fe, ppm max.	0.01
46	Copper as Cu, ppm max.	0.0015
	Hydrazine(residual) ppm max.	0.015
	Conductivity (cation) @ 77°F-micro-mho/cm max.	0.3
	pH at 77°F for austenitic heaters	8.7-9.1
46 45	Chlorides, ppm max.	0.009

42 Limited duration operation with impurity level limits increased by a factor of two is allowable for periods not to exceed 24 hours in special instances. These special instances are defined to include: condensate polishing system perturbations, such as those immediately associated with a termination of regeneration. These instances shall not exceed 6% of
41 the total time.

Corrosion impurities may enter the feedwater system through condenser leakage and/or poor makeup water. To guard against damage from such sources, the feedwater and steam drum water are maintained at levels stated in the above table by full flow demineralization and continuous steam drum drainage or blowdown to a maximum of 10%. (See Section 10.4.7).

To determine the feedwater and recirculating water quality, in-line analyses for conductivity and sodium content are performed for the introduction point into the steam generator system and at the evaporator inlet. The steam drum water is monitored at the drain or blowdown line for the same chemicals. The condenser hot-well is monitored for conduc-
48 tivity and sodium ions to guard against condenser leakage. The demineralizer effluent is guarded against impurities break-through by in-line measurements of silica, conductivity and sodium. Finally, the feedwater train is monitored downstream of the deaerator for pH and oxygen content to prevent potential corrosion of this portion of the steam system. An alarm will be coupled with the most critical in-line measurements to signal departure from specified levels.

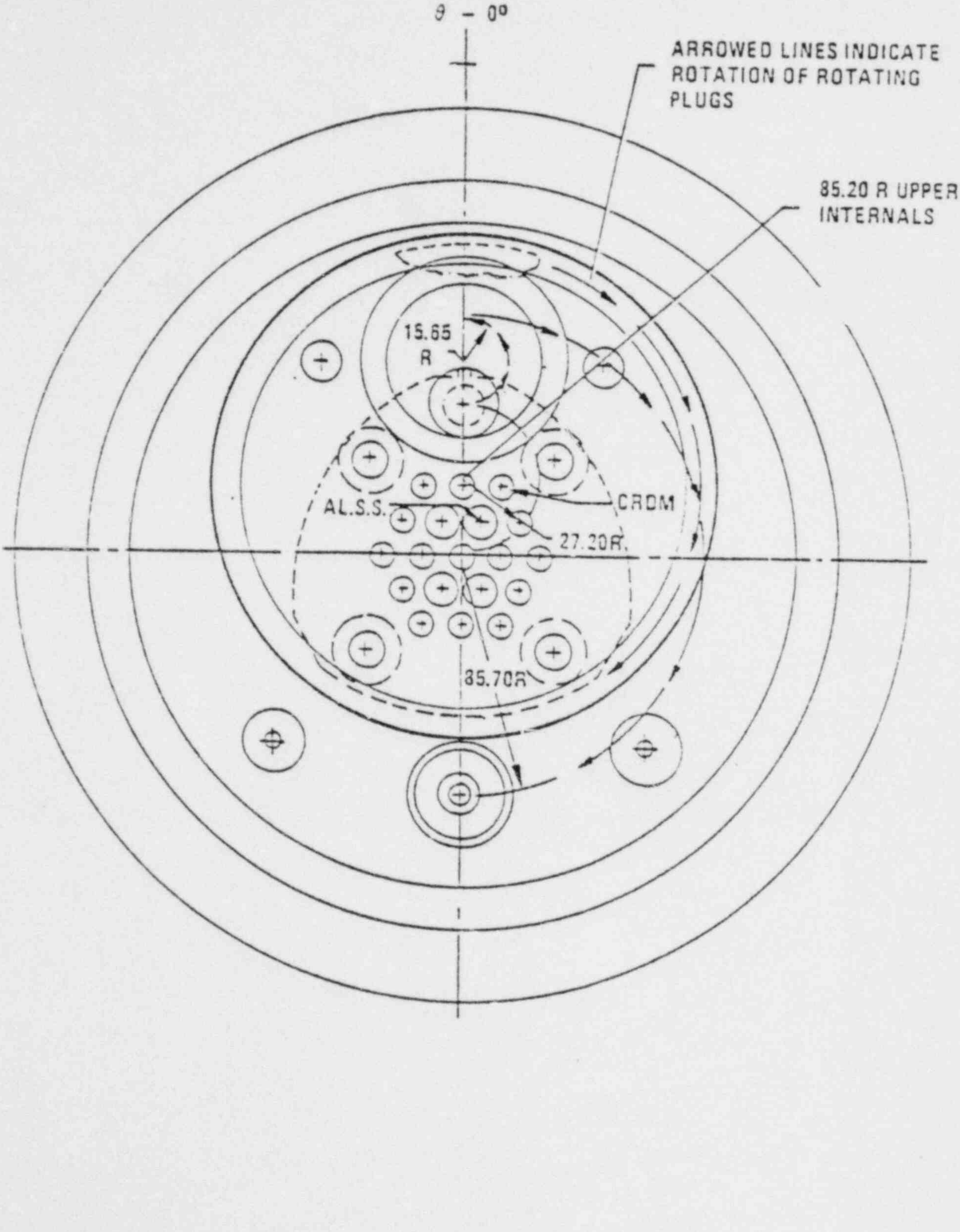
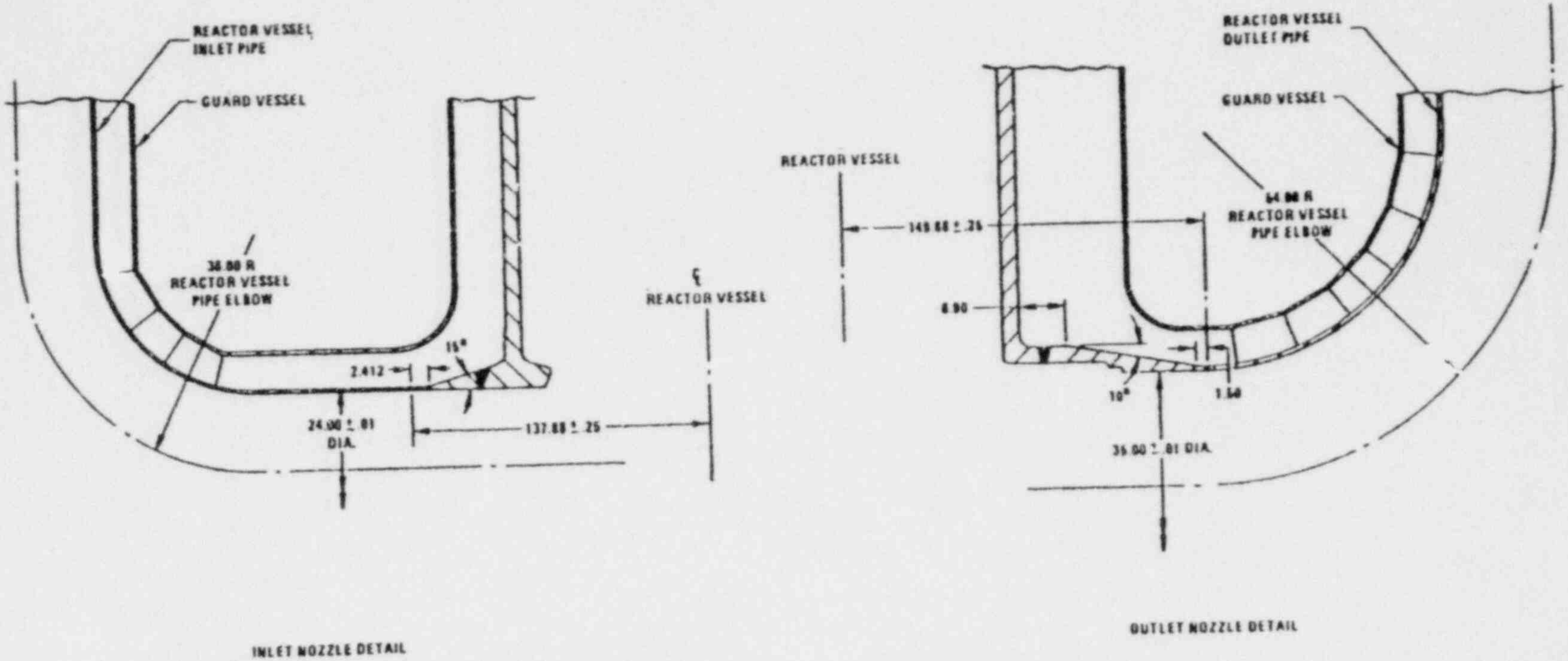


