00836

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

November 7, 1980

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

Dear Mr. Eisenhut:

AMENDMENT NO. 57 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. 57 to the Preliminary Safety Analysis Report pursuant to 50.34(a) of 10 CFR Part 50. This Amendment No. 57 includes: updates to Section 7, "Instrumentation and Controls"; Section 11.6, "Offsite Radiological Monitoring Program"; Section 15.A, "CRBRP Radiological Source Term for Assessment of Site Suitability"; and other updates and revisions.

A Certificate of Service, confirming service of Amendment No. 57 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:80:332

Enclosure

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this 30th day of October, 1980.

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My Commission Expires April 28, 1984

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PAGE REPLACEMENT GUIDE FOR

AMENDMENT 57

CLINCH RIVER BREEDER REACTOR PLANT

PRELIMINARY SAFETY ANALYSIS REPORT

(DOCKET NO. 50-537)

Transmitted herein is Amendment 57 to the Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report, Docket 50-537. Amendment 57 consists of new and replacement pages for the PSAR text.

Vertical lines on the right hand side of the page are used to identify question response information and lines on the left hand side are used to identify new or changed design information.

The following attached sheets list Amendment 57 pages and instructions for their incorporation into the Preliminary Safety Analysis Report.

AMENDMENT 57

PAGE REPLACEMENT GUIDE

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16.6-11, 12	16.6-11

Amendment 57

Question/Response Supplement

This Question/Response Supplement contains an Amendment 57 tab sheet to be inserted following Q-i (Amendment 56, Aug. 1980) page. Page Q-i (Amendment 57, Nov. 1980) is to follow the Amendment 57 tab.

Replacement pages for the Question/Response Supplement are listed below.

Replacement Pages

Remove These Pages	Insert These Pages
Q7-1	Q7-1

1.5.1.1.4. Criteria of Success

41 The latch Component Test in Sodium has been completed. The Inconel gripper/Inconel coupling head performed in accord with specifications for 4 times the required number of cycles. The components are considered acceptable. 39

41 The prototype system test will confirm the capability to function reliably throughout its design life.

1.5.1.1.5. Fallback Positions

41 For the latch mechanism, other designs could be utilized as a fallback position. During the SCRS concept selection phase a linkage-type latch was identified as a potential fallback latch.

39

1.5.1.2 Direct Heat Removal

41 26

41 1.5.1.2.1 Purpose

The Direct Heat Removal (DHR) service provides a supplementary means for removing long term decay heat for the remote case when none of the steam generator decay heat removal paths are available. This system must be able to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant boundary are not exceeded. This supplementary decay heat removal is performed by a cooling system incorporated into the sodium make-up/overflow system with plant conditions as specified in Chapter 5 (Section 5.6.2). The principal uncertainty of the make-up/overflow cooling system is short circuiting of the make-up flow with the reactor vessel. Short circuiting would occur if the inlet fluid flows directly to the overflow line without cooling the reactor core. Tests are needed to design the system to ensure short circuiting does not compromise core cooling.

41 1.5.1.2.2 Program

This program is conducted by Hanford Engineering Development Laboratory at the Integral Reactor Flow Model Test Facility. A 1/21 scale outlet plenum model test was used initially to test promising OHR candidate designs for the outlet plenum. Of concern is the location of the make-up and overflow nozzles to reduce short circuiting of make-up flow. This test will conceptually determine overflow nozzle locations.

Confirmation testing of the selected make-up and overflow concept was
53 successfully completed in the Phase I testing of the Integral Reactor Flow Model.

41 1.5.1.2.3 Schedule

41

CY	74	75	76	77	78	79	80	81
				Assembly of Reactor Vessel Shell Courses				
Make-up/Overflow Location Tests in 1/21 Scale Outlet Plenum Model	———	▲ completed						
Confirmation Tests and Evaluation of Make-up/Overflow Concept		———	———	——— ▲	completed			

41 41 53

41 1.5.1.2.4 Success Criteria

The tests demonstrated that the distribution of the make-up flow
26 53 in the outlet plenum was adequate to assure that the DHR service will function to remove decay heat following a reactor shutdown. This system must be capable of removing heat at a rate such that specified acceptable fuel design limits and design conditions of the reactor coolant boundary are not exceeded.

53 1.5.1.2.5 Results of Tests

53 The 1/21 scale outlet plenum and the HEDL IRFM model tests have been completed. The results show that short circuiting of make-up flow to the over-
53 flow nozzle is limited to approximately 5%. The test and results are discussed in more detail in Response 001.580.

57 1.5.1.3 Blanket Failure Threshold

41 1.5.1.3.1 Purpose

The CRBRP is being designed to operate with a limited number of failed
41 fuel and blanket rods. This requires demonstration that operation with failed
57 blanket rods exposed to sodium does not result in rod-to-rod failure propagation. This program investigates the potential of blanket material/ sodium reaction to cause swelling, flow blockages, and rod-to-rod failure
57 propagation in blanket assemblies.

57

It needs to be further demonstrated that a relatively small local flow
57 blockage will not lead to failure in a substantial number of blanket
rods. Tests performed for core fuel rod bundle geometries (Ref. 4), indicated
that such propagation is highly unlikely. However, the geometry, thermal
57 conditions and flow conditions in the blanket assemblies are sufficiently
different from that in core fuel assemblies to warrant an independent evalua-
57 tion of flow blockage effects. The variation in coolant flow rates to
blanket assemblies cover a wide flow velocity range from laminar through
transition to turbulent modes of flow. At low flow rates and with steep
temperature gradients across assemblies, buoyancy effects could become a
significant contributor to the temperature and flow distribution within the
blanket assembly.

Efforts are therefore planned to: (1) evaluate the failed
57 blanket rod performance; specifically to verify the performance of
blanket rods with failed cladding and blanket material exposed to sodium,
57 (2) to verify the effect of the high oxygen-to-metal ratio and density on the
probabilities of blanket material sodium reaction, swelling, and flow
57 blockages, (3) to evaluate the cooling rate behind a solid or porous local
flow blockage with tightly arranged rod bundles with pitch to diameter
ratios <1.1.

41 1.5.1.3.2 <u>Program</u>

This program will be conducted by Westinghouse at its ARD facility.

1) <u>Failed Blanket Rod Tests</u>

53

The scope of the blanket rod RBCB (Run Beyond Cladding Breach) portion of the LMFBR Reference Fuel Irradiation Program Includes the design, irradiation, and examination of EBR-II tests and/or TREAT experiments. The scope includes the acquisition, evaluation, analysis and reporting of results to:

- o demonstrate performance capability of breached blanket rods at beginning, middle, and end-of-life,
- o test the capability to accommodate design transients at end-of-life,
- o provide insight into the effect of reactor shutdown on breached blanket rods, and
- o establish a theoretical understanding of post-breach bahavior through predictive iterations based on and supported by experiments.

Information developed from the RBCB task will support the following specific areas:

1. Plutonium contamination of sodium.
2. Allowable operating time after breach.
3. Operating procedures for reactor shutdown and restart. 57

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4. Delayed neutron detector values for removal of breached rods.

5. Operational transient margins of breached rods.

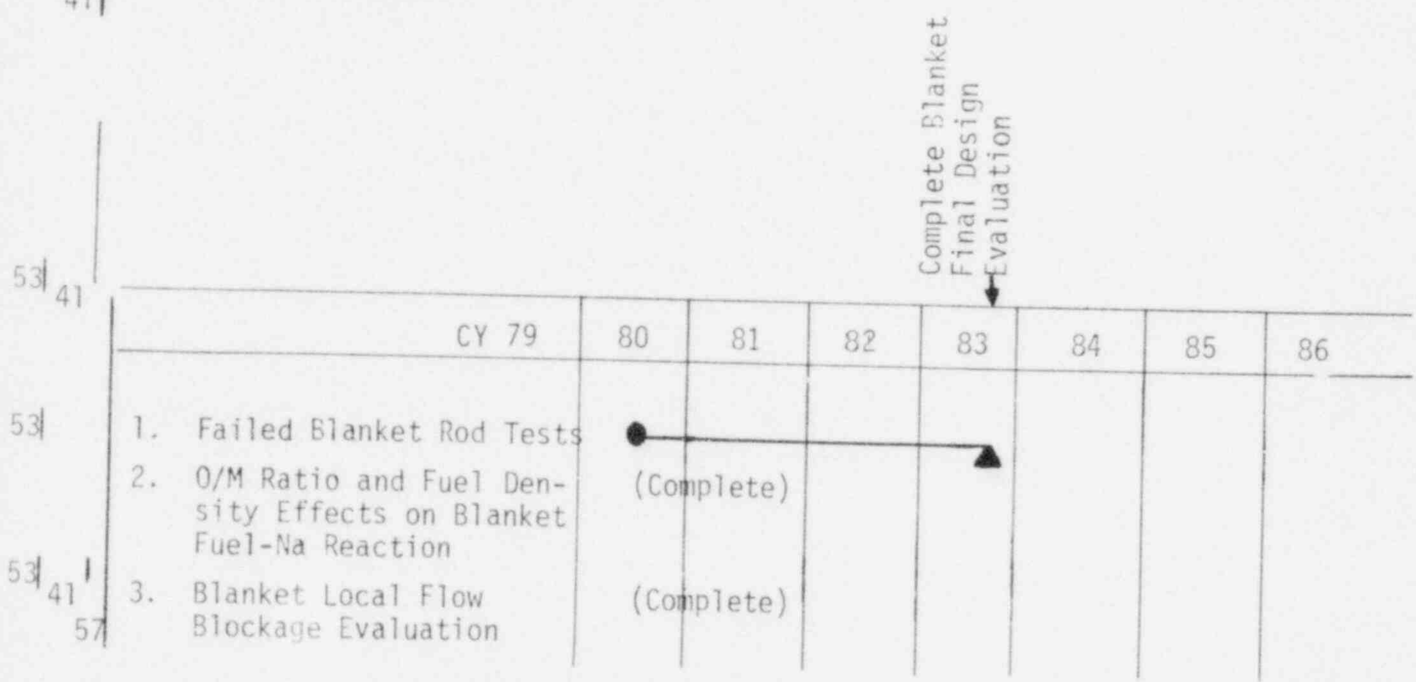
2) O/M Ratio and Density Effects on Blanket Fuel-Na Reaction

These effects were evaluated as part of the test program on fuel-sodium reaction phenomena conducted by General Electric Company. The results of this program are given in Reference 12.

3) Blanket Assembly Local Flow Blockage Evaluation

The effect of a local flow blockage in a blanket rod bundle have been evaluated with a water flow test and will be documented in a future 57 report.

41 1.5.1.3.3 Schedule



41 1.5.1.3.4 Criteria of Success

57 The program will demonstrate that operation with failed blanket rods exposed to sodium does not result in rod-to-rod propagation and that a relatively small local flow blockage will not lead to failure 57 in a substantial number of blanket rods.

41 1.5.1.3.5 Fallback Position

57 In the event that operating with failed blanket assemblies cannot be shown to be satisfactory from a public safety viewpoint, the reactor may be required to shutdown when the blanket material is exposed to the sodium.

1.5.1.4 Sodium-Water Reaction Pressure Relief Test

41 1.5.1.4.1 Purpose

The principal concern associated with the large water to sodium leak in steam generators is potential system damage, principally to the IHX by propagation of transient pressure waves through the Intermediate Heat Transport System (IHTS). The objective of the Sodium Water Reaction Pressure Relief Subsystem (SWRPRS) is to relieve pressures from the IHTS and thereby protect the primary coolant boundary from damage in the region of the primary sodium to intermediate sodium heat transfer interface.

The approach to design of CRBRP IHX is to assume a conservatively large design basis water to sodium leak and to use a validated calculational method to predict pressure loads on the IHX. It is a design requirement that the IHX be able to withstand the sodium-water reaction pressures without compromising the primary coolant boundary.

A survey of available existing analytical methods was completed to select the best method for improvements consistent with CRBRP requirements. The TRANSWRAP computer program (Ref. 5) was selected for use in the CRBRP analysis. An improved version of this code was used to establish loads on the IHX for the reference design IHTS piping and component arrangement and the reference design SWRPRS. A design basis leak was assumed to consist of a Double-Ended Guillotine failure (DEG) of a steam generator tube followed immediately by the equivalent of six additional DEG tube failures. The seven tube DEG failure is not intended to represent a realistic event, but rather it provides a basis for calculating conservatively large pressure loads for the design of IHX and the pressure relief system. Results of analyses using this basis are reported in Section 5.5.3.6.

To increase confidence in assuring integrity of the primary coolant boundary even during a large sodium-water reaction, the development program will provide technical information which is not available for inclusion in the PSAR. The safety related objectives of the development program are:

a) to validate the computer program used to predict pressures in the IHX during a postulated sodium water reaction, and

b) to confirm that effects of the design basis leak assumed for determining pressures in the IHX are conservative.

41 1.5.1.4.2 Program

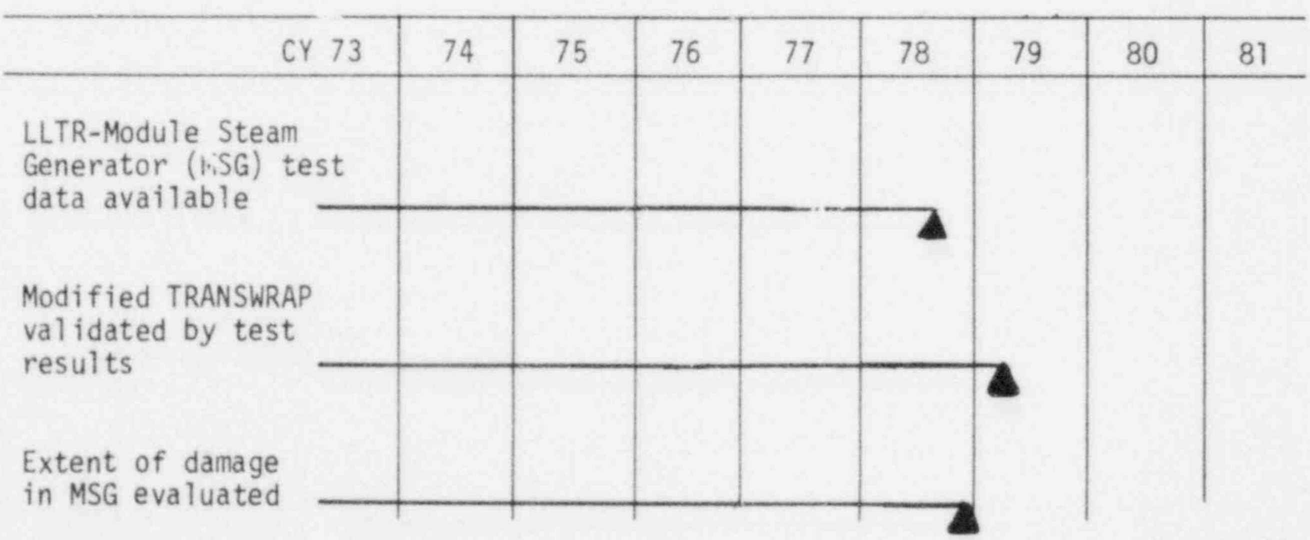
As part of the Steam Generator Development Program, AI has constructed the Large Leak Test Rig (LLTR). The test programs
41 included pulling apart a notched tubular specimen in the sodium filled test article to simulate a DEG failure. A steam/water mixture was forced through
44 41 the burst tube into the sodium. For most tests, surrounding tubes contained stagnant, pressurized steam/water mixtures. In general, the development
44 effort provided technical information regarding the design of pressure relief systems to handle unexpected water-to-sodium leaks.

Measured values of pressure at various locations in the test rig are being compared with calculated pressures obtained using the modified TRANSWRAP computer program to analyze the test rig and test article. It now appears that the computer code predicts values of pressure that are either in agreement with measurements or are conservatively large relative to measured pressures for the test rig and test article. Thus, it appears that the analysis of CRBRP for
41 sodium-water reaction pressures using this code are being conservatively
44 accomplished. This conclusion is still under review and evaluation and therefore subject to adjustment as the remaining test data are examined.

Examination of the test article following intentional bursting of
44 a single tube gives some indication of the nature and extent of damage propagation to other tubes. It is expected that the tests will demonstrate
47 that the calculated loadings from sodium-water reactions are conservative.

41 1.5.1.4.3 Schedule

44 41



47 41

41

44 41

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1.5.2.12.3 Schedule

41

CY	75	76	77	78	79
A. In-Vessel Model	1 Δ	2 Δ			3 Δ
B. Melt-Thru Model	4 Δ	5 Δ			6 Δ
C. Experimental Support					
C.1 Experimental Studies in In-Vessel Core Material Transport			7 Δ	8 Δ	
C.2 CDA Dynamic Loads Simulation in a Scale Model				9 Δ 10 Δ	

Milestones:

1 = Preliminary Design
2 = Improved Model
3 = Final Model
4 = Preliminary Model
5 = Improved Model
6 = Final Model
7 = Development of a Flashing Source
8 = Flashing Source Test Report
9 = CDA Dynamic Loads Simulation in a Scale Model Phase One Tests Complete
10 = CDA Dynamic Loads Simulation Phase Two Tests Complete 19

1.5.2.12.4 Criteria of Success

The model development and its associated confirmation by test results must confirm the margins of conservatism in the present analysis of radiological source terms which do not take full credit for removal mechanisms.

1.5.2.12.5 Fallback Positions

If the development program does not provide specific confirmatory information on schedule, continued use will be made of generic information extrapolated to CRBRP conditions to provide conservative estimates of radiological
30 source terms.

5

1.5.3 References

1. HEDL-SA-771, "Fuel Pin Transient Behavior Technology Applied to Safety Analysis," Presentation to AEC Regulatory Staff, 4th Regulatory Briefing on Safety Technology, November 19-20, 1974.

2. Haugen, E. B., "Probabilistic Approaches to Design," Wiley Book Co., New York (1968).

3. Chang, E. C. C., and C. B. Brown, "Functional Reliability of Structures," Journal of the Franklin Institute, September, 1973.

4. Fontana, M. H., et al., "Effect of Partial Blockages in Simulated LMFBR Fuel Assemblies," Proc. Fast Reactor Safety Meeting CONF-740401-P3, 1139 (1974).

5. Bell, C. R. "TRANSWRAP - A Code for Analyzing the System Effects of Large-Leak Sodium-Water Reactions in LMFBR Steam Generators," Proc. Fast Reactor Safety Meeting, CONF-740401-P1, 124 (1974).

6. Division of Reactor Research and Development, USAEC, RDT Standard M3-7T, "Austenitic Stainless Steel Welded Pipe (ASME SA-358 with Additional Requirements)" November, 1974.

7. Marr, W. W., et al., "Subassembly-to-Subassembly Failure Propagation: Thermal Loading of Adjacent Subassembly," Proc. Fast Reactor Safety Meeting, CONF-740401-P2, 598 (1974).

8. Van Erp, J. B., et al., "An Evaluation of Pin-to-Pin Failure Propagation in LMFBR Fuel Subassemblies," ibid, 615 (1974).

9. Erdman, C. A., et al., "Improvements in Modeling Fuel-Coolant Interactions and Interpretation of the S-11 TREAT Test," ibid, 955 (1974).

34 10. Letter from A. Buhl to R. Boyd, "Transmittal of Additional Information on SRI Scale Model Tests", SE-1788, October 26, 1976.

41 11. Letter from R. P. Denise (NRC) to L. W. Caffey (ERDA) May 6, 1976.

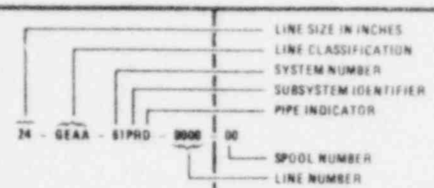
57 12. GEFR-00424, UC-79B, "A Physicochemical Model for Predicting Sodium Reaction Swelling in Breached LMFBR Fuel and Breeder Elements", R. W. Caputi, M. G. Adamson, and S. K. Evans, March, 1979.

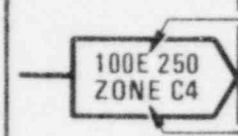
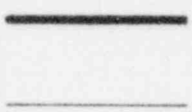
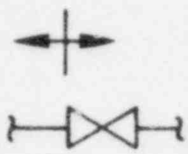
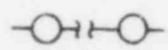
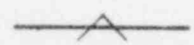
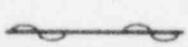
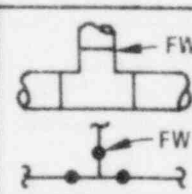
CHAPTER 1 - APPENDIX A

Flow Diagram Symbols

TABLE OF CONTENTS

LINE DESIGNATIONS



			LINE CONTINUES ON DWG INDICATED IN ZONE INDICATED
	PRINCIPAL FLOW ROUTE ALTERNATE FLOW ROUTE		CUT PIPE HERE TO REMOVE
	PIPELINE SPECIFICATION CHANGE		ELECTRICALLY HEATED PIPE (FLOW SHEET ONLY)
	ELECTRICAL SIGNAL		BURIED PIPE
	PNEUMATIC SIGNAL		PNEUMATIC SUPPLY
	CAPILLARY TUBING FILLED SYSTEM		HYDRAULIC SIGNAL
	RADIATION OR SONIC SIGNAL WITHOUT WIRING OR TUBING		FIELD WELD FIELD WELD IN SINGLE LINE DRAWING

57

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LINE CLASSIFICATIONS

1. PIPING CLASSES ARE DESIGNATED BY A FOUR LETTER CODE. THE FIRST LETTER INDICATES THE PRIMARY VALVE AND FLANGE RATING; THE SECOND LETTER THE TYPE OF MATERIAL; THE THIRD LETTER THE CODE TO WHICH THE PIPING IS DESIGNED AND THE FOURTH LETTER INDICATES THE SYSTEM FLUID

THE DESIGNATIONS ARE AS FOLLOWS:

FIRST LETTER: PRESSURE RATING

57 A – 4500 ANSI
B – 2500 ANSI
C – 1500 ANSI
D – 900 ANSI
E – 600 ANSI
F – 400 ANSI
G – 300 ANSI
H – 150 ANSI
J – 125 ANSI B16.1
K – 175 WOG UNDERWRITERS LABORATORIES, INC.
L – 250 ANSI B16.1
X – GRAVITY RATING

SECOND LETTER: MATERIAL

A – ALLOY
B – CARBON STEEL
C – STAINLESS STEEL (TP 304)
D – COPPER
E – STAINLESS STEEL (TP 316H)
F – CARBON STEEL – COPPER BEARING
G – CARBON STEEL – LINED
H – CAST IRON
J – CONCRETE PIPE
K – VITRIFIED CLAY PIPE
L – CARBON STEEL – IMPACT TESTED
M – DURIRON
N – CARBON STEEL – GALVANIZED
P – CAST IRON – CEMENT LINED
Q – ASBESTOS – CEMENT
57 U – PCV CHROME

THIRD LETTER: DESIGN CODE

A – NUCLEAR POWER PLANT COMPONENTS, ASME B&PV CODE SEC. III CLASS I
B – NUCLEAR POWER PLANT COMPONENTS, ASME B&PV CODE SEC. III CLASS II
C – NUCLEAR POWER PLANT COMPONENTS, ASME B&PV CODE SEC. III CLASS III
D – POWER PIPING CODE ANSI B31.1.0-1967
F – NATIONAL FIRE PROTECTION ASSOCIATION CODE
G – NATIONAL PLUMBING CODE
H – POWER BOILERS, ASME B&PV CODE SEC I
J – AMERICAN WATER WORKS STANDARDS

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Amend. 57
Nov. 1980

LINE CLASSIFICATIONS (Continued)

FOURTH LETTER: SYSTEM FLUID

A – SODIUM	K – AIR
B – STEAM	L – LIUQID NITROGEN
C – FEEDWATER	M – MIXED CAPS GASES
D – NAK	N – LIQUID ARGON
E – DOWTHERM	P – ACID
F – ARGON	R – CAUSTIC
G – NITROGEN (GAS)	S – ALCOHOL
H – CHILLED WATER	T – HYDROGEN
J – WATER	V – CARBON DIOXIDE
	W – OIL

SYSTEM/SUBSYSTEM/PIPE INDICATOR

1. FIRST TWO DIGITS IDENTIFY THE SYSTEM IN WHICH THE PIPE IS LOCATED
2. THE NEXT TWO ALPHAS IDENTIFY THE SUBSYSTEM
3. THE LETTER "D" INDICATES THAT THE ITEM IDENTIFIED IS PIPING.

LINE NUMBER

1. THE LINE NUMBER IS DESIGNATED BY SEQUENTIALLY ASSIGNED SERIAL NUMBERS FOR RUNS OF PIPE WITHIN THAT SYSTEM (THE MAXIMUM OF FOUR DIGITS)

SPOOL NUMBER

1. A SEQUENTIALLY ASSIGNED SERIAL NUMBER DENOTING PIPE SPOOLS WITHIN THE PIPE LINE. THE ASCENDING ORDER OF THE SEQUENCE IS IN THE DIRECTION OF THE FLUID FLOW.

SPECIAL CASES

1. IN THE PIPELINE IDENTIFICATION SYSTEM THE TWO-DIGIT FIELD INDICATING NOMINAL PIPE DIAMETER MAY BE EXPANDED TO EXPRESS FRACTIONS, e.g. 2½ INCH PIPE IS POSTED AS 2.5.

2. THE PIPELINE IDENTIFICATION MAY BE ABBREVIATED ON DRAWINGS BY REDUCING THE THREE-DIGIT SEQUENTIAL SERIAL NUMBERS FROM 001, 002, 003, ETC. TO 1, 2, 3, ETC.

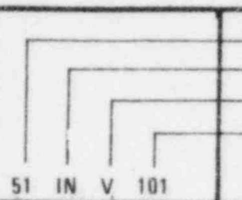
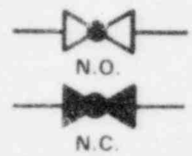
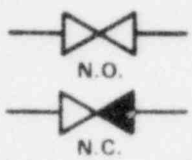
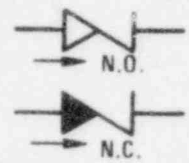
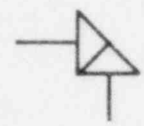
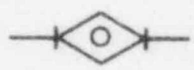
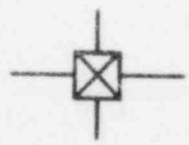
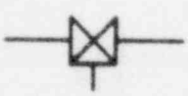
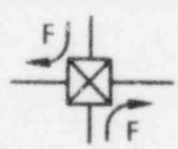
3. PIPELINE NUMBERS ON PIPING DRAWINGS MAY BE FURTHER ABBREVIATED BY ADDING A GENERAL NOTE STATING THAT "ALL PIPELINE NUMBERS APPEARING ON THIS DRAWING INCORPORATE THE SYSTEM/SUBSYSTEM DESIGNATION XXXX UNLESS OTHERWISE NOTED". THE SYSTEM/SUB-SYSTEM CHARACTERS MAY THEN BE OMITTED FROM PIPELINE IDENTIFICATION NUMBERS POSTED ON THE DRAWING.

4. THE PIPELINE IDENTIFICATION FOR PIPING DRAWINGS WILL BE ABBREVIATED FOR PIPELINE LISTS BY ELIMINATING THE 2-DIGIT FIELD INDICATING NOMINAL PIPE SIZE AND THE 4-DIGIT FIELD INDICATING PIPELINE CLASSIFICATION.

   THE NOMINAL PIPE SIZE AND THE PIPELINE CLASSIFICATION WILL BE IDENTIFIED AS DATA ELEMENTS IN THE PIPELINE LIST, BUT NOT AS PART OF THE PIPELINE NUMBER.

5. IDENTICAL PIPE LINES IN DIFFERENT LOOPS WITHIN A SYSTEM ARE IDENTIFIED BY LETTER SUFFIX (A, B, C, etc) 57

VALVE BODY SYMBOLS

	SYSTEM NUMBER SUBSYSTEM IDENTIFIER VALVE SEQUENTIAL SERIAL NUMBER		GLOBE
	GATE		STOP CHECK
	CHECK		ANGLE
	AUTOMATIC CHECK POSITIVE CLOSING		BUTTERFLY
	NEEDLE		BALL
	PLUG		FOUR-WAY
	THREE-WAY		FOUR-WAY INDICATING FAILURE PATH

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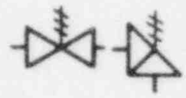
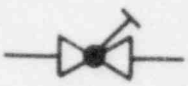
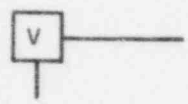
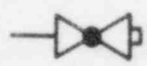
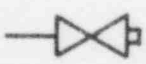
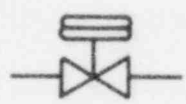
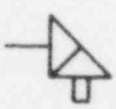
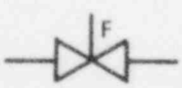
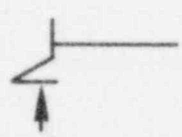
VALVE BODY SYMBOLS (Continued)

	THREE WAY - INDICATING FAILURE PATH		SAFETY OR RELIEF
 (SAUNDERS TYPE)	DIAPHRAGM		AUTOMATIC BALL DRIP CHECK
	"Y" PATTERN GLOBE (BLOWDOWN)		AIR VENT
	MANUALLY OPERATED TEST		HOSE GLOBE
	HOSE GATE		BELLOWS SEAL
	HOSE ANGLE		PINCH
	FREEZE SEAL		ANGLE CHECK (STOP CHECK)

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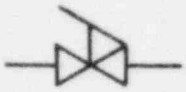
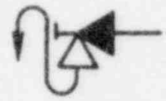
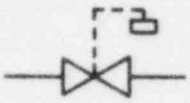
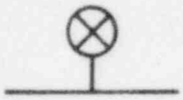
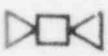
VALVE BODY SYMBOLS (Continued)

	SLIDE OR BLAST GATE		FIRE HOSE, EXPOSED
	FLOAT OPERATED		POST INDICATOR
	HYDRANT		PREACTION VALVE
	DRAG DISC VALVE		DELUGE VALVE
	ALARM CHECK VALVE		

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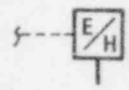
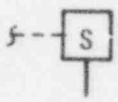
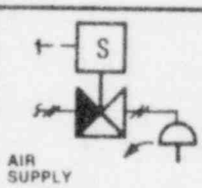
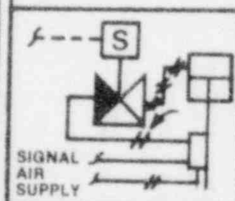
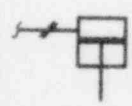
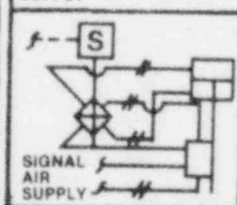
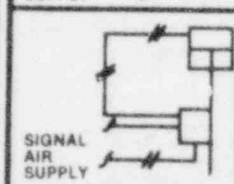
VALVE ACTUATORS

	MOTOR OPERATED		HAND OPERATED (Mounted at top side or bottom of valve assembly)
	HYDRAULIC		ELECTRO-HYDRAULIC
	SOLENOID OPERATED		PNEUMATIC
	DIAPHRAGM: SPRING OPPOSED, WITHOUT POSITIONER OR OTHER PILOT		DIAPHRAGM; PRESSURE-BALANCED
	1,2,3 DIAPHRAGM: SPRING OPPOSED, OVERRIDING PILOT VALVE THAT PRESSURIZES DIAPHRAGM WHEN ACTUATED		DIAPHRAGM: SPRING OPPOSED, ASSEMBLED WITH PILOT, ONE CONTROLLED INPUT
	1,2,3 SINGLE ACTING CYLINDER: CONVERTER, OVERRIDING PILOT VALVE THAT PRESSURIZES DIAPHRAGM WHEN ACTUATED		SINGLE ACTING CYLINDER WITHOUT POSITIONER OR OTHER PILOT
	2,3 DOUBLE ACTING CYLINDER WITH POSITIONER, CONVERTER, OVERRIDING PILOT VALVE		DOUBLE ACTING CYLINDER WITHOUT POSITIONER OR OTHER PILOT
	2,3 SINGLE ACTING CYLINDER WITH POSITIONER	1. NORMALLY SHUT PORT IS "FILLED IN." 2. OTHER COMBINATIONS ARE POSSIBLE AND WHEN USED SHALL FOLLOW THE FORMAT ESTABLISHED BY THESE EXAMPLES. 3. ITEMS NOT SHOWN ON P&ID	

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VALVE ACTUATORS (Continued)

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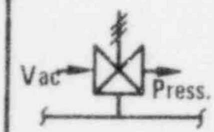
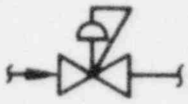
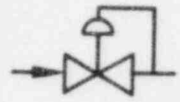
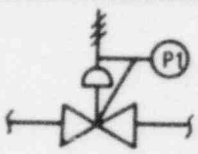
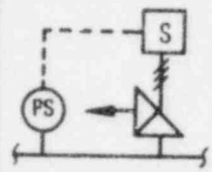
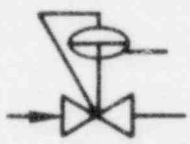
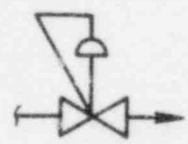
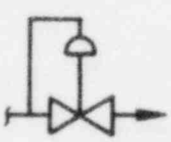
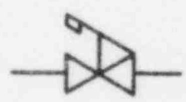
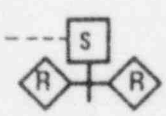
			PRESSURE AND VACUUM RELIEF VALVE WITH INTEGRAL PILOT. NOTE: ACTUATOR SYMBOL: ≢ FOR PRESSURE RELIEF OR SAFETY VALVES ONLY: DENOTES A SPRING WEIGHT OR INTEGRAL PILOT
	PRESSURE REDUCING REGULATOR SELF-CONTAINED		PRESSURE REDUCING REGULATOR WITH EXTERNAL PRESSURE TAP
	PRESSURE REDUCING REGULATOR WITH INTEGRAL OUTLET PRESSURE RELIEF VALVE AND OPTIONAL PRESSURE INDICATOR (TYPICAL AIR SET)		PRESSURE RELIEF OR SAFETY VALVE, ANGLE PATTERN, TRIPPED BY INTEGRAL SOLENOID
	DIFFERENTIAL PRESSURE REDUCING REGULATOR WITH INTERNAL AND EXTERNAL PRESSURE TAPS		BACK PRESSURE REGULATOR SELF-CONTAINED
	BACK PRESSURE REGULATOR WITH EXTERNAL PRESSURE TAPS		QUICK OPENER
	SOLENOID RESET (OPTIONAL)		

Amend. 57
Nov. 1980

ABBREVIATIONS ASSOCIATED WITH VALVES

N O.	NORMALLY OPEN	N C	NORMALLY CLOSED
L O	LOCKED OPEN	L C	LOCKED CLOSED
F O	FAIL OPEN	F C	FAIL CLOSED
F L	FAIL LOCKED	F I	FAIL INTERMEDIATE
F A I	FAIL AS IS	A O	AIR OPENS
A C	AIR CLOSES	SOV	SOLENOID OPERATED VALVE
AOV	AIR OPERATED VALVE	MOV	MOTOR OPERATED VALVE
		NRV	NON RETURN VALVE

SPECIAL CASES

1. THE VALVE IDENTIFICATION SYSTEM MAY BE EXPANDED TO DESIGNATE VARIOUS TYPES OF CONTROL VALVES WITH A 4-DIGIT CODE DEFINING THE VALVE FUNCTION. THE VALVE FUNCTION CODE APPEARS IN THE "INSTRUMENTATION IDENTIFICATION TABLE" IN COLUMNS HEADED "CONTROL VALVE" AND SELF "ACTUATED VALVE", e.g.

   FCV SELF ACTUATED FLOW CONTROL VALVE
   FRV FLOW RATIO CONTROL VALVE
   LCV SELF ACTUATED LEVEL CONTROL VALVE
   PDV PRESSURE DIFFERENTIAL CONTROL VALVE
   TDCV SELF ACTUATED TEMPERATURE DIFFERENTIAL CONTROL VALVE

2. THE THREE-DIGIT SEQUENTIAL SERIAL NUMBER IN THE VALVE IDENTIFICATION SYSTEM POSTED AS 001, 002, 003, ETC. ON VALVE LISTS MAY BE SIMPLIFIED TO READ 1, 2, 3, ETC.

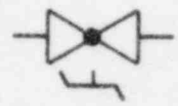
3. BY ADDING A GENERAL NOTE TO DRAWINGS STATING THAT ALL VALVE IDENTIFICATION NUMBERS APPEARING ON THE DRAWING "ARE PREFIXED BY SYSTEM/SUB-SYSTEM XXXX UNLESS OTHERWISE NOTED", THE VALVE IDENTIFICATION SYSTEM MAY BE FURTHER SIMPLIFIED ON PIPING DRAWINGS BY ELIMINATION OF THE FOUR-DIGIT SYSTEM/SUB-SYSTEM CHARACTERS.

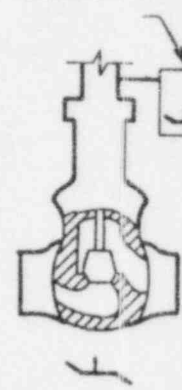
4. ON DRAWINGS INTENDED FOR FABRICATION/ERECTION REVERSE VALVE INSTALLATIONS SHALL BE CONSPICUOUSLY IDENTIFIED ON THE DRAWING BY A SPECIAL NOTE, e. g. "INSTALL BACKWARDS", INDICATING THAT THE VALVE SHALL BE POSITIONED WITH THE FLOW ARROW ON THE VALVE BODY
57 DIRECTED AGAINST NORMAL SYSTEM FLOW.

VALVE ORIENTATION SYMBOLOGY

SEAT – BRIDGE MARK

THE VALVE ORIENTATION SYMBOL FOR CONVENTIONAL "Y"-PATTERN GLOBE VALVES ( ) DENOTES FLOW UNDER THE PLUG WHEN THE PROCESS FLUID IS MOVING FROM LEFT TO RIGHT AND OVER THE PLUG WHEN FLOW IS IN THE REVERSE DIRECTION. THE SEAT-BRIDGE MARK CLEARLY DENOTES **VALVE STEM-SEAT ORIENTATION.** RELATIVE TO SYSTEM FLOW FOR ALL GLOBE VALVE.



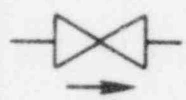


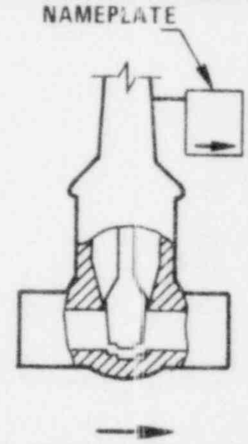
CONVENTIONAL GLOBE VALVE

FLOW DIRECTION ARROW

ALL OTHER VALVES EXCEPT THE VALCOR LIQUID METAL VALVES SHALL UTILIZE A FLOW DIRECTION ARROW (⟶) TO DENOTE THE ORIENTATION OF THE VALVE IN THE LINE.

THE ORIENTATION SYMBOL WILL BE LOCATED BELOW THE VALVE SYMBOL ON THE PIPING & INSTRUMENT DIAGRAMS AND INCLUDED ON THE VALVE NAMEPLATES AT THE TIME OF MANUFACTURE.

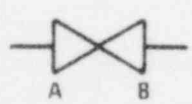


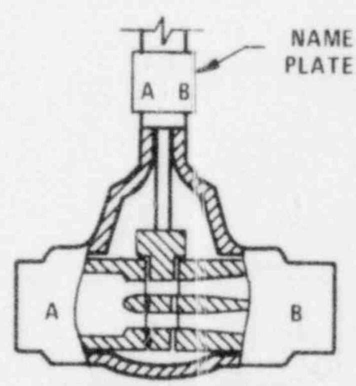


CONVENTIONAL GATE VALVE

STRAIGHT PATTERN LIQUID METAL VALVE (LMS)

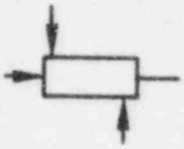
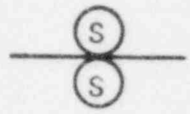
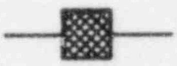
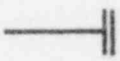
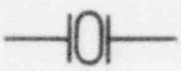
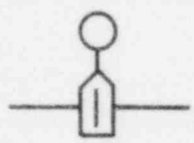
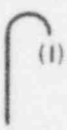
THE VALCOR LIQUID METAL VALVE SHALL IDENTIFY THE INLET PORT AS PORT "A" AND THE OUTLET PORT AS PORT "B". THE DESIGNATION SHALL BE ON THE VALVE BODY & VALVE NAMEPLATE.





STRAIGHT PATTERN LIQUID METAL VALVE (LMS)

SPECIALTY SYMBOLS

	IN LINE SELFCLEANING STRAINER		Y TYPE STRAINER
	SIMPLEX STRAINER		DUPLEX STRAINER
	FILTER		TEMPORARY STRAINER
	BLIND FLANGE		PLUG OR CAP
	EXPANSION JOINT		SPECTACLE FLANGE
	RESTRICTING ORIFICE		ORIFICE PLATE IN QUICK CHANGE FITTING
	VENT (I) = INDOORS (O) = OUTDOORS		VENT WITH FLAME ARRESTER
	TAPERED EXPANSION JOINT		

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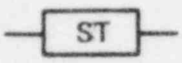
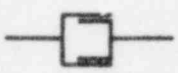
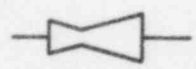
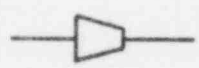
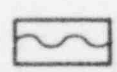
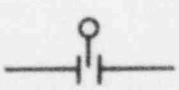
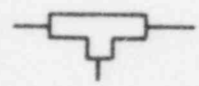
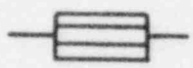
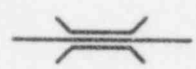
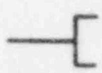
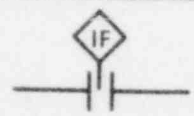
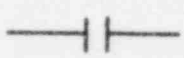
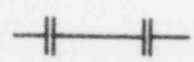
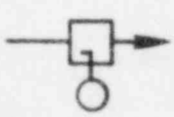
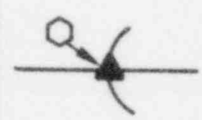
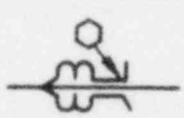
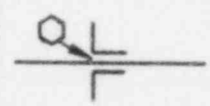
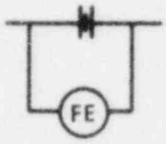
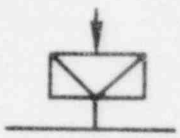
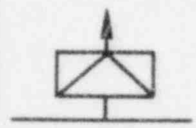
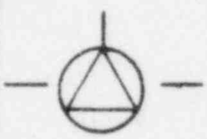
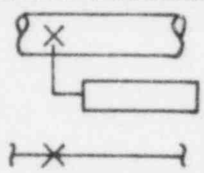
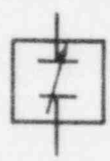
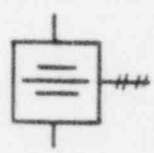
SPECIALTY SYMBOLS (Continued)

	STEAM TRAP		DRIP PAN ELBOW (FOR STEAM RELIEF VALVE)
	LOOP SEAL		SPRAY NOZZLES
	IN-LINE SIGHT FLOW GLASS		FLEXIBLE CONNECTION
	QUICK DISCONNECT		FLOW NOZZLE (TUBE OR VENTURI)
	BEARING		REDUCER OR INCREASER
	FLOOR DRAIN, HUB OR TRENCH		CHEMICAL SEAL
	CIRCULAR OR HAMMER BLIND		VAPOR TRAP, DRY

SPECIALTY SYMBOLS (Continued)

	VAPOR TRAP, WET		FUNNEL DRAIN
	EXHAUST HEAD		MIXING TEE
	FLOW STRAIGHTENING VANES		MAGNETIC FLOWMETER E.M. = ELECTROMAGNETIC P.M. = PERMANENT MAGNET
	SERVICE CONNECTION		INSULATING FLANGE
	FLANGE CONNECTION		SPOOL PIECE
	PITOT OR PITOT VENTURI TUBE		ANCHORED CONTAINMENT PENETRATION
	BELLOWS SEAL CONTAINMENT PENETRATION		LINED CELL PENETRATION

SPECIALTY SYMBOLS (Continued)

	ORIFICE PLATE WITH VENA CONTRACTA RADIUS OR PIPE TAPS		ATTEMPERATOR
	SAFETY HEAD		SPRINKLER NOZZLE
	RUPTURE DISK VACUUM RELIEF		RUPTURE DISC PRESS RELIEF
	GAS SEPARATOR		PIPE SUPPORT PIPE SUPPORT IDENTIFICATION NO. PIPE SUPPORT FOR SINGLE LINE DRAWING
	ACOUSTIC (NOISE) DETECTOR		VARIABLE SPEED COUPLING
			CLUTCH & BRAKE

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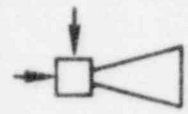
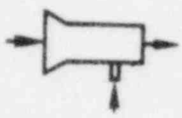
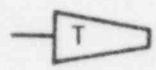
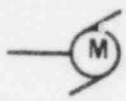
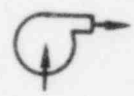
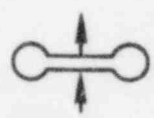
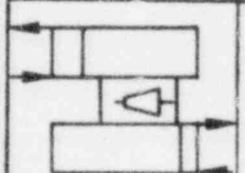
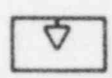
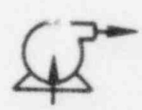
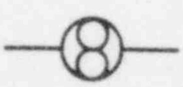
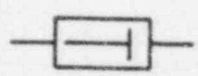
EQUIPMENT SYMBOLS

	DIESEL DRIVE		EDUCTOR OR EJECTOR
	DESUPERHEATER		TURBINE DRIVE
	ELECTRIC MOTOR DRIVE		CENTRIFUGAL PUMP
			POSITIVE DISPLACEMENT PUMP
57 	CENTRIFUGAL LIQUID CHILLER		PLUGGING METER
	GENERAL PURPOSE VERTICAL PUMP		GENERAL PURPOSE HORIZONTAL PUMP
	ROTARY PUMP		PISTON PUMP

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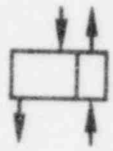
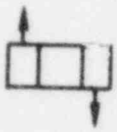
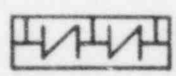
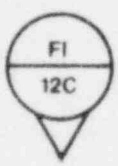
EQUIPMENT SYMBOLS (Continued)

	ELECTROMAGNETIC PUMP		HEAT EXCHANGER (LIQUID - LIQUID)
	HEAT EXCHANGER (LIQUID - GAS)		WATER TRAP
	FLOW SENSOR		SHOWER HEAD
	EYE WASH		FLOW BALANCING VALVE
	VERTICAL PUMP		MIXER
	WATER HAMMER ARRESTER		BACKFLOW PREVENTER
	ROTOMETER		—INLINE PUMP

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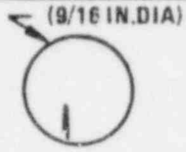
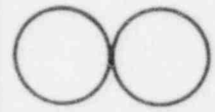
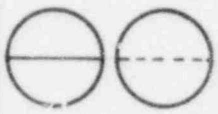
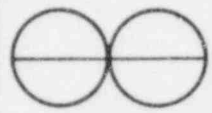
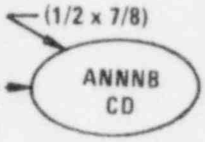
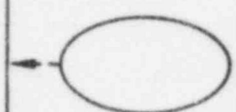
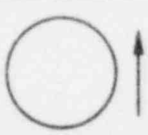
INSTRUMENT DESIGNATIONS

(9/16 IN.DIA) 	RELAY OR LOCAL INSTRUMENT INCLUDING TRANSMITTER FOR SINGLE MEASURED VARIABLE		LOCAL INSTRUMENT FOR TWO MEASURED VARIABLES OR MORE THAN ONE FUNCTION
 FACE BACK	PANEL MOUNTED INSTRUMENT FOR SINGLE MEASURED VARIABLE		PANEL MOUNTED INSTRUMENT FOR TWO MEASURED VARIABLES OR MORE THAN ONE FUNCTION
(1/2 x 7/8)  ANNNB CD	VARIABLE INTO DATA SYSTEM WHERE: A = TYPE OF VARIABLE OR MEASUREMENT NNN = SERIAL NUMBER (LOOP OR CHANNEL NUMBER WHENEVER POSSIBLE) B = TYPE OF INPUT SIGNAL CD = OPTIONAL PARALLEL OR REDUNDANT MEASURE-MENTS AND PLANT PR[illegible]N SIGNALS WHERE T[illegible]NAL "B" IS DEFINED [illegible] A – ANALOG D – DIGITAL E – EVENT (CONTACT SENSE) P – PULSE (CONTACT INTERRUPT)		CONTROL SIGNAL FROM DATA SYSTEM
	ARROW INDICATES DIRECTION IN WHICH RELAY RESPONDS TO A FAULT. ARROW UP = FORWARD LOOKING ARROW DOWN = REVERSE LOOKING	9/16 IN. DIA. WALL  PDIT 101 AD	INSTRUMENT BALLOON WITH INSTRUMENT NUMBER (WALL OF BALLOON MAY BE RUPTURED TO ACCOMMODATE INSTRUMENT NUMBER).

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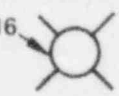
INDICATING LIGHTS

O – OPEN 7/16  C – CLOSED	O - OPEN C - CLOSED H - HIGH L - LOW	COLORS W - WHITE G - GREEN R - RED A - AMBER	

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INSTRUMENT DESIGNATIONS (CONTINUED)

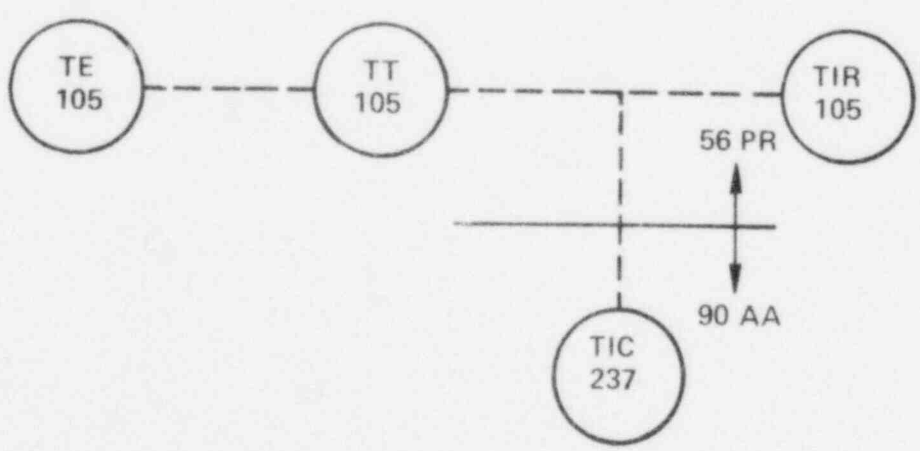
INSTRUMENT IDENTIFIER IS AS FOLLOWS:

NMABCDEFGXYZHJ

WHERE:

NM	ARE TWO NUMBERS IDENTIFYING THE SYSTEM DESIGN DESCRIPTION (SDD)
AB	ARE TWO LETTERS REPRESENTING THE SDD SUBSYSTEM
CDEFG	ARE FIVE LETTERS REPRESENTING THE FUNCTION OF THE INSTRUMENTATION CONSISTENT WITH ISA-S5.1
XYZ	ARE THREE NUMBERS TO REPRESENT LOOP OR CHANNEL NUMBER (ASSIGNED BY THE COGNIZANT ENGINEER
HJ	ARE TWO OPTIONAL LETTER(S) TO INDICATE REDUNDANT OR PARALLEL MEASUREMENTS WITHIN A LOOP OR CHANNEL. ALL LETTERS A THROUGH Z CAN BE USED WITH "P" AND "S" RESERVED FOR PLANT PROTECTION SYSTEM.
NOTE:	THE SDD AND THE SUBSYSTEM IDENTIFYER ARE NOT PLACED INSIDE THE INSTRUMENT BALLOON ON A DRAWING BUT ARE IDENTIFIED BY A NOTE ON THE DRAWING. IF MORE THAN ONE SYSTEM IS REPRESENTED ON A DRAWING, THE CONVENTION OF ISA-S5.1 SHALL APPLY. (INSTRUMENT BALLOONS IN EXAMPLE ARE ENLARGED RATHER THAN RUPTURED TO ACCOMMODATE INSTRUMENT NUMBER)

FOR EXAMPLE:



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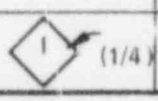
RELAY AND CONVERSION DEVICE

I/P	CURRENT TO PNEUMATIC	E/P	VOLTAGE TO PNEUMATIC
E/I	VOLTAGE TO CURRENT	P/I	PNEUMATIC TO CURRENT
R/I	RESISTANCE TO CURRENT	E/H	VOLTAGE TO HYDRAULIC
P/E	PNEUMATIC TO VOLTAGE	$\triangle$	DIFFERENCE
F	FREQUENCY METER	GE	GENERATOR EXCITER

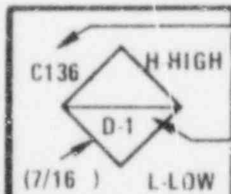
RELAY DESIGNATIONS (SEE ANSI C37.2)

21x TIMER (TO BE USED WITH 21Z2)
21Z1 PHASE DISTANCE RELAY, ZONE 1
21Z2 FOR ZONE 2
21Z3 FOR ZONE 3
ETC.
27 UNDER VOLTAGE RELAY
40 FIELD CURRENT RELAY
42 CONTRACTOR
46 PHASE BALANCE CURRENT RELAY
49 THERMAL RELAY
50 INSTANTANEOUS OVERCURRENT RELAY
50BF BREAKER FAILURE CURRENT DETECTOR RELAY
50CM CURRENT MONITORING RELAY
50FD PHASE FAULT DETECTOR
50G INSTANTANEOUS GROUND OVERCURRENT RELAY
51 TIME OVERCURRENT RELAY
51G TIME GROUND OVERCURRENT RELAY
51N NEUTRAL INDUCTION TIME OVERCURRENT
51V GENERATOR INVERSE TIME OVERCURRENT WITH VOLTAGE RESTRAIN RELAY
59 OVERVOLTAGE RELAY
60 FUSE FAILURE RELAY
62 TIME DELAY RELAY
63 FAULT PRESSURE RELAY
67 DUAL POLARIZED DIRECTIONAL GROUND RELAY
74 ALARM RELAY
79 RECLOSER
81 FREQUENCY RELAY
83 DROPOUT RELAY
85 CARRIER AUXILIARY RELAYS
85R CARRIER RECEIVER RELAY FOR LINE RELAY CHANNEL
85T CARRIER TRANSMITTER RELAY FOR LINE RELAY CHANNEL
85TTS CARRIER TRANSFER TRIP RELAY SEND
86 HAND RESET LOCKOUT RELAY
87 DIFFERENTIAL RELAY
94 HIGH SPEED TRIPPING RELAY
OSC OSCILLOGRAPH ELEMENT
TT TRANSFER TRIP
57 BFBU BREAKER FAILURE BACKUP

RELAY AND CONVERSION DEVICE (Continued)

$\pm$	BIAS	AVG	AVERAGE
AVG-R	AVERAGE - REJECT	1:1	BOOST
% OR 1:3	GAIN OR ATTENUATE	X	MULTIPLY
$\div$	DIVIDE	f(x)	FUNCTION GENERATOR
REV	REVERSING	A/D OR D/A	ANALOG TO DIGITAL OR DIGITAL TO ANALOG
$\int$	INTEGRATE	Σ	SUMMER
D OR d/dt	DERIVATIVE OR RATE	<	SELECT LOWER
>	SELECT HIGHER	CP	COMPUTER
LIM	LIMITER	VOT	2 OUT OF 3 VOTER
1 - 0	AUTOMATICALLY CONNECT, DISCONNECT OR TRAMSFER ONE OR MORE CIRCUITS	n/m	SELECTOR eg 1/4-1 OUT OF 4
	INTERLOCK		PANEL MOUNTED PATCHBOARD OR MATRIX CONNECTION

ANNUNCIATOR

	PANEL NO. ANNUNCIATOR POINT WINDOW NO.		AUDIBLE ALARM

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FIRST LETTER		SECOND & SUCCEEDING LETTERS														
		DISPLAY DEVICES							CONTROLLING DEVICES							
MEASURED VARIABLE	SYMBOL FOR MEASURED VARIABLES	INDICATING	RECORDING	INTEGRATING INDICATOR (See Note 4)	SCAN (See Note 6)	LOCAL ALARM (See Note 9) LOW	HIGH	LOW HIGH	INDICATING	RECORDING	BLIND	CONTROL VALVE	SELF ACTUATED VALVE	FINAL CONTROL ELEMENT (See Note 8)	SWITCH (See Note 5)	PRIMAR[Y] ELEME[NT]
TYPICAL SYMBOL	( )	( )I	( )R	( )QI	( )J( )	( )AL	( )AH	( )AHL	( )IC	( )RC	( )C	( )V	( )CV	( )Z	( )S( )	( )E
ANALYSIS (See Note 1)	* A	AI	AR		AJ ( )	AAL	AAH	AAHL	AIC	ARC	AC	AV		AZ	AS( )	AE
BURNER FLAME	B	BI	BR		BJ ( )	BAL					BC	BV			BS( )	BE
CONDUCTIVITY	C	CI	CR		CJ ( )	CAL	CAH	CAHL	CIC	CRC		CV		CZ	CS( )	CE
DENSITY	* D	DI	DR		DJ ( )	DAL	DAH	DAHL	DIC	DRC		DV		DZ	DS( )	DE
VOLTAGE (EMF)	* E	EI	ER		EJ ( )	EAL	EAH	EAHL	EIC	ERC	EC			EZ	ES( )	EE
FLOW	F	FI	FR	FQI	FJ ( )	FAL	FAH	FAHL	FIC	FRC	FC	FV	FCV	FZ	FS( )	FE
FLOW RATIO	FF	FFI	FFR		FFJ( )				FFIC	FFRC	FFC	FFV		FFZ		
GAGING (DIMENSIONAL)	G	GI	GR		GJ ( )	GAL	GAH	GAHL	GIC	GRC	GC	GV		GZ	GS( )	GE
HAND	H								HIC		HC	HV	HCV	HZ	HS( )	
CURRENT	I	II	IR	IQI	IJ ( )	IAL	IAH	IAHL	IIC	IRC	IC			IZ	IS( )	IE
POWER	J	JI	JR	JQI	JJ ( )	[illegible]	JAH	JAHL	JIC	JRC	JC			JZ	JS( )	JE
TIME	K	KI	KR	KQI	KJ ( )	KAL	KAH	KAHL	KIC	KRC	KC			KZ	KS( )	
LEVEL	L	LI	LR		LJ ( )	LAL	LAH	LAHL	LIC	LRC	LC	LV	LCV	LZ	LS( )	LE
MOISTURE	M	MI	MR		MJ ( )	MAL	MAH	MAHL	MIC	MRC	MC	MV		MZ	MS( )	ME
LEAK DETECTOR	N	NI	NR									NV				NE
TORQUE	O	OI	OR		OJ ( )	OAL	OAH	OAHL	OIC	ORC	OC	OV		OZ	OS( )	OE
PRESSURE	P	PI	PR		PJ ( )	PAL	PAH	PAHL	PIC	PRC	PC	PV	PCV	PZ	PS( )	PE
PRESSURE DIFFERENTIAL	PD	PDI	PDR			PDAL	PDAH	PDAHL	PDIC	PDRC	PDC	PDV	PDCV	PDZ	PDS( )	
QUANTITY OR EVENT	Q	QI	QR	QQI	QJ ( )	QAL	QAH	QAHL	QIC	QRC	QC	QV		QZ	QS( )	
RADIATION	R	RI	RR	RQI	RJ ( )	RAL	RAH	RAHL	RIC	RRC	RC	RV		RZ	RS( )	RE
SPEED OR FREQUENCY	S	SI	SR	SQI	SJ ( )	SAL	SAH	SAHL	SIC	SRC	SC			SZ	SS( )	
TEMPERATURE	T	TI	TR		TJ ( )	TAL	TAH	TAHL	TIC	TRC	TC	TV	TCV	TZ	TS( )	TE SE[E] NOTE
TEMPERATURE DIFFERENTIAL	TD	TDI	TDR			TDAL	TDAH	TDAHL	TDIC	TDRC	TDC	TDV	TDCV	TDZ	TDS( )	
MULTI-VARIABLE	U	UI	UR		UJ ( )	UAL	UAH	UAHL	UIC	URC	UC	UV		UZ	US( )	
VISCOSITY	V	VI	VR		VJ ( )	VAL	VAH	VAHL	VIC	VRC	VC	VV		VZ	VS( )	VE
WEIGHT	W	WI	WR	WQI	WJ ( )	WAL	WAH	WAHL	WIC	WRC	WC	WV		WZ	WS( )	WE
UNCLASSIFIED (See Note 3)	X	XI	XR		XJ ( )	XAL	XAH	XAHL	XIC	XRC	XC	XV		XZ	XS( )	XE
VIBRATION	Y	YI	YR												YS( )	YE
POSITION	Z	ZI	ZR		ZJ ( )	ZAL	ZAH	ZAHL	ZIC	ZRC	ZC			ZZ	ZS( )	ZE

POOR ORIGINAL

ENSING DEVICES							
BLIND TRANSMITTER (See Note 13)	INDICATING TRANSMITTER	LOCAL OBSERVATION GLASS	TEST CONNECTION	RELAY OR CONVERTER (BLIND) (See Note 2)	UNDESIGNATED FUNCTION	BUFFER	CONTROL STATION NOTE 10
( )T	( )IT	( )G	( )P	( )Y	( )X	( ) B	( )K
AT	AIT		AP	AY	AX		AK
BT		BG	BP	BY	BX		
CT	CIT		CP	CY	CX		
DT	DIT		DP	DY	DX		
ET	EIT			EY	EX		
FT	FIT	FG	FP	FY	FX	FB	
GT	GIT						
IT	IIT			IY	IX		
JT	JIT			JY	JX	JB	
KT	KIT			KY	KX		
LT	LIT	LG	LP	LY	LX	LB	
MT	MIT		MP	MY	MX		
OT				OY	OX		
PT	PIT		PP	PY	PX	PB	
PDT	PDIT					PDB	
QT	QIT			QY	QX		
RT	RIT		RP	RY	RX	RB	
ST	SIT			SY	SX		
TT	TIT		TP SEE NOTE 7	TY	TX	TB	
						TDB	
				UY	UX	UB	
VT	VIT			VY	VX		
WT	WIT			WY	WX		
XT	XIT			XY	XX		
				YY	YX		
ZT	ZIT			ZY	ZX		

NOTES

1. 'A' IS USED FOR ALL ANALYTICAL VARIABLES, FOR EXAMPLE: O_2, H_2O, CO_2 pH, OCTAINE IMPROVEMENT, CHROMATOGRAPH ANALYZING ONE OR MORE STREAMS FOR ONE OR MORE COMPOUNDS, BOILING POINT, FREEZING POINT, COMBUSTIBLES ETC. THE CHEMICAL FORMULA RECOGNIZED SYMBOL (SUCH AS pH) OR A DESCRIPTION DENOTING THE FUNCTION OF THE ANALYZER SHOULD BE NOTED ON THE P & ID OUTSIDE THE INSTRUMENT SYMBOL.

2. THE DESCRIPTION OR SYMBOL DENOTING THE FUNCTION OF THE RELAY 'Y' SHOULD BE SHOWN ON THE P & ID.

3. 'X' IS USED TO REPRESENT ANY SPECIAL VARIABLE, AND MAY BE DEFINED AS REQUIRED. FOR EXAMPLE: MASS FLOW RECORDERS WHICH RECEIVE A SIGNAL FROM A MULTIPLYING RELAY WHICH COMBINES THE PRODUCT OF DENSITY AND FLOW. THIS ITEM IS NOT TO BE CONFUSED WITH 'U' MULTI VARIABLE SYMBOL.

4. WHEN 'Q' IS USED AS A SECOND OR SUCCEEDING LETTER IT DENOTES AN INTEGRATING MODIFIER. FOR EXAMPLE: 'FQI' IS AN INDICATING FLOW INTEGRATOR (OR TOTALIZER). NOTE THAT THE INTEGRATING FUNCTION SHALL BE SHOWN WITH SEPARATE IDENTIFICATION. FOR EXAMPLE: FQI/FRS OR FR/FQIS.

5. STARTUP AND SHUTDOWN DEVICES ARE USUALLY BLIND, BUT MAY BE INDICATING OR RECORDING. IF SO, ADD 'I' OR 'R' AFTER MEASURED VARIABLE. FOR EXAMPLE: FIS, TRS. SWITCH FUNCTIONS SHALL BE FURTHER MODIFIED BY 'L' FOR LOW AND 'H' FOR HIGH.

6. THE DESIGNATION 'AJ( )' MAY DENOTE A SCANNING ANALYZER INDICATOR, RECORDER, TRANSMITTER, ETC. BY USING THE DESIGNATION AJI, AJR, AJT, ETC. RESPECTIVELY.

7. 'TW' DENOTES AN EMPTY THERMOWELL. 'TE' DENOTED A THERMOWELL WITH THERMOCOUPLE OR RTD IN A THERMOWELL OR A SURFACE MOUNTED SHEATHED THERMOCOUPLE.

8. FOR DEVICES OTHER THAN CONTROL VALVES, SUCH AS HYDRAULIC COUPLING, VARIABLE SPEED DRIVES, ETC.

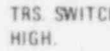
9. HIGH-HIGH ALARMS WILL BE DESIGNATED '( ) AHH' AND LOW-LOW ALARMS '( ) ALL'. FOR EXAMPLE: LAHH DENOTES 'HIGH-HIGH LEVEL ALARM'.

10. ( ) K DENOTES A CONTROL STATION FOR AN ANALYZER OR OTHER INSTRUMENT SUCH AS AK FOR A CHROMATOGRAPHY PROGRAMMER OR INFRARED CONTROL STATION WHICH IS SEPARATE FROM THE ANALYZER ITSELF.

11. WHERE SPECIAL DESIGNATION IS REQUIRED, PILOT LIGHTS SHALL BE IDENTIFIED WITH THE PARTICULAR VARIABLE LETTER, FOLLOWED BY SECOND LETTER 'L'.

12. PRESSURE RELIEF VALVES AND RUPTURE DISKS SHALL BE IDENTIFIED AS 'PSV' AND 'PSE' RESPECTIVELY.

13. ET REPRESENTS A POTENTIAL TRANSFORMER 57

*  WHEN A, D OR E APPEAR IN THIS SYMBOL A = ANALOG; D = DIGITAL; E = EVENT; INPUT TO COMPUTER

INSTRUMENTATION IDENTIFICATION TABLE

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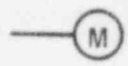
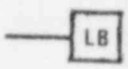
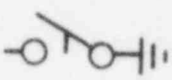
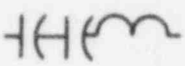
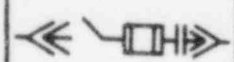
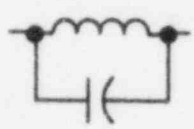
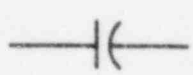
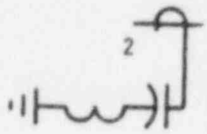
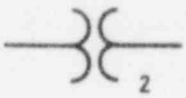
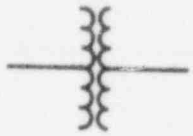
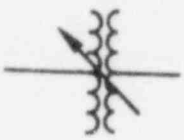
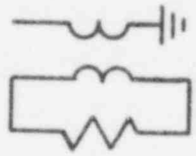
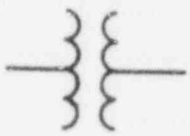
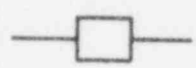
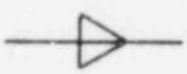
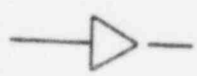
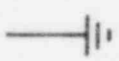
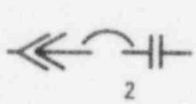
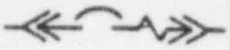
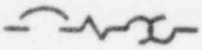
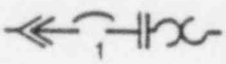
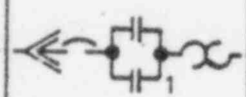
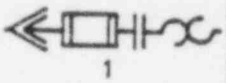
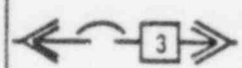
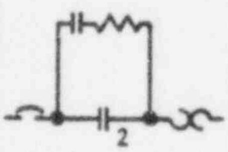
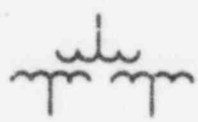
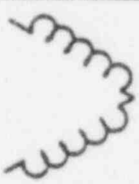
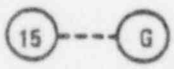
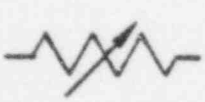
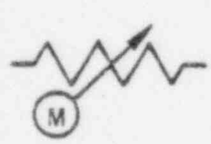
ONE LINE & ELEMENTARY DIAGRAMS

	GANG OPERATED HORN GAP SWITCH		HIGH VOLTAGE PRIMARY FUSE
	GANG OPERATED DISC SWITCH		MOTOR OPERATOR
57 	LOAD BREAK		SINGLE POLE
	GROUNDING SWITCH		LIGHTNING ARRESTER
	COUPLING CAPACITOR FOR CARRIER CURRENT		MEDIUM VOLTAGE COMBINATION FUSED DISCONNECT SWITCH & MOTOR CONTROLLER. FULL VOLTAGE. NON-REVERSING
	COUPLING CAPACITOR WITH POTENTIAL DEVICE		CARRIER CURRENT WAVE TRAP
 2	GROUND DETECTION CURRENT TRANS. NUMBER INDICATES QUANT. RATIO AS NOTED		CAPACITOR
57  1	GROUND DETECTION CURRENT TRANS. ZERO SEQUENCE TYPE. NUMBER INDICATES QUANTITY RATIO AS NOTED.		