Figure 5.2-2 Closure Head Top Schematic View



Amend. 48
Feb. 1979

FIGURE 5.2-1B REACTOR VESSEL INLET AND OUTLET NOZZLE AND PIPE DETAIL

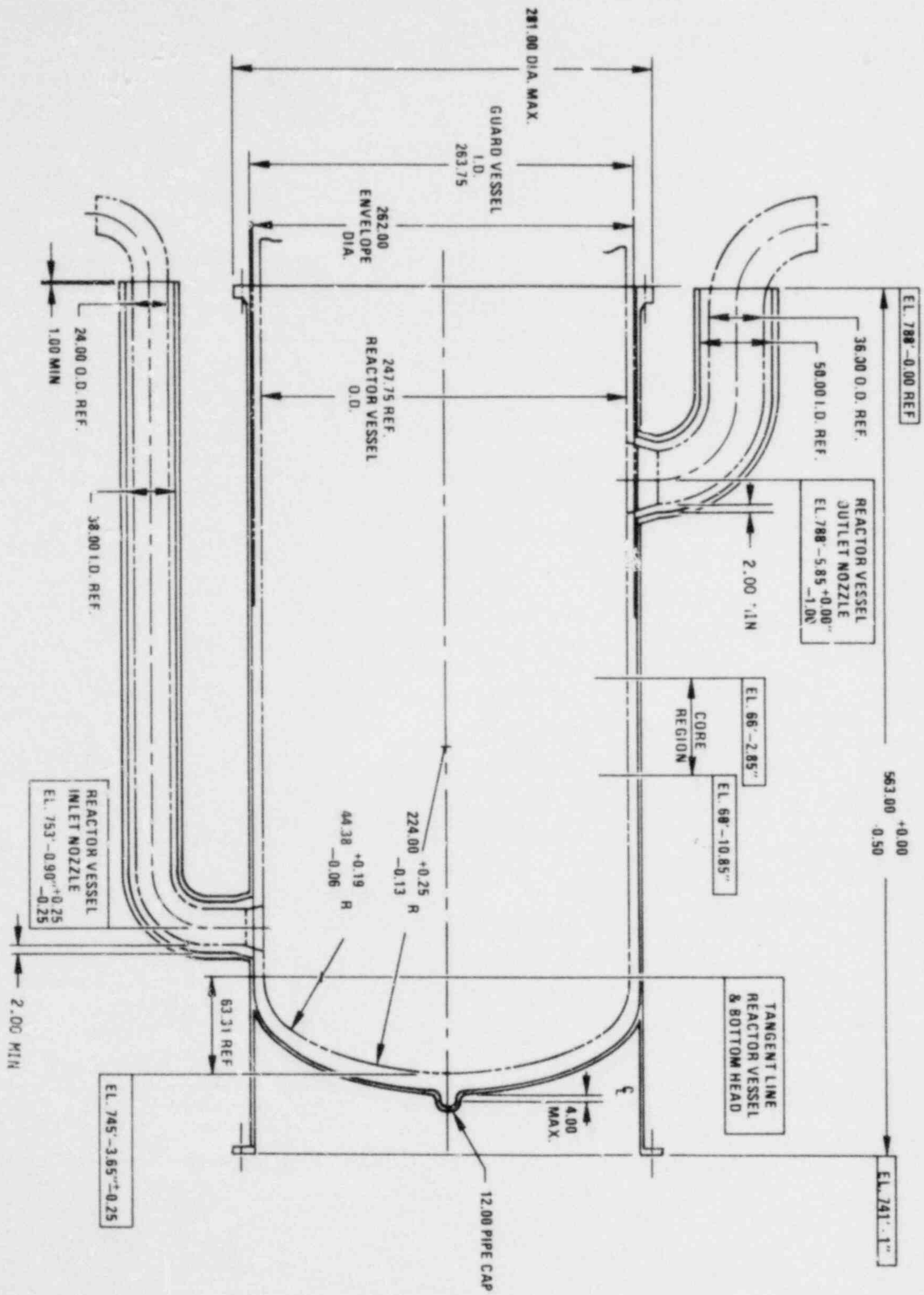


Figure 5.2-1A Reactor Vessel Guard Vessel

0285-1

Amend. 46
August 1978

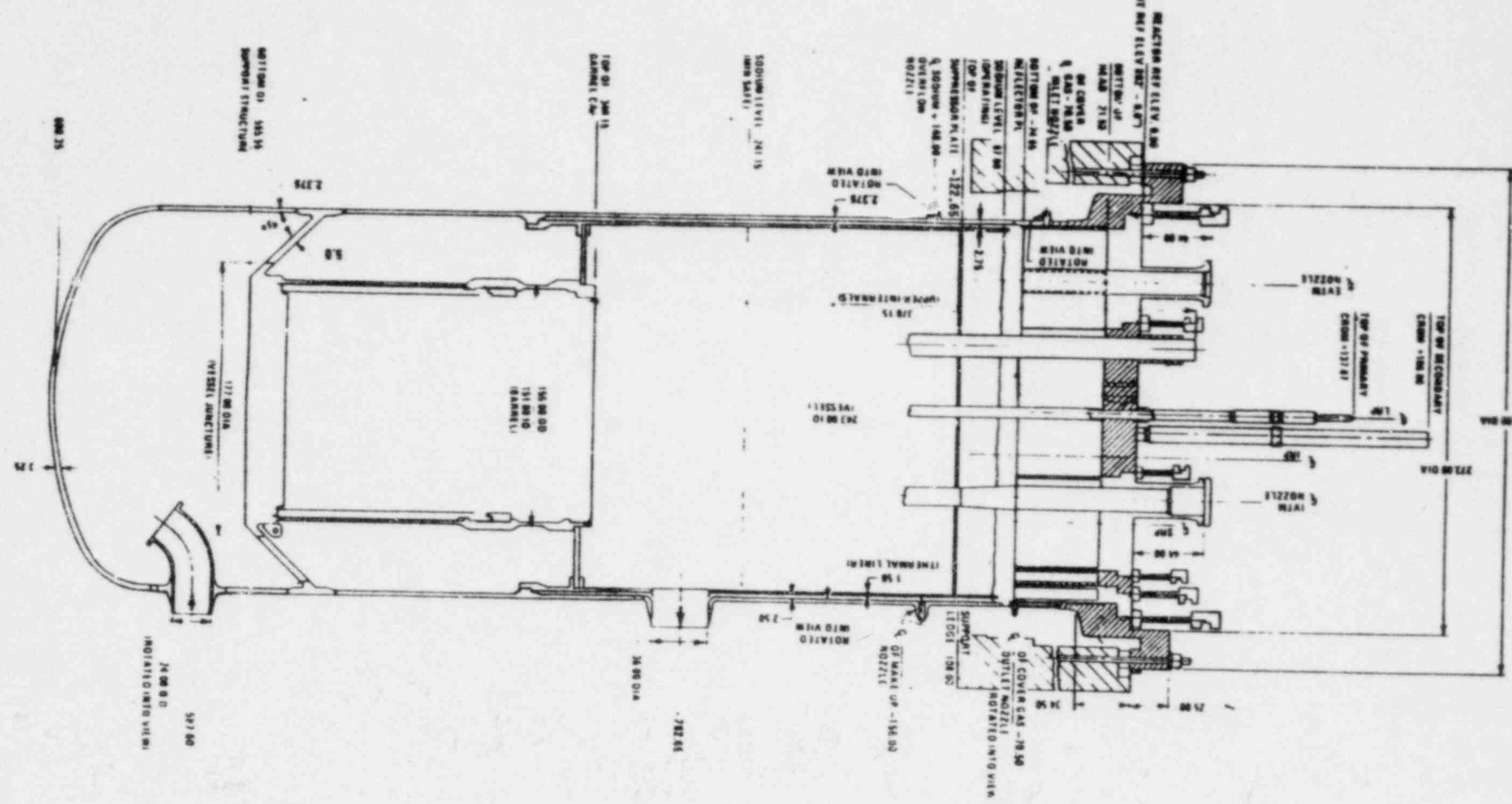


FIGURE 5.2-1 EXPANDED SCHEMATIC OF REACTOR VESSEL AND CLOSURE HEAD

Due to the inverse lifetime temperature behavior (fuel assembly rod temperatures decrease and radial blanket assembly rod temperatures increase with life, see Section 4.4.3.3), the maximum allowable cladding temperature has been set at different values in the fuel and radial blanket assemblies to meet lifetime objectives. The radial blanket assemblies temperatures are higher at end-of-life and internal pressures are significantly less in the radial blanket asssembly rods than in the fuel assembly rods.