ONE LINE & ELEMENTARY DIAGRAMS (Continued)

	BUSHING TYPE POTENTIAL DEVICE NUMBER INDICATES QUANT. RATING AS NOTED		BUSHING TYPE CURRENT TRANSF. NUMBER INDICATES QUANT. RATIO AS NOTED
	POTENTIAL TRANSF. NUMBER INDICATES QUANT. RATING AS NOTED		PWR. TRANSF. SIZE & RATING AS NOTED. AA-OPEN DRY TYPE, GA-SEALED DRY TYPE, DA-NATURAL COOLING, FA-FORCED AIR (FAN) COOLING, FOA-FORCED OIL-AIR (PUMP & AIR) COOLING
	REGULATING PWR. TRANSF. SIZE RATING AS NOTED		GROUND TRANSF. & RESISTOR, SIZE & RATING AS NOTED
57 	GENERAL TRANSFORMER		REACTOR, SIZE & RATING AS NOTED
	DISCONNECT LINKS		HIGH VOLTAGE CIRCUIT BREAKER, INTERRUPTING CAP., SIZE & RATING AS NOTED
	STRESS CONE		CABLE TERMINATOR POTHEAD
	GROUND CONNECTION		COMB.3POLE AIR CIRCUIT BREAKER & MAGNETIC CONTACTOR NUMBER INDICATES SIZE

ONE LINE & ELEMENTARY DIAGRAMS (Continued)

	DRAWOUT DISCONNECT DEVICE		3-POLE CIRCUIT BREAKER DRAWOUT TYPE
	PWR. OPER. AIR CIRCUIT BREAKER DRAWOUT TYPE, MAGNETIC OVER-CURRENT TRIP. AF-AMP FRAME, AT-AMP TRIP, M-MANNUALLY OPER. BREAKER, E-ELECTRICALLY OPER. BREAKER, NA-NON-AUTOMATIC, U-UNDERVOLTAGE ATTACHMENT		MOLDED CASE AIR CIRCUIT BREAKER 3 POLE W/THERMAL & MAGNETIC TRIP
	COMB. 3 POLE AIR CKT. BKR., FULL VOLT., NON-REVERSING SINGLE SPEED STARTER W/THERMAL OVERLOAD ELEMENTS. DRAWOUT TYPE NO. INDICATES NEMA SIZE.		COMB. 3 POLE AIR CKT. BKR. & FULL VOLT. REVERSING SINGLE SPEED MAG. STARTER WITH THERMAL OVERLOAD ELEMENTS DRAWOUT TYPE, NO. INDICATES NEMA SIZE.
	COMB. 3 POLE FUSED DISCONNECT SW. & FULL VOLTAGE NON-REVERSING SINGLE SPEED STARTER, W/THERMAL OVER-LOAD ELEMENTS. NO. INDICATES SIZE.		AIR CIRCUIT BKR. DRAWOUT TYPE SOLID STATE TRIPPING DEVICE NO. INDICATES QUANTITY OF POLES
	PRIMARY RESISTOR REDUCED VOLTAGE STARTER. NO. INDICATES SIZE		THREE WINDING, 3ϕ PWR. TRANSFORMER, SIZE & RATING AS NOTED
	PHASE SHIFTING TRANSFORMER		MOTOR – NO. INDICATES HORSEPOWER
	MOTOR GENERATOR SET SIZE & RATING AS SHOWN		SYNCHRONOUS MOTOR NO. INDICATES HORSEPOWER
	RHEOSTAT - MANUALLY OPERATED		RHEOSTAT-MOTOR OPERATED

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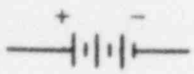
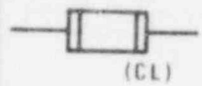
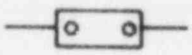
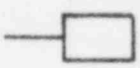
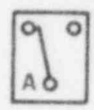
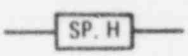
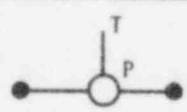
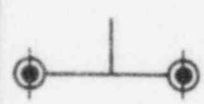
ONE LINE & ELEMENTARY DIAGRAMS (Continued)

	BATTERY (MULTICELL)		FUSE - CL INDICATED CURRENT LIMITING
	SHUNT		RESISTOR
 RTD	RESISTANCE TEMP. DETECTOR		TRANSFER SWITCH M-MANUAL OPERATED, A-AUTOMATICALLY OPERATED
	ELECTRICAL EQUIP. SPACE HEATER	 480V, 100A 3Φ 4W.	POWER RECEPTACLE
	GENERATOR – SIZE & RATING AS NOTED		CONTROL SW. W/RED & GREEN IND. LIGHTS. P INDICATES PERMISSIVE. T INDICATES COMBINATION CONTROL/TEST SW.
	START – STOP MAINTAINED CONTACT PUSHBUTTON		START – STOP MAINTAINED CONTACT PUSHBUTTON W/RED & GREEN IND. LIGHTS
	START – STOP MOMENTARY CONTACT PUSHBUTTON		RED & GREEN INDICATING LIGHTS

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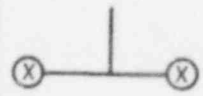
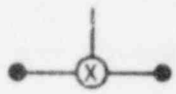
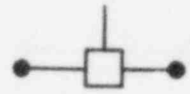
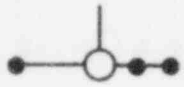
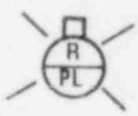
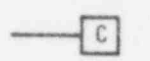
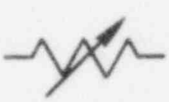
ONE LINE & ELEMENTARY DIAGRAMS (Continued)

	START – STOP MOMENTARY CONTACT PUSHBUTTON W/RED & GREEN INDICATING LIGHTS		HAND–OFF AUTOMATIC SELECTOR SWITCH
	SELECTOR SWITCH W/RED & GREEN INDICATING LIGHTS		TEST SWITCH W/RED & GREEN INDICATING LIGHTS
	LOCKOUT PUSHBUTTON		CONTROL SWITCH W/RED GREEN & AMBER INDICATING LIGHTS
	INDICATING LIGHTS WITH DROPPING RESISTOR. LETTER INDICATES COLOR		BELL ALARM CONTACT
	ANNUNCIATOR POINT		COMPUTER INPUT OR OUTPUT SIGNAL
	VARIABLE RESISTOR	N O	NORMALLY OPEN
N C	NORMALLY CLOSED	---- E ----	ELECTRICAL INTERLOCK

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ONE LINE & ELEMENTARY DIAGRAMS (Continued)

----■----	MECHANICAL INTERLOCK	---- K ----	KEY INTERLOCK
	MANUAL MOTOR STARTER, I POLE W/THERMAL OVERLOAD	C ● ————	CLEAR INDICATING LIGHT
A	AMMETER	V	VOLT METER

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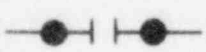
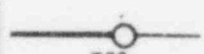
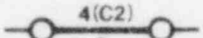
ADDITIONALLY FOR ELEMENTARY BLOCK DIAGRAMS

	SOLID CIRCLE DENOTES TERMINATION FOR INTERNAL WIRING		OPEN CIRCLE DENOTES TERMINAL POINT FOR EXTERNAL WIRING
	NUMBER ON TOP INDICATES CABLE NUMBER. NUMBER BELOW INDICATES CONDUCTOR NUMBER AND COLOR CODE OF THE CABLE ABOVE. EXAMPLE-NUMBER 3 EQUALS "RED".		

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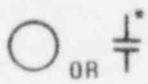
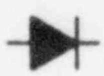
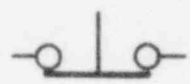
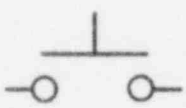
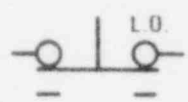
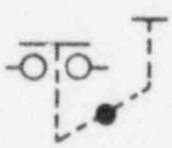
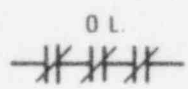
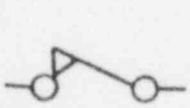
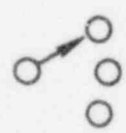
ONE LINE & ELEMENTARY DIAGRAMS (TRANSFORMER CONNECTIONS)

	3 ϕ ZIG-ZAG UNGROUNDED		3 ϕ ZIG-ZAG GROUNDED
	3 ϕ, 3W DELTA UNGROUNDED		3 ϕ, 3W DELTA GROUNDED
	3 ϕ, 4W DELTA UNGROUNDED		3ϕ, 4W DELTA GROUNDED
	3 ϕ OPEN DELTA		3ϕ OPEN DELTA, GND. AT COMMON PT.
	3 ϕ OPEN DELTA, GND. AT MID POINT		3ϕ BROKEN DELTA
57 	3 ϕ WYE OR STAR, UNGROUNDED		3ϕ WYE OR STAR, GROUNDED NEUTRAL
	3 ϕ, 4W WYE OR STAR UNGROUNDED		3ϕ, 4W WYE OR STAR, RESISTANCE GND. NEUTRAL

ELEMENTARY & 1 LINE DIAGRAMS (DEVICES)

	ASTERISK INDICATES PLACEMENT OF TYP. ABBREVIATION OF CONTROL DEVICE	AS	AS-AMMETER SW. VS-VOLTAGE SW. SS-SYNCHRONIZING SW. MSS-METERING SEL. SW.
(3) 	ASTERISK INDICATES PLACEMENT OF TYP. ABBREVIATION OF METER OR INSTRUMENT PREFIX R - RECORDING No. DENOTES QUANT.		ASTERISK INDICATES PLACEMENT OF TYP. ABBREVIATION OF RELAY OR DEVICE

ELEMENTARY & 1 LINE DIAGRAMS (SWITCH CONTACT & MISCELLANEOUS SYMBOLS)

	DIODE		PUSHBUTTON-MOMENTARY CONTACT, NORMALLY CLOSED
	PUSHBUTTON-MOMENTARY CONTACT, NORMALLY OPEN		PUSHBUTTON LOCKOUT
	PUSHBUTTON MAINTAINED CONTACT		CONTACTS OF OVERLOAD DEVICES
	TORQUE LIMIT SWITCH		SELECTOR SWITCH (2 OR 3 POSITION)
	THERMAL ELEMENT		PROTECTIVE RELAY OR SOLENOID

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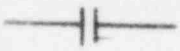
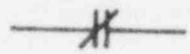
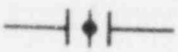
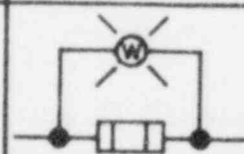
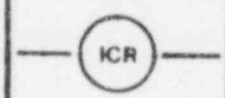
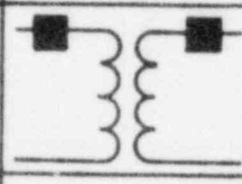
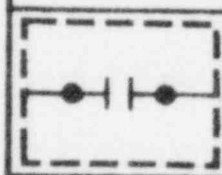
ELEMENTARY & 1 LINE DIAGRAMS (SWITCH CONTACT) (Continued)

	NORMALLY OPEN CONTACT (N.O.)		NORMALLY CLOSED CONTACT (N.C.)
	MOTOR OPER. VALVE POS. LIMIT SWITCH		INDICATING TYPE FUSE
	CONTACTOR OR AUXILIARY RELAY OPERATING COIL		TRANSFORMER WITH POLARITY SIGN
	A DEVICE LOCATED IN A DIFFERENT COMPARTM[illegible] WITHIN THE SWITCH [illegible] OR MOTOR CONTRO[illegible]NTER		

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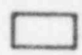
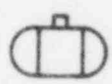
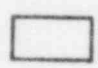
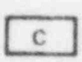
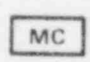
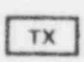
POWER, GROUNDING & LIGHTING PLANS

	LIGHTING PANEL		PANEL– MISCELLANEOUS
	POWER DISTRIBUTION PANEL		MOTOR HORIZONTALLY MTD.
	MOTOR VERTICALLY MTD.		TRANSFORMER– SIZE & RATING AS NOTED
	CONTACTOR		MOTOR STARTER OR CONTROLLER
	TRANSFER SWITCH SIZE & TYPE AS NOTED		DISCONNECT OR SAFETY SWITCH– SIZE & TYPE AS NOTED

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POWER, GROUNDING & LIGHTING PLANS (Continued)

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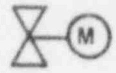
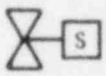
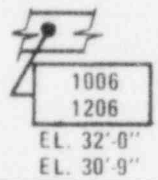
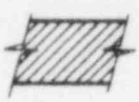
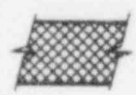
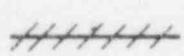
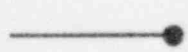
	POWER RECEPTACLE 100A, 480V, 3Ø, 4W		MOTOR OPERATED VALVE
	SOLENOID OPERATED VALVE		UNIT HEATER
	RADIANT HEATER		CABLE TRAY OR LADDER SYSTEM W/NUMBERS FOR COMPUTER CABLE LOADING. ELEV. ARE TO BOTTOM OF TRAY.
	5 KV BUS DUCT		BUS DUCT OVER 5 KV
	DIRECT BURIAL CABLE		RIGID CONDUIT RUN EXPOSED
	RIGID CONDUIT EMBEDDED IN CONCRETE		RIGID CONDUIT RUN CONCEALED
	RIGID CONDUIT RUN BELOW EL. SHOWN		FLEXIBLE CONDUIT
	CONDUIT OR CABLE TURNING UP OR TOWARDS OBSERVER		CONDUIT OR CABLE TURNING DOWN OR AWAY FROM OBSERVER

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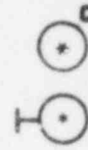
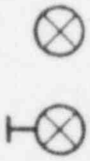
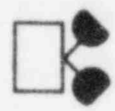
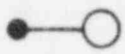
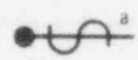
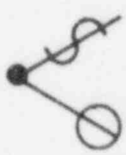
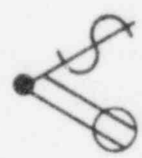
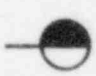
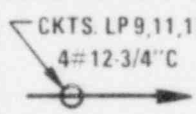
POWER, GROUNDING & LIGHTING PLANS (Continued)

 MTG. HGT. ABOVE FINISHED FLOOR	* INDICATES A LETTER WHICH IDENTIFIES FIXTURE TYPE AS SPECIFIED ON LIGHTING FIXTURE SCHEDULE	 CEILING C-2 WALL MTD.	LIGHTING FIXTURE WITH INCANDESCENT OR MERCURY LAMPS. *INDICATES A LETTER WHICH IDENTIFIES FIXTURE TYPE AS SPECIFIED ON FIXTURE SCHEDULE. "C-2" INDICATES POWER SUPPLIED FROM LIGHTING PANEL "C", CIRCUIT NO. "2".
 CEILING WALL	EXIT LIGHTING FIXTURE	 CEILING WALL	FLUORESCENT LIGHTING FIXTURE
	BARE LAMP FLUORESCENT STRIP		AC/DC EMERGENCY LIGHTING UNIT
	GOOSENECK LIGHTING STANCHION & FIXTURE		STREET LIGHTING FIXTURE
	FLOODLIGHT FIXTURE	 a 3	SINGLE POLE TOGGLE SWITCH a - IND. ASSOCIATED CONTROLLED FIXTURES 3 - IND. 3-WAY SWITCH
	SWITCH & SINGLE CONVENIENCE RECEPTACLE COMBINATION		SWITCH & DUPLEX CONVENIENCE RECEPTACLE COMBINATION
	RECEPTACLE - SINGLE CONVENIENCE, VERTICAL SLOTS, 120V, 20A, 3W, GNDED		RECEPTACLE - DUPLEX CONVENIENCE, VERTICAL SLOTS, 120V, 20A, 3W, GNDED
	RECEPTACLE - SINGLE PHASE, HORIZONTAL SLOTS, 208V, 20A, 3W, GNDED	 CKTS. LP 9,11,15 4#12-3/4"C	HOMERUN TO PANELBOARD - ALL UNMARKED CONDUITS ARE 3/4" & CONTAIN 2 #12 UNLESS OTHERWISE NOTED

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POWER, GROUNDING & LIGHTING PLANS (Continued)

T, P	UNDERFLOOR DUCT W/JUNCTION BOX T-TELEPHONE DUCT P-POWER DUCT	J J	J-IND. JUNCTION BOX TB-IND. TERMINAL BOX PB-IND. PULL BOX ADD BOX NUMBER IF REQUIRED
PB STA.	PUSHBUTTON STATION		

POWER, GROUNDING & LIGHTING PLANS (GROUNDING)

	GROUND ROD		GROUND CONNECTION THERMIT WELD PROCESS
——G——	ANNEALED BARE STRANDED COPPER GND. CABLE RUN EXPOSED, SIZE AS INDICATED	UP	GROUND CABLE RISER UP
DN	GROUND CABLE RISER DOWN	---G---	ANNEALED, BARE STRANDED COPPER GND. CABLE RUN CONCEALED, SIZE AS INDICATED
	GROUND CABLE RISER FROM UNDERMAT GND. GRID PER	——G——→	GROUND CABLE RISER, ___FT. LONG, TERMINATED AT GRADE FOR FUTURE CONNECTION
	PILE WITH GROUND WIRE		GROUND TEST BOX
——TG——→	CONNECTION TO TRAY GROUND	——IG——→	CONNECTION TO INSTRUMENT GROUND

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CONTROL DEVICE CONTACTS (ELEMENTARY)

	—| |—* INDICATES PLACEMENT OF CONTROL DEVICE ABBREVIATION (SAME AS BELOW)	CONTROL DEVICE CONTACTS (ELEMENTARY)		
	FLS *COIF FLS *OOIF	FLOW SWITCH	TS *CORT TS *OORT	TEMPERATURE SWITCH
	COS	CUT-OUT SWITCH	LS *CORL LS *OORL	LEVEL SWITCH
57	PS *CORP PS *OORP	PRESSURE SWITCH	CS	CONTROL SWITCH
	T	THERMOSTAT	LMS	LIMIT SWITCH
	TDC TDO	TIME DELAY CLOSE TIME DELAY OPEN	*C01F	INDICATES CLOSES ON INCREASE OF FLOW
	DPS	DIFFERENTIAL PRESSURE SWITCH	*001F	INDICATES OPENS ON INCREASE OF FLOW
	EPS	ELECTRO-PNEUMATIC SWITCH	*CORP	INDICATES CLOSES ON RISING PRESSURE
	H	HUMIDISTAT	*OORP	INDICATES OPENS ON RISING PRESSURE
	PMS	PERMISSIVE SWITCH	*CORT	INDICATES CLOSES ON RISING TEMPERATURE
	INST	RELAY INSTANTANEOUS CONTACT	*OORT	INDICATES OPENS ON RISING TEMPERATURE
57	*OORL	INDICATES OPENS ON RISING LEVEL AS TABULATION OR TO APPLICABLE DESCRIPTION COLUMNS.	*CORT	INDICATES CLOSES ON RISING LEVEL

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DEVICE ABBREVIATIONS (ELEMENTARY)

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SE	SPECIAL EQUIPMENT FURN. BY MFGR.	PE	PNEUMATIC-ELECTRIC RELAY
PT	PRESSURE TRANSMITTER	TE	TEMPERATURE ELEMENT
FI	FLOW INDICATOR	FT	FLOW TRANSMITTER
TDDO	RELAY TIME DELAY DROPOUT	LT	LEVEL TRANSMITTER
T/C	THERMOCOUPLE	TDPU	RELAY TIME DELAY PICKUP
TZ	TRANSDUCER		

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POWER, GROUNDING & LIGHTING PLANS (DEVICE ABBREVIATIONS)

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XFMR	TRANSFORMER	I	INTERLOCK
SWGR	SWITCHGEAR	STR	STARTER
MCC	MOTOR CONTROL CENTER	HPO	HEALTH PHYSICS OFFICE
PC	POWER PANEL (AC)	RHCP	REHEATER CONTACTOR PANEL
PD	POWER PANEL (DC)	ATP	AUTOMATIC TEMPERATURE CONTROL PANEL
LC	LIGHTING PANEL (AC)	PT	POTENTIAL TRANSFORMER
LD	LIGHTING PANEL (DC)	MTS	MANUAL TRANSFER SWITCH
GND	GROUND	DB	DIRECT BURIAL CABLE
C	CONDUIT	FPC	LOCAL FIRE PROTECTION PUMP CONTROLLER
EP	EXPLOSION PROOF	CT	CURRENT TRANSFORMER
WP	WEATHER PROOF	POS	POSITIVE
VT	VAPOR TIGHT	NEG	NEGATIVE
EC	EMPTY CONDUIT	RL	REMOTE LOCATION
DT	DUST TIGHT	SPT	SEQUENTIAL PROGRAM TIMER
WT	WATERTIGHT (SUBMERSIBLE)	MR	MULTIPLE RATIO
RP	RELAY PANEL	IP	ISOLATED PHASE BUS DUCT
EBB	ELECTRICAL BENCH BOARD	BD	BUS DUCT
MBB	MECHANICAL BENCH BOARD	CP	CONTROL PANEL
LTU	LINE TUNING UNIT	PE	PHOTOELECTRIC

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COMMUNICATIONS

* *	SPEAKER-*REPRESENTS LETTER CORRESPONDING TO SPEAKER AND/OR AMPLIFIER TYPE	*	HANDSET-*REPRESENTS LETTER(S) CORRESPONDING TO HANDSET TYPE
*	SPEAKER AMPLIFIER. *REPRESENTS LETTER CORRESPONDING TO SPEAKER AMPLIFIER TYPE	*	SPEAKER. *REPRESENTS LETTER CORRESPONDING TO SPEAKER TYPE

SPEAKER TYPE (USED WITH FOLLOWING DESIGNATIONS)

A DIRECTIONAL TRUMPET. 85° SOUND DISPERSION WITH 15" HORN (APPROX.); 20" BELL DIAMETER (APPROX.) & 30W. DRIVER.

B PAGING/TALK BACK SPEAKER. 105° SOUND DISPERSION. 9" HORN (APPROX.) 10" BELL DIAMETER (APPROX.)

C WALL MOUNTED CONE SPEAKER ASSEMBLY, WALNUT FINISHED SPEAKER BAFFLE WITH 8 Ohm. 8" DIAMETER (APPROX.) CONE SPEAKER AND VOLUME CONTROL.

D CORRIDOR TYPE, BI-DIRECTIONAL BAFFLE WITH 8" DIAMETER SPEAKER AND VOLUME CONTROL.

E FLUSH, WALL OR PANEL MOUNTED CONE SPEAKER ASSEMBLY WITH PROJECTING BAFFLE. 8" DIAMETER SPEAKER WITH VOLUME CONTROL.

F FLUSH, WALL OR PANEL MOUNTED CONE SPEAKER ASSEMBLY WITH FLUSH BAFFLE. 8" DIAMETER SPEAKER

G MULTI-DUTY WEATHERPROOF HIGH-FREQUENCY SPEAKER. 120° SOUND DISPERSION.

H DUAL, WIDE ANGLE HORN SPEAKER. 120° × 60° SOUND DISPERSION. 10" HORN (APPROX.) 18" × 9" BELL DIAMETER (APPROX.) WITH 30W. DRIVER.

SPEAKER AMPLIFIER (USED WITH FOLLOWING DESIGNATIONS)

M SPEAKER AMPLIFIER ASSEMBLY PUSH-PULL, CLASS B. 12WATT AMPLIFIER WITH INDIVIDUAL VOLUME CONTROL; BAKED ENAMEL ON ZINC CHROMATE PHOSPHATE BONDED ENCLOSURE.

N SPEAKER AMPLIFIER ASSEMBLY (AMPLIFIER TYPE SAME AS M) WEATHERPROOF ENCLOSURE.

HANDSET TYPE (USED WITH FOLLOWING DESIGNATIONS).

Q DESK-TOP STATION WITH REMOTE HANDSET SPEAKER. SPEAKER AMPLIFIER. VOLUME CONTROL & ENCLOSURE.

K WALL STATION WITH HANDSET. SPEAKER AMPLIFIER & ENCLOSURE.

R DESK EDGE STATION WITH SUBSET. REMOTE HANDSET. SPEAKER AMPLIFIER & ENCLOSURE.

S FLUSH PANEL STATION WITH SUBSET. REMOTE HANDSET. SPEAKER AMPLIFIER & ENCLOSURE.

T WEATHERPROOF WALL STATION: WITH HANDSET. SPEAKER AMPLIFIER & WEATHERPROOF ENCLOSURE.

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TELEPHONE

D	DIRECT DISPATCH TELEPHONE		COMMERCIAL TELEPHONE
	LOAD DISPATCH INTERCOM SYSTEM	T/S	TELEPHONE SWITCHBOARD
SP	SOUND POWERED TELEPHONE JACK	P D W	PRIVATE AUTOMATIC EXCHANGE (PAX) TELEPHONE D – INDICATES DESK MTD W – INDICATES WALL MTD
			SOUND PROOF ENCLOSED HANDSET

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COMMUNICATIONS (CONT.)

RADIATION MONITORS

	AREA		PARTICULATE
	HAND & FOOT		LIQUID MONITOR
	GAS MONITOR		FRISKER MONITOR
	LAUNDRY INSPECTION MONITOR		PARTICULATE/GASEOUS MONITOR
	PARTICULATE SAMPLER		PARTICULATE/GASEOUS/IODINE MONITOR
	TRITIUM SAMPLER		ALPHA MONITOR

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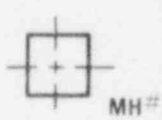
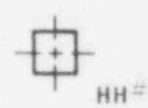
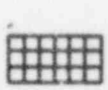
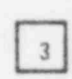
UNDERGROUND DISTRIBUTION PLANS

	MANHOLE		HANDHOLE
	DUCTBANK		SEISMIC JOINT
	SINGLE PHASE TRANSFORMER SIZE & RATING AS NOTED		THREE PHASE TRANSFORMER SIZE & RATING AS NOTED
	STREET LIGHTING REGULATOR RATING AS NOTED		

HAZARD MONITORS – LOOP/LOGIC USE

HF VAPOR AE	HF ACID VAPOR MONITOR	FIRE – IR AE	INFRARED FLAME DETECTOR
FIRE – I AE	IONIZATION SMOKE DETECTOR	FIRE – T AE	THERMAL HEAT DETECTOR – FIXED TYPE
FIRE – P AE	PHOTOELECTRIC SMOKE DETECTOR	FIRE – TL AE	THERMAL HEAT DETECTOR – LINE TYPE
FIRE – UV AE	ULTRAVIOLET FLAME DETECTOR	FIRE – TR AE	THERMAL HEAT DETECTOR – RATE OF RISE TYPE

HAZARD MONITORS – SYSTEM ARRANGEMENT USE

AREA I	IONIZATION SMOKE DETECTOR	IR	INFRARED FLAME DETECTOR
DUCT I	IONIZATION SMOKE DETECTOR (ON HVAC DUCTING)	FIXED T	THERMAL HEAT DETECTOR – FIXED TYPE
P	PHOTOELECTRIC SMOKE DETECTOR	RATE T	THERMAL HEAT DETECTOR – RATE OF RISE TYPE
UV	ULTRAVIOLET FLAME DETECTOR	LINE T	THERMAL DETECTOR – LINE TYPE

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4.1 SUMMARY DESCRIPTION

The Clinch River Breeder Reactor Plant (CRBRP) uses a mixed (Pu-U) oxide fueled, sodium cooled fast reactor having a total thermal output of 975 Mwt. A schematic of the reactor is shown in Figure 4.2-36. The reactor vessel, the closure head, the inlet nozzles and the core barrel are identified in this figure. The core support plate and the support cone form the principal pressure boundary inside the vessel. The fuel, control, blanket and removable shield assemblies are supported by the core support plate which also supports a fixed radial shield. Each of these reactor assemblies has two load pad areas which match the elevation of the core former rings. The rings are supported by the core barrel which is welded to the core support plate.

The upper internals structure, located above the core, is
supported from the intermediate rotating plug of the vessel closure and
keyed to the upper core former ring permitting vertical motion while re-
straining lateral and rotational motion. The structure laterally stabilizes
primary and secondary control rod shroud tubes. In case of a loss of hydraulic
balance, the upper internals structure acts as a secondary holddown device.
The four support columns of the upper internal structure have jacks for lifting
57 the upper internals structure with its keys clear of the core former ring and
reactor assemblies for refueling. The in-vessel transfer machine rotates with
the upper internals structure for removing and replacing of reactor assemblies
at refueling.

A vortex suppressor plate is provided just below the sodium pool surface to minimize gas entrainment in the sodium exiting from the outlet plenum. Fuel transfer and contingency storage positions are provided in the annulus formed between the core barrel and the reactor vessel thermal liner.

The active length of the core is 36 inches and the equivalent diameter is 79.5 inches. The fuel region consists of a single enrichment zone with a total fissile plutonium loading of $\sim$1500 Kg. The reactor control systems include 9 primary and 6 secondary control rods. The two systems are independent and diverse. Both the systems are capable of shutting down the reactor from full power to hot standby conditions. The core mid-plane details are shown schematically in Figure 4.3-1.

4.1.1 Lower Internals

The lower internals structure positions and restrains the reactor
assemblies. The main components of the structure are: the core support
structure composed of the core support plate and core barrel, horizontal
baffle, core former rings, fixed radial shielding, lower inlet modules,
bypass flow modules and fuel transfer and storage assemble. These compon-
51 ents are shown in Figure 4.2-36. Most of these components are also shown
in Figure 4.2-37.

The core support cone, a component of the reactor vessel, is an inverted truncated conical shell that connects the thick circular perforated core support plate with the reactor vessel wall. These two welded-in components (plate and cone) form the upper boundary for the reactor vessel inlet high pressure plenum.

The core barrel is a thick wall right circular cylinder that extends upward from the outer edge of the core support plate to the top plane of the reactor assembly outlet nozzles.

The horizontal baffle forms the upper boundary of the annular region between the core barrel and the reactor vessel and separates sodium in the outlet plenum region from bypass flow sodium below the baffle. It limits leakage to the outlet plenum and heat flow to the bypass sodium to provide for cooling components within the annulus and the reactor vessel above the baffle while minimizing flow which bypasses the core.

Supported inside the core barrel are upper and lower core former rings. These rings are contoured inside to the outline formed by the outer surfaces of the upper and lower load pads of the outer row radial shield assemblies. A small gap is provided in the cold condition between the rings and shield assemblies to allow a small amount of free bow at power operation. This gap facilitates replacement of reactor assemblies at refueling.

Fitted to the inside of the core barrel is fixed radial radiation shielding for protecting the barrel from structural damage from neutron fluence.

There are 61 lower inlet modules for the core. A lower inlet module is shown in Figure 4.2-40. They are surrounded by six bypass flow modules. The lower inlet modules are inserted into lined holes in the core support plate. The bypass flow modules (Figure 4.2-41A) are installed on the core support plate. Each lower inlet module holds and distributes sodium coolant flow to the inlet nozzles of seven reactor assemblies. The bypass flow modules also receive low pressure coolant from the lower inlet modules and distributes the coolant to the removable radial shield assemblies.

The core lattice of equilateral triangular pitch is established by the core support plate and the lower inlet modules. The inlet nozzles of the reactor assemblies are held on this lattice by their respective modules which allow some on-plant rotation of the nozzles as part of the core restraint system. Lateral above-core restraint of the reactor assemblies is provided by the core former rings, reference Figure 4.2-37, which are located at the upper and lower hard-faced pad elevations of the reactor assemblies. These rings act on the outer row radial shield assemblies and contribute to the stable control of reactivity as the reactor power and
51 coolant temperature are changed. In addition, they provide the lateral support required to withstand seismic events.

orific to provide coolant flow to the bypass flow module/removable radial shielding and the core barrel/reactor vessel annulus. Although the modules are 30 year life components, they will be designed to be removable. The inlet modules have the following principal functional requirements:

a) Support, vertically position and restrain downward, position and restrain horizontally the reactor assemblies during assembly, operation and refueling of the reactor using hydraulic balance features where required.

b) Distribute and provide coolant to the various reactor assemblies (fuel assemblies, blanket assemblies, control rod assemblies, and removable radial shielding).

c) Provide a source of low pressure bypass coolant to the bypass flow module/removable radial shield assemblies and the core barrel/reactor vessel annulus.

d) Provide features to assure correct placement of the reactor assemblies in a safe location.

e) Maintain a pressure boundary between the high pressure region and low pressure region within the reactor vessel and limit the leakage flow across the boundary.

f) Provide a low impedance flow path through the LIM for the Secondary Control Rod System bypass flow.

g) Provide retention of loose debris greater than 0.25 in. in diameter to preclude blockage of the reactor assembly rod bundles.

h) Provide for the retention of the modules during normal reactor operation using hydraulic balance features where required.

i) Assure that the LIM can be removed through the Upper Internals Structure and the IVTM port.

j) Maintain nominal primary coolant flow and preclude any adverse change of flow paths.

k) Provide a capability to use multiple coolant flow sources in the core support structure module liner.

4.2.2.1.1.3 Bypass Flow Modules

The Bypass Flow Modules (BPFM) shown in Figure 4.2-41A, are supported by the Core Support Structure (CSS). Subsequently, the BPFM
51 supports and positions the Fixed Radial Shield (FRS) and Removable Radial

Shield (RRS) assemblies. The BPFM is designed as a permanent component. The BPFM receives low pressure coolant from the inlet modules and distributes the coolant to the RRS assemblies. A total of 6 BPFM provides a flow to a total of 264 RRS and each is designed to meet the following functional requirements.

a) Support and position a total of 44 removable radial shield assemblies per module during reactor assembly, operation, and refueling.

b) Distr[illegible]nd provide coolant to each of the radial shield as[illegible]ported.

c) [illegible]ve mechanical discrimination features to insure p[illegible] only removable radial shield assemblies into any of the receptacles in the bypass flow modules.

d) Provide a redundant flow path for the coolant which feeds the RRS assemblies.

e) Support and position the FRS.

f) Provide a thirty year life (22.5 full power years) with no planned maintenance.

4.2.2.1.1.4 Fixed Radial Shielding

The fixed radial shielding is located inside the core barrel beyond the radius of the removable radial shielding and rests on the bypass flow module and beneath the lower core former. The fixed radial shielding is designed for the 30 year plant life. The functional requirements for this component are:

a) In conjunction with the removable radial shielding assemblies, the fixed radial shielding will provide radiation protection for the core barrel and reactor vessel. This shielding will contribute to the overall reactor shielding system. The minimum ductility provided by the combination of the fixed and removable radial shielding is 10% residual ductility for the core barrel and reactor vessel. This value is based on the total elongation at end of design life and includes effects due to stretch conditions.

b) Operate for a thirty year life at seventy-five percent plant capacity factor. The minimum ductility limits (based on total elongation) for the fixed radial shielding are the following:

Shielding Material at Attachments	10%

57|51

b) Requirement - Operate for a thirty year life at seventy-five percent plant capacity factor which also includes effects due to stretch conditions. The minimum ductility limits (based on total elongation) for the fixed radial shielding are the following:

Shielding Material at Attachments	10%

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Bases - The combination of removable and fixed radial shielding must be designed such that the fixed radial shielding remains ductile within the limits stated above. The ability of the material to yield locally would be reduced if the ductility of the material dropped below the stated limits. This situation would produce uncertainties concerning the integrity of the structure.

The ductility of the structural regions of the fixed radial shielding must be maintained at a level which will insure that the ductile failure mode analysis used in analyzing the design remains valid.

4.2.2.1.2.5 Fuel Transfer and Storage Assembly

a) Requirement - Provide features which allow cooling of the stored fuel assemblies.

Bases - The removed fuel assembly cannot be allowed to increase in temperature such that the gas pressure would build up and result in cladding failure. Additionally, a removed fuel assembly inherently contains useful information which could be compromised or destroyed if the fuel temperature subsequent to removal is permitted to exceed significantly the peak operating temperature. Thermal analysis has indicated that cooling of the transfer and storage assembly is required to remove the heat generated in a spent fuel assembly.

b) Requirement - Provide interface features that minimize leakage with the horizontal baffle.

Bases - The outlet plenum is filled with the hot core effluent, while the plenum underneath the horizontal baffle is filled with the cooler core inlet sodium. The upper portion of the fuel transfer and storage assembly is a part of the horizontal baffle and provides the guidance for insertion of the core component pot and the stored fuel into the fuel transfer and storage
51 assembly. If significant leakage flow were permitted in between

the fuel transfer and storage assembly and the horizontal baffle it could lead to possible insufficient material fatigue strengths to withstand the potential hot and cold flow oscillation sweeping the interface surfaces between the horizontal baffle and the fuel transfer and storage assembly.

c) Requirement - Provide interface features which permit differential thermal expansion between the fuel transfer and storage assembly and the horizontal baffle.

Bases - The core barrel and the fuel transfer and storage assembly will have somewhat different average temperatures. The different temperatures, which occur over a long length, could induce stresses in both the core barrel and the fuel transfer and storage assembly if some provision is not included to allow free differential expansion between the two components.

4.2.2.1.2.6 Horizontal Baffle

a) Requirement - Limit heat flow from the outlet plenum sodium to the core barrel/reactor vessel annulus sodium to an acceptable level.

Bases - Bypass flow in the core barrel/reactor vessel annulus cools components within and on the periphery of the annulus as well as the vessel and vessel liner. Therefore, heat flow across the baffle must be within limits which will provide acceptable bypass cooling flow temperature to components which utilize bypass flow for cooling.

b) Requirement - Form an effective hydraulic barrier between the core barrel/reactor vessel annulus and the outlet plenum sodium.

Bases - Random mixing of hot sodium flowing through and exiting through the fuel region and sodium rising in the annulus between the core barrel and the reactor vessel creates unstable flow conditions characterized by thermal plumes and/or widespread turbulence. In addition, bypass flow must be directed to provide coolant to the reactor vessel and liner above the baffle.

c) Requirement - Insure that temperature oscillations on metal surfaces in or adjacent to the coolant flow paths are compatible with the horizontal baffle service life requirements.

Bases - Thermal stratification exists in the flow exiting the core. The flow through the radial blanket will be cooler than that in the main core region. The radial blanket flow will tend to be split off at elevation just above the core barrel (base of the UIS). This cooler flow will mix with that circula-
51 ting in the outlet plenum, and thermal striping can result.

Loads from weight, hydraulic pressure drop and seismic acceleration are transmitted by the support plate to the reactor vessel. Sizing analysis for internal pressure, flow blockage, control rod drop, and seismic loads indicate that under normal operating loads with flow blockage the inlet module meets the ASME Section III criteria for primary stresses.

Six bypass flow modules, surrounding the lower inlet modules, distribute low pressure coolant received from the lower inlet modules to the removable radial shield assemblies. The bypass flow modules provide receptacles to accept the removable radial shield assemblies that are not positioned in the lower inlet modules.

51 The details of the FRS are provided in Section 4.2.2.2.1.4.

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The general design rule of 5.0% minimum residual ductility insures that non-ductile fracture will not occur during short term loadings in reactor internal structures. This criterion is based upon the minimum residual total elongation of 10.0% and the established relationship between total and uniform residual elongation of $\varepsilon_t = \varepsilon_u + 5\%$ as noted in Table 4.2-53. This relationship is based upon the end-of-life tensile test data in Tables 4.2-54 through 4.2-57 and data from References 178, 179 and 180. It is conservatively based upon a data set showing the least uniform elongation for a total elongation of 10.0%. An evaluation of all current data indicates that when the degradation on ductility is greatest at a particular fluence level the uniform elongation tends to be a greater fraction of the total than this relationship indicates. Since this limit is based upon uniaxial test data a correction for the multiaxial state of stress for actual reactor component conditions is required. This correction can be performed using scientific paper 67-JDO-CODES-P1, "Applied Mechanics in the Nuclear Industry Applications of Stress Analysis". For a typical thermal stress conditions which causes an equibiaxial stress state the 5.0% would be reduced to 0.9%. The elongation available to insure ductile behavior can be determined by considering the factor of safety, consistent with the ASME Code Section III factor of safety protecting against ultimate failure. The use of the factor of safety of 3.0 would reduce the elongation for a equibiaxial state of stress to 0.30%.

The applied strain considered relevant to this elongation limit is the maximum value of the three principle strains and represents an accumulation of elastic plus plastic strain at the end of life. These limits would apply at a minimum to membrane plus bending strains regardless of whether the loading is primary or secondary. Thermal transient strains in reactor internal components are less than the 0.30% membrane plus bending. Therefore, from the tensile data base that is presently available, the ductility required at the end-of-life in reactor internal components is sufficient to insure their integrity when 10% residual total elongation is available and the criteria described is applied. In locations where significant fatigue damage occurs in the low cycle regime, which is also affected by the ductility of the material, corrections to the fatigue design curves are applied using accepted theories of fatigue design curve construction which are based upon reduction in area.

A test program is presently in place which will experimentally characterize the fracture toughness of reactor component materials when subjected to a fast-neutron irradiation environment. This program includes tests of smooth, notched and welded specimens. The establishment of the fracture toughness and fatigue crack propagation characteristics will provide a basis for confirmation of the described criteria or the substitution of a
57 more refined criteria.

4.2.2.2.1.2 Lower Inlet Module

Sixty-one inlet modules support and position the reactor assemblies on the core support plate. These modules distribute the coolant to the following reactor components: fuel assemblies,
51 blanket assemblies, removable shield assemblies, control rod

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assemblies, core barrel, pressure vessel thermal liner, fuel transfer and storage assembly and horizontal baffle. Each module fits into a liner integral with the core support plate and supports and positions seven reactor assemblies while providing orificing that is unique to specific reactor assembly locations. Figure 4.2-41 shows 1/6 of the core and indicates the relative position of fuel and radial blanket assemblies and orifice zones.

The module stem acts as a strainer which collects and prevents loose debris from directly blocking the various reactor assemblies.

Mechanical discrimination features are designed into each module to assure placement of the reactor assemblies into core lattice positions that will not result in assembly undercooling. Angular alignment of each module for its lattice position is assured by an alignment pin between the module liner and the module. The modules are welded 304 stainless steel structures. There are several internal configurations, excluding discrimination differences, due to the differing flow requirements of the reactor assemblies.

4.2.2.2.1.3 Bypass Flow Module

The bypass flow modules shown in Figure 4.2-41A, are functionally similar to the lower inlet modules in that they provide support and position removable radial shield assemblies and direct low pressure flow to cool these assemblies. There are a total of 6 identical modules designed to rest on the core support plate and conform to the periphery formed by the 61 lower inlet modules. A flow pipe attached to the bottom of a bypass flow module mates with a hole in the core support plate. This provides a flow path for the coolant between the lower inlet module and bypass flow module.

The bypass flow modules distribute 1.22% of the total nominal reactor flow to 264 removable radial shield assemblies, 44 of which are in each module. Flow enters each of the six bypass flow modules through a bottom entry port. Each bypass flow module is hydraulically interconnected to the adjacent two bypass flow modules giving multiple flow sources for all the RRS assemblies served by the bypass flow modules. The removable radial shield assemblies fit into receptacles integral with the bypass flow modules. These receptacles are designed with a mechanical discrimination feature to assure placement of only the removable radial shield assemblies into the bypass flow module.

The low pressure existing within the region of the outer removable radial shielding results in negligible hydraulic forces and consequently a hydraulic balance system is not required. The assemblies are simply slip fitted into the receptacle permitting
51 easy insertion and removal.

4.2.2.2.1.4 Fixed Radial Shield

The fixed radial shield is a segmented annular ring of type 316 stainless steel located between the removable radial shielding and the core barrel as shown in Figure 4.2-42. The segments rest on the bypass flow modules and extend upward to the lower core former structure. The segments are laterally positioned by captured pins at the lower end to the bypass flow modules and at the upper end to the former structure. The pinning
36 arrangement accommodates differential thermal expansion and results in the

fixed radial shield being a simple unrestrained structure. The fixed radial shield weight is carried by the bypass flow modules and the seismic loads are transmitted through the core former structure and bypass flow modules to the core support structure. The fixed radial shield in conjunction with the removable shielding protects the core barrel and vessel from radiation damage to assure the retention of ductility for a design lifetime of thirty years.

4.2.2.2.1.5 Fuel Transfer and Storage Positions

Reactor refueling requires bringing new fuel into the reactor vessel and removing the spent fuel. The fuel is brought in and out through the vessel head by an ex-vessel transfer machine and is handled inside the vessel by an in-vessel transfer machine. A fuel transfer position is required to set down an assembly so that one machine can decouple and move out of the way so the other machine can grapple the assembly to continue the fuel handling operation.

The component surveillance program necessitates placement of specimens outside the core barrel. The fuel transfer, fuel storage, and surveillance specimen positions are provided by the five wells in the reactor vessel/core barrel annulus.

The wells are fabricated of Type 304 stainless steel and are attached to the core barrel and the horizontal baffle. Thus all dead weight and earthquake loads are transmitted to the core support structure.

4.2.2.2.1.6 Horizontal Baffle

The horizontal baffle shown in Figure 4.2-44 forms the upper boundary of the core barrel/reactor vessel annulus and physically separates hot sodium in the outlet plenum from the cooler bypass flow sodium in the core barrel/reactor vessel annulus. The baffle maintains the temperature of the sodium in the core barrel/reactor vessel annulus close to reactor inlet temperature to reduce temperature differences across components below the baffle and to provide for decay heat removal from the irradiated reactor assemblies, stored in the fuel transfer and storage assembly. In addition, the boundary formed by the baffle forms a part of the flow path which diverts bypass flow between the reactor vessel and reactor vessel thermal liner, through uniformly spaced holes in the thermal liner below the baffle, to provide cooling for the reactor vessel and reactor vessel thermal liner. A small pressure differential must be maintained across the horizontal baffle to provide the head for this flow. The pressure differential, approximately 0.5 psi, causes leakage through the seals at the edges of the horizontal baffle base plate and at the FT&SA inlet port nozzles. However, the leakage is limited to 12.5% of bypass flow to insure that sufficient cooling flow
51 is provided to the vessel and vessel thermal liner.

The horizontal baffle design incorporates a single 1.5 inch thick simply supported base plate restrained on the outer diameter by a segmented ring outer edge attachment. Each outer ring segment includes a top ring segment, a spacer block and a bottom ring segment, all of which are bolted to the vessel liner flange with a single bolt that extends through the ring segments and spacer block and into the vessel liner flange. At the inner diameter the base plate is supported on a ledge on the core barrel wall. It is held vertically by a con-
57 tinuous retaining ring and located radially by a spacer ring. Circumferential motion of the base plate relative to the core barrel is restrained through a key. Radial movement of the base plate is not restricted and wear resistant surfaces of Haynes 273 are provided on both sides of the plate at the inner and outer diameters and on the ring segments to accommodate relative radial and angular rotation displacements due to thermal and seismic effects at the outside diameter and angular rotation displacements due to thermal effects at the inside diameter.

The base plate normally operates with a 150-200°F temperature difference through the thickness. Since the upper surface is hotter, the plate will tend to develop an upwardly convex spherical curvature. The plate edges, however, are restrained vertically to the relative vertical thermal displacement between the vessel thermal liner flange and the core barrel ledge. As a result of the vertical restraint, a thermally induced vertical downward force will act on the vessel liner flange and an equal upward force will act on the core barrel.

These vertical reaction forces provide a positive seal at the base plate outer and inner diameters. During down transients, the direction of the holddown forces can reverse due to the reversal of the through-the-thickness temperature gradient. The core barrel ledge will be in compression (down load) and the upward load at the vessel liner flange is carried through the top ring segments.

The portion of fuel transfer and storage assembly associated with the baffle consists of five penetrations through the base plate at a radius of 85.62 inches with an inlet port nozzle at each penetration. The penetrations allow access to the portion of the FT&SA located in the core barrel/reactor vessel annulus. The nozzles are fabricated from Inconel 718 because of the thermal striping (alternate washing of a metal surface with hot and cold fluid) anticipated in the FT&SA inlet
41 ports.

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The horizontal baffle, except for the Inconel 718 FT&SA inlet
41 port nozzles, is fabricated from Type 316 stainless steel.

Flow patterns in the region immediately above the core have been investigated in water table tests. These tests have shown that a torroidal flow pattern exists in the mixing chamber located directly above the core. A large portion of the stream to stream temperature differences are reduced in this chamber before the flow exits. Temperatures in these flow streams differ substantially, hence the mixing adjacent to the inner surface of the mixing chamber 57 41 results in thermal striping. The material selected for the exposed surfaces in the mixing chamber must therefore have an endurance stress limit in excess of the maximum anticipated stress amplitude produced by fluid mixing. This requirement led to the selection of Inconel 718 for the exposed surfaces of mixing chamber components.

4.2.2.2.1.8 Core Restraint System

Design of the CRBRP core restraint system is based upon the limited free bow concept. Essential features of this concept are illustrated in Figure 4.2-47. Fixed peripheral formers provide lateral support to the core assemblies at two locations above the active core. A third support at the core support plate elevation completes the lateral support configuration.

Relief of restraint loads for refueling in the limited free bow core restraint concept is achieved by allowing the core assemblies limited freedom for unrestrained bowing during the core startup and shutdown transients.

The amount of free bow permitted is controlled by sizing the gaps between core assembly load pads, and between the peripheral load pads and the adjacent core formers. The upper bound of the allowable core and former gaps is defined by a conservative analysis of the effect on critical core components of a step compaction of the core through the range of free motion permitted by the gap configuration. The resulting core step reactivity insertion is not allowed to produce transient heating rates in the fuel which would result in the fuel pin upset condition damage limits being exceeded.

It is evident that the core restraint system in its entirety includes all the reactor assemblies plus elements of the core support structure and the upper internals structure. Only the core formers, their associated retention and positioning hardward and the removable radial shield assemblies are categorized as core restraint hardware.

The core formers are comprised of profile milled segments assembled into continuous rings, as illustrated in Figure 4.2-46 and centered in the core barrel cavity by means of radial shims. The above core load plane 41 former, called the lower core former, is mounted on a ledge machined in the inner diameter of the core barrel. A spacing cylinder provides holddown for the lower core former and support for the top load plane former called the upper core former. The upper core former has six lugs that fit slots in the top of the core barrel to transmit seismic and other loads to the core barrel. A series of L-shaped keys are circumferentially slipped into the groove on the inside of the core barrel, between each of the six lugs, and trapped by means of a radially oriented dowel pin on either side of each slot. These L-shaped keys prevent vertical displacement of the core formers away from the load 49 planes.

4.2.2.2.1.9 Removable Radial Shield

The radial shield assemblies are made up of stainless steel rods held within thin walled hexagonal ducts. These assemblies are designed to be as flexible as possible in order not to contribute to the off-power restraint loads. A close-fitting support block is inserted inside the duct at the ACLP to provide axial restraint for the shield rods and to absorb seismic loads that are transmitted through the ACLP to the core former.

4.2.2.2.1.10 Maintainability

All the reactor internals except for the reactor assemblies, are designed for a 30 year life. However, provision has been made to permit removal of the lower inlet modules to assure full plant life and malfunction recovery capability. All items of core support structure equipment with any significant potential for maintenance are located in the removable lower inlet modules. Items having some potential for maintenance include:

1. The reactor assembly receptacles, subject to insertion and removal of reactor assemblies.

2. Strainers and orifices, subject to coolant induced changes, such as wear or partial plugging.

4.2.2.2.1.11 Surveillance and In-Service Inspection

Surveillance

Material surveillance coupons are contained within special assemblies located in removable radial shield positions and a fuel transfer and storage assembly. In addition to these special assemblies, irradiated removable shield assemblies will be available for material surveillance.

4.2.2.3 Design Criteria

The design criteria presented in this section are those that were in effect at the time analyses were performed. These analyses will be updated as required, to reflect the 1974 edition of the ASME Code, which provides the basic design criteria for
51 these components.

4.2.2.3.1 Lower Internals Structures (LIS)

The LIS components and Core Former Structure (CFS) are evaluated as nuclear components in accordance with the rules of:

The ASME Boiler and Pressure Vessel Code, Section III.

Where these rules cannot be applied, the following rules are invoked:

a. Code Case 1592, Class I Components in Elevated Temperature Service, Section III

b. RDT F 9-4 Components at Elevated Temperature (Supplement to ASME Code Cases 1592, 1593, 1594, 1595, and 1596)

c. RDT F 9-5, Guidelines and Procedures for Design of Nuclear Systems Components at Elevated Temperatures (Non-mandatory)

d. The special purpose strain controlled high-cycle fatigue criterion discussed in Section 4.2.2.3.2.2 may be applied to 304 and 316 austenitic stainless steel at temperatures
57 up to 1100°F.

Material properties not given in the Code are taken from the Nuclear Systems Materials Handbook, TID-26666, and Section 4.2.2.3.3.1 below.

4.2.2.3.1.1 Core Support Structure (CSS)

The CSS was analyzed using the following additional rules:

a. The 1974 Edition of the Code, Subsection NB and selected portions of Subsection NG with Addenda through Summer 1975.

b. RDT Standard E 15-2NB, November 1974, (Supplement to ASME Code Section III, Subsections NA and NB).

c. Modification to the high temperature design rules for Austenitic Stainless Steel - same as para. 4.2.2.3.2.2f.

4.2.2.3.1.2 Lower Inlet Module (LIM), Bypass Flow Module (BPFM), and Core Former Structure (CFS)

The 1974 Edition of the Code with Addenda through Winter 1976 were used for the LIM, BPFM, and CFS analyses.

4.2.2.3.1.3 Horizontal Baffle (HB), Fuel Transfer & Storage Assembly (FT&SA), and Fixed Radial Shield (FRS)

The HB, FT&SA, and FRS are internals structures and are not covered by mandatory Code rules, but the Owner's designee has required that the rules stated in 4.2.2.3.1 be applied to the design and analysis of these components. The HB and FT&SA use the 1974 Edition of the Code with Addenda through Winter 1976, and the FRS uses the 1977 Edition of the Code with Addenda through Winter 1977.

4.2.2.3.2 Upper Internals Structure (UIS)

Code criteria applicable to the analysis of the UIS are divided into two categories as follows:

4.2.2.3.2.1 Class 1 Appurtenances

Those portions of the UIS support columns located within the boundary of code jurisdiction for appurtenances and the IVTM port plug cap shall be analyzed as Class 1 appurtenances in accordance with:

a. 1974 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB with addenda through Winter 1974 and,

b. RDT Standard E15-2NB, November 1974.

4.2.2.3.2.2 Internal Structure

Even though the existing ASME Code does not provide rules for the analysis of components operating at temperatures in excess of 800°F, those portions of the UIS located within the primary pressure boundary shall be analyzed as Class 1 components in accordance with the following:

For temperatures below 800°F the 1977 Edition of the ASME Boiler and Pressure Vessel Code, Section III, with addenda through Summer 1977, Subsections NA and NG shall be used.

For temperatures in excess of 800°F the following shall be used:

a. The 1977 Edition of the ASME Boiler and Pressure Vessel Code, Section III, with addenda through Summer 1977.

b. Code Cases 1592, 1593, and 1594.

c. RDT E15-2NB (Supplement to Section III).

d. RDT F9-4 (Supplement to Code Cases 1592, 1593, 1594, and 1596).

e. The Nuclear Systems Materials Handbook, TID 26666, "Inconel Alloy 718", Technical Bulletin T-39, International Nickel Company and the Alloy 718 design fatigue curve, Figure 4.2-48 shall be used.

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$$D = \sum \frac{n}{N_d} + \sum \frac{t}{T_D}$$

To meet criterion: $D_e \leq D$

(7) A cycle limiting criterion is required to verify the applicability of the modified rule. The effective number of allowable design cycles is:

$$n_e = n\left(\frac{D}{D_e}\right)$$

Where n is the total number of significant strain cycles between hold periods. Low amplitude high cycle strain fluctuations (such as normal power fluctuations) need not be considered in n if they are elastic excursions that result in negligible fatigue damage.

For the modified rule to be applicable, n_e shall not exceed 3000 for type 316 stainless steel nor 6000 for type 304 stainless steel.

Modification of Creep Damage Rules

In cases where a local stress concentration exists, the creep-fatigue damage evaluation may be modified as described herein.

(1) The material is austenitic stainless steel Type 304 or 316 solution treated.

(2) The structure does not require a Code Stamp under existing Code rules.

(3) Simplified or rigorous inelastic analysis is used.

(4) Stress rupture test data of the same type of stress concentration with similar geometric proportions tested at prototypic temperatures are used as a basis for modification of the Code Strength. The test temperature may be higher than the service temperature in order to more closely simulate the actual component lifetime and the stress level.

(5) The notched stress rupture data shall be from specimens which are comparably or more severely loaded than the component, i.e., membrane loading of a notched specimen should be more severe than a gradient loading.

(6) The stress rupture test data include data up to 1/60 of the component lifetime at prototypic temperatures or the equivalent when a short-time
49 high temperature combination is used to simulate the desired long-time service environment.

(7) Subject to the above limitations, the creep damage may be calculated in accordance with F9-4T and Code Case 1592 as modified. The modification is to use a peak stress to rupture design curve based upon the stress to rupture design curve in Code Case 1592 adjusted for the influence of a non-linear stress state caused by the presence of a geometric stress concentration as with the following:

Step 1 - Determine the smooth speciment stress rupture strength curve by tests of the same material at the temperature of interest.

Step 2 - Determine the stress rupture strength curve with the presence of the geometric stress concentrations under the same conditions in (1) with specimens of the same heat of material with the same histories. Analytically determine the peak stress relative to the net stress thus defining the stress rupture strength in terms of "peak stress" vs. time to rupture.

Step 3 - Ratio Code Case 1592 stress to rupture design curve by the ratio of Step 1 divided by Step 2. This must be done for at least 3 points in time with a separation in time of at least two orders of magnitude. In cases where the strength ratio varies with lifetime, the lesser of the value extrapolated to the component lifetime or the experimental value for the longest duration tests shall be used.

(8) The total creep-fatigue damage is determined by adding to the creep damage and fatigue damage calculated in accordance with T-1411, -1412, -1413, and -1414 of Code Case 1592.

(9) The allowable creep-fatigue damage (D) is determined from the lesser of the values from Figure T-1420-2 of Code Case 1592 (See Figure 4.2-47a) and an average of test values from creep-fatigue interaction tests of notched specimens.

(10) The greater of the damage using the modified rule and the damage using the stress unaltered by the stress concentrations and the Code Case 1592
49 stress to rupture design curve shall be used.

High Cycle Strain Controlled Fatigue Limits

For those 304 and 316 Stainless Steel components which are outside ASME Code jurisdiction, the fatigue damage for strain controlled cyclic deformations in excess of $1 \cdot 10^6$ cycles may be evaluated using allowable strain ranges obtained from Figure 4.2-47B, provided metal temperatures do not exceed 1100^{o}F. Fatigue life reduction factors must be applied indepen-
57 dently for slow strain rates and hold times, in accordance with ASME Code requirements.

Equivalent strain range shall be used to evaluate whether or not the allowable strain range limits have been met. The equivalent strain range value for entering the curve shall be calculated in accordance with the procedures specified in ASME Code Case 1592 except that one of the following formulations shall be used:

Formula 1. When the elastic and plastic components of the total strain range are not known, the equivalent strain range shall be calculated as:

$$\Delta\varepsilon_{equiv.} = \frac{\sqrt{2}}{2(1+\nu)}\left[\Delta(\varepsilon_1 - \varepsilon 2)^2 + \Delta(\varepsilon 2 - \varepsilon 3)^2 + \Delta(\varepsilon 3 - \varepsilon 1)^2\right]^{1/2}$$

where ν = Poisson's ratio for elastic strains.

Formula 2. When the elastic and plastic components of the total strain range are known, the equivalent strain range shall be calculated as the sum of equivalent elastic strain range.

$$\Delta\varepsilon^{e}_{equivalent} = \frac{\sqrt{2}}{2(1+\nu)}\left[\Delta(\varepsilon_1 - \varepsilon_2)^2 + \Delta(\varepsilon 2 - \varepsilon_3)^2 + \Delta(\varepsilon_3 - \varepsilon_1)^2\right]^{1/2}$$

and equivalent plastic strain range

$$\Delta\varepsilon^{p}_{equivalent} = \frac{\sqrt{2}}{3}\left[\Delta(\varepsilon_1 - \varepsilon 2)^2 + \Delta(\varepsilon 2 - \varepsilon_3)^2 + \Delta(\varepsilon_3 - \varepsilon_1)^2\right]^{1/2}$$

where ν = Poisson's ratio for elastic strains.

Formula 3. The following formula is included as an alternative to formula 2 as it represents the method of calculating total equivalent strain range employed in some computer routines, e.g. ANSYS. The total equivalent strain range is calculated as:

$$\Delta\varepsilon\ equiv. = \frac{\sqrt{2}}{2(1+\nu_g)}\left[\Delta(\varepsilon_1 - \varepsilon_2)^2 + \Delta(\varepsilon_2 - \varepsilon_3)^2 + \Delta(\varepsilon_3 - \varepsilon_1)^2\right]^{1/2}$$

where ν_g is a generalized Poisson's ratio found as:

$$\nu_g = 0.5 - (0.5-\nu)\ (E_S/E)$$

where ν = Poisson's ratio for elastic strains
E_s = The material secant modulus prior to the last plastic strain increment
57 E = The material elastic modulus

4.2.2.3.3 Additional Material Properties

4.2.2.3.3.1 Inconel 718 Fatigue Properties

The Alloy 718 design fatigue curve, Figure 4.2-48 proposed for inclusion in the NSM Handbook as interim data, shall be used until superceded. The effects of the fabrication processes and service environment
41 on the structural integrity of the UIS shall be considered. The effect

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27. K. Q. Bagley, E. Edmonds, H. J. Powell, and others, "Fuel Element Behavior Under Irradiation in DFR", in Fuel and Fuel Elements for Fast Reactors, Vol.1, pp. 87-100, International Atomic Energy Agency, Vienna, 1974.

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29. "Oxide Fuel Element Development Quarterly Progress Report for Period Ending June 30, 1974", WARD-OX-3045-14, September, 1974. (Availability: USAEC Technical Information Center).

29a. HEDL-TME-73-46, "Flow Induced Vibration Tests of a Prototypic FFTF Control Rod Absorber Pin Bundle Assembly", June 1973. 16

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51 * References annotated with an asterisk support conclusions in the Section. Other references are provided as background information.

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TABLE 4.2-52

REFERENCES SUPPORTING CRBRP FUEL ROD LOADING
(See Response to Question 241.68)

Loading Category	Exemplary References: FFTF Design	EBR-II Irradiation	U.S. LMFBR Program	Foreign LMFBR Program
1. Fuel-Cladding Differential Expansion	59	62	64, 65	67, 68
2. Fission Gas Released from Fuel	59	61, 63	64, 65	67, 68
3. Differential Thermal & Irradiation Induced Expansion	59	---	64	67, 68
4. Support System Interaction				
A. Fuel Rod-Spacer	59	63	64	67
B. Bundle-Duct	59	63	64	---
5. Flow Induced Vibration	59	---	66	67
6. Accident Loading				
A. Fission Gas Ejection	60	---	65	69
B. Fuel Coolant Interaction	60	---	---	69

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TABLE 4.2-53

REACTOR INTERNALS IRRADIATION ENVIRONMENT & DESIGN CRITERIA

CRBRP Structural Component	Sub-Component	Material	Irradiation Environment: Irrad. Temp. Range(a) Lower Bound (^oF)	Irrad. Temp. Range(a) Upper Bound (^oF)	Maximum(b) Total Fluence (n/cm^2)	Average Neutron Energy (Mev)	Design Criteria: Minimum Residual Total Elongation (%)	Energy(c) Dependent Fluence Limit (n/cm^2)
Core Former Structure	Support Ring	SS304	700	950	6.3×10^{21}	0.03	10	4.1×10^{22}
	Support Ring Weld	SS308	700	950	6.3×10^{21}	0.03	10	9.4×10^{21}
	Lower Former Ring	SS316	700	900	1.6×10^{22}	0.08	10	3.5×10^{22}
Fixed Radial Shield	Segment at Core Midplane	SS316	700	900	3.6×10^{22}	0.08	10	3.6×10^{22}
	Lower Retention Pin	INC718	700	700	3.6×10^{20}	0.04	10	5.0×10^{21}
Lower Inlet Module	Holddown Ring	SS304	700	800	5.4×10^{21}	0.06	10	2.3×10^{22}
	Receptacle	INC718	700	800	4.1×10^{21}	0.05	10	5.0×10^{21}
	Body/Stem Weld	SS308	700	800	2.5×10^{19}	0.03	10	1.0×10^{22}
Bypass Flow Module	Top Forging	SS304	700	800	1.2×10^{21}	0.04	10	3.3×10^{22}
	Locking Collar	INC718	700	800	1.5×10^{21}	0.04	10	5.0×10^{21}
	Seismic Lug Weld	SS308	700	800	5.4×10^{20}	0.04	10	1.0×10^{21}

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57 | 54

FIGURE 4.2-43 DELETED

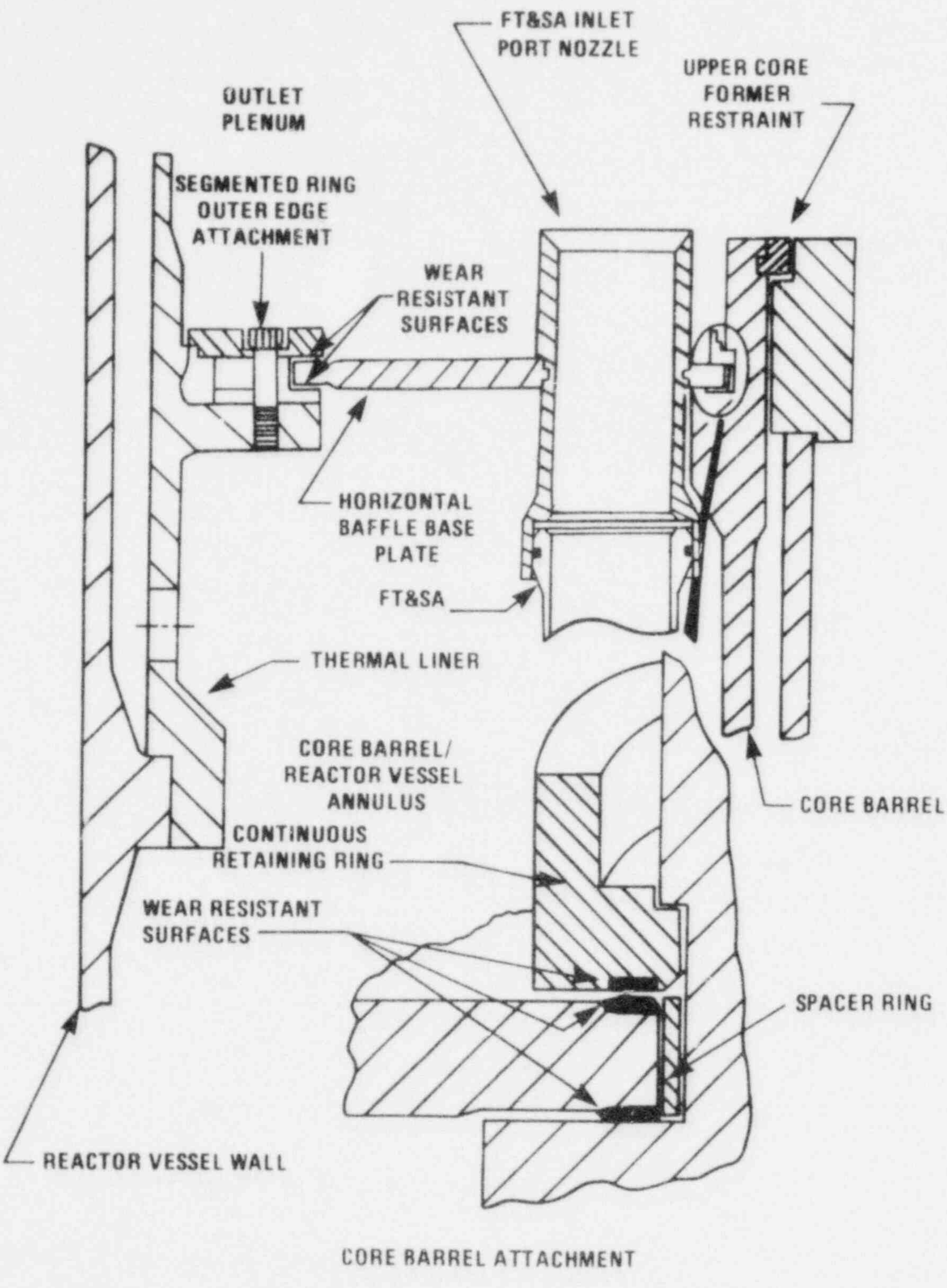


Figure 4.2-44. Horizontal Baffle

3152-1

Amend. 57
Nov. 1980

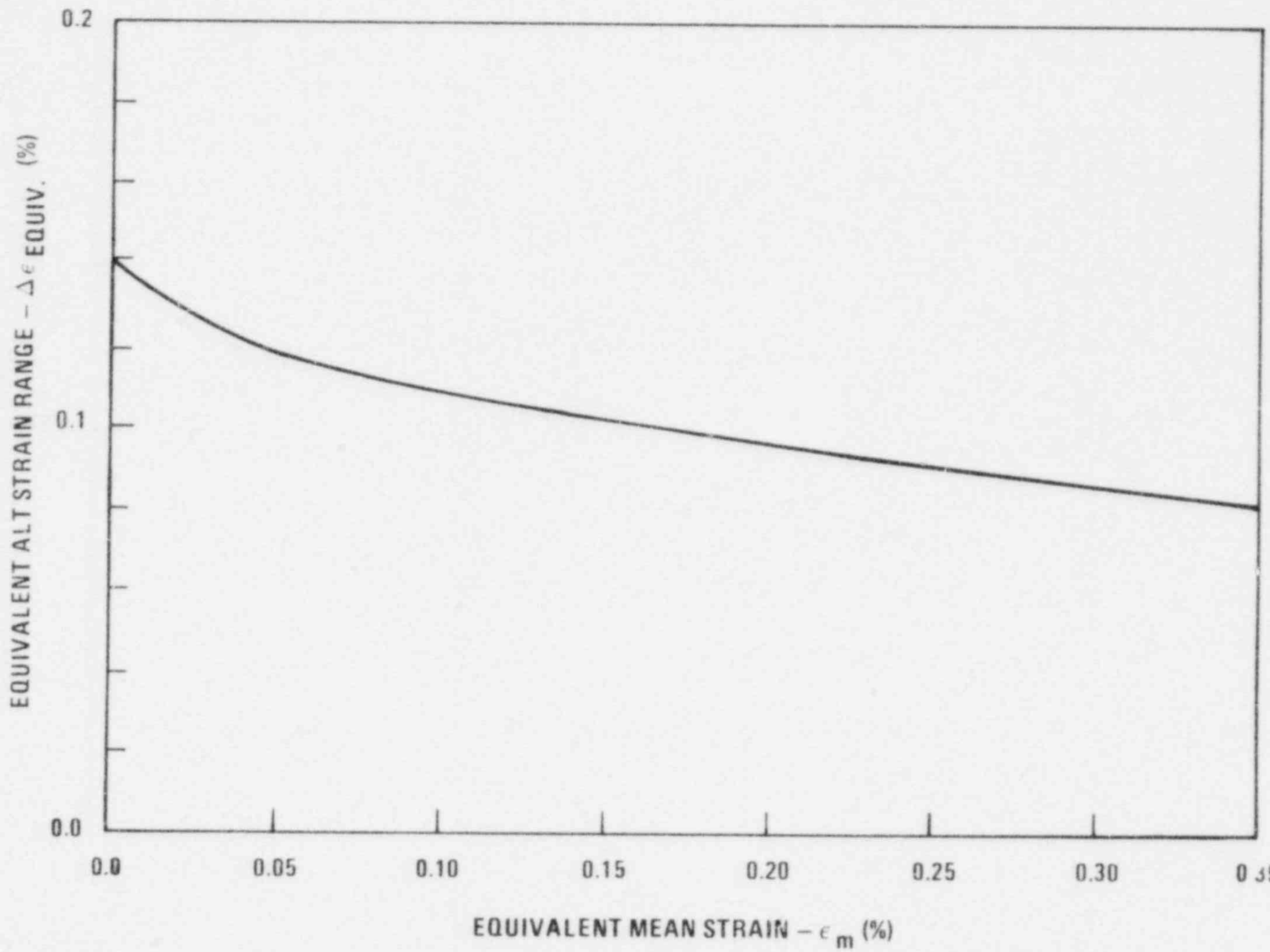


Figure 4.2-47B Allowable Alternating Strain Range Versus Mean Strain for T ≤ 1100°F