The criterion in selecting the number and location of orificing zones was to equalize, as uniformly as practical, the maximum cladding temperature in the fuel assemblies and in the radial blanket assemblies separately, and at the same time to achieve a simple and economic design by minimizing the number of discriminators.

Figure 4.4-9 presents the selected orificing arrangement: the fuel assemblies are divided in five flow zones and the radial blanket assemblies in four. Also shown in the figure are the individual flow rates for the fuel, radial blanket and primary and secondary control assemblies.

The radial blanket shuffling scheme is shown in Figure 4.4-10. As indicated, three types of assemblies (A, B, C) go through a double shuffle with two years successive residence in the inner, middle and outer row; two types (D and E) go through a single shuffle with three years residence in each position, while assembly F remains at its location during the entire six years lifetime. The criterion in shuffling the blanket assemblies to a lower power position was to not exceed the limiting linear power rating of 20 KW/ft which would cause fuel centerline melting.

Also illustrated in Figure 4.4-10 is the management scheme of the fuel assemblies where approximately one-third of them are replaced at each annual refueling.

In determining the orificing scheme, the envelope of end-of-life (EOL) conditions has been considered both for fuel (end of third year residence of each assembly) and radial blanket (end of second, or third or sixth year residence time in each position) assemblies.

In comparison to a scheme which considers the envelope of beginning-of-life (BOL) conditions, the adopted scheme will show somewhat higher cladding temperatures at BOL, but in lower temperatures at EOL and consequently in lower ultimate cladding strain (which is the limiting criterion to meet burnup and lifetime objectives, see Section 4.4.1), since the fission gas pressure is maximum at end-of-life (see Section 4.4.3.3).

The fuel/radial blanket assemblies flow split (Section 4.4.2.4.3) of 80/12% of total reactor flow, respectively, was selected to meet both prescribed fuel burnup and radial blanket residence time goals.

48

4.4.2.4.2 Fuel and Radial Blanket Assemblies Orificing Criteria

The orificing scheme for the fuel and radial blanket assemblies has been selected on the basis of equalizing the maximum cladding midwall temperature at equilibrium cycle end-of-life conditions. The end-of-life temperatures were utilized since they are the most effective in determining the total cladding strain integrated over the assembly life, which is the ultimate parameter in establishing whether the burnup and residence time objectives are met. In addition, the fission gas pressure, which is the principal contributor to cladding strain, is maximum at end-of-life. The radial blanket assemblies cladding temperature is also maximum at discharge conditions due to the progressive production of plutonium.

Equilibrium core rather than first core conditions were adopted as the basis for orificing since a) the CRBRP will operate at equilibrium for more than 80% of its lifetime, b) at full power, temperatures are lower at first core than at equilibrium core conditions, and c) operation at reduced power is possible during the initial years if necessary.