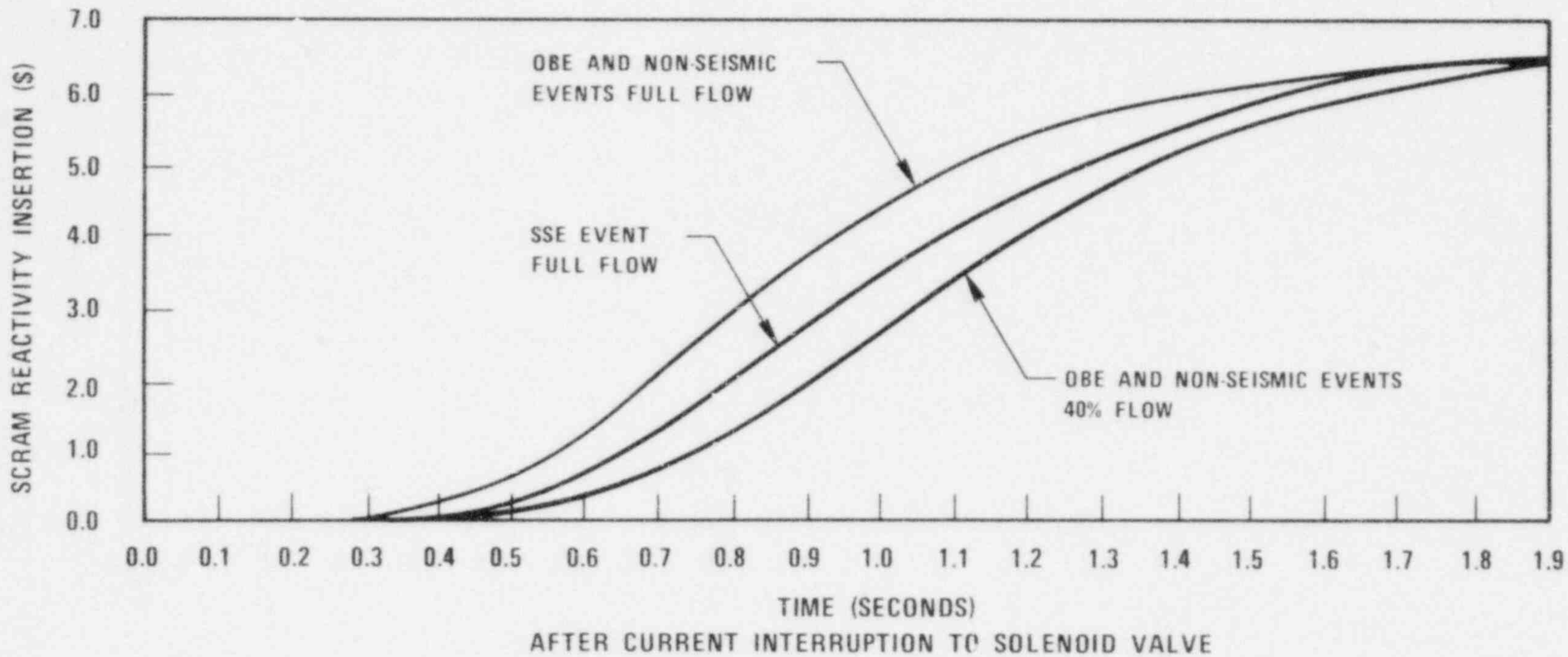


Figure 4.2-94 Secondary Control Rod System Minimum SCRAM Insertion Requirements

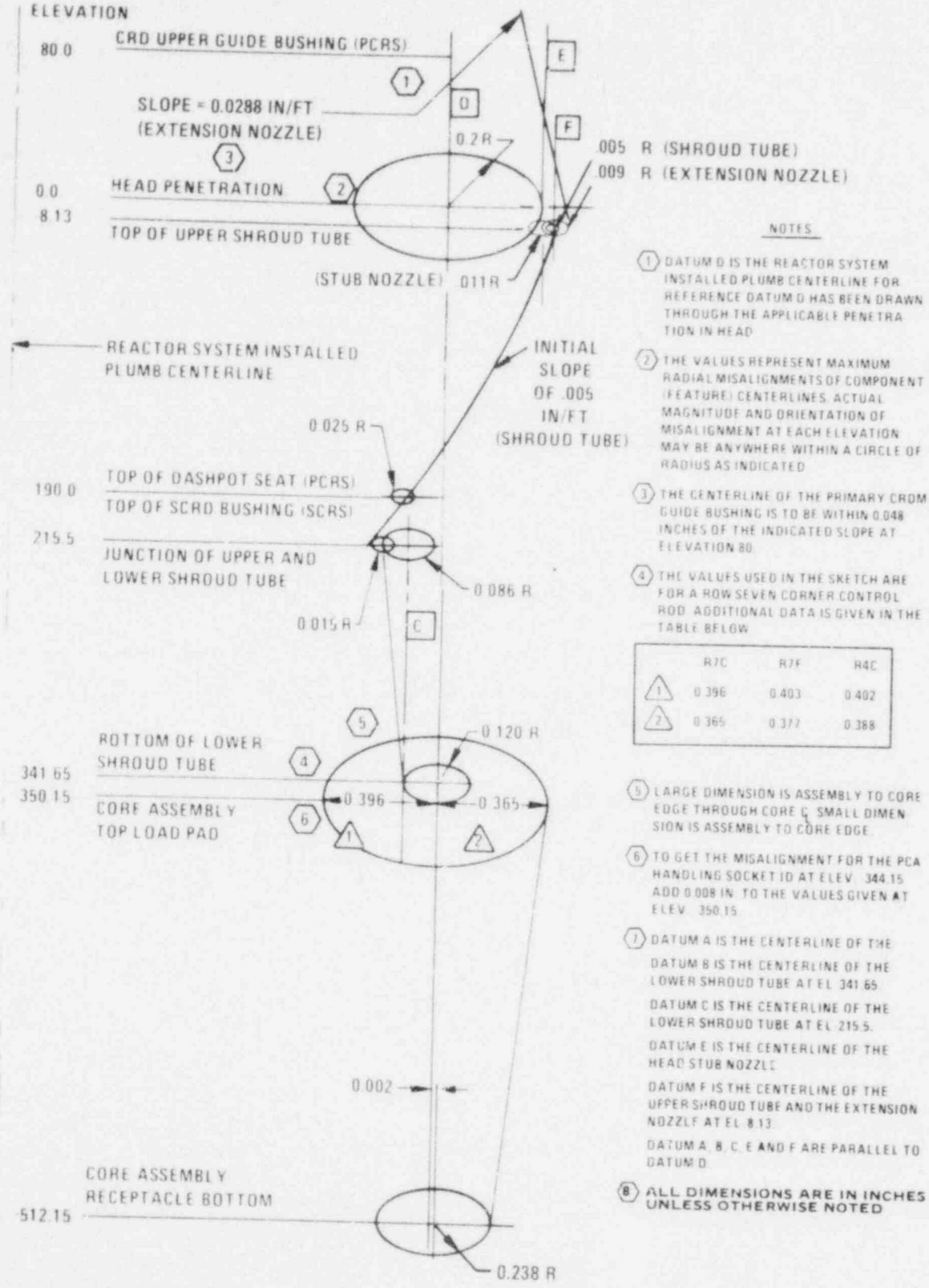
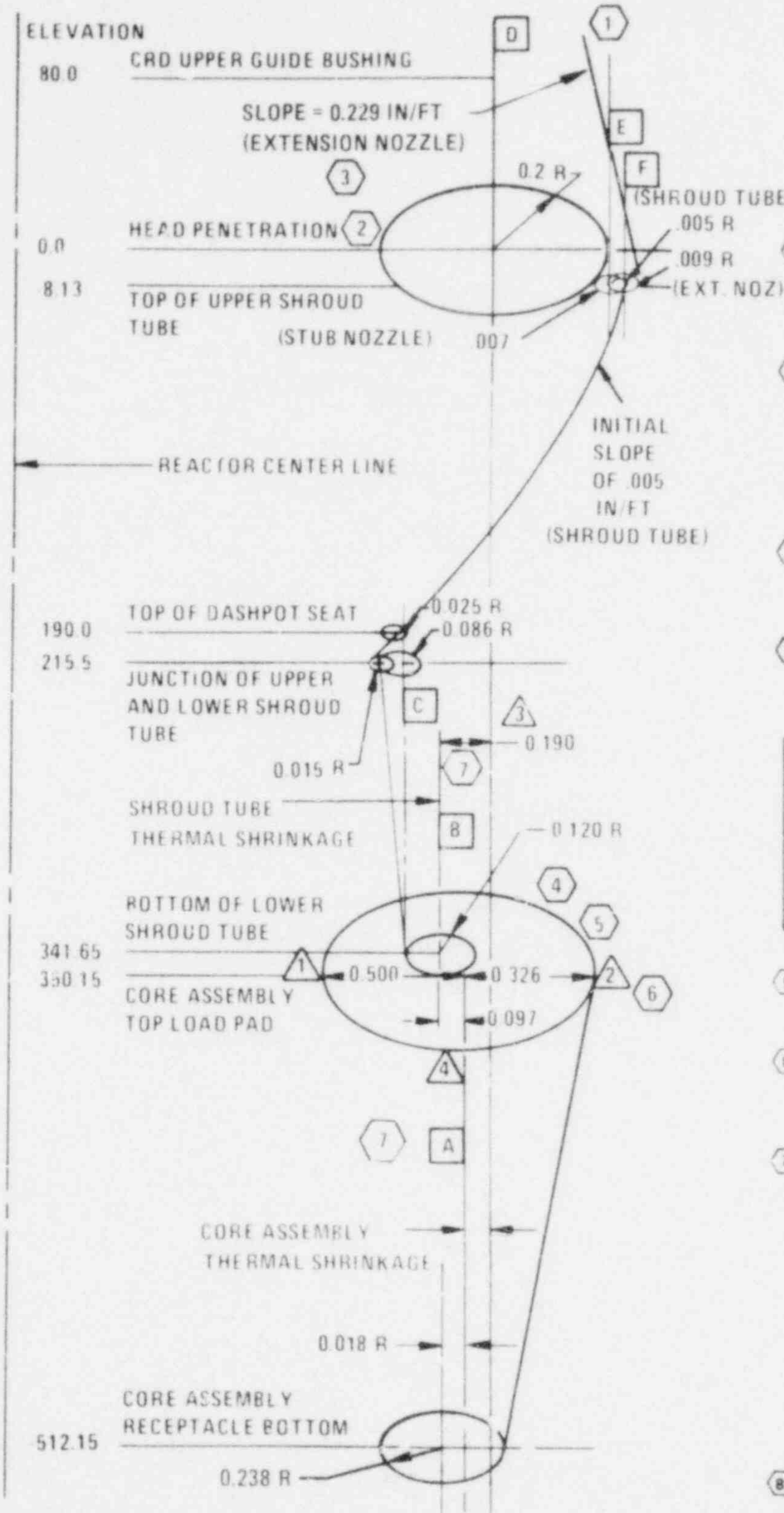


Figure 4.2-95A Control Rod System Operating Misalignment Envelope

3975-4a



NOTES

(1) DATUM D 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	0.502	0.466
2	0.326	0.334	0.381
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

(8) ALL DIMENSIONS ARE IN INCHES UNLESS OTHEPWISE NOTED

Figure 4.2-95B Control Rod System Refueling Misalignment Envelope

3975-5a

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

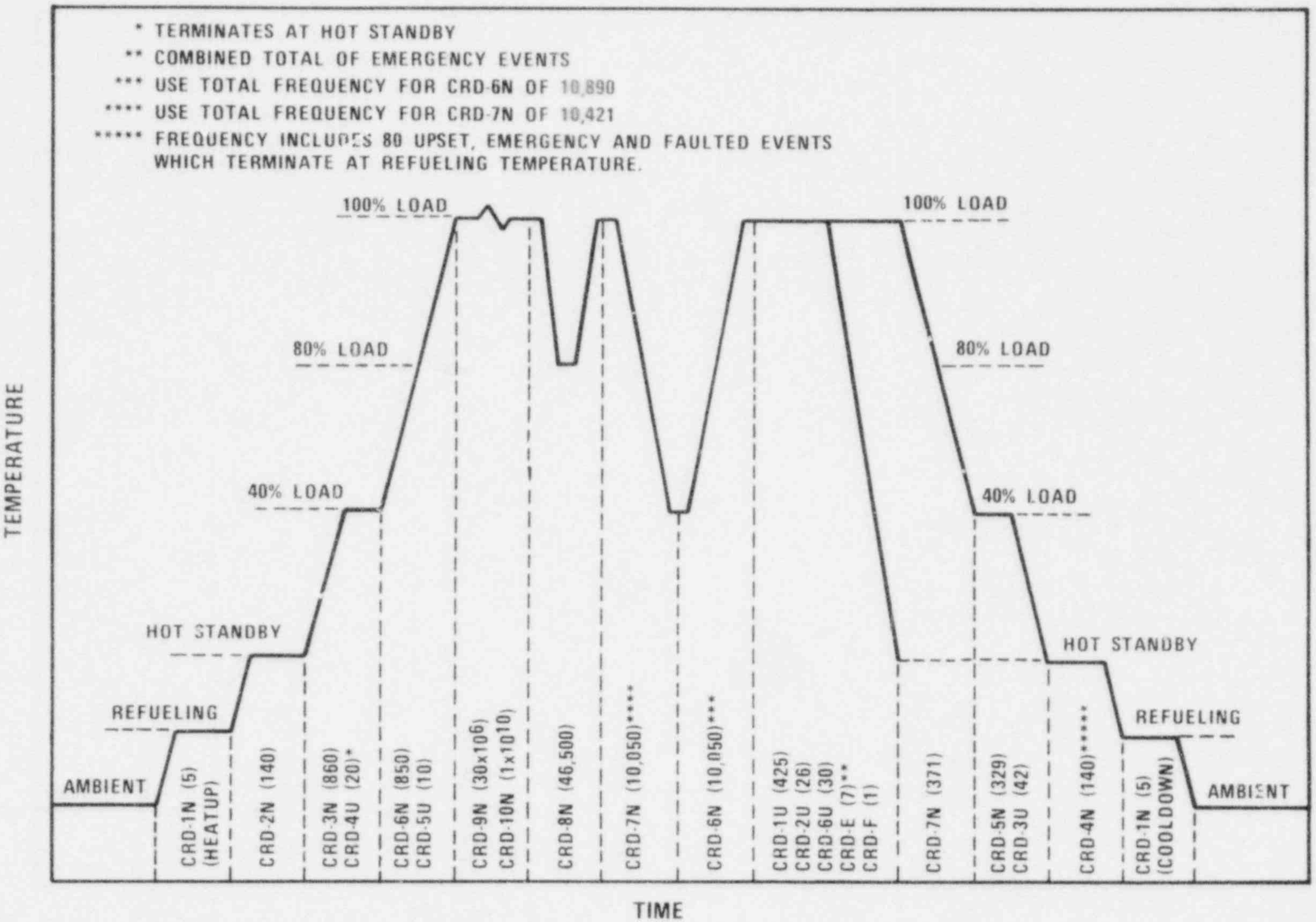


Figure 4.2-96 Control Rod Driveline 30 Year Histogram

Bypass Flow

Figure 4.4-7 shows the bypass orificing which regulates the bypass flow that is used for radial shield and vessel thermal liner cooling. This bypass orificing is located in the 24 peripheral inlet modules. In 18 of the modules, the bypass orificing feeds a controlled portion of the flow downward into the low pressure plenum below the module where it reverses direction and flows upward and is then directed radially outward through the core support structure low pressure manifold to the reactor vessel/core barrel annulus. In the other 6 modules, the flow is directed upward to the bypass flow module where it is distributed to those radial shield assemblies (264) outside the inlet module region of the core.

Radial Shielding

The radial shielding is made up of fixed shielding attached to the core barrel and removable shielding supported by the core support structure as described in Section 4.2. The removable shielding is cooled by external flow through the inter-assembly gaps as well as internal flow through the assemblies. The orifices in the assembly are sized to provide adequate cooling to meet assembly lifetime requirements. External cooling alone was found to be adequate for the fixed shielding.

Vessel Cooling Flow

The flow from the annulus between the reactor vessel and the core barrel passes upward into an annulus formed by the reactor vessel and the vessel thermal liner. A horizontal baffle is installed between the liner and the core barrel; it minimizes leakage from the liner-barrel annulus into the outlet plenum. From the vessel-liner annulus the coolant discharges into the outlet plenum region above the suppressor plate. Two percent of total flow enters the vessel-liner annulus; a fraction of this flow appears as leakage at the outlet nozzle and the
57 makeup nozzle penetrations in the thermal liner.

Leakage

Seals between the core support structure and core inlet module liner, and other mechanical interfacing locations are sources of leakage. This leakage is estimated to be 1.05% of total flow and is assumed to
51 pass upward through the core interstitial and peripheral region without contributing to the cooling of any reactor component.

Outlet Plenum

All fuel, blanket, control, and a portion of the radial shield assembly flow discharges into the upper internals structure. The coolant first enters a mixing chamber before entering the chimneys(Figure 4.4-8). The chimneys duct the flow vertically upward and discharge the flow into the upper region of the vessel outlet plenum. The flow is directed into the upper region of the plenum to minimize flow stratification in this region during a reactor trip transient.

The flow from some of the removable radial shields which are located outside of the peripheral skirt of the upper internal structure discharge directly into the outlet plenum. Also, ~14% of total reactor flow from the fuel, blanket, control and radial shield assemblies bypasses the chimneys through the gap between the top of the core assemblies and the skirt of the upper internals structure and discharges directly into the outlet plenum.

The coolant leaves the reactor vessel outlet plenum through three 36-inch diameter outlet nozzles.

4.4.2.5 Fuel and Blanket Assemblies Orificing

4.4.2.5.1 Orificing Philosophy, Approach and Constraints

Core orificing, i.e., flow allocation to the various fuel and blanket assemblies is an important step in the core thermal-hydraulic design. Since the assembly temperatures are directly dependent on the amount of flow and since the flow allocation is the only thermal-hydraulic design parameter which can be varied, within certain limits, by the designer, it logically follows that the core T&H design and performance is only as "good" as the core orificing. Therefore, much attention in the CRBRP core T&H design has been placed on core orificing.

Previous experience has indicated that a successful orificing should account "a priori" for all the various aspects to be considered through the design, in order to avoid time consuming and costly iterations when the analyses are well in progress. Thus, a systematic orificing approach was developed, which accounted for lifetime/burnup, transient, upper internals temperature constraints. This new approach represented a change in philosophy and a significant improvement over the previous maximum temperature equalization method. Characteristic features of this approach are determination of the limiting temperatures (see Section 4.4.2.5.2) for all types of assemblies and simultaneous orificing of the fuel and blanket assemblies. Finally, both first and second core conditions were investigated in determining the orificing constraints and the most restrictive in either core was used in deriving the orificing configuration. This guaranteed, a priori, that the thermal-
51 hydraulic performance would satisfy the constraints considered in both cores.

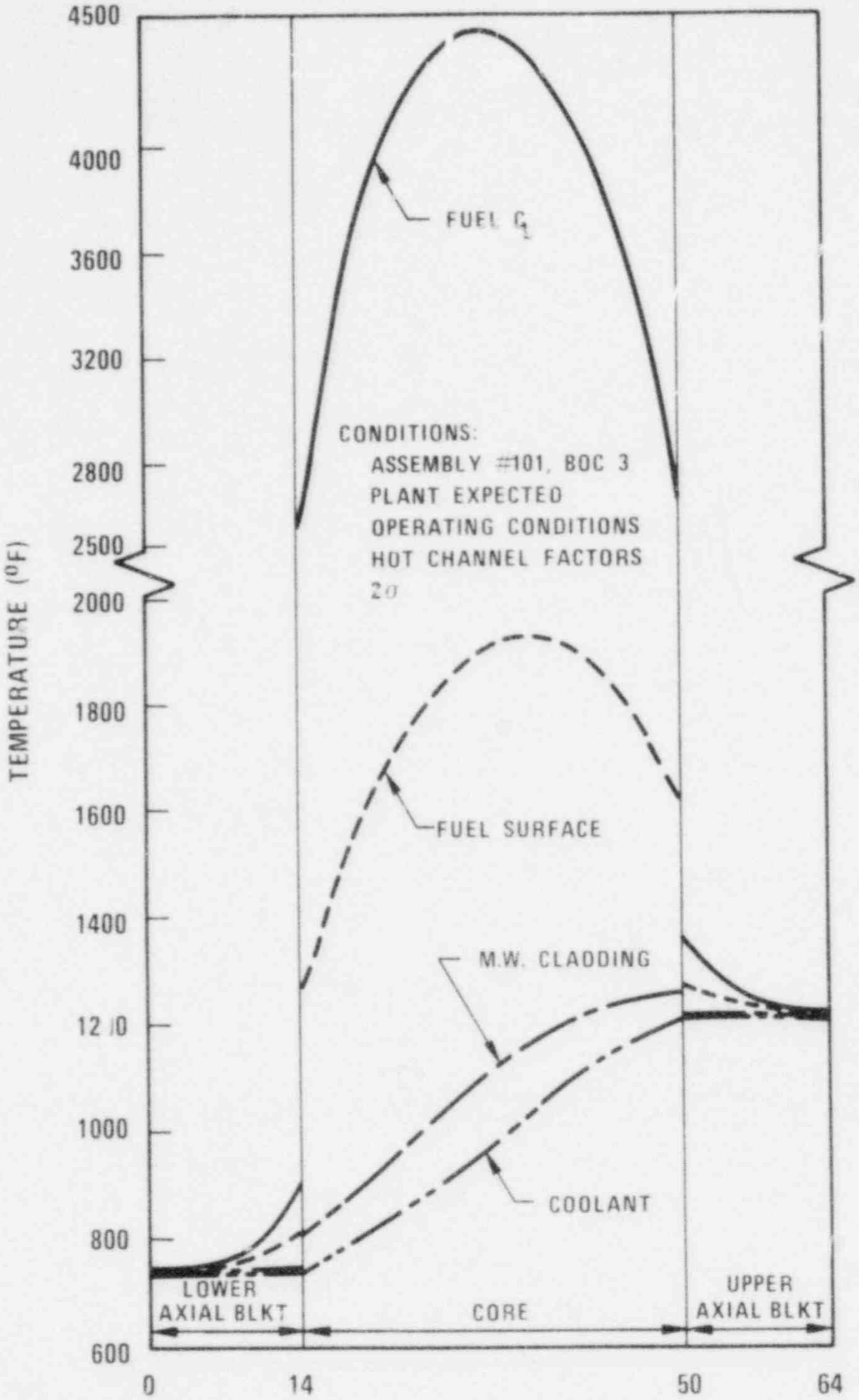
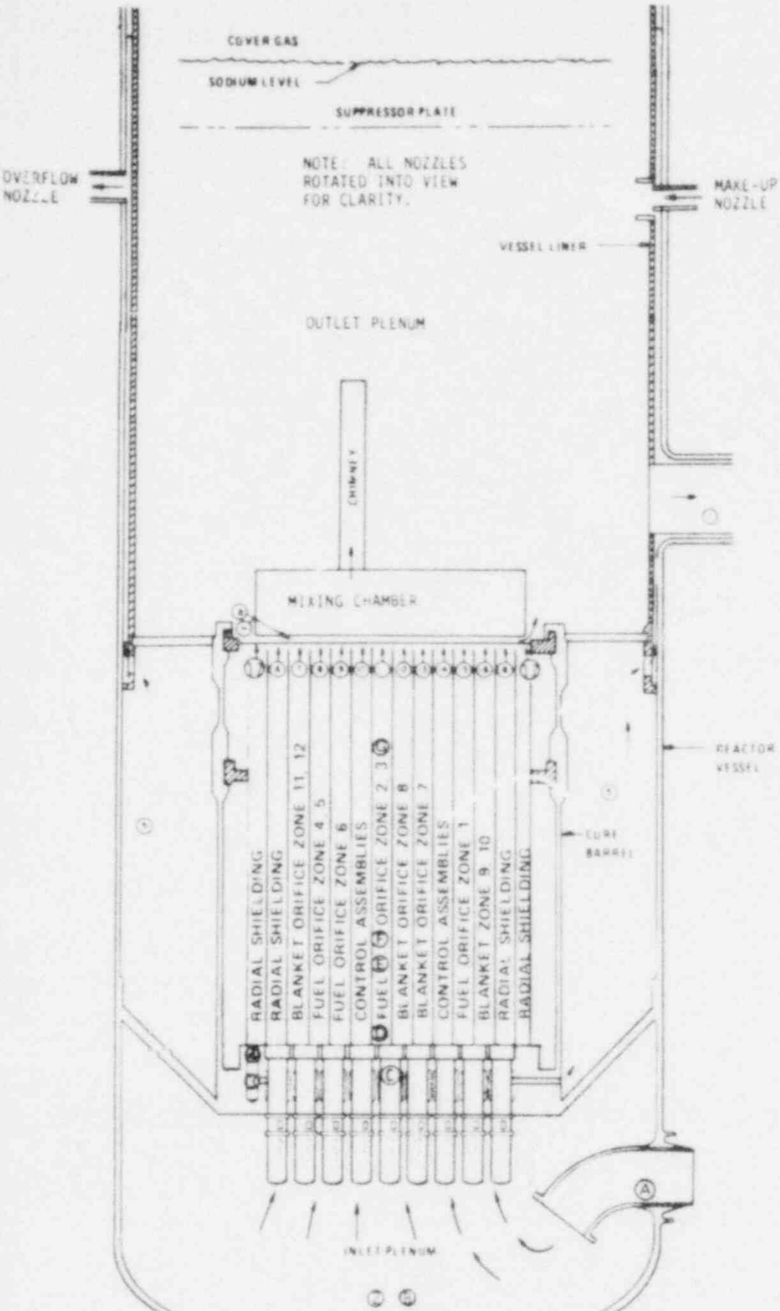


Figure 4.4-1 Typical Axial Temperature Profile in CRBRP Fuel Rod

1668-69



CRBR FLOW ALLOCATION
HETEROGENOUS CORE - CYCLE 4
REACTOR ΔT - 265°F TOTAL FLOW - 41.446 x 10^6 LB/HR

COMPONENT		LOCATION	NUMBER OF ASSEMBLIES	FLOW LB/HR x 10^6	FLOW % TOTAL
1. FUEL ZONE	1	2-15	39	7.295	17.60
	2	2-11	54	9.388	22.68
	3	2-11	21	3.451	8.33
	4	2-9	18	2.718	6.56
	5	2-8	24	3.532	8.52
	6	2-9	6	1.063	2.56
2. TOTAL FUEL			162	27.458	66.25
3. INNER BLANKET	7	2-13	57	4.983	12.02
	8	2-12	19	1.459	3.52
4. TOTAL INNER BLANKET			76	6.442	15.54
5. OUTER BLANKET	9	2-16	12	0.742	1.79
	10	2-16	36	1.726	4.17
	11	2-7	48	1.672	4.03
	12	2-7	36	0.920	2.22
6. TOTAL OUTER BLANKET			132	5.060	12.21
7. RADIAL SHIELD ASSEMBLIES					
a) INNER		2-6	42	0.120	0.29
b) OUTER		2-17	264	0.435	1.05
8. TOTAL RADIAL SHIELD			306	0.555	1.34
9. CONTROL ASSEMBLIES					
a) PRIMARY		2-10	4	0.424	1.02
b) SECONDARY UPFLOW		2-14	5	0.101	0.24
DOWNFLOW		2-3		0.328	0.79
10. TOTAL CONTROL ASSEMBLIES (UPFLOW)			15	0.525	1.26
11. BYPASS & LEAKAGE					
a) AROUND CORE		2-5		0.933	2.25
b) THROUGH CORE				0.872	1.85
12. TOTAL BYPASS & LEAKAGE				1.805	3.40
13. REACTOR VESSEL THERMAL LINER				0.829*	2.00*
14. TOTAL				41.446	100.0

*DO NOT INCLUDE IN FLOW TOTAL (INCLUDED IN ITEM 11)

CRBRP REACTOR COMPONENT
LOADS AT T&H DESIGN CONDITIONS

COMPONENT	LOCATION	HYDRAULIC LOAD LB
INLET NOZZLE	A-B	2623
CORE SUPPORT STRUCTURE	B-D	7.37 x 10^6
HORIZONTAL BAFFLE PLATE	S-H	32127
INLET MODULE	B-C	810
INLET NOZZLE	C-D	1831
SHIELD AND ORIFICE	D-E	555
ROD BUNDLE INLET	E-F	20
ROD BUNDLE	F-G	820
ROD BUNDLE EXIT	G-16	58
OUTLET NOZZLE	H-I	6650

*THE MAXIMUM HYDRAULIC LOAD OCCURS DURING STEADY STATE OPERATION. LOADS DURING UPSET, EMERGENCY AND FAULTED EVENTS ARE LESS SEVERE

FIGURE 4.4-2 CRBRP Flow and Hydraulic Loads
1668-71

Figure 4.4-2 CRBRP Flows and Hydraulic Loads

1668-71a

Amend. 57
Nov. 1980

POOR ORIGINAL

Riser Elastomer Seals

The balance of the seals on the riser assembly operate at temperatures
41 below 125°F.

Upper Internals Structure Jacking Mechanism

The UIS jacking mechanism utilizes metal buffered seals in the 400°F areas. These seals are part of the mechanical assemblies. The seals will be removed with components at the appropriate maintenance period. Elastomer
41 seals are located in the cooler regions, have a service life of five years,
57 and will be replaced using hands-on maintenance, or with special tools.

Liquid Level Monitor Ports Plugs

Four of these components, operating at 400°F, are located on the reactor vessel head and provide receptacles for holding the liquid level monitors. Three small port plugs are attached to the top surfaces of the closure head rotating plugs by partial penetration welds, two on the intermediate and one on the large rotating plug. One large port plug is bolted to the top surface of the large rotating plug and is sealed to the plug by double metal "O" rings. The seals remain attached to the port plug during installation and removal. An inerted cask will be used to install and remove the liquid 25
level monitor while at 400°F, requiring no hands-on operation. Because the port plug remains stationary relative to the head assembly, the metal "O" rings
57 beneath the plug flange are not expected to require maintenance.

5.2.1.4 Guard Vessel

The guard vessel provides for the retention of the primary sodium coolant in the event of a leak in the portion of the primary coolant boundary which it surrounds. The guard vessel geometry assures reactor vessel outlet nozzle submergence after such a leak which will maintain continuity in operating primary coolant loops to provide core cooling. The guard vessel also provides a uniform annulus for in-service inspection of the reactor vessel, with clearances that preclude contact with the reactor vessel and piping under accident conditions. Insulation for the reactor vessel and a heating system for the reactor vessel to be used prior to sodium fill and during prolonged shutdown are also mounted upon the guard vessel.

The maximum and minimum widths of the radial gap between the guard vessel and the reactor vessel have been conservatively calculated, taking into account all relevent factors such as tolerances on the diameters of the two vessels, permissible out-of-roundness of the two vessels, possible deviations from straightness due to manufacture and subsequent operation, thermal expansion, initial deviations in the alignment of the two vessels, etc. The transporter for the television camera will be designed to accommodate itself to this maximum possible range of gaps as it moves in the space between the two vessels. 25

5.2.1.5 Reactor Vessel Preheat

The Reactor Vessel Preheat System will control the dry heat-up and cool down of the Guard Vessel, Reactor Vessel and Internals between ambient (70°F) and 400°F and if required will provide make-up heat for that lost to the Reactor Cavity during prolonged shutdowns.

The heat will be provided by tubular electrical heaters mounted between the Guard Vessel and insulation. These heaters will be arranged circumferentially around the Guard Vessel and will be grouped and controlled in zones of uniform heat output. Temperature sensing devices will monitor the Guard Vessel temperature in each of these zones and provide the necessary feedback for power level adjustments in the heaters.

The heaters will be mounted to the same framework which supports the Guard Vessel insulation. Ceramic offsets will be used to offset the framework and heaters from the Guard Vessel surface. The heaters and framework will therefore be electrically isolated from the Guard Vessel. Convective barriers, reflective sheaths and the Guard Vessel insulation will be used to optimize heat input to the Guard Vessel and minimize losses to the Reactor Cavity.

Preliminary preheat, startup, and shutdown analyses have been performed on the Reactor Vessel and Guard Vessel to determine the temperature differences which will result in opening and/or closure of the annular gap between the two vessels. By necessity the preheat analysis is very preliminary since no firm preheat procedure has yet been developed. Figures 5.2-4 through 5.2-6 show the temperature differences between the Reactor Vessel and Guard Vessel in the inlet and outlet plenum regions for the three transients in question. As shown the largest positive temperature difference between the Reactor Vessel and the Guard Vessel occurs in the outlet plenum region during startup (335°F) while the largest negative temperature difference occurs in the outlet plenum region during shutdown (-214°F). The nominal radial gap between the reactor vessel and guard vessel is 8 inches at assembly and at the end of preheat. This gap decreases to approximately 7.6 inches minimum during start-up and increases to approximately 8.3 inches maximum during shutdown. During preheat the gap also increases but to a lesser value than during shutdown due to the smaller maximum temperature difference.

Variations in the axial gap between the bottom of the reactor vessel and the inner surface of the guard vessel are noted between the states shown in the table. Thus the largest axial gap is 11.0 inches at the dry cold condition and the smallest gap is 6.2 inches at the end of the heating phase of preheat. 9

5.2.2 Design Parameters

56 Overall schematic views of the reactor vessel, closure head assembly, inlet and outlet piping, and guard vessel are shown in Figures 5.2-1, 1A and 1B. The top view is given in Figure 5.2-2.

41 17 load transmitted to the support ledge is reduced due to the large mass
presented by the spring-coupled closure head/reactor vessel system compared
to the mass of the postulated sodium slug.

36 5.2.2.2 <u>Closure Head</u>

The closure head consists of three rotating plugs which will be
constructed of SA 508 Class 2 steel. Each plug contains a major penetration
eccentric to its outside diameter. These rotating plugs are interconnected
17 by means of a series of plug risers. Sealing between the plugs is accomplished
by sodium dip seals and double inflatable seals of elastomer material. At
36 its top, the large rotating plug has an outer diameter of 257.38 in., and an
inner diameter of 176.50 in. The large rotating plug provides access to the
vessel interior for the ex-vessel transfer machine and the core coolant
liquid level monitors. The intermediate rotating plug (175.50 in. O.D. and
68.94 in. I.D.) provides access to the vessel interior for the control rod
41 drivelines, upper intervals support columns, and the liquid level monitors.
The small rotating plug (67.94 in. O.D.) provides access to the vessel interior
for the In-Vessel Transfer Machine. The thickness of each rotating plug is
41 22.0 in. Rotation of the plugs will be accomplished by a gearing and bearing
system attached to the plug risers. The nozzles for each penetration will
41 be constructed of an austenitic stainless steel.

Each rotating plug is provided with a system of mechanical locks and electrical interlocks which prevent plug rotation during reactor operation and refueling when plug rotation is not desired.

The mechanical locks include the following:

a. Each plug includes a separate positive lock to assure that the plug cannot be moved, and will not drift from its normal operating position during reactor operation. This lock will be installed to prevent relative rotation between each bull gear and its outer riser whenever the control rod drivelines are connected. The locks shall be manually installed at the end of each refueling cycle, and will be removed only during the refueling period when plug rotation is necessary.

b. The plug drives are designed to be self locking to react to any seismic torque occurring during refueling, which could rotate the plugs and thus damage a fuel or blanket assembly during removal from the core.

The electrical interlocks include the following:

a. During reactor operation, the plug drive and control system keyswitch is in the OFF position, the control system is deenergized, and there is no power to the plug drive motors.

24

b. Electrical interlocks are provided to prevent the plugs from being inadvertently rotated by their drive system unless the upper internals are raised and locked, and the IVTM and EVTM are in a safe condition.

56 24 c. An electrical interlock is also provided to prevent vertical operation of the IVTM or operation of the EVTM over the HAA during operation of the plug drive system.

Each rotating plug has attendant thermal and radiological shielding extended to a depth of 74.65 in. beneath the top of each plug forging. The shielding is composed of a series of plates fabricated from carbon steel,
17 and stainless steel. The cover gas between each set of plates attenuates thermal conduction and thereby acts to decrease the heat flux imparted to the rotating plug. A heating and cooling system is provided to maintain the closure head at 400°F (nominal) as well as providing heating and cool-
17 ing for other small head mounted subassemblies.

A gas entrainment suppressor plate assembly is positioned beneath
45 17 the head thermal and radiological shielding at a depth of 122.65 in. beneath the top of each rotating plug. It protects the head shielding from being contacted by the core coolant and minimizes the amount of cover gas entrained in the core coolant. The assembly is designed to accommodate all normal,
17 upset, emergency, and faulted conditions.

56 25

42 In plan view, the subassembly consists of 33 plates at the same elevation with horizontal gaps between them. (Fig. 5.2-3) These plates have penetrations in line with the head penetrations to allow the passage of the head mounted components into the outlet plenum. Each plate is supported by means of a central support column affixed to the lower shield plate. These central columns, when possible, consist of tubes which surround closure head penetrations. Support columns which do not surround penetrating equipment will be capped to minimize the amount of cover gas entrained. The support columns will be inserted through oversized penetrations in the lower shield plate, accurately positioned and then attached to the top surface of the lower plate by means of bolting. The support columns will be attached to the suppressor plate by means of welding. This attachment weld is located above the region of the suppressor plate where high thermal gradients occur by using a plate with an extruded weld neck. The top end of the support column, which protrudes through the lower shield plate is composed of 2¼ Cr-1Mo. material to minimize the differential expansion with the carbon steel shield plate. The lower, in sodium, portion is austenitic stainless
57 42 steel. The use of a single support provides adequate support while lessening 2
the thermal stresses by permitting the plates to flex freely under the expected thermal gradient.

The riser has been designed to maintain a maximum temperature of $125^{o}F$ in the region of the elastomer seals. Thermal analysis has been completed for this design which shows that this temperature ($125^{o}F$) is maintained by natural circulation cooling of the riser structure. 25

5.2.2.3 Guard Vessel

The guard vessel is a bottom-supported, right circular cylindrical vessel surrounding the reactor vessel. It will be fabricated from SA240 Type 304 stainless steel. The purpose of the guard vessel is to assure outlet nozzle submergence in the event of a leak in the 2

Figure 5.2-8 gives enlarged views of the elastomer seal areas
to show the sealing arrangement. The upper view shows typical seals for
the top end of the riser. The inflatable dynamics seals are mounted
into a removable seal retainer ring, which is sealed to the riser top
by elastomer o-ring seals. The space enclosed by the two inflatable 20
seals is a static buffer.

The CRBRP inflatable seals are the same in cross-section and
made of the same material (nylon-reinforced nitrile rubber) as the FFTF-
IVHM inflatable seals. Extensive testing was conducted to qualify the
IVHM inflatable seal at an operating temperature of 150°F (Reference
41 5.2-6). Additional performance testing has been conducted (Reference
5.2-7) on an FFTF-IVTM type inflatable seal to assure these seals will
function reliably under the CRBRP speed and pressure operating conditions.
This testing was also conducted at 150°F.

Specific objectives of the CRBRP program included buffer
cavity leakage and seal drag measurements under combinations of seal and
buffer cavity pressure, measurement of gas diffusion through the seal,
breakaway seal drag as a function of dwell times, effect of lubrication and
long duration (life) cycling on the wear characteristics of the seal, and
the effect of horizontal and vertical runout of the seal runner on seal
performance. Testing to support the present design adequacy under dynamic
operation is nearing completion. The verification of elastomer life at
125°F is documented in Reference 5.2-8. The life of the elastomers for
upward of 5 years in this temperature and environment has been demonstrated 25
by tests.