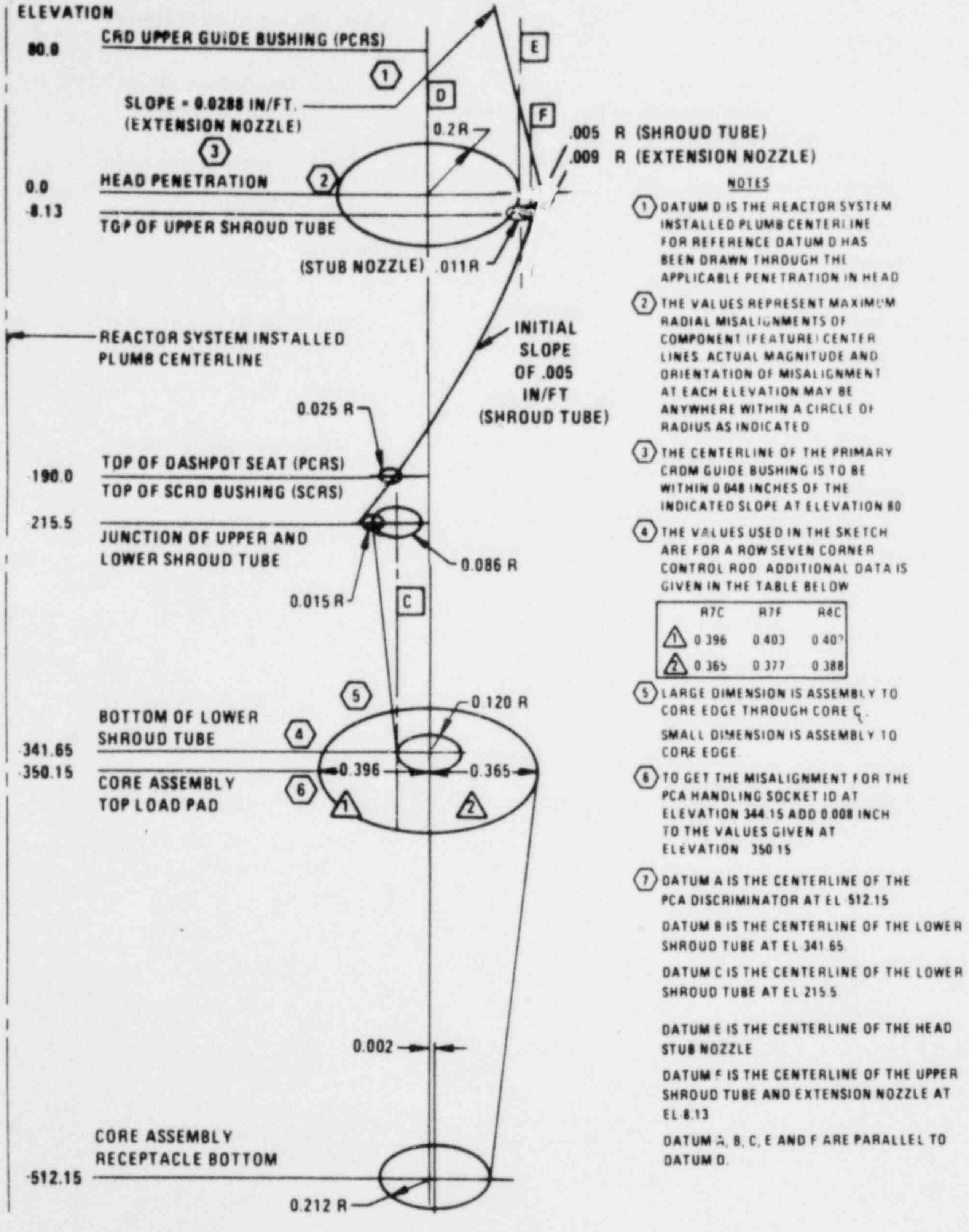
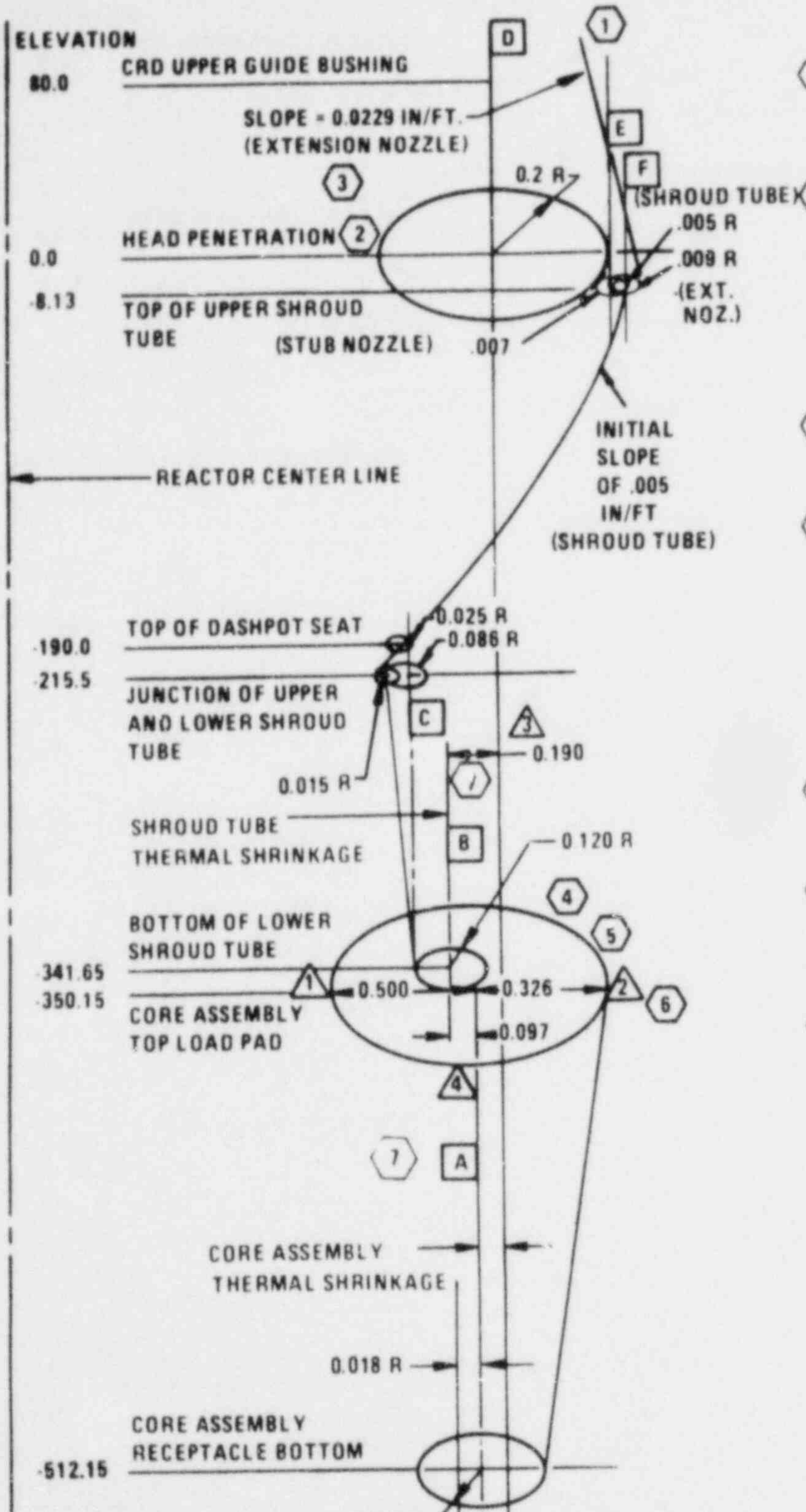


Figure 4.2-95B

1197-1 CONTROL ROD SYSTEM MAXIMUM MISALIGNMENT SOURCES FOR THE REACTOR OPERATING CONDITION



NOTES

(1) DATUM IS THE REACTOR SYSTEM INSTALLED PLUMB CENTERLINE FOR REFERENCE DATUM D HAS BEEN DRAWN THROUGH THE APPLICABLE PENETRATION IN HEAD

(2) THE VALUES REPRESENT MAXIMUM RADIAL MISALIGNMENTS OF COMPONENT (FEATURE) CENTERLINES ACTUAL MAGNITUDE AND ORIENTATION OF MISALIGNMENT AT EACH ELEVATION MAY BE ANYWHERE WITHIN A CIRCLE OF RADIUS AS INDICATED TEMPERATURE ASSUMED TO BE 400°F