The seal retainer ring is sealed to the inner riser by two solid
elastomer o-ring seals. These have an inerted, static buffer space between
them which acts similar to the one described above. The seal follower is
sealed to the outer riser in the same way, but with the space between seals 20
49 purged with a small argon flow instead of static buffering.

The primary source of experimental data for CRBRP seal design is
the on-going Cover Gas Seal Development Program being performed by Atomics
International (AI). Testing includes static, rotary, and reciprocating
seal leakage tests, compression set tests, gas permeability measurements,
and seal lubricant evaluations including elastomer compatibility, thermal
stability and friction testing. Elastomers tested include silicone,
ethylene-propylene, urethane, nitrile (Buna N) and butyl rubber supplied
by three different seal verdors. Test temperatures ranged from 100 to
300°F depending upon the elastomer being tested. In addition to the
numerous quarterly reports which have been published, two summary reports
41 have been issued (References 5.2-8 and 5.2-9). The information contained
in these report amply demonstrate the ability of several elastomers to
meet CRBRP design requirements at 125°F. These tests demonstrate that 25
elastomer life of upward of 5 years in this temperature and environment
is achievable.

The closure head has several penetrations that permit access to the reactor interior. Penetrations will be sealed in one of two ways. One is to provide two seals with a pressurized buffer space between them. In the event of a failure of the inner seal, that between the reactor interior and the buffer space, buffer gas will leak into the vessel. In the event of a failure in the outer seal, buffer gas will leak into the head access area.

In each case, the leak will be detected and repaired before a leak can occur between the reactor interior and the head access area. The other method is to use a hermetic seal.

Figure 5.2-8 shows the method of sealing the riser bases to
the vessel head. Soft coated metallic o-rings with an inerted, purged
space between them are used in the same way as the elastomers at the 20
49 riser top. A continuous metal "C" ring or canopy seal is welded at both
ends completely around the periphery to provide a hermetic seal at the base
36 of the large outer riser. The base of the small and intermediate outer
risers are welded directly to the closure head and therefore have no leak
57 path at the juncture.

General approach to seal selection is:

1. For openings which are at high temperature and/or require long life, metallic seals, hermetic or double buffered, are used.

2. Seals which operate at low temperature and can be replaced relatively frequently are elastomer sealed.

1

Inspection of materials for the primary pressure boundary (Reactor
Vessel and Closure) will be in accordance with the RDT material standards for
the particular materials. Inspection during fabrication will be in accor-
41 27 dance with Section III of the ASME Code and RDT E15-2 NB-T, Class I Nuclear
Components. Inspection of materials and inspection during fabrication of
the Guard Vessel will be in accordance with ASME Code, Class I and RDT E15-
41 27 2NB-T. The overall inspection and test plans for the three structures will
be prepared by the fabricator and approved by the purchaser prior to fabri-
17 cation.

5.2.7 <u>Packing, Packaging & Storage</u>

Applicable requirements to assure adequate quality during shipping
43 and storage will be in the respective equipment specifications rather than
in RDT Standards.

The specifications will require that packaging and packing be
adequate to protect items while at the suppliers' facilities, during trans-
31 portation to the delivery point and during storage at the site.

The specifications will, where appropriate, provide requirements for
sealing the openings in the components, purging the components and/or their
31 containers, selecting and using desiccants, selecting and using materials
contacting the components which are suitably free of chlorides, flourides,
lead, copper, zinc, cadmium, sulfur, mercury, etc.

During storage, the equipment will be maintained in a dry gas environ-
57 31 ment, where appropriate, to protect it from contamination. The purge gas,
container integrity, etc., will be monitored to assure compliance with
previously prepared procedures.

57 Protective measures to be taken during interim storage and con-
struction will be provided by the construction contractor. 25

REFERENCES FOR SECTION 5.2

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4. S. Schrock, S. Shiels, C. Bagnall, "Carbon - Nitrogen Transportation in Sodium Systems", Summaries of Technical Papers, CONF-76-05-3-SUM

5. "Heat Treating, Cleaning and Forming", Metals Handbook, Vol. 2, ASME 1964, pp. 149-166. 22

6. Atomics International Report No. TR-707-810-004, "Test Report (Development) IVHM Reactor Refueling Plug Dynamic Seal Test", May 8, 1974.

7. Atomics International Report No. TR-707-810-009, "Test Report (Development) CRBRP Rotating Plug Inflatable Seal", July 9, 1975.

8. Atomics International Report No. AI-AEC-13145, "Design Guide for Reactor Cover Gas Elastomeric Seals", March 7, 1975.

9. Atomics International Report No. AI-AEC-13146. "Penetration, Leakage, and Compression Set Testing of Elastomeric Seals for LMFBR Use", April 2, 1975. 25

TABLE 5.2-1

SUMMARY OF CODE, CODE CASES AND RDT
STANDARDS APPLICABLE TO DESIGN AND MANUFACTURE
OF REACTOR VESSEL, CLOSURE HEAD AND GUARD VESSEL

			Closure Head*		
	Component/Criteria	Reactor Vessel*	Pressure Boundary	Internals (as appropriate)	Guard Vessel
57	Section III ASME Code, 1974 Edition	Addenda thru Winter '74	Addenda thru Winter '74	Addenda thru Winter '74	Addenda thru Summer '75
		Class 1	Class 1	Class 1	Class 2*
	ASME Code Cases	1521-1,1592-2,1593-0,1594-1,1595-1, 1596-1,1682,1690	1682,1690	1521-1 1592-4,1593-1	1592,1593,1594 if elected by supplier 1521-1 & 1682
57	RDT Standards Mandatory	E8-18T, 2/75 E15-2NB-T, 11/74 Amend thru 1/75	E15-2NB-T, 11/74 Amend thru 6/75	E15-2NB-T, 11/74 Amend thru 6/75	E15-NB-T, 11/74 Amend thru 6/76
		F2-2, 8/73 Amend thru 3/74	F2-2, 8/73 Amend thru 7/75	F2-2, 8/73 Amend thru 7/75	F2-2, 8/73 Amend thru 7/75
		F3-6T, 12/74	F3-6T, 12/74**	F9-4, 9/74	F3-6T, 10/75
		F6-5T, 8/74 Amend thru 2/75	F6-5T, 8/74 Amend thru 2/75		F6-5T, 8/74 Amend thru 11/75
		F7-3T, 11/74	F7-3T, 6/75		F7-3T, 6/75
42		F9-4T, 9/74	M1-1T, 3/75 M1-2, 3/75 Amend thru 7/75		F9-4, 9/74

57 *For those reactor vessel and closure head components internal to the pressure boundary special purpose high cycle fatigue curves and creep damage rules have been developed as discussed in Appendix 5.2A.

TABLE 5.2-1 (Continued)

	Component/Criteria	Reactor Vessel	Closure Head: Pressure Boundary	Closure Head: Internals (as appropriate)	Guard Vessel
57	RDT Standards	M1-1T, 3/75			
		M1-2T, 4/75	M1-4T, 3/75		
		M1-4T, 3/75	M1-6T, 4/75 Amend 1-7/75		
		M1-6T, 4/75	M1-10T, 3/75		
		M1-10T, 3/75	M1-11T, 3/75 Amend.1-7/75		
		M1-11T, 3/75	M1-17T, 3/75		
		M1-17T, 3/75	M2-2T, 12/74		
		M2-2T, 12/74	M2-7T, 3/75**		
		M2-5T, 1/75 Amend 1-2/75	M3-10T, 7/75		
		M2-7T, 2/75	M7-4T, 3/75		
		M2-18T, 4/76			
57		M2-21T, 12/77			
		M3-6T, 3/75			
		M3-7T, 4/75			
		M5-1T, 11/74			
		M5-2T, 5/73			
		M5-3T, 12/74			
		M5-4T, 1/75			
57		M6-3T, 2/75			
		M6-4T, 2/75			
		M7-3T, 11/74			
42	Non-Mandatory	F9-5T, 9/74		F9-5T, 9/74	F9-5T, 9/74

* Functionally designated Class 2, and constructed to rules for Class 1, but not hydrostatically tested or code stamped.

42 ** Except for the three rotating plugs, for which the applicable issues are: F3-6T, 3/69 for LRP & SRP; F3-6T, 5/74 for IRP. M2-7T, 2/69 for LRP & SRP; M2-7T, 2/74 for IRP.

TABLE 5.2-3

MATERIALS FROM WHICH THE REACTOR VESSEL, CLOSURE HEAD AND GUARD VESSEL ARE FABRICATED

Reactor Vessel	Product Form	Material	Comment
Support Ring	Ring Forging	SA 508 Class 2	
Vessel Flange	Ring Forging	SA 508 Class 2	
1 Transition Shell	Plate	SB 168	Inconel 600
17 Shell Cources	Plate	SA 240, Type 304	Austenitic stainless steel
Core Support Ring	Forging	SA 182, Type F304	Austenitic stainless steel formed into arcs and welded
Core Support Cone	Plate	SA 240, Type 304	
Inlet Plenum	Plate	SA 240, Type 304	
Thermal Liner	Plate	SA 240, Type 316	Austenitic stainless steel
1 Thermal Liner Support Ring	Forging	SA 182, Type F304	Austenitic stainless, formed into segments and welded
Nozzles	Forging	SA 182, Type F304	
Closure Head			
Rotating Plugs	Forging	SA-508, Class 2	
Penetration Nozzles	Forging	SA-182, Type F304	Austenitic stainless steel
Shield Plates	Plate	SA-516, Grade 60	
Shielding Support Skirts	Plate	SA-516, Grade 60	Plate formed into arcs and welded
Reflector Plates	Plate	SA-240, Type 304	Austenitic stainless steel
Reflector Plate Supports	Forging or Pipe	SA-182, Type F304 or SA-312, Type 304	Austenitic stainless steel
Suppressor Plates	Plate	SA-240, Type 316H	Austenitic stainless steel
Suppressor Plate Support Column	Forging	SA-182, Type F316H and SA-336, Gr F22	Austenitic stainless steel Lower portion welded to 2¼ C. - 1 Mo. upper portion
Spacer Bars	Bar	SA-387, Class 2, Grade 22	
Margin Ring	Bar	SA-540, Class 1, Grade B-24	
Margin Ring Keeper	Bar	SA-533, Class 1	
57 Suppressor Plate Column Caps	Plate or Forging	SA-387, Grade 22, Class 2 or SA-336, Grade F22	

TABLE 5.2-3 (Cont'd.)

	Guard Vessel	Product Form	Material	Comment
1				
	Vessel top flange	Bar, Plate, Forging	SA 479, SA 240, SA 182, Type 304	
	Vessel	Plate	SA 240 Type 304	
	Vessel to support skirt ring	Bar, Forging	SA 479, SA 182 Type 304	
	Support Skirt	Plate	SA 240 Type 304	
	Support Flange	Plate	SA 240 Type 304	
	Nozzles	Plate, Forging	SA 240, SA 182 Type 304	
	Guard Pipe Flanges	Bar, Plate	SA 479, SA 240 Type 304	
	Guard Pipe	Welded Pipe, Plate	SA 409, SA 240 Type 304	
17	Guard Pipe Elbows	Welded Fitting, Plate	SA 403, SA 240 Type WP 304	
	Cleanout Nozzle	Forging	SA 182 Type F 304	
	Cleanout Nozzle Cap	Forging, Plate	SA 182, SA 240 Type F 304	
41				

56 TABLE 5.2-4 HAS BEEN DELETED

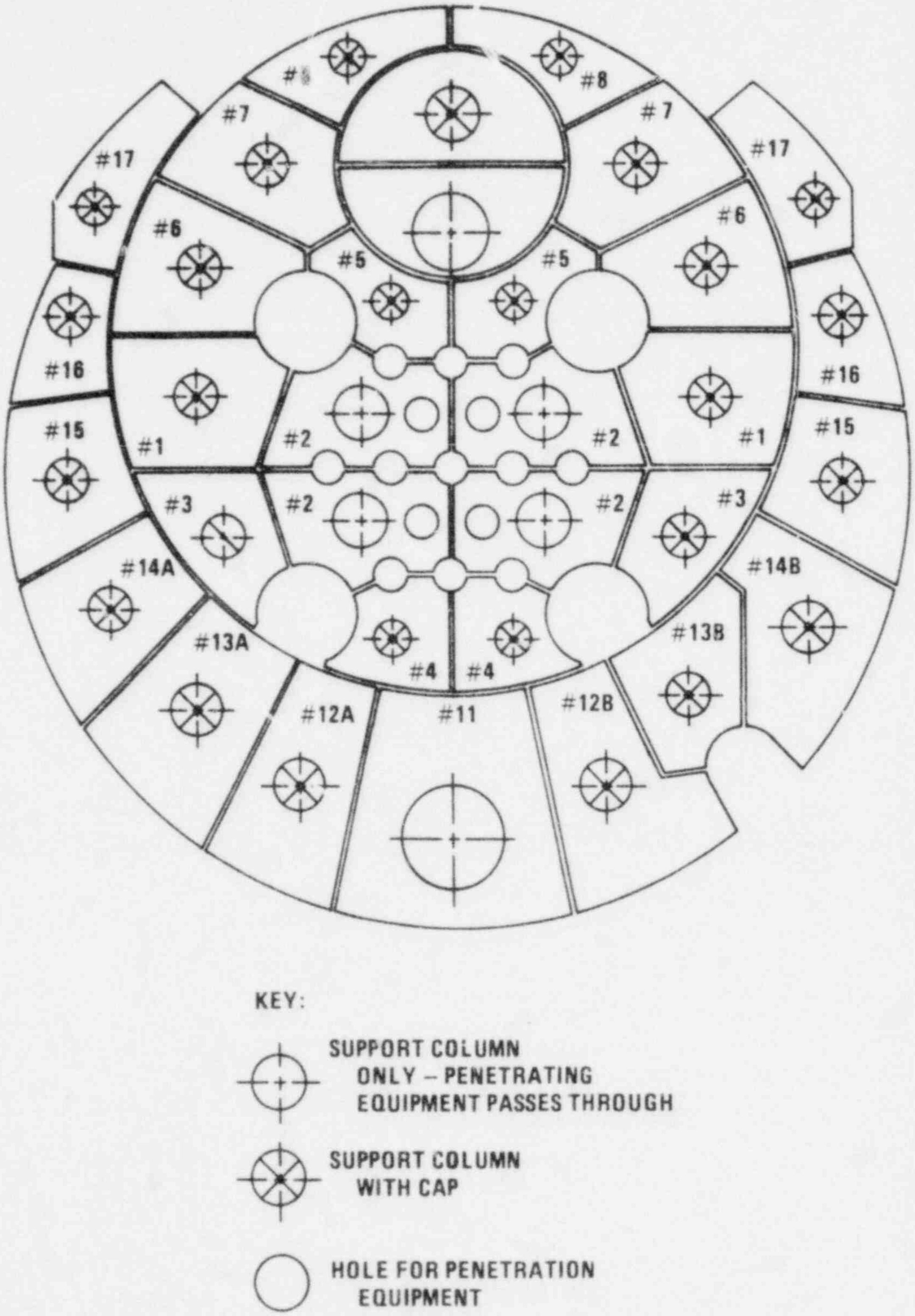

Figure 5.2-3 Suppressor Plate Plan View

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3975-1

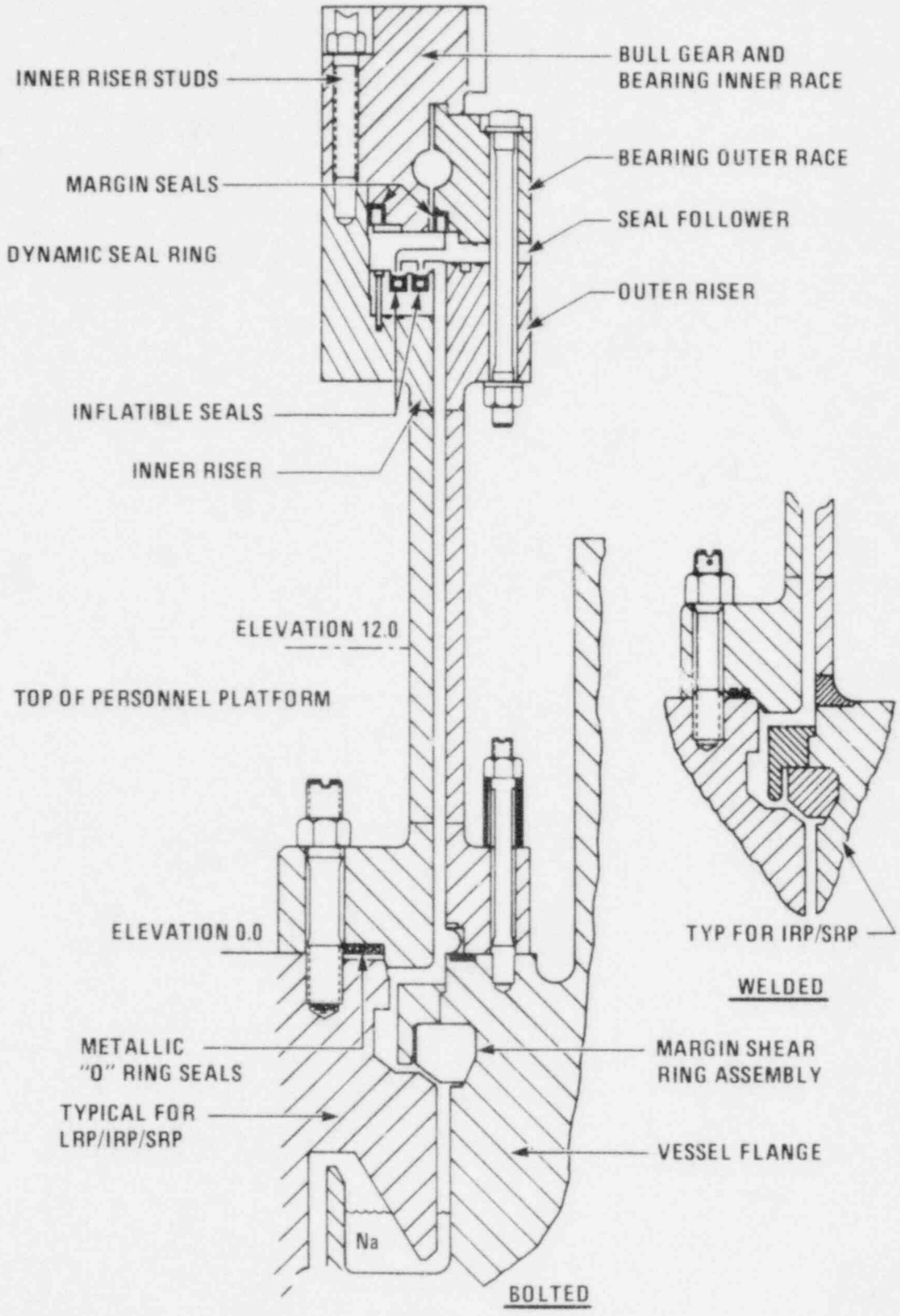
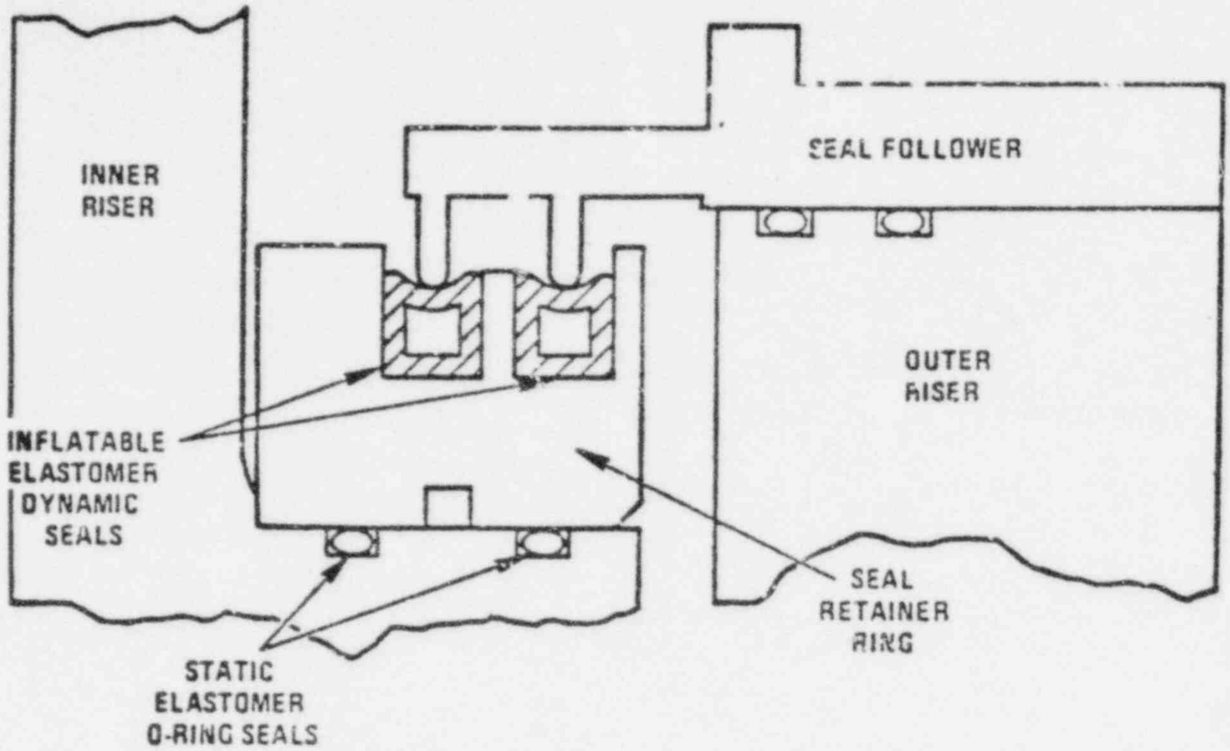
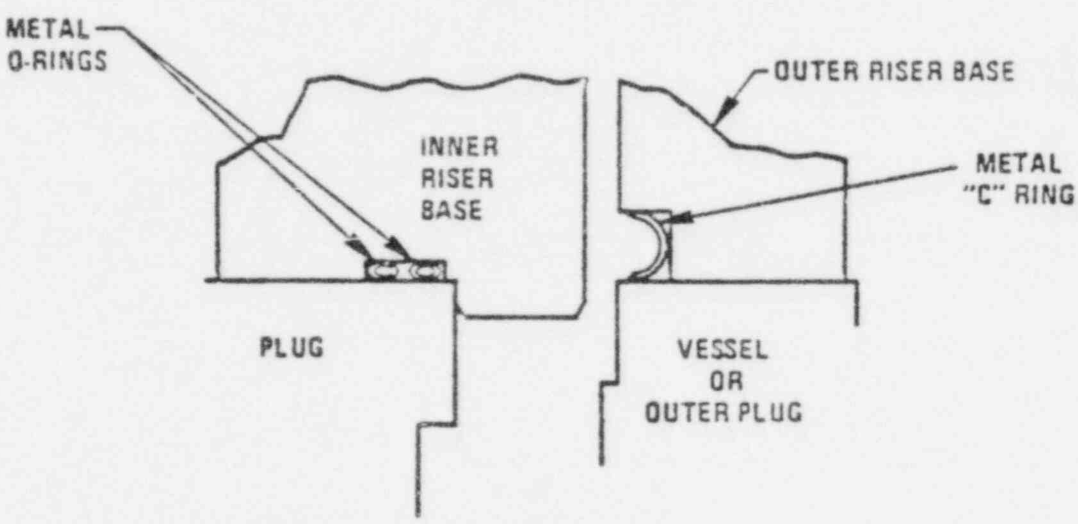


Figure 5.2-7. Typical Riser Assembly



TYPICAL RISER TOP SEALS



TYPICAL RISER BASE SEALS

Figure 5.2-8 Riser Sealing Details

5.3 PRIMARY HEAT TRANSPORT SYSTEM (PHTS)

5.3.1 Design Bases

5.3.1.1 Performance Requirements 17

The Primary Heat Transport System extracts reactor generated heat and delivers this heat to the intermediate coolant system under all normal and off-normal operating conditions. Specific performance requirements include: 17

Heat Transport and Flow Performance

a. Transport of reactor generated heat (975 MWt) through the primary to intermediate coolant system while maintaining an adequate flow rate for controlling reactor temperature conditions within limits which preclude damage to the reactor vessel, fuel and reactor internals.

b. Regulation of heat transport system flow in response to plant process control over the full operating power range of 40 to 100 percent reactor thermal power.

c. Transfer of decay heat to the intermediate coolant system by pony motor operation or natural circulation under all normal and off-normal conditions including failure of a heat transport system component or loop. Specifically, there will be capability to remove decay heat by natural circulation with two or three loops following operation on two loops at the appropriate power. 17

d. Containment of primary sodium coolant and radioactive fission products within the primary coolant system by providing a boundary for primary coolant confinement and a separation of the primary and intermediate coolant systems all within the confines of the containment building.

e. Transport of reactor generated heat to the intermediate coolant system with two-loop operation at the appropriate power input. 17

f. Provide a sodium coolant system which can be easily filled, vented and drained.

g. Support of operation in a hot stand-by condition - nominally 7 1/2 to 10% of full flow at a normal temperature of 600°F.

Structural Performance

a. Design, fabrication, erection and testing of the PHTS components which comprise the sodium boundary shall be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1,
57 Nuclear Power Plant Components, 1974 Edition. The addenda to

Section III that are applicable for the various components are given in Section 5.3.1.2. The applicable ASME Code Cases (1592, 1593, 1594, 1595, 1596) for elevated temperature components shall also be employed. In addition, the RDT Standards E15-2NB-T and F9-4T with revisions as given in Section 5.3.1.2 shall apply, along with the RDT Standards F2-2, F3-6T and F6-5T
27 57 of appropriate revisions coincident with the component contract date of concern.

b. The natural frequencies of all components will, where possible, avoid resonance with all expected pump driving frequencies. Where not possible, the component design shall insure that structural damage will not occur as a result of resonance.

c. Structural design shall provide for dry HTS piping and component
41 heat up at a rate of 3°F/hr.

d. Structural design shall provide for a system fill under conditions of full vacuum with system components at an average temperature of 400°F and hot spot temperatures of 600°F.

Transients

a. The primary heat transport system shall be designed to accommodate the thermal transients resulting from the normal, upset, emergency and faulted conditions described in Appendix B of this document.

b. The system shall be designed such that a normal or upset event does not adversely affect the useful life of any HTS component.

c. Following an emergency condition, resumption of operation must be possible following repair and re-inspection of the components, except that the primary coolant pumps (damaged or undamaged) must maintain capability to provide pony motor flow following all emergency conditions except in the affected loop for pump mechanical failure.

d. During and following a faulted condition, the heat transport system
must remain sufficiently intact to be capable of performing its 33
decay heat removal function, including maintenance of primary coolant pump pony motor flow.

e. The primary heat transport system will accommodate, without loss of decay heat removal capability, the pressures imposed on the intermediate system, by a major sodium water reaction and the sodium hammer resulting from a primary loop check valve closure caused by the most severe flow degradation, such as a primary pump seizure.

f. The primary heat transport system will accommodate, without loss of integrity of the primary coolant boundary, the pressure transients calculated to be the result of a hypothetical accident releasing energy in the reactor vessel. 33 These "Third Level Design Loadings" for various parts of the system are given in Section 15.1.1.3.

Seismic Loads

The PHTS components and supports (ASME Class 1) under the jurisdiction of the ASME Code, Section III, shall be designed to accommodate the load combinations (including seismic loads) prescribed herein without producing total combined stresses, total strains (for all loading cycles) and cumulative creep/fatigue damage (for all loading cycles) in excess of those allowed by the Code. No component of stress or strain of an individual loading condition shall be included which would render the combination non-conservative. Transient loadings shall be included as required by the Code. For elevated temperatures, Code Case 1592
27 57 supplemented by RDT Standard F9-4T will apply.

The ASME Code Class 1 PHTS seismic Category I components shall be designed to withstand the effects of the SSE and of the OBE and remain functional. The OBE will be considered as an upset condition and the SSE 1
5 as a faulted condition. For the low-temperature portion of the design analysis, the rules in the ASME-III code proper will be followed. Accordingly, the OBE shall be included in the analysis to the Code "Design Condition" requirements and limits. For the elevated-temperature portion of the design analysis, the requirements set forth in Code Case 1592 will be followed. Therefore, the OBE shall be included in the analysis to the Code "Upset Condition" requirements and limits. The reason for this difference is that the rules of the Code proper do not require an assessment of Primary Membrane stress plus Bending except in the "Design Condition" case.

Complete details for loading combinations of normal and transient condition loadings with seismic loads are provided in Section 3.9.1.5.

Thermal and Hydraulic design basis parameters are given in Table 5.3-1. Component structural design pressures and temperatures are given in Table 5.3-2. Pressures and temperatures used for structural evaluation at steady state operating conditions are given in Table 5.3-3. 35

5.3.1.2 Applicable Code Criteria and Cases

The PHTS pressure containing components shall be designed, fabricated, erected, constructed, tested and inspected (in compliance with 10CFR50, Section 50.55a) to the standards listed below:

COMPONENT	APPLICABLE STANDARD AND CLASS
PHTS Pump	ASME III, Class 1
PHTS IHX	ASME III, Class 1
PHTS Check Valve	ASME III, Class 1
PHTS Piping (including flowmeter, thermowells and pressure tap penetrations, etc.)	ASME III, Class 1

All primary heat transport system components shall be analyzed as Cla 1 nuclear components in accordance with the rules indicated by the documents in the following:

COMPONENT	ASME Code[1] EDITION & ADDENDA	RDT-E15-2NB-T[2] EDITION & AMENDMENTS	RDT F9-4T[3] EDITION	CODE CASE 1592[4] REVISION
PHTS IHX	1974 plus Summer 1974 Addenda	November 1974 Amendment 1	September 1974	1592-1
Primary Pump Guard Vessel	1974 plus Summer 1974 Addenda	November 1974 Amendments 1, 2, 3	September 1974	1592-4
IHX Guard Vessel	1974 plus Summer 1974 Addenda	November 1974 Amendments 1, 2, 3	September 1974	1592-4
PHTS Piping	1974 plus Addenda to Summer 1975	November 1974 Amendments 1, 2, 3	January 1976	1592-7
PHTS Check Valve	1974 plus Addenda to Summer 1975	November 1974 Amendments 1, 2	September 1974	1592-2

NOTES: 1) ASME Boiler and Pressure Vessel Code, Section III

2) RDT E15-2NB-T (Supplement to Section III)

3) RDT F9-4T (Supplement to Code Case 1592)

4) ASME Code Case 1592, "Class 1 Nuclear Components
57 in Elevated Temperature Service"

57 The "Nuclear Systems Materials Handbook", (Ref. 1)
shall be used to obtain material properties data not available from the above
sources. As required in RDT F9-4T, the use of additional or alternative
material properties shall require the approval of the purchaser. Code
Case 1521, "Use of H-Grades of SA-240, SA-479, SA-336, and SA-358, Sec-
tion III," may be used for H-Grades of Type 304 and 316 austenitic stainless
steels. RDT F9-5T, Sept. 1974 (Section 6) provides alternative procedures for
satisfying the strain limits of Appendix T of Code Case 1592 which are acceptable
27 to the purchaser. Section 6 of RDT F9-5T, Sept. 1974 also provides time/temper-
36 atures below which the primary plus secondary and peak stress limits of Section
III may be used in place of the limits of Appendix T of Code Case 1592. The
scope of the analysis of Code Case 1592 shall be used even if the limits from
Section III are used. For example, the primary plus secondary stress intensity
range due to emergency as well as normal plus upset conditions is limited.

In addition, Code Cases 1593, for fabrication and installation of elevated temperature components, 1594 for their examination, 1595 for their testing, and 1596 for their overpressure protection shall apply for the primary heat transport system components.

5.3.1.3 Surveillance Requirements

Changes in fracture toughness of the PHTS piping and components may be caused by carburization, plastic creep straining and the thermal environment though not radiation effects.

The need for surveillance of stainless steel piping and components for changes in fracture toughness will be determined by ongoing programs.

If a requirement is identified by ongoing programs, a surveillance program will be designed in accordance with the philosophy of 10CFR50, Appendix H.

5.3.1.4 Materials Considerations

5.3.1.4.1 Basis for High Temperature Design/Analysis

Rules governing the construction of Class 1 components which are to experience temperatures above those now provided in Section III shall be constructed in accordance with the following considerations:

a. The rules for materials in NB-2000 shall apply except as modified by Code Case 1592, and

The guard vessels for the IHX and primary pump components shall
44 be fabricated from Type 304 Austenitic Stainless Steel.

The material specifications for the construction of the primary heat transport system components are given in Tables 5.3-4 thru 5.3-8.

The supplier may substitute other ASME-approved material than that specified in Table 5.3-4 thru 8 after review and approval by purchaser prior to use or procurement. If the supplier elects to use other ASME-approved materials, the supplier's request for approval shall contain topical reports demonstrating the adequacy of the selected alternate material.

Selection of alternate materials shall be based on the mechanical properties, metallurgical stability, sodium compatibility, and response to radiation under the applicable design and environmental conditions. When recommending the use of alternate materials, the supplier shall document the justification which shall include, as a minimum, a summary of available test or experience data and a discussion of the adequacy of the recommended materials relative to 304 or 316 stainless steel or other purchaser-specified alternate.

5.3.1.4.3 Additional Requirements

The following requirements are modifications or additions to the requirements of the materials specifications identified in Tables 5.3-4 thru 5.3-8.

Hydrostatic Test of PHTS Piping

The PHTS piping will be hydrostatically tested in accordance with the ASME Code, Section III, Article NB6000 and Code Case 159' (supplemented by RDT E15-2NBT, October 1975), as described below. The lengths of straight pipe will be tested by the manufacturer at his facility to satisfy the Code material specifications. The thermowell body sub-assemblies will be tested under external pressure, by the manufacturer. The IHX Vent-line flow restrictor will be tested by its manufacturer. All other items are considered by the Code to be materials and will not be tested separately. Rather, each completed PHTS piping sub-assembly (spool) will be hydrostatically tested by the spool fabricator at his facility prior to their being shipped to the plant site. Thus, all piping items such as fittings and branch connections will be tested, as well as the spool welds, prior to installation. Specific procedures have not yet been written, but they will be in compliance with the Code requirements noted above as they
57 apply to the design conditions at each location in the system.

Strength Tests of HTS Components

Hydrostatic or pneumatic strength tests shall be conducted in accordance with the ASME Code Section III Code Case 1595-1 and implement
27 any requirements of RDT Std E15-2NB-T, Oct. 1975. 16

Water shall be used as the test medium for hydrostatic tests unless it cannot be shown that residual water can be completely removed. If complete water removal cannot be accomplished, a substitute liquid may be used. For example, the cold leg check valve hydrostatic test will be performed using liquid Freon as the test medium. In any case where a suitable liquid cannot be used, a pneumatic strength test will be employed in accordance with the Code. Guard vessels are not pressure tested.

In addition to hydrostatic and/or pneumatic strength tests, helium leak tests of certain liquid metal containing parts or assemblies will be required. Particularly, tube to tube sheet welds, tube bundles, single pass welds and thin sections (e.g., bellows, rupture discs and seal welds) will be helium leak tested in accordance with the ASME Code, Section V, Article 10. Helium leak tests will be performed after a pneumatic test so that any porosity or minute defect will be exaggerated thereby increasing the sensitivity of the helium leak test. If a component to be hydrostatically tested will also be helium leak tested, the helium leak test will be performed first to preclude water molecules from plugging minute leak paths which the helium leak test otherwise would detect.

Gas Pressure Test - Leak Test

Prior to sodium fill of the reactor and the PHTS, a system pneumatic test will be performed in accordance with 6320 of ASME Code Case 1595-1. This Code Case requires that the test pressure shall not exceed the lowest of the maximum test pressures allowed for any of the components in the system which in the case of the CRBRP would be 18 psig (1.2 times the 15 psig design pressure of the reactor vessel upper plenum). The principal purpose of the test is to leak check field welds. Individual components and piping spool pieces will have been pressure tested prior to installation in accordance with individual specification requirements as noted above. This test will be conducted to supplement visual penetrant and radiographic examination of field welds by checking for gas leaks.

A trace gas (helium) may be added to the test medium or a bubble test may be performed. These tests may be supplemented by a gross leak rate test. The various options that may be used in preparing a specific test procedure have not been evaluated, but will be included in the FSAR.

16

Chemical Analysis

The materials specified in Tables 5.3-4 thru 5.3-8 for the primary heat transport system components shall conform to the chemical compositions specified by the applicable RDT Standard and the additional requirements noted.

27

Shell Assembly

The shell, which is the main IHX enclosure, is fabricated from Types 304 and 316 stainless steel. The shell is welded directly to
27 the lower edge of the cylindrical hanging support through the "Z" junction.

The shell cylinder is fabricated from three cylindrical sections. The use of Type 316 stainless steel reduces the thermal and creep ratcheting problems, thus
27 providing a shell design without the need for a thermal liner in the high temperature region. The bottom portion of the shell consists of a shell cylinder, lateral support ring, hemi-head and primary outlet nozzle, all of Type 304
27 stainless steel. The lateral support ring has sixteen spacer guides attached to it which serve as guides and restraints for the tube bundle lower tubesheet.

To preclude potential cover gas accumulation in the IHX, there is continuous venting sodium from the top of the IHX shell to the primary pump tank below the minimum safe sodium level. This sodium flow will carry any gas evolved in the IHX to the pump tank where the gas will migrate to the cover gas space.

Tube Bundle Assembly

The tube bundle assembly consists of two major sub-assemblies: (1) the bundle, consisting of tubesheets, tubes, support plates, tierods and spacers, outer shroud, hemi-head, downcomer, strongback and by-pass seal, and (2) the channel assembly consisting of replaceable bellows, upper head, intermediate outlet nozzle, intermediate vent, inner and outer channel
27 cylinders, upper downcomer pipes, and "Z" junction forging.

Tube Bundle Subassembly

The tube bundle contains 2850 7/8 inch O.D. x 0.045 inch minimum wall tubes spaced on a 1-5/16 inch triangular pitch. The tubes are joined to both the upper and lower tubesheets. The tube-to-tubesheet joints are made by front face fillet welding the tube ends to specially prepared stubs in the tube sheet face. An automatic T.I.G. welding procedure is used to join tubes to tubesheets. The tube ends will be expanded into the tubesheet holes. The upper and lower tubesheets are designed with a minimum inner and outer rim
27 so as to permit better thermal response with the perforated portion.

The strongback pipe, 34-5/8 inches diameter, is welded to the upper
tubesheet and extends down to approximately 6 inches above the lower tube-
27| sheet. The O.D. of the strongback is machined to provide little distance
between it and the inner support plates reducing by-pass flow away from the
tubes. The lower portion of the strongback contains three slots which engage
matching keys attached to the downcomer. The slot and key combination pro-
27| vides torsional and lateral stiffness for the lower tubesheet-head complex
during shipping and operation. A mechanism (gas trap) is incorporated
into the lower downcomer region to prevent gas from being entrapped in the
annulus between the strongback and downcomer.

The outer shroud is provided as the outer bound of the flow in the
heat transfer zone of the bundle. The uppermost portion of the outer shroud
is the distribution cylinder which forms the entrance zone to the heat trans-
fer tubes. This distribution cylinder is designed to insure uniform circum-
ferential flow into the bundle. This perforated entrance cylinder is welded
to the upper tubesheet. Eight additional rings and eight cylindrical sections
welded together make up the remainder of the outer shroud complex. The machined
rings serve as stiffeners. One ring is utilized as a primary bypass seal attach-
ment point. The ring at the primary inlet nozzle is a support point for saddle
and strap supports during shipment. The outer shroud terminates 1 to 2 inches
below the lowermost support plate to allow for exit flow from the heat transfer zone
57| to the primary outlet nozzle.

The tubes in the tube bundle are supported by nineteen support plates.
27| The uppermost support plate extends to the I.D. of the shell. This plate forms
the upper boundary of the inlet plenum. The plate limits direct impingement of
sodium upon the face of the upper tubesheet. The second support plate is a
complete plate, i.e., it extends from the O.D. of the strongback to the I.D.
of the outer shroud. This plate contains flow holes preferentially drilled to
further ensure uniform bundle flow. The lowermost plate is also complete and
27| aids in uniform distribution through the tube bundle exit. The sixteen inter-
mediate support-baffles are overlapping "doughnut" baffles which contain uniform
flow holes resulting in a combination of cross and axial flow.

The plates are supported by eighteen tierods and spacers, 6 at the
I.D., 6 intermediate and 6 at the O.D. The tierods are threaded into tapped
holes in the shell side of the upper tubesheet. This is a standard method of
support for heat exchangers. The lower tubesheet and hemispherical head is
a floating assembly. The head is welded to the lower tubesheet and serves
as a pressure boundary between the intermediate and primary systems. A
baffle flow ring is provided to uniformly distribute the intermediate flow

Although equipment sizing calculations, plant transient analysis and safety evaluations are not based on expected operating conditions, predictions of fuel life are computed based on nominal parameters plus 2 standard deviations of the reactor inlet and outlet temperatures based on the above mentioned Monte Carlo technique.

31 It is to be emphasized that accident analysis, structural evaluation of permanent plant components and component sizing are based on a fixed set of parameters conservatively chosen and are not based on the expected temperatures and flows resulting from a statistical analysis. 25

The thermodynamic and physical properties of the fluids and materials are based on References 1 and 2.

5.3.3.1.1 Structural Evaluation Plan (SEP)

A structural evaluation plan for each segment of the Heat Transport System is not presently available. However, to facilitate the orderly review and verification of the stress report to be provided to the owner by the manufacturer, structural evaluation plans shall be prepared for each major Class 1 component of the primary heat transport system. The SEP for each component shall provide a description of the methods of analysis which the manufacturer contemplates using in various phases of the structural analysis. The plan shall indicate the degree to which the manufacturer anticipates using elastic, simplified inelastic and rigorous inelastic methods of analysis in design iterations and for demonstrating compliance with the requirements of RDT Standard F9-4T. The manufacturer shall identify any computer programs to be used and shall describe, or provide the basic theory of the program, and identify the assumptions involved in their use. The manufacturer shall also

57 5.3.3.1.2 Stress Analysis Verification

The SEP for each major Class 1 component of the primary heat transport system (i.e., primary piping, primary pump, IHX, guard vessel*, and primary check valve) will specify that a checklist be provided for identifying the anticipated analytical requirements for each component under normal, upset, emergency and faulted plant conditions. A sample of the structural design and analysis checklist to be used for each component is given in Figure 5.3-17.

Structural evaluation details of how the manufacturer of each primary system Class 1 component intends to demonstrate compliance with structural requirements shall be as described in the following categories.

Duty Cycle

To confirm the structural integrity of the PHTS equipment and piping, the sequence of the application of loads or cycles must be selected to provide the most conservative loading history of the applicable events. This is especially important if inelastic analysis methods are being used. Per the PHTS Equipment Specifications, the designer or manufacturer must determine the most conservative sequence of the application of the start-up to shutdown cycles using simplified analysis techniques subsequent to the detailed thermal transient and stress analysis. The equipment designer
57 in the ASME Code stress report must substantiate and document the load history used in the analysis of the equipment for the most severe cases. The application of material properties to be used in structural analysis of the PHTS equipment and piping is in accordance with ASME Code Case 1592 and RDT Standard F9-4T, and this application will give conservative results; sensitivity studies are not required. For example, the standard requires average yield stress properties to be used when calculating residual stress rupture damage. 19

Failure Modes

The manufacturer shall identify the locations and failure modes which are expected to be dominant and the load conditions (pressure, thermal, seismic, etc.) associated therewith. Any failure modes not identified in the
27 57 ASME Code Section III, Code Case 1592 and/or RDT Standard F9-4T to be guarded against for specified loads shall be identified.

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41 *Guard vessels, although classified as ASME III, Class 2, will be designed and analyzed as ASME Class 1 components but will not be code stamped.

References

1. TID-26666, "Nuclear Systems Materials Handbook", Hanford Engineering
57 Development Laboratory, 1974.

1a. "Sodium Components Development Program, Mass Transfer Investigations in Liquid Metal Systems, Quarterly Progress Report No. 9, March-May 1969," GEAP-10036, June 1969.

1b. W. T. Lee, "Biaxial Stress-Rupture Properties of Austenitic Stainless Steels in Zirconium-Gettered Sodium," NAA-SR-12353, October 1967.

1c. R. C. Andrews, R. H. Hiltz, L. H. Kirschler, and R. J. Udavcak, "Limited Comparison of the Relative Merits of 304 Versus 316 Stainless Steel for Liquid Metal Service, Topical Report No. 10." MSAR-67-216, December 1967.

1d. I. A. Rohrig, "Residual Elements and Their Effect on Applications of Austenitic Stainless Steels in the Power Industry," in Effects of Residual Elements on Properties of Austenitic Stainless Steels, pp. 78-89, American Society for Testing and Materials, 1967.

1e. P. D. Goodell, T. M. Cullen, and J. W. Freeman, "The Influence of Nitrogen and Certain Other Elements on the Creep-Rupture Properties of Wholly Austenitic Type 304 Steel," Trans. Am. Soc. Mech. Engrs. 89, Series D, pp. 517-524 (1967).

1f. L. Colombier and J. Hochmann, Stainless and Heat Resisting Steels, St. Martin's Press, New York, 1967.

1g. D. E. Sonon and G. V. Smith, "Effect of Interstitial Impurities, Grain Size and Temperature on Plastic Flow of Type 330 Stainless Steel," Trans. American Soc, Metals 58, pp. 353-359 (1965).

1h. R. B. Gunia and G. R. Woodrow, "Nitrogen Improves Engineering Properties of Chromium-Nickel Stainless Steels," J. Mater. 5 pp. 413-430 (1970).

1i. P. N. Flagella, R. Hundal, and R. Ammon, "Effect of a Liquid-Sodium Environment of the Mechanical Properties of Type 316 Stainless Steel," Trans. Amer. Nucl. Soc. 14, pp. 589-590 (1971).

1j. R. Hundal and P. N. Flagella, "The Effect of Sodium Exposure on Stress Relaxation for Type 316 Stainless Steel at 1200°F and 1325°F,"
paper presented at the ASM/AIME Materials Engineering Congress, 17
Chicago, Illinois, October 1973.

TABLE 5.3-3

STEADY STATE OPERATING CONDITIONS FOR STRUCTURAL EVALUATION

System Location	Refueling		Hot Standby		40% Power		80% Power		100% Power	
	T (°F)	P psig	T (°F)	P psig	T (°F)	P psig	T (°F)	P psig	T (°F)	P psig
Reactor Outlet Nozzle	400	6.8	600	6.6	910	6.1	974	5.4	1015	4.9
Reactor Inlet Nozzle	400	18.5	600	18.1	610	33	674	95	715	134
Pump Inlet	400	9.1	600	8.9	910	8.2	974	6.9	1015	6.0
Pump Discharge	400	14.4	600	14.3	910	34	974	115	1015	168
IHX Inlet	400	6.5	600	6.4	910	26	974	104	1015	155
Check Valve Inlet	400	1.3	600	1.4	610	19	674	88	715	133

NOTES:

1. "100%" power is actually 115% of rated power or 1121 MWt. The flow rate in each loop = 14.03 x 10^6#/hr = 1.015 times design flow.

2. "80%" power is actually 897 MWt or 115% of rated 80% power (780 MWt). The loop flow = 11.22 x 10^6#/hr.

3. "40%" power is actually 40% of rated power (390 MWt) which is 35% of the 115% power condition. The loop flow is 4.84 x 10^6#/hr.

TABLE 5.3-4

INTERMEDIATE HEAT EXCHANGER MATERIALS SPECIFICATION

	Product Form	RDT Standard*	Grade
	Plate	M5-1 Nov. 1974	304 & 316
	Forgings	M2-2 Dec. 1974	F304
		M2-4 Nov. 1974	F8 & F8 m
	Tubing	M3-2 Dec. 1974	TP 304H
	Pipe	M3-3 Nov. 1974	TP 304
	Bars	M7-3 Nov. 1974	
	Bolting	M6-1 Feb. 1975, M6-3 Feb. 1975	
	Nuts	M6-4 Feb. 1975	
	Springs	M8-1 May 1975	
27	Studs	M6-3 Feb. 1975	B8M
	Diaphragm	M5-1 Nov. 1974	TP 316
	Spool Pieces	M3-6 Apr. 1976	TP 316
57	Flange	M2-4 Nov. 1974	

The following additional chemistry controls apply:

Carbon - 0.04 to 0.08% for material <0.25 in. thick (Types 304 and 316)

- 0.05% minimum for material ≤0.25 in. thick (Type 304H)

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*RDT Materials Standards apply only to those parts forming portions of the pressure retaining boundaries or that are exposed to liquid sodium or sodium containing environments.

TABLE 5.3-7

PRIMARY HEAT TRANSPORT SYSTEM PIPING

MATERIALS SPECIFICATIONS[1]

ITEM	PRODUCT FORM	MATERIAL GRADES COVERED
Pipe	Welded [2] Seamless [2]	Types 316H, 304H, 304, 2 1/4 Cr-1 Mo Types 316H, 304H, 304, 2 1/4 Cr-1 Mo
Fittings	Welded [2] Seamless [2]	Types 316H, 304H, 304, 2 1/4 Cr-1 Mo Types 316H, 304H, 304, 2 1/4 Cr-1 Mo
Branch Connections	Forging	Types 316H, 304H, 304, 2 1/4 Cr-1 Mo
Thermowells	Fabrication	Type 316H, 2 1/4 Cr-1 Mo
Flued Heads, Pipe Support	Forging plus Fabrication	Types 316H, 304H, 304
Hangers	Fabrication	Various - as allowed by ASME Code
Snubbers	Fabrication	Various - as allowed by ASME Code
Clamps	Fabrication	Type 304
Auxiliary Steel & Hardware	Fabrication	Various - as allowed by AISC Code
IHX Vent-Line Flow Restrictor	Fabrication	Type 316H
Shop Fabrication of Pipe Sub-Assemblies (Spools)	Fabrication	Types 316H, 304H, 304, 2 1/4 Cr-1 Mo

(1) The CRBRP Materials Specifications are based on ASME Code Section III and RDT Standards requirements. They are used for all large diameter sodium piping in both Primary and Intermediate Heat Transport Systems.

57 (2) Welded and seamless products to these specifications are interchangeable for the intended service.

TABLE 5.3-8

IHX AND PRIMARY PUMP GUARD VESSEL
MATERIALS SPECIFICATION*

44

Product Form	ASME Specification	Grade
Stainless Steel		
Forgings	SA 336	F8
Plate	SA 240	304
Bolting	SA 540	B22 Class 2

44

*No RTD Materials Standards apply.

44

TABLE 5.3-9

PRIMARY HEAT TRANSPORT SYSTEM COMPONENT
WELD FILLER MATERIALS SPECIFICATIONS

	Weld Material Form	RDT Standards	Filler Material Classification
27	Stainless Steel	M1-1 March 1975	E 308L-15 or 16
	Covered Electrodes		E 308-15 or 16
			E 316L-15 or 16
			E 316-15 or 16
			E 16 - 8 - 2
27	Stainless Steel	M1-2 July 1975	ER 308, ER 308L
	Bare Rods and Electrodes		ER 316, ER 316L
			ER 16 - 8 - 2

WELD MATERIALS FOR GUARD VESSELS

	Weld Material Form	RDT Standards	Filler Material Classification
	Stainless Steel Covered Electrodes	None	E 308-15
57	Stainless Steel Bare Rods and Electrodes	None	ER 308

TABLE 5.3-10

PRIMARY REACTOR COOLANT PRESSURE BOUNDARY VALVES AND PUMPS

	Component	Status Active/Inactive	Normal Operating Mode*
	PHTS Pumps	Active	Running
40	PHTS Check Valve	Inactive	NO
	PHTS Fill and Drain Valves†	Inactive	LC
	Reactor Coolant Make-up Pumps†	Inactive	Running
	PHTS High Point Vent Valve†	Inactive	LC

*NO - Normally Open
LC - Locked Closed
†These pumps and valves are part of the primary coolant pressure boundary, but are not actually parts of the heat transport system. These components are discussed in Section 9.3 and 9.5.

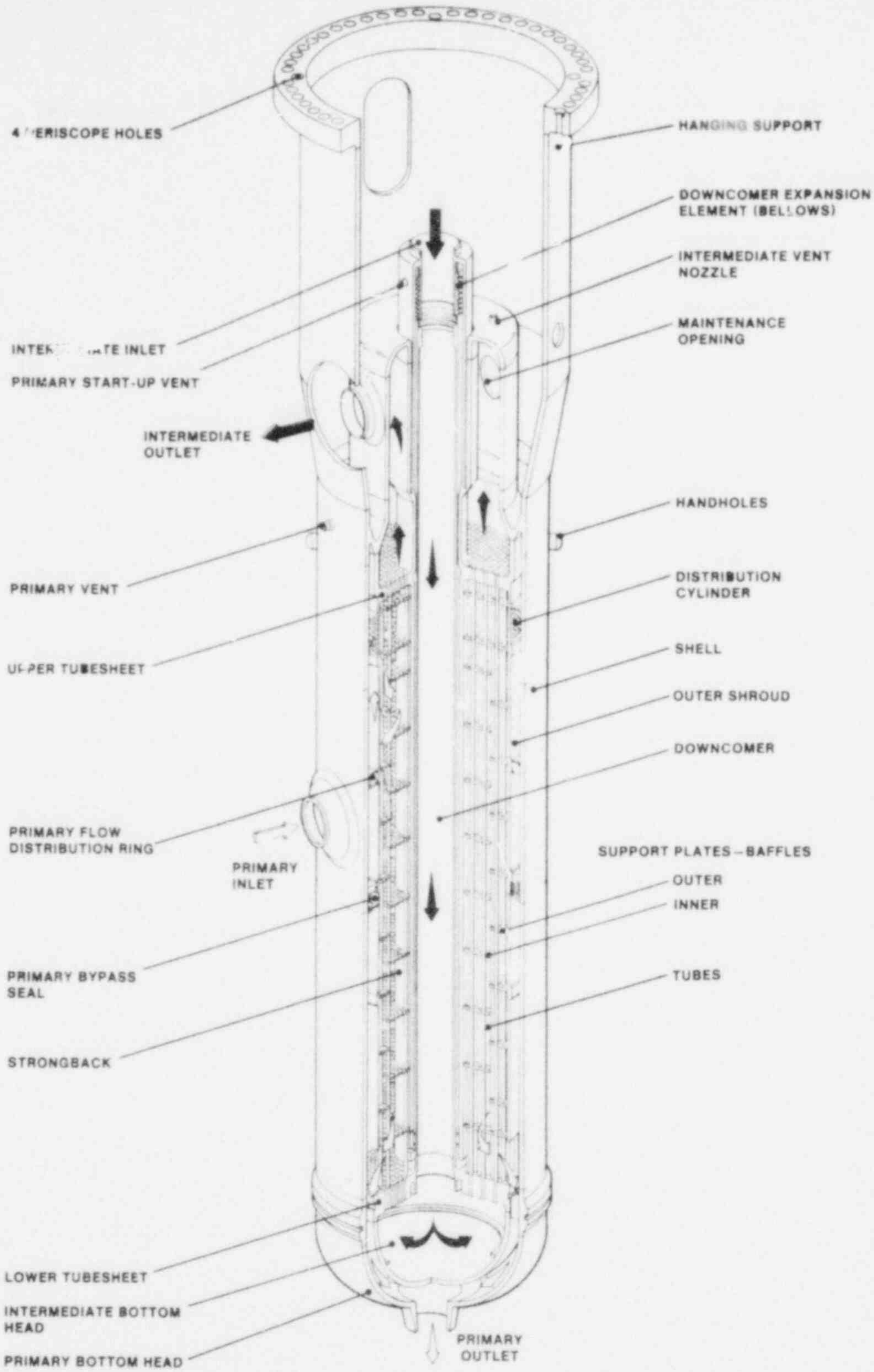

Figure 5.3-15. Intermediate Heat Exchanger (IHX)

4290-1

CHAPTER 7.0 INSTRUMENTATION AND CONTROLS

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7.0 INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This chapter includes a description of the instrumentation and Control Systems provided for the CRBRP. Particular emphasis is placed on the description of safety related systems, which include the Plant Protection System and the safety related display instrumentation required to maintain the plant in a safe shutdown condition. The Plant Protection System includes all equipment to initiate and carry to completion reactor heat transport and balance of plant shutdown, decay heat removal and containment isolation. Safety related display instrumentation assures that the operator has sufficient information to perform required manual safety functions and monitor the safety status of the plant. Major control systems not required for safety are described and analysis is included to demonstrate that even gross failure of those systems does not prevent Plant Protection System action. Analysis is also included to demonstrate that the requirements of the NRC 43 General Design Criteria, IEEE Standard 279-1971, applicable NRC Regulatory Guides and other appropriate criteria and standards are satisfied.

7.1.1 Identification of Safety Related Instrumentation and Control Systems

Table 7.1-1 lists the Safety Related Instrumentation and Control Systems and includes the definition of Safety Related Equipment from Section 3.2.1. The entire Plant Protection System, including the Reactor Shutdown System, the Containment Isolation System and the Shutdown Heat Removal System is safety related. The Reactor Shutdown System input variables are described in Section 7.2. The Containment Isolation Instrumentation and Control System 26 described in Section 7.3 while the Shutdown Heat Removal System Instrumentation and Control System is described in Section 7.4 and Section 7.6. The instru-33 mentation which provides input signals to the Plant Protection System is also safety related and is described in Section 7.5. Safety Related Display Instrumentation, which assures that the operator has sufficient information to monitor the safety status of the plant and maintain it in a safe shutdown condition, is discussed in Sections 7.5 and 7.9. Other safety related instru-33 mentation and control systems including Emergency Chilled Water System, Emergency Plant Service Water System, and Fuel Handling and Storage Interlocks 57 are described in Section 7.6.

7.1.2 Identification of Safety Criteria

In addition to meeting the requirements of the CRBRP General Design Criteria (refer to Section 3.1), the safety related I&C systems will be designed to meet the applicable requirements of the Regulatory Guides and IEEE Standards listed in Tables 7.1-2 and 7.1-3. The means of compliance with the guides and standards applicable to all safety related instrumentation and control equipment are described in paragraphs 7.1.2.2 through 7.1.2.11.

Compliance with guides or standards applicable to specific I&C systems or equipment are described in the paragraphs related to those systems. In addition to meeting the requirements of the Regulatory Guides and IEEE Standards, the safety related equipment will be designed to meet the applicable requirements of the RDT Standards listed in Table 7.1-4. The instrument error and other performance consideration are addressed in the description of individual subsystems.

7.1.2.1 Design Basis

The Plant Protection System (PPS) includes the Reactor Shutdown System (RSS), the Containment Isolation System and the Shutdown Heat Removal Systems.

The Reactor Shutdown System consists of a Primary and a Secondary System either of which is designed to initiate and carry to completion trip of the control rods and sodium coolant pumps to prevent the results of postulated fault conditions from exceeding the allowable limits. Table
45 4.2-35 shows the basis for Primary and Secondary RSS performance for the defined fault categories. The performance limits for the fuel and cladding are identified in Section 4. The Reactor Shutdown Systems are described in Section 7.2.

The Containment Isolation System (CIS) is designed to react automatically to prevent or limit the release of radioactive material to the outside environment. The system acts to isolate the interior of the containment by closing the containment isolation valves in the event that radioactive material is released within the containment. Radiation monitors within the containment boundary are used to activate the CIS. A description of this system is given in Section 7.3.

The Shutdown Heat Removal Instrumentation and Control System is designed to provide assurance against exceeding acceptable fuel and reactor coolant system damage limits following normal and emergency shutdowns. The description of this instrumentation and control is given in Section 7.4 for the removal through the auxiliary steam/water system (Steam Generator Auxiliary Heat Removal System (SGAHRS) and Outlet Steam Isolation System (OSIS)) and Section 7.6 for removal through the NaK to air system (Direct Heat Removal
26 57 System (DHRS)).

Sufficent instrumentation and associated display equipment will be provided to permit effective determination of the status of the reactor at any time. Section 7.5 provides a description of the instrumentation provided. In Section 7.9, a description of the control room, control room
57 layout, operator-control panel interface, instrument and display groupings and habitability are given.

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7.1.2.2 Independence of Redundant Safety Related Systems

To assure that independence of redundant safety related equipment is preserved, the following specific physical separation criteria are imposed for safety related instrumentation.

- All interract PPS wiring shall be run in conduits (or equivalent) with wiring for redundant channels run in separate conduits. Only PPS wiring shall be included in these conduits. Primary RSS wiring shall not be run in the same conduit as secondary RSS wiring. Wiring for the CIS may be run in conduits containing either primary RSS wiring or conduits containing secondary RSS wiring, but never intermixed. 24

- Wiring for other safety related systems may be run in conduits containing either primary RSS wiring or conduits containing secondary RSS wiring, but never intermixed, provided that no degradation of the separation between primary and secondary RSS results. 24

- Wiring for redundant channels shall be brought through separate containment penetrations with only PPS wiring brought through these penetrations. Primary RSS wiring shall not be brought through the same penetration as secondary RSS wiring. Wiring for the CIS and other safety related systems will be brought through the same penetration as the RSS wiring with which it is 24
57 routed.

- Instrumentation equipment associated with redundant channels shall be mounted in separate racks (or completely, metallically enclosed compartments). Only PPS channel instrumentation shall be mounted in these racks. Primary RSS equipment shall not be
57 located in the same rack as Secondary RSS equipment.

- The physical separation between conduits, penetrations, or racks containing redundant instrument channels shall be specified on an individual case basis to meet the requirements of Regulatory Guide 1.75. This separation shall provide assurance that credible single events do not simultaneously degrade redundant channels
57 or redundant shutdown systems.

- The wiring from a PPS buffered output which is used for a non-PPS purpose may be included in the same rack as PPS equipment. The PPS wiring shall be physically separated from the non-PPS wiring. The amount of separation shall meet the requirements of IEEE 384-1974.

- Electrical power for redundant PPS equipment shall be supplied from separate sources such that failure of a single power source does not cause failure of more than one redundant channel. The power sources and associated wiring shall be separated, as specified in Section 8.

The criteria for cable tray fill, cable derating, cable routing in congested or hostile areas, fire detection and protection in cable areas, and cable markings are defined in Section 8. Separation of redundant safety related equipment within the control boards is described in Section 7.9.

7.1.2.3 Physical Identification of Safety Related Equipment

The Plant Protection System equipment will be identified distinctively as being in the protection system. This identification will distinguish between redundant portions of the protection system such that qualified personnel can distinguish whether the equipment is safety related and, if so, which channel. Color coding, cabinet and wire labeling and other techniques as appropriate will be used.

7.1.2.4 Conformance to Regulatory Guides 1.11 "Instrument Lines Penetrating Primary Reactor Containment" and 1.63 "Electric Penetration Assemblies in Containment Structures for Watercooled Nuclear Power Plants"

There are no instrument lines as defined in Regulatory Guide 1.11 which penetrate primary reactor containment. All electric penetration assemblies in the containment vessel will be designed, constructed and installed in accordance with Regulatory Guide 1.63 and IEEE Standard 317-1972.

7.1.2.5 Conformance to IEEE Standard 323-1974 "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations 1

All Class IE equipment will be qualified to confirm the adequacy of the equipment design under normal, abnormal, and postulated accident conditions for the performance Class IE functions. This will be accomplished through a disciplined program of quality assurance and testing. Type testing, operating experience and analysis will be used to assure that for each type of Class IE equipment the design and manufacturing processes are such that there is a high degree of confidence that the equipment of the same type will perform as required.

7.1.2.6 Conformance to IEEE Standard 336-1971 "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations" 1

The installation, inspection and testing of the instrumentation, electrical and electronic equipment during construction will conform to the requirements of IEEE Standard 336-1971. The quality assurance program for the safety related instrumentation and control equipment will conform to the requirements of Regulatory Guide 1.30. Refer to Chapter 17 for a description of the quality assurance program. 1

7.1.2.7 Conformance to IEEE Standard 338-1971 "Periodic Testing of Nuclear Power Generating Station Protection System" 1

Capability for periodic testing is provided to ensure that the Plant Protection System and safety related instrumentation and equipment meet the necessary performance objectives. This testing capability includes provisions for on-line-testing, testing during shutdown, and calibration as appropriate.

The basic objective of the testing is to assure that the Plant Protection System is performing within its specifications; or conversely, to detect functional failures of redundant components or degradation of important performance features. Further, neither the testing of the equipment nor the features incorporated to permit testing will compromise the inde-
57 pendence of redundant components. Since both Reactor Shutdown Systems use 3 redundant channels, on-line testing is possible. Specific testing features are described with the associated hardware.

7.1.2.8 Conformance with Regulatory Guide 1.22 "Periodic Testing of Protection System Actuation Functions"

Plant Protection System actuation devices are periodically tested with the RSS during reactor operation. Since the RSS scram circuit breakers and scram solenoid valves are arranged in 2 out of 3 coincidence logic, scram circuit breakers or solenoid valves may be individually tested during reactor operation. The HTS breakers are arranged to allow testing by using a test breaker to maintain power to the coolant pumps while testing the breaker. While the Reactor Shutdown Systems can be tested through the scram circuit breakers or solenoid valves, the primary and secondary rod release capability cannot be tested during reactor operation since dropping a single control rod will initiate a reactor scram. Scram actuator and control rod insertion times will be functionally tested every time the plant is shut down for refueling. The Containment Isolation System actuated equipment can be tested during reactor operation since containment isolation can take place without necessitating a reactor scram. Shutdown Heat
57 Removal actuation system can be tested on line.

Specific details concerning test and calibration for each protection system channel will not be available until the FSAR. However, the capability for on line functional testing will be provided for all protection system channels. The design will satisfy the requirements of Section 4.10 of IEEE 279-1971 and the recommendations of Regulatory Guide 1.22. The functional performance of a protection system channel will, in general, be tested by inserting a signal in the instrument channel as near as is practical to the sensor. For all tests requiring disconnection of the sensor or modification of the analog input signal, the channel will be placed in a safe condition by tripping the channel comparator associated with the protection system channel under test. Exception to this rule
57 is made for the functional test of the PPS nuclear flux channels. Since addition of the nuclear flux test signal to the analog signal always drives the channel under test toward a safe (i.e., tripped) condition, the comparator output is not placed in trip before the functional test begins. All 16

protection system instrument channels are functionally tested by varying the magnitude of the test signal through the trip point to verify that the comparator trips, then readjusting its magnitude to reset the comparator. After this functional test is completed, the test signal is removed from the instrument channel, and the instrument channel operation
57 is restored. Calibration checks to assure that the protection system channel meets its performance requirements will be accomplished at periodic intervals during regularly scheduled shutdowns. Actuated equipment will, in general, be testable on line. In cases where this is not practical (for example, a control rod cannot be dropped during operation without scramming the reactor), the recommendations of EICSB 22 will be met. 16

7.1.2.9 Conformance to Regulatory Guide 1.47 "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"

Administrative procedures for indicating the bypass or inoperable status of portions of the protection system or supporting systems will be supplemented by a system that automatically indicates at a system level the bypass or deliberately induced inoperability of the Protection System, systems actuated or controlled by the Protection System, or supporting systems that must be operable for the Protection and related systems to perform their safety related functions. An indication of each bypass or deliberately induced inoperability will be displayed in the control room in accordance with Regulatory Guide 1.47.

7.1.2.10 Conformance to Regulatory Guide 1.53 "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems"

Any single failure within the protection system will not prevent proper protection action at the system level when required. The Plant Protection System is periodically tested so that failures are detected. Test schemes will be designed to duplicate as closely as possible the operation being tested. Design precautions include two independent, redundant diverse shutdown systems, each capable of shutting down the reactor; physical independence between redundant channels; and physical barriers utilized for separation between redundant channels. The use of fire retardant materials in construction, fire retardant cable and wire insulation jackets, and physical separation between redundant circuits is relied upon to prevent or mitigate the consequences of a fire.

7.1.2.11 Conformance to Regulatory Guide 1.62 "Manual Initiation of Protective Functions"

The Plant Protection System will provide for manual actuation of each protective action at the system level. The manual initiation of a protective action will result in a Protection System response identical to automatic actuation of the same protective action. For example, manual trip buttons permit operator initiation of reactor scram and containment isolation. The amount of equipment common to both manual and automatic initiation is minimized. Manual initiation of protection actions is designed to go to completion once initiated. No single failure within automatic,
57 manual, or common portions of protective subsystems will prevent initiation of a protective system action by manual or automatic means.

7.1.2.12 <u>Regulatory Guide 1.89 "Qualification of Class 1E Equipment for Nuclear Power Plants"</u>

IEEE Standard 323-1974 will be applied to the safety related instrumentation and control equipment as indicated in Section 7.1.2.5.

This guide further recommends the use of a source term that is
equivalent to one based on the failure of all safety related equipment
designed to prevent or mitigate the condition from which the source
term is derived. The purpose of qualification is to assure that the
safety related equipment will perform under the environmental conditions
57 to which it may be subjected. It is highly inconsistent to require
the qualification of equipment to radiation levels which could not be
reached even as a result of complete failure of the very equipment being
57 qualified. As such, Regulatory Guide 1.89 source term requirement will
not be imposed on the safety-related instrumentation and control systems
and components.

Radiation environments to be considered in qualifying safety-related instrumentation and control systems and components will be determined considering the source term during and/or after the applicable design basis events, the spatial location, shielding and equipment. 22

TABLE 7.1-1

SAFETY RELATED INSTRUMENTATION AND CONTROL SYSTEMS*

Reactor Shutdown Systems

Includes all RSS sensors, signal conditioning calculation units, comparators, buffers, 2/3 logic, scram actuators, scram breakers, scram solenoid
57 valve power sources, control rods, HTS shutdown logic, coolant pump breakers, and mechanical mounting hardware (equipment racks).

Containment Isolation System

Includes radiation monitoring sensors, signal conditioning, comparators, 2/3 logic, containment isolation valve actuators and valves.

57 Shutdown Heat Removal System Instrumentation and Control System

Includes initiating sensors, signal conditioning, calculation units, comparators, logic, auxiliary feedwater pump actuators and controls including feedwater turbine pump, PACC DHX actuators and controls, steam relief valve actuators and valves; sensors, signal conditioning, logic and
26 57 actuators related to shutdown heat removal functions of DHRS including control of sodium and NaK pumps and air blast heat exchangers; and sensors, signal conditioning, logic and actuators related to removal of heat from the EVST.

Other Safety Related Instrumentation and Control

Includes Instrumentation and Controls for portions of the following functions to assure the plant is maintained in a safe shutdown condition:

- Emergency Chilled Water System
- 33 Emergency Plant Service Water System
- Instrumentation necessary to assure plant is maintained in safe
  49 shutdown status (See Table 7.5-4)
- Fuel Handling and Storage Safety Interlocks
- Heating, Ventilating, and Air Conditioning System
- Recirculating Gas Cooling System 1

*The Clinch River Breeder Reactor Plant (CRBRP) structures, systems, and components important to safety are to be designed to remain functional to the event of a Safe Shutdown Earthquake (SSE). These plant features are also designated as safety-related features in the SAR. These include, but are not limited to, those structures, systems and components which are necessary:

a. To assure the integrity of the Reactor Coolant Boundary;
b. To shut down the reactor and maintain it in a safe shutdown condition;
c. To prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10CFR100.

NOTE: Class IE equipment loads are identified in Chapter 8.

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TABLE 7.1-2

LIST OF REGULATORY GUIDES APPLICABLE TO SAFETY RELATED INSTRUMENTATION AND CONTROL SYSTEMS

1.6	Independence Between Redundant Power Sources and their Distribution Systems (as discussed in Sections 8.3.1.2 and 8.3.2.2)
1.12	Instrumentation for Earthquakes
1.17	Protection of Nuclear Power Plants Against Industrial Sabotage
1.22	Periodic Testing of Protection System Actuation Functions
1.28	Quality Assurance Program Requirements (Design and Construction)
1.29	Seismic Design Classification
1.30	Quality Assurance Program Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment
1.32	Use of IEEE Std 308-1971 "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations"
1.40	Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems
1.62	Manual Initiation of Protective Actions
1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants
1.64	Quality Assurance Program Requirements for the Design of Nuclear Power Plants
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants
1.75	Physical Independence of Electric System
1.78	Control Room Habitability During Chemical Release (as discussed in Section 6.3).
1.89	Qualification of Class IE Equipment for Nuclear Power Plants (as discussed in Section 7.1.2.5).