(3) THE CENTERLINE OF THE PCRDM GUIDE BUSHING AT ELEVATION 80 SHALL BE WITHIN 0.048 INCHES OF THE INDICATED SLOPE

(4) THE VALUES USED IN THE SKETCH ARE FOR A ROW SEVEN CORNER CONTROL ROD ADDITIONAL DATA IS GIVEN IN THE TABLE BELOW

	R7C	R7F	R4C
1	0.500	0.507	0.466
2	0.326	0.334	0.391
3	0.190	0.165	0.096
4	0.097	0.084	0.049

(5) LARGE DIMENSION IS ASSEMBLY TO CORE EDGE THROUGH CORE ℄ SMALL DIMENSION IS ASSEMBLY TO CORE EDGE

(6) TO GET THE MISALIGNMENT FOR THE PCA HANDLING SOCKET ID AT ELEV 344.15 ADD 0.008 IN. TO THE VALUES GIVEN AT ELEV 350.15

(7) DATUM A IS THE CENTERLINE OF THE PCA DISCRIMINATOR AT EL 512.15

DATUM B IS THE CENTERLINE OF THE LOWER SHROUD TUBE AT EL 341.65

DATUM C IS THE CENTERLINE OF THE LOWER SHROUD TUBE AT EL 215.5

DATUM E IS THE CENTERLINE OF THE HEAD STUB NOZZLE AT EL 8.13

DATUM F IS THE CENTERLINE OF THE UPPER SHROUD TUBE AND THE EXTENSION NOZZLE AT EL 8.13

DATUM A, B, C, E AND F ARE PARALLEL TO DATUM D

Figure 4.2-95A

CONTROL ROD SYSTEM MAXIMUM MISALIGNMENT SOURCES FOR THE REACTOR REFUELING CONDITION

1197-2

Assumption 6 (above) was the basis for defining a "worst case" set of input misalignments for the control rod system components. This "worst case"
48 is represented on Figure 4.2-95A, and was established in accordance with the reference control rod system drawings and documents.

Forces were determined for eight different locations of the control rod along its translational travel. Salient features of these locations and the retardation forces (equal to the coefficient of friction times the sum of the absolute values of the reaction forces) are given in Table 4.2-43. It can be seen that except for the full-in position, there is only a small difference in the drag forces. The full-in drag force is effective only over approximately the last six inches of travel and its magnitude is still compatible with meeting the required scram time.

Torsional misalignment of the control rod system results from manufacturing tolerance on twist of the control assembly outer duct. Conservative assumptions of initial line on line contact at both the torque taker and control assembly wear pads were made for the full-in position; thus, the torsional misalignment increases as the control assembly is withdrawn and is maximum at the full-out location.

It was conservatively assumed that the control rod shaft below the coupling makes contact with the outer sleeve so that only a 4-in. length of the shaft is effective in the torsional spring stiffness calculation. Based on the above, the maximum attainable torsional misalignment retardation force was calculated to be 34-lb_f at the fully withdrawn positions.

An estimate of the upper bound retardation force resulting from system misalignments was made by considering the torsional and lateral cases together.

The maximum total retardation force from lateral misalignment occurs when the control rod is near full insertion and the control rod shaft coupling rubs against the inside diameter of the scram arrest flange. The magnitude of the drag force for this position, shown on Table 4.2-43, is 190 lbs, the out motion limiter pawl drag load must be
47 added to the internal misalignment drag force in the translating assembly.
This drag load results from the spring loaded pawls ratchetting along the leadscrew as it scrams. The magnitude of this force is less that 20 lb_f based on analysis. An additional drag force can potentially occur as a result of control assembly duct bowing. Design requirements for control assembly clearances (see Table 4.2-36) assure that this drag force is less than 25-lb_f. For the preliminary analyses, this maximum value has been applied for all rod withdrawal positions.

The resulting total drag forces for the PCRS are shown in Table 4.2-43. Drag forces are less than 180-lb_f over the first 31-in. of rod insertion from an initially fully withdrawn rod position of 37-in. Over the last 6-in., the total drag force increases to 225-lb_f.

4.2.3.3.1.2 Control Assembly Bowing Analysis

Bowing of both the movable control rod and the fixed outer duct must be considered in the control assembly bowing analyses. This section summarizes analyses which are being performed to assure that adequate clearances are provided to prevent large drag forces from duct bowing.

The outer duct is restrained laterally (see Section 4.2.2) at the inlet nozzle, the above core load plane and the top load plane in a manner identical with the adjacent fuel assemblies.

The control rod moves inside of the outer duct and is guided by Inconel 718 hexagonal wear pads provided at its upper and lower ends as shown in the control assembly schematic (Figure 4.2-104). A nominal diametral clearance of 0.1-in. exists between the wear pads and the outer duct inner wall and a diametral clearance of 0.24 exists between the inner and outer duct. The control rod shaft coupling diametral clearance with the handling socket scram arrest flange is 0.38-in. over the last six inches of rod insertion. At withdrawn distances greater than six inches, the diametral clearance increases to 1.2-in.

The control rod inner duct is essentially free to bow with only the thin flexible shaft restraining the duct bowing. As a consequence, the potential for high drag forces as a result of inner duct bowing is minimized by the shaft flexibility until extreme bow conditions result in three point contact of the pin bundle with the outer duct. The clearance requirements of Section 4.2.2.1.2 establish a 25-lb maximum drag force increase from bowing as a design basis and require that three point pin bundle contact be excluded by appropriate inner to outer duct clearance. The results of a preliminary bowing analysis for Row 4 and Row 7 control assembly outer ducts at operating temperature are shown in Figure 4.2-110. The analysis was performed in conjunction with the core restraint system analyses (see Section 4.2.2) using the ANSYS structural analysis code and includes: the effects of interaction with other adjoining assemblies; the effects of thermal gradients across the assemblies; and the effects of thermal creep and irradiation swelling. Displacements shown in the plotted curve represent the differential bowing and thermally-induced displacements of the outer duct. The relative displacement between the top and bottom of the duct is due primarily to differences in thermal expansion between the upper internals structure (which engage the control assembly handling sockets) at core outlet temperature and the core support structure (which engages the inlet nozzle) at core inlet temperature.

Figure 4.2-111 shows a schematic for possible sequences of progressive bowing of both the inner and outer ducts. Duct bowing tends to progress from sketch (a) to sketch (g) of Figure 4.2-110 as the duct fluences increase.

Bowing of the inner duct assembly is induced by transverse thermal growth and irradiation-induced swelling. Maximum transverse thermal gradient results during power operation, with the pin bundle fully inserted and off

Cell Liners and Liner Support System	Carbon Steel*
Piping	Carbon Steel & Stainless Steel
Pipe Insulation and Canning Material	TBD
Pipe Supports and Auxiliary Steel	Carbon Steel
Conduit	Carbon Steel
Embedments	Carbon Steel & Stainless Steel
Equipment (pumps, tanks, etc.)	TBD
Trace Heaters	TBD

Various candidate materials for sealing cell penetration are currently undergoing evaluation and testing. Those materials selected will be specified in a future amendment. 31

3A.1.8 Inner Barrier

The Inner Barrier is a limited leakage barrier which consists of the reactor closure head, the reactor cavity, and the PHTS pipeway cells. The Inner Barrier is designed so that the reactor closure head will maintain its leak tightness (less than 1000 SCC/sec for a minimum of 1000 sec after initiation of the hypothetical accident.) The Reactor Cavity (RC) will limit leakage to 100 volume percent per day at 15 psid until venting of the reactor cavity to above the operating floor occurs. The Inner Barrier is sealed from the PHTS cells and overflow cells by limited leakage seals. The seals are designed to limit leakage under temperatures that might be expected as discussed in Reference 10 of Section 1.6. Electrical penetrations running out of the cavity and pipeways are sealed with limited leakage type electrical penetrations.

Intercommunication between the RC and the Reactor Containment Building (RCB) above the operating floor is provided by a venting system including two separate vent paths. The isolation between the reactor cavity and RCB atmosphere to retain the inerted atmosphere of the reactor cavity is accomplished by a rupture disc for each vent line. In addition a normally open valve, remotely operated from the RCB or the Control Room, is provided to isolate the reactor cavity in the unlikely event of failure of the rupture disc during normal plant operation or minor accident. The rupture disc will burst at 15 psi
32 pressure differential from the reactor cavity to the RCB atmosphere.

A reactor cavity liner venting system is provided as discussed in Section 3.8.3.1.1.

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*With the exception of the reactor cavity floor and lower cavity wall liner plates which may be chrome-moly steel. 31

(Appendix A) was employed to model the entire driveline, dashpot cup and piston, scram guide tube and control assembly duct to obtain the forces arising from lateral misalignments. A representation of this model is shown on Figure 4.2-109. Torsional misalignment forces were evaluated by determining the maximum rotation to which the control rod is subjected and the rotational stiffness of the rod

Due to the large number of gap elements necessary in a control rod lateral misalignment force model (see Figure 4.2-109) a pseudo-static method of solution was used (non-linear transient dynamics mode of ANSYS).

For verification, the displacement solution was input to the control rod model in a static analysis mode. The static and dynamic (pseudo-static) force solutions were in agreement.

Conservative assumptions salient to the lateral misalignment force analysis were as follows:

1) The control rod system is at a uniform temperature. This corresponds to the "worst case" refueling condition for misalignments
48 (see Figure 4.2-95A).

2) Linear interpolation of the maximum misalignment values at the top and bottom of the guide tube and control assembly is valid.

3) A conservative coefficient of friction of 1.5 is utilized (see Section 4.2.3.1.3).

4) The dashpot cup is centered on its seat initially.

5) The stiffness of the disconnect actuating rod internal to the control rod driveline is negligible when compared with the outer tube.

6) The sum of the absolute values of the reaction forces acting on the control rod will be greatest (largest retardation forces) for the two-dimensional case with the greatest curvature in the elastic curve of the control rod.

Three types of elements were employed in the lateral misalignment model (Figure 4.2-109) as follows:

1) Three-Dimensional Pipe - To model the control rod as a "beam."

2) Two-Dimensional Interface - To model the potential interference points where reaction leads may develop between the control rod and adjacent restraints.

3) One-Dimensional Sliding Interface - To model the dashpot cup/seat interaction.

4.2.3.1.4 Positioning Requirements

The positioning requirements for the control rod systems are:

1. Both the primary and secondary control rod system shall each provide two independent position indication systems and a means for verification of coupling and disconnect between the driveline and control rod.

2. Each control rod system shall provide capability for measurement of scram insertion times for individual control rods.

3. One of the position indication systems for each control rod system shall have a minimum indication accuracy of ±0.5 inch for the full-in and full-out position of the control rods and ±1.25 inches over the full control rod stroke. These accuracies apply to the positions of the translating assemblies (drivelines)
48 relative to the CRDM housings.

4. One of the primary control rod system position indication systems shall provide an accuracy of ±0.15 inch for the leadscrew relative to the full insertion position.

5. One of the secondary control rod system position indication systems shall provide an accuracy of ±0.5 inch at the full-in,
48 withdrawn operating and refueling positions.

Two independent position indication systems are provided for each system to give positive verification of control rod position and a means to check operation of each system by comparison with the other system. These systems are expected to monitor the positions of the control rod drivelines (leadscrews). Consequently, an additional indicator is provided to verify connection and disconnection operations between the driveline and control rod.

Testing capability for control rod scram performance is planned for all plant conditions between cold shutdown and full power conditions. Measurements of individual control rod scram insertion times are required to ensure this capability and to provide periodic checks for abnormal control rod performance.

Position accuracy of ±0.5 inch at full insertion is provided to verify the fully inserted positions for reactor shutdown and to assure insertion positions for control rod disconnect and subsequent refueling operations. Accuracy in the fully withdrawn position is specified to assure adequate positioning for potential scrams from the parked position. These positions are also used for safety interlocks with the reactor control and refueling systems.

The primary control rod system is used for establishing criticality and subsequent power control operations. While the rod position indication

is not fed back directly to the reactor control system, the operator utilizes the position data to evaluate the plant and to interpret reproducibility of reactivity control. The relative position indication accuracy of ±0.1 inch leads to reactivity reproducibility of approximately 1¢ for the highest worth rod in the primary system. In addition, the position indication is utilized for logic interlocks and alarm as described in Section 7.7.1.3.

4.2.3.1.5 Structural Requirements

Control Rod Drive Mechanisms

The primary and secondary control rod drive mechanisms are designed to the following classes of components:

1. ASME Boiler and Pressure Vessel Code, Section III, 1974 edition,
39 Class 1. For the primary control rod system, the mechanism
motor tube, motor tube hold-down ring, nozzle extensions and
position indicator housing form a part of the pressure retaining
boundary. For the secondary control rod system, the extension
nozzle, the hold-down ring, the upper portion of the mechanism
39 housing, and the connector plate form a portion of the pressure
retaining boundary.

2. Seismic Category I. The control rod systems are required to remain functional and shut down the reactor in the event of an SSE. (See Section 3.2.1 for detailed discussion).

3. Safety Class I. The control rod systems are categorized as Class I because of their control and shutdown functions. (See Section 3.2.2 for detailed discussion).

39 The primary control rod drive mechanisms shall be designed to the
load conditions of Table 4.2-37 and shall meet the structural requirements of
Section III of the ASME Pressure Vessel Code together with applicable code
cases and amendments to the code by RDT Standards. The portion of the
Secondary Control Rod System that is coded in accordance with the ASME B&PV
code and hence forms a part of the pressure retaining boundary shall be
designed to the load conditions of Table 4.2-37. The structural requirements
39 of Section III of the ASME Pressure Vessel Code together with applicable code
cases and amendments to the code by RDT Standards shall be met.

The governing stresses in the mechanism are the time independent effects of primary mechanical loads, secondary thermal loads and fatigue. Use of the methods of these codes together with consideration of material effects such as carbon and nitrogen depletion, thermal aging, and environmental correction factors to account for material interaction with sodium leads to conservative structural designs of the mechanisms.

The primary and secondary control rod drive mechanisms shall have a
design life of 30 years. This lifetime is consistent with the design life-
time of the reactor. Sufficient shielding shall be provided where appropriate
to assure adequate strength to meet the structural criteria over the required
lifetime. Interim maintenance will be required in order to achieve this
39 lifetime.

The magnesium oxide floor aggregate is assumed to provide no lateral support to the embedded steel sections. The size and spacing of the embedded rolled steel sections are designed such that stresses and strains in the beam web and the liner plate fall within the limits specified in Table 3.8-1 of Appendix 3.8-B. 37

45 37 A liner vent system will be installed to limit the pressure behind the liner generated by the heatup of structural concrete during a sodium spill. The liners will be designed to withstand the pressure under the maximum liner temperature.

47 The steam generated below the floor liner by the heat up of the structural concrete will be vented through the MgO aggregate bed and through holes in the webs of the support beam to collective points along the periphery of the cell. Plugging of this region is precluded by the use of a large number of vent holes in the beams. In areas other than the reactor cavity, the steam from the floors will be released with the steam from the walls and ceilings into the liner vent system piping. Effects on stiffness caused by liner corrosion will be accounted for in the liner plate/anchors analysis. Equipment supported on the floor liner will be provided with special supports to transmit the loads directly to the structural slab. During construction and maintenance the floor liner will be protected from loading as specified in Section 3.1.1 of Appendix 3.8-B. Diagrams of the cell liner configurations are shown in Figures 3A.8-4, 3A.8-5 and 3A.8-6.

48 The vent path for the cell liner wall and ceiling system is provided by a 1/4" continuous air gap as shown in Figures 3A.8-4 and 3A.8-5. The air gap is maintained during construction and the life of the plant by:

48 a) The physical presence of the thermal plastic (ethafoam) placed during construction of the prefabricated wall and ceiling panel units.

48 b) The bond strength of the liner anchor embedment (Nelson Stud) into the insulating concrete (to prevent compression of the thermal plastic during liner erection).

In the event of a sodium spill in a lined cell, the thermal plastic (ethafoam) will contract from the heated liner and provide the air gap. Local plugging of the air gap is precluded since the air gap is continuous over the entire surface area of the lined cell. Therefore, there are no effects on the liner or liner anchors due to pressure buildup. 37

Liners will not ordinarily be exposed to sodium. The structural
48 concrete will be protected by an insulating concrete or MgO layer between the steel liner and the structural concrete. During accident conditions, some spalling of this non-structural concrete insulation may occur. However, this is considered acceptable since liner failure due to spalling of the insulating concrete is prevented by embedding liner anchors into the structural concrete.

The inner cells are reinforced concrete structures with steel liners designed to maintain a leak-tight barrier during normal operating conditions. Piping penetrations to inert cells are designed to prevent leakage, and are sealed by any of the following methods depending upon individual design requirements:

a) packing between pipe and a pipe sleeve which is welded to the cell liner.

b) flued head or flexible bellows attachments welded to pipe and pipe sleeve with sleeve seal welded
37 to cell liner.