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TABLE 7.1-3

LIST OF IEEE STANDARDS APPLICABLE TO
SAFETY RELATED INSTRUMENTATION AND CONTROL SYSTEMS

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE-317-1972	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE-323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class IE Electric Equipment for Nuclear Power Generating Stations
57 IEEE-323-A-1975	Supplement to the Foreword of IEEE 323-1974
IEEE-334-1971	IEEE Trial-Use Guide for Type Tests of Continuous-Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations
IEEE-336-1971	IEEE Standard: Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE-338-1971	IEEE Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
46 IEEE-344-1975	IEEE Std. 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE-352-1972	IEEE Trial-Use Guide: General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE-379-1972	IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE-384-1974	IEEE Trial Use Standard Criteria for Separation of Class IE Equipment and Circuits
IEEE-420-1973	Trial-Use Guide for Class IE Control Switchboards for Nuclear Power Generating Stations

TABLE 7.1-4
LIST OF RDT STANDARDS APPLICABLE TO
SAFETY RELATED INSTRUMENTATION AND CONTROL SYSTEMS

	C1-1T	Instrumentation and Control Equipment Grounding and Shielding Practices, January, 1973, Amendment 1, January 30, 1975.
27	C4-5T	Permanent Magnet Flowmeter for Liquid Metal Piping Systems, April, 1974.
44	C5-1T	Inductive Level Measurement Sensor for Use in Liquid Metal, March, 1975
	C6-1T	NaK Transmission High-Temperature Pressure Transmitter for Liquid Metal Service, March, 1971, Amendment 1, May 1971-Amendment 2, November 1971-Amendment 3, October 1973-Amendment 4, January, 1974-Amendment 5, June 1974.
	C7-6T	Thermocouple Material and Thermocouple Assembly, February, 1975.
	C10-1T	Thermocouple Signal Transmitter, November 1971.
	C15-3T	Current Pulse Preamplifier for use with Fission Counters, August 1971, Amendment 1, June 1973-Amendment 2, October 1974.
	C15-5T	Fission Type Neutron Detector, December, 1971-Amendment 1, October, 1973.
	C15-6T	Logarithmic Mean Square Voltage (MSV) Intermediate Range Neutron Flux Monitoring System, July, 1971
	C15-7T	Gamma Compensated Ionization Chamber Assembly (Fixed Electrical Compensation) July ,1971, Amendment 1-August 1973-Amendment 2, March 1974.
	C15-8T	Direct Current Power Range Neutron Flux Monitoring System, July, 1971.
44		
	C15-10T	Logarithmic Count Rate Source Range Neutron Flux Monitoring System, July, 1971.
	C16-1T	Supplementary Criteria and Requirements for RDT Reactor Protection Systems, December, 1969.
	C16-2T	Protection System Logic, April, 1972-Amendment 1, June, 1973.
27	C16-3T	PPS Buffers, October, 1971-Amendment 1, December 1971.

TABLE 7.1-4 (Continued)

44	C16-4	Plant Protection System Comparators, April 1972 - Amendment 1, June 1973
57		
	C17-5T	Metal-Sheathed, Mineral-Insulated Cable Bulk Material February 1973- Amendment 1, April 1974.
	E6-5T	Collapsible-Rotor, Roller Nut Control Rod Drive Mechanism for Sodium Service, March 1971, Amendment 1, December 1972- Amendment 2, August 1973- Amendment 3, September 1974.
	F2-2	Quality Assurance Program Requirements, August 1973, Amendment 1, December 1973 - Amendment 2, March 1974 - Amendment 3, July 1975.
41 44	F2-4T	Quality Verification Program Requirements, December 1974
	F3-2T	Calibration System Requirements, February 1973.
	F3-39T	Testing of High Temperature Cable for Nuclear Detectors, August 1971.
27	F7-2T	Preparations for Sealing, Packaging, Packing, and Marking of Components for Shipment and Storage, February 1969, Amendment 1, October 1971 - Amend 2, September 1972.

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7.2 REACTOR SHUTDOWN SYSTEM

7.2.1 Description

7.2.1.1 Reactor Shutdown System Description

The Reactor Shutdown System (RSS) consists of two independent and diverse systems, the Primary and Secondary Reactor Shutdown Systems, either of which is capable of Reactor and Heat Transport System shutdown. All anticipated and unlikely events can be terminated without exceeding the specified limits by either system even if the most reactive control rod in the system cannot be inserted. In addition, the Primary RSS acting alone can terminate all extremely unlikely events without exceeding specified limits even if the most reactive control rod in the system cannot be inserted. To assure adequate independence of the shutdown systems, mechanical and electrical isolation of redundant components is provided. Functional or equipment diversity is included in the design of instrumentation and electronic equipment. The Primary RSS uses a local coincidence logic configuration while the Secondary RSS uses a general coincidence. Sufficient redundancy is included in each system to prevent single random failure degradation of either the Primary or Secondary RSS.

As shown in the block diagram of the Reactor Shutdown System, Figure 7.2-1, the Primary RSS is composed of 24 subsystems and the Secondary RSS is composed of 16 subsystems. Figure 7.2-2A is a typical Primary RSS instrument channel logic diagram. Each protective subsystem has 3 redundant sensors to monitor a physical parameter. The output signal from each sensor is amplified and converted for transmission to the trip comparator in the control room. Three physically separate redundant instrument channels are used. When necessary, calculational units derive additional variables from the sensed parameters with the calculational units inserted in front of the comparators as needed. The comparator in each instrument channel determines if that instrument channel signal exceeds a specified limit and outputs 3 redundant signals corresponding to either the reset or trip state. The 3 outputs of each comparator are isolated and recombined with the isolated outputs of the redundant instrument channels as inputs to three redundant logic trains. The recombination of outputs is in a 2 out of 3 local coincidence logic arrangement.

1

Operating bypasses are necessary to allow RSS functions to be bypassed during main sodium coolant pump startup, ascent to power, and two loop operation. Operating bypasses are accomplished in the instrument channels. For bypasses associated with normal three loop operation, the bypass cannot be instated unless certain permissive conditions exist which assure that adequate protection will be maintained while these protective functions are bypassed. Permissive comparators are used to determine when bypass conditions are satisfied. When permissive conditions are within the
57 allowable range, the operator may manually instate the bypass. If the

permissive condition goes out of the allowable range, the protective function is automatically reinstated. The trip function will remain reinstated until the permissive conditions are again satisfied and the operator again manually initiates the bypass. Operator manual bypass control is not effective unless the bypass comparator indicates that permissive conditions are satisfied. A functional diagram of the Primary and Secondary bypass permissive logic is shown in Figure 7.2-2AA.

Two loop bypasses are established under administrative control by changing the hardware configuration within the locked comparator cabinets. These bypasses are also under permissive control such that the plant must be shutdown to establish two loop operation and if the shutdown loop is
57 activated the bypass is automatically removed.

Bypass permissives are part of the Plant Protection System (PPS), and are designed according to the PPS requirements detailed elsewhere in this section of the PSAR.

Continuous local and remote indication of bypassed instrument channels will be provided in conformance with Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety 15
Systems".

Figure 7.2-2B is a logic diagram of the Primary RSS logic trains. The outputs from the comparators and 2/3 functions are inputs to a 1 out of 24 general coincidence arrangement. The output of the 1/24 is an input to a 1 out of 2 with the manual trip function to actuate the scram breakers. The scram breakers are arranged in a 2 of 3. When 2 or more logic trains actuate the associated scram breakers, power to the control rods is open circuited and the control rods are released for insertion to shutdown position with spring assisted scram force. Open circuiting the control rod power initiates Heat Transport System shutdown.

In the Secondary RSS, the sensed variables are signal conditioned and compared to specified limits by equipment which is different from the Primary RSS equipment. The secondary logic is configured in general rather than local coincidence to provide additional protection against common mode failure. Each instrument channel comparator outputs its trip or reset signal to a 1 of 16 logic module. The 3 redundant secondary instrument channels from each subsystem feed 3 redundant logic trains, which are coupled to the secondary scram actuators. Figure 7.2-2D is a logic diagram for the Secondary RSS logic.

The Secondary RSS consists of 16 protective subsystems and monitors a set of parameters diverse from the Primary RSS as shown in Table 7.2-1. However, since a measure of nuclear flux is necessary in both the Primary and Secondary RSS, nuclear flux is sensed with compensated ionization chambers in the primary while fission chambers are used in the secondary. The Primary RSS monitors primary and intermediate pump speed while the Secondary RSS monitors primary and intermediate coolant flow. Similarly, the steam flow to feedwater flow ratio is used in the Primary RSS while the steam drum level is sensed for the Secondary RSS.

Figure 7.2-2C is a typical Secondary RSS instrument channel logic diagram. Each protective subsystem has 3 redundant sensors to monitor a physical parameter. The output signal from each sensor is conditioned for transmission to the trip comparator located in the control room. Redundant instrument channels are used. When necessary, calculational units are placed in front of the comparators to derive additional variables. The output of the comparators are input to redundant logic trains in a general coincidence arrangement.

Bypass of secondary comparators is implemented in the same fashion as in the primary system except that different equipment is used to provide the permissive comparator function.

57 Figure 7.2-2D is a logic diagram of the Secondary RSS logic trains. The outputs from the instrument channels are input to a 1/16 general coincidence arrangement. The 1/16 output controls the solenoid power sources through isolated outputs. Isolated outputs are also provided to initiate Heat Transport System shutdown. A trip latch-in function is provided to assure that once initiated, the scram will go to completion. The remaining
43 redundant logic trains provide the other two signals for the 2/3 function.

Figure 7.2-2 shows the RSS interface with the Heat Transport System (HTS) pump breaker control. Two HTS pump breakers are connected in series for each HTS pump. Each HTS breaker receives input from the Primary RSS and Secondary RSS pump trip logic. Upon receipt of a reactor trip signal from either Primary or Secondary RSS, the HTS pump breakers open to remove power from the primary and intermediate pumps.

Provisions are made to allow testing of the HTS breaker actuation function during reactor operation. A test breaker is used to bypass the main HTS breaker during a test condition. Test signals are then inserted through the Primary or Secondary RSS pump trip logic to open the main HTS breaker. Mechanical interlocks are provided on the bypass breakers to prevent more than one main HTS breaker in any loop from being bypassed at a time. Control interlocks are provided which make the breaker test inputs ineffective unless the bypass breakers are properly installed. Main HTS breaker and test breaker position status is supplied as part of the
57 RSS status display on the main control panel.

The RSS subsystems do not directly require the reactor operator or control system to implement a protective action. However, manual control devices to manually initiate each protective function are included in the
57 design of the Plant Protection System.

Where signals are extracted from the Plant Protection System, buffers are provided. These buffers are designed to meet the requirements
27 of IEEE-279-1971, RDT Standards C16-1T, Dec. 1969 and C16-3T, Dec. 1971. The buffers prevent the effects of failures on the non-PPS side from affecting the performance of the PPS equipment. The buffers are considered part of the PPS and meet all PPS criteria.

System Testability

Both Reactor Shutdown Systems are designed to provide on-line
57 testing capability. For the Primary RSS, overlapping testing is used. The sensors are checked by comparison with redundant sensor outputs and related measurements. Each instrument channel includes provisions for insertion of a signal on the sensor side of the signal conditioning electronics and test points to measure the performance at the comparator (or calculational unit) input. Where disconnection of the sensor is unavoidable for test purposes, the comparator is tripped when disconnected. The instrument channel electronics including trip comparators and bypass permissive comparators are tested for ability to change value to beyond
57 the trip point and provide a trip input to the logic. The comparators and logic are tested by the PPS Monitor. A set of pulsed signals are inserted from the Monitor into the comparators associated with one subsystem and the logic output is checked by the Monitor to assure that logic trip occurs for the correct combinations of comparator trips. The logic and scram breakers are tested by manually tripping one logic train
57 and observing that the corresponding breakers trip. HTS breakers are

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tested by maintaining power to the pump through a bypass circuit breaker and manually inserting a test signal to the pump trip logic.

For the Secondary RRS insertion of the test signal into a channel causes the entire train (comparator, logic, and scram solenoid valves) to trip. Testing of the pump breaker trip is identical to that for the
57 Primary RRS.

System Instrumentation

57 The instrumentation used by the RSS to detect the occurrence of off-normal plant conditions includes:

- Neutron Flux

  The Primary RSS uses three compensated ionization chamber power range nuclear sensors evenly spaced around the
  57 reactor vessel. The Secondary RSS uses three fission chamber wide range nuclear sensors evenly spaced around the reactor vessel. See Section 7.5.2 for detector details.

- Reactor Inlet Plenum Pressure

  57 The Primary RSS uses six pressure detectors, two per HTS primary loop, located as close as practical to the reactor vessel inlet plenum in the elevated primary cold leg piping. Each set of two detectors comprises an instrument channel. The outputs of the two detectors in each loop are auctioneered. The resultant output signal is provided to the comparator. See Section 7.5.2 for detector details

- Sodium Pump Speed

  57 The Primary RSS uses three redundant tachometers per primary and intermediate HTS pump to measure pump speed. See Section 7.5.2 for detector details.

- Sodium Flow

  57 The Secondary RSS uses six permanent magnet flowmeters to measure HTS sodium flows. One flowmeter is located in each of the primary and intermediate cold legs. Each flowmeter provides three redundant measurements of loop flow. See Section 7.5.2 for detector details.

- Reactor Vessel Sodium Level

  The Primary RSS uses four sodium level detectors evenly spaced within the reactor vessel. Three of these
  57 detectors provide redundant active signals to the RSS. The fourth detector is used as a spare. See Section 7.5.3 for detector details.

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- Undervoltage Relay

57 The Primary RSS uses nine undervoltage relays, three per coolant loop pump bus. The undervoltage relays are located on the HTS pump buses.

- Steam Flow

57 The Primary RSS uses three redundant steam mass flow signals per loop. Each steam supply loop has one venturi flowmeter, three differential pressure sensors, three temperature sensors, and three pressure sensors located between the super-
57 heater exit and the main steam header. Three redundant steam mass flow signals are generated by pressure and temperature compensation of the venturi flowmeter analog signal. See Section 7.5.2 for detector details.

- Feedwater Flow

One venturi flowmeter, three differential pressure sensors, and three temperature sensors per steam generator loop supply the Primary RSS with three redundant temperature-compensated feedwater mass flow signals. See Section 7.5.2 for detector
57 details.

- IHX Primary Outlet Sodium Temperature

57 The Primary RSS uses three redundant thermocouples, mounted in three thermowells, per loop to measure the sodium temperature in the primary cold leg. See Section 7.5.2 for detector details.

- Steam Drum Level

The Secondary RSS uses three redundant reference column level
57 sensors to determine the water level in each steam drum. The level sensor is density compensated. See Section 7.5.2 for detector details.

- Evaporator Outlet Sodium Temperature

The Secondary RSS uses three redundant thermocouples per loop,
57 mounted in three thermowells. These thermocouples are provided to monitor the sodium temperatures in each intermediate cold leg. See Section 7.5.2 for detector details.

47 6

- Sodium-Water Reaction

57 The Secondary RSS uses three redundant pressure sensors located in the reaction products vent line immediately downstream from each rupture disk to detect if the rupture disks have blown. See Section 7.2.1 for details.

The configuration of the instrumentation in the protective subsystems is described in Section 7.2.1.2.

57 Primary Reactor Shutdown System Logic

The Primary RSS logic is implemented using integrated circuits to minimize the scram delay time. Other advantages include minimizing power consumption and space required and maximizing testability, maintainability
57 and reliability. The Primary RSS logic is arranged as shown in Figure 7.2-3.

57 In each logic train, twenty-four 2/3 coincidence logic circuits feed a 1/24 module, whose output is coupled to the final actuation logic and rod actuators by a transistorized power amplifier. When only one comparator of any or all protective functions is tripped, the logic signal output remains positive (reset). When any two comparators of a protective function trip and provide a negative logic signal to the protective logic, the output of the corresponding 2/3 module also trips to a negative logic signal. This negative logic signal in turn trips the 1/24 logic module which outputs a negative logic signal to the final actuation logic and removes power from the scram
57 breaker undervoltage coil.

Light emitting diodes and phototransistors are utilized to provide complete electrical isolation at strategic points through the Primary RSS
57 logic. There is no electrical connection between the comparator output and protective logic input. Consequently, an internal electrical fault in a single instrument channel or comparator cannot propagate to the other channels, protective functions, or logic trains of the protective system. Each logic train is electrically isolated from the other so that protective action can be initiated regardless of any internal electrical fault in a single logic train.

The equipment needed to implement the 24 protective subsystems of the Primary RSS includes the sensors, signal transmitters and amplifiers or equivalent, calculational units, comparators, logic isolators, 2/3 logic modules,
57 1/24 logic modules, logic drivers, final scram actuation circuitry and breakers,
43 buffers, permissives and bypasses. A three section equipment cabinet is used to house the equipment for each of the three instrument channels including the calculational units, comparators, power supplies and buffers. A two section equipment

cabinet is used to house the equipment for each of the three logic trains and
single equipment cabinets house signal conditioning equipment for each channel.
This arrangement of equipment within cabinets provides the necessary mechanical
36 separation of redundant equipment.

Secondary Reactor Shutdown System Logic

The Secondary RSS logic consists of the 16 protective
subsystems arranged in a general coincidence configuration, as shown in
Figure 7.2-4. In this arrangement, the outputs of instrument channel A
comparators are directly coupled to a 1/16 logic circuit in logic train A,
as are the outputs of instrument channel B with logic train B and the
57 outputs of instrument channel C with logic train C.

When the sensed parameter in an instrument channel exceeds its
setpoint (trips), the comparator outputs a zero (trip) signal to the 1/16
logic module, which in turn outputs a zero (trip) signal to the scram latch
and scram solenoid valves (see Figure 7.2-2D). The 1/16 logic module output
voltage changes to zero regardless of the output of the other comparators.
The output of the 1/16 logic modules are combined in a 2/3 coincidence by the
scram 3 solenoid valves located within the Secondary RSS rod. Electrical
43 57 isolation of the logic output to the solenoid drivers and Heat Transport System
shutdown logic (HTS pump breakers) is shown in Figure 7.2-2D. Redundant iso-
lated outputs are provided from each Secondary RSS logic train to the
Secondary RSS pump trip logic where they are combined in a 2/3 logic. Trip
signals are provided to the HTS pump breakers when 2/3 of the redundant
57 Secondary RSS channels are in a tripped condition.

57 The equipment of the Secondary RSS includes sensors,
signal conditioning equipment (transmitters), calculational units, com-
43 parators, 1/16 logic modules, solenoid drivers, secondary final actuation
logic and actuators, buffers, permissives and bypasses. The equipment
is designed using hardware which is diverse from that used in the Primary
RSS. Since each instrument channel is uniquely associated with a logic
train, a four section equipment cabinet houses each of the instrument
43 57 channels comparators, logic trains and solenoid drivers. Single equipment
cabinets are used to house signal conditioning equipment for each channel.
This arrangement of equipment within separate, completely metallically
enclosed cabinets provides the necessary mechanical separation between
redundant equipment.

41

Channel Output Monitoring

Channel output monitoring is included to provide the operators with
early indication of anomalous instrumentation performance. This equipment
is not safety related. If the output of one channel differs from either of the
57 redundant channels by more than a preset amount, the channel output monitoring
circuitry alarms this condition.

7.2.1.2 Design Basis Information

57 The RSS initiates and carries to completion Reactor,
Heat Transport and Balance of Plant Shutdown if any of the off-normal plant
conditions listed in Table 7.2-2 occur. The table also shows the frequency
classification of the postulated fault, and the first Primary and Secondary
57 RSS subsystems which act to terminate the fault. As detailed in Chapter 15,

the RSS design describe below provides the performance necessary to appro-
priately limit the results of the postulated events. Table 7.2-1 shows the
57 Primary and Secondary RSS subsystems which use the instrumentation described
previously to determine the off-normal conditions and trip the plant.

57 7.2.1.2.1 Primary Reactor Shutdown System Subsystems

High Flux

57 The High Flux subsystem (Figure 7.2-5) initiates trip for
positive reactivity insertions at or near full power. This subsystem assures
that sustained operation will not occur with the fuel near incipient center-
line melting. As shown on the figure, the subsystem compares the compensated
ion chamber output signal with a fixed setpoint and initiates trip when the
signal exceeds the setpoint. Analysis of the performance of this subsystem is
based on worst case time response of the instrument of 50 milliseconds and
worst case trip point of 115% of full power. This subsystem is never
bypassed.

Flux-Delayed Flux

The Flux-Delayed Flux subsystem (Figure 7.2-5) initiates trip for
rapid sustained reactivity disturbances which occur anywhere in the load range.
Two subsystems are provided; one for positive flux rates and one for negative
flux rates. These subsystems prevent undesired thermal transients caused by
rapid changes in power with flow held constant. As shown on the figure,
the flux signal is compared with the output of a long lag circuit
whose input is flux. To initiate trip for increasing reactivity
57 disturbances, the flux signal is a negative input to the comparator. For
decreasing reactivity disturbances, the flux signal is positive and the out-
put of the lag circuit is negative. The operation of this subsystem is such
that the trip point is dependent on the initial condition, rate of power
change, and magnitude of power changes. For a given initial power, there is
a threshold magnitude of power change to cause a trip. A step change of
smaller magnitude than the threshold value will not cause a trip. Power
changes greater than the threshold value will initiate a trip with a lower
total power change than a slower ramp rate. The trip equation constants
are adjusted to provide the necessary protection for the
range of normal power conditions without significantly impairing the plant
operations. Worst case values of the constants, instrument response times
and repeatabilities are used in analyzing the performance of the subsystem.
The positive flux rate subsystem is never bypassed. The negative flux rate
subsystem must be bypassed for plant startup. Nuclear flux is used as a per-
missive signal. If nuclear flux is less than 20% of full power flux a bypass
can be manually instated. The bypass is automatically removed when power is
57 increased above the permissive level.

Flux-Pressure

57 The Flux-Pressure* subsystem (Figure 7.2-5) initiates
trip for positive reactivity excursions or reductions in primary flow over
the load range. Two pressure sensors are used for each redundant channel of
the system. This arrangement assures appropriate redundancy while providing
effective plant operational characteristics since pressure sensor replacement

57 *Formerly referred to as Flux-√Pressure subsystem.

cannot be carried out on-line. The use of the high auctioneer automatically accommodates failure of a sensor. All six pressure sensor outputs are compared in the channel output monitoring circuitry to provide early indication of anomalous performance to the operating personnel. The subsystem performance is a function of initial operating level and is analyzed using worst case values for the instrumentation and electronics including response time for the pressure instrumentation of 150 milliseconds. This subsystem must be bypassed for plant startup. Nuclear flux is used as a permissive signal. If if nuclear flux is less than 10% of full power flux a bypass can be manually instated. The bypass is automatically removed when power is increased above
57 the permissive level.

Pump Electrics

The Pump Electrics Subsystem (Figure 7.2-6) provides protec-
40 tion for loss of pumping power for one, two, or three HTS loops. Three subsystems are included, one for each of the coolant pump buses. In each subsystem, three undervoltage relays, one on each phase, are used as redundant channels. If two of three are tripped in any subsystem, reactor trip ensues. A time delay is used to allow the plant to continue through momentary power outages. The subsystem is analyzed using worst case values including 500 millisecond total delay time. For two loop operation, a manual bypass is instated
40 under administrative control by changing the hardware configuration. Two loop bypasses are also under permissive control. Nuclear flux must be less than 10% of full power flux at the time of instating and the primary HTS pump speed in the shutdown loop must be less than 15% of the speed producing nominal full flow or the
57 two loop bypass is automatically removed.

Primary-Intermediate Speed Mismatch

57 The Primary-Intermediate Speed Mismatch subsystems (Figure 7.2-6) initiate trip for imbalances in heat removal capability between the primary and intermediate circuits within a heat transport loop. Three subsystems are included, one for each HTS loop. As shown in the figure, the primary and intermediate speed signals are normalized and subtracted. The absolute value of this difference is compared with a fixed bias and a linear ratio of the primary speed to determine trip initiation. The actual trip point is dependent on initial conditions. Worst case values are used for analysis including a 20 millisecond tachometer time constant. These subsystems must be bypassed to start the plant. The permissive signal used is the nuclear flux. If the nuclear flux is less than 10% full power flux, the subsystem can be bypassed manually. The bypass for the shutdown loop is automatically removed as power is increased. For two loop operation, provisions are made to bypass the function associated with the shutdown loop. A manual bypass is instated under admininstrative control by changing the hardware configuration. Two loop bypasses are also under permissive control. Nuclear flux must be less than 10% of full power flux at the time of instating and the primary HTS pump speed in the shutdown loop must be less than 15% of the speed producing nominal full flow or the [illegible]
57 loop bypass is automatically removed.

Reactor Vessel Level

The Reactor Vessel Level subsystem (Figure 7.2-5) prevents reactor operation unless the sodium level in the reactor vessel is at least 6 inches above the suppressor plate. The output of the level sensor is compared with a fixed setpoint to determine the need for a reactor trip. Worst case values are used in the analysis of the performance of this subsystem including a sensor time constant of 0.5 second. This subsystem is never bypassed.

Steam-Feed Flow Mismatch

57 The Steam-Feed Flow Mismatch subsystem (Figure 7.2-7) initiates reactor trip to prevent continued operation with large imbalances between the steam and feedwater flow for each HTS loop. One of these subsystems is included in each HTS loop. These subsystems protect the steam generators and drums against unacceptable thermal transients. As shown in the figure, each subsystem compares the steam flow and feedwater flow, both of which are multiplied by appropriate constants, in two individual comparators. If the difference between the two values exceeds the setpoint in either of the comparators, a trip is initiated. Increasing steam flow and decreasing feedwater flow fault events are sensed by the first comparator. The second comparator senses decreasing steam flow and increasing feedwater flow fault events. Analysis of this function is based upon worst case parameter values. This subsystem must be bypassed for plant startup. A permissive is included which allows manual bypass of this subsystem for nuclear power less than 10%. Two loop bypass provisions are also included for the shutdown loop. Two loop bypasses are also under permissive control. Nuclear flux must be less than 10% of full power flux at the time of instating and the primary HTS pump speed in the shutdown loop must be less than 15% of the speed producing
57 nominal full flow or the two loop bypass is automatically removed.

IHX Primary Outlet Temperature

The IHX Primary Outlet Temperature subsystem (Figure 7.2-7) compares the sodium temperature in the primary cold leg of each IHX to a fixed set point. A reactor trip is initiated if the sodium temperature exceeds this set point. These subsystems assure that temperature increases in an intermediate loop sodium resulting from steam side fault events or intermediate flow reductions do not increase the reactor coolant temperature. There is one IHX primary outlet temperature subsystem per HTS loop. These subsystems are never bypassed.

57 7.2.1.2.2 Secondary Reactor Shutdown System Subsystems

Modified Nuclear Rate

The Modified Nuclear Rate subystems (Figure 7.2-8) initiate trip for rapid sustained reactivity disturbances which occur in the load range. Two subsystems are provided. One for positive flux rates and one for negative flux rates. These subsystems prevent undesired thermal transients caused by rapid changes in power with flow held constant. The reactor trip is based on flux rate measurements from the fission counters. A permissive is included which allows manual bypass of the negative rate subsystem for nuclear power
57 less than 10%. The positive rate subsystem is never bypassed.

Flux-Total Flow

The Flux-Total Flow Subsystem (Figure 7.2-8) provides protection against increasing and decreasing flow and power events over the 40 to 100% load range. The primary flows of the three HTS loops are summed and multiplied by an appropriate gain. A nuclear power signal obtained from the fission counters is subtracted in the comparator from the total flow value and this difference is compared to a fixed set point. If the difference exceeds the set point, then a reactor trip is initiated. Analysis of this subsystem is based on worst case parameter values, including a 500 msec. time delay for the flow detectors. This subsystem is never bypassed.

Startup Nuclear

The Startup Nuclear subsystem (Figure 7.2-8) obtains a wide range log
57 channel measurement of nuclear power from the fission counters and compares
it to a fixed-set point. If nuclear power is greater than the set point,
a reactor trip is initiated. A permissive module is provided which allows
manual bypass of this subsystem upon the verification of the operation of
57 tne wide range linear channel. This subsystem provides protection against
positive reactivity disturbances occurring during startup.

Primary to Intermediate Flow Ratio

The Primary to Intermediate Flow Ratio subsystems (Figure 7.2-8) pro-
tect against an imbalance in the heat removal capability of the primary and
intermediate loops. The heat removal capability of a particular loop is
determined by measurement of the sodium flow within the loop. The Secondary
57 RSS includes two of these subsystems, Primary Flow High and Primary
Flow Low. In the Primary Flow High subsystem, the output of the high
primary flow auctioneer is compared to the summation of the outputs from the
low intermediate flow auctioneer and a signal proportional to the total
primary flow. When the high primary flow auctioneer signal exceeds the low
intermediate flow auctioneer signal by an amount proportional to the total
primary flow, a reactor trip is initiated.

Similarly in the Primary Flow Low subsystem, a comparison is made
between low primary flow and high intermediate flow. When the high
intermediate flow auctioneer signal exceeds the low primary flow auctioneer
signal by an amount proportional to the total primary flow, a reactor trip
is initiated. These subsystems are manually bypassed during plant startup.
41 The permissive signal used is based on reactor power. If reactor power is
less than 10%, the subsystems can be manually bypassed.

Steam Drum Level

57

The Steam Drum Level subsystem (Figure 7.2-9) measures steam drum
water level and compares it to a fixed setpoint. A reactor trip is initiated
whenever the drum water level decreases below this fixed setpoint. There
are three of these subsystems, one per HTS loop. Analysis of these sub-
systems are based upon worst case perimeter values. For two loop operation,
a manual bypass is instated under administrative control by changing the
hardware configuration. Two loop bypasses are also under permissive control.
Nuclear flux must be less than 10% of full power flux at the time of
instating and the primary flow in the shutdown loop must be less than 15%
of full flow or the two loop bypass is automatically removed.

Evaporator Outlet Sodium Temperature

The Evaporator Outlet Sodium Temperature subsystems (Figure 7.2-10) compare the sodium temperature at the outlet of the evaporator in each HTS loop to a fixed set point. If this temperature exceeds the set point, a reactor trip is initiated. There are three of these subsystems, one per loop. These subsystems detect a large class of events which impair the heat removal capability of the steam generators. These subsystems are never bypassed.

Sodium Water Reaction

The Sodium Water Reaction subsystems (Figure 7.2-10) detect the
occurrence of a sodium water reaction within a superheater or evaporator
module. There are three of these subsystems, one per loop. Each subsystem
57| receives nine signals from the sensors in the reaction products vent lines of a
steam generator. These subsystems are never bypassed.

7.2.1.2.3 Essential Performance Requirements

In order to implement the required protective functions within the appropriate limits, PPS equipment must meet several essential performance requirements. These essential performance requirements and the PPS equipment to which they apply are summarized below.

The PPS instrumentation will meet the essential performance require-
57| ments of Table 7.2-3. This table defines the minimum accuracy and time
constants which will result in acceptable performance of the PPS.

Analysis of worst case PPS functional performance is based on the values given in Table 7.2-3.

The maximum delay between the time a protective subsystem indicates
the need for a trip and the time the rods are released is 0.200 second.
| This time includes the delays due to the calculational units, comparators, logic,
41 scram breakers, and control rod release.

The maximum delay between the time a protective subsystem indicates the need for a trip and the time the HTS sodium pumps are tripped is 0.500 second. This time also includes the delays due to the logic and HTS scram breakers.

The PPS is designed to meet these essential performance requirements over a wide range of environmental conditions and credible single events to assure that environmental effects do not degrade the performance

of the PPS. The environmental extremes are documented in Reference 13 of
57 PSAR Section 1.6. Provisions are incorporated within the PPS which provide a defense against the following incidents:

- Environmental Changes

All electrical equipment is subject to performance degradation due to major changes in the operating environment. Where practical, PPS equipment is designed to minimize the effects of environmental changes; if not, the performance at the environmental extremes is used in the analysis.

Measures have been taken to assure that the PPS electronics are capable of performing according to their essential performance requirements under variations of temperature. The range of temperature environment specified for all the electronic equipment considered here is greater than is expected to occur during normal or abnormal conditions. Electronics do not fail catastrophically when these limits are exceeded even though this is the assumed failure mode. The detailed design of the circuit boards, board mounting and racks includes free ventilation to minimize hot
57 spots. Ventilation is a result of natural convection air flow.

The PPS is designed to operate under or be protected from a wider range of relative humidity than that produced by normal or postulated accident conditions.

Vibration and shock are potential causes of failure in electronic components. Design measures, including the prudent location of equipment, minimize the vibration and shock experienced by PPS electronics. The equipment is qualified to shock and vibration specifications which exceed all normal and off-normal occurrences.

The PPS comparators and protective logic are designed to operate over a power source voltage range of 108 to 132 VAC and a power source frequency range of 57 to 63 HZ. The maximum variation of the source voltage is expected to be ±10%. More extreme variations in the power source may result in the affected channel comparator or logic train outputting a trip signal. In addition, testing and monitoring of PPS equipment is used, where appropriate, to warn of impending equipment degradation. Therefore, it is not expected that changes in the environment will cause total failure of an instrument channel or logic train, much less the simultaneous failure of all instrument channels or logic trains.

The majority of the PPS electronics is located in the control building, and is not subjected to a radioactive environment. Any PPS equipment located in the radioactive areas (such as the head access area) will be designed to withstand the level of activity to which it will be subjected, if its function is required.

- Tornado

  The PPS is protected from the effects of the design basis tornado by locating the equipment within tornado hardened structures.

- Local Fires

  All PPS equipment, including sensors, actuators, signal conditioning equipment, wiring, scram breakers, and cabinets housing this equipment is redundant and separated. These characteristics make any credible fire of no consequence to the safety of the plant. The separation of the redundant components increases the time required for fire to cause extensive damage and also allows time for the fire to be brought to the attention of the operator such that corrective action may be initiated. Fire protection systems are also provided as discussed in Section 9.13.

- Local Explosions and Missiles

  All PPS equipment essential for reactor trip is redundant. Physical separation (distance or mechanical barriers) and electrical isolation exists between redundant components. This physical separation of redundant components minimizes the possibility of a local explosion or missile damaging more than one redundant component. The remaining redundant components are still capable of performing the required protective functions.

- Earthquakes

  All PPS equipment, including sensors, actuators, signal conditioning equipment, wiring, scram breakers and structures (e.g., cabinets) housing such equipment, is classed as Seismic Category I. As such, all PPS equipment is designed to remain functional under OBE and SSE conditions. The characteristics of the OBE and SSE used for the evaluation of the PPS are found in Section 3.7.

7.2.2 Analysis

The Plant Protection System meets the safety related channel performance and reliability requirements of the NRC General Design Criteria,
57 RDT Standard C16-1T, IEEE Standard 279-1971, applicable NRC Regulatory Guides and other appropriate criteria and standards.

General Functional Requirement

The Plant Protection System is designed to automatically initiate appropriate protective action to prevent unacceptable plant or component damage or the release or spread of radioactive materials.

Single Failure

No single failure within the Plant Protection System nor removal from service of any component or channel will prevent protective action when required.

57 Two independent, diverse reactor shutdown systems are provided, either of which is capable of terminating all excursions without allowing plant parameters to exceed specified limits. Each system uses three redundant instrument channels and logic trains. The Primary RSS is configured
57 using local coincidence logic while the Secondary RSS uses general coincidence logic. To provide further assurance against potential degradation of protection due to credible single events, functional and/or equipment diversity are included in the hardware design.

Bypasses

Bypasses for normal operation require manual instating. Bypasses will be automatically removed whenever the subsystem is needed to provide protection. The equipment used to provide this action is part of the PPS. Administrative procedures are used to assure correct use of bypasses for infrequent operations such as two loop operation. If the protective action of some part of the system has been bypassed or deliberately rendered inoperative, this fact will be continuously indicated in the control room.

Multiple Setpoints

Where it is necessary to change to a more restrictive setpoint to provide adequate protection for a particular normal mode of operation or set of operating conditions, the PPS design will provide automatic means of assuring that the more restrictive setpoint is used. Administrative procedures assure proper setpoints for infrequent operations.

For CRBRP, power operation on two-loops will be an infrequent occurrence, and will only be initiated from a shutdown condition. While the reactor is shutdown, the PPS equipment will be aligned for two-loop operation which will include set down of the appropriate trip points. Sufficient trip point set down is being designed into the PPS equipment to adequately cover the possible range (conceptually from 2% to 100%) of trip point adjustment required. In addition, administrative procedures (specifically the pre-critical checkoff) will be invoked during startup to ensure that the proper PPS trip points have been set.

The analysis of plant performance during two-loop operation has not been completed to date. Therefore, the exact trip point settings for two-loop operation cannot be specified at this time. However, the range of trip point settings indicated above is adequate to ensure that trip points appropriate for the anticipated lowest two-loop operating power can be achieved.

In summary, the design of the PPS equipment trip point adjustments and other features for two-loop operation coupled with the anticipated two-loop operating power level and administrative procedures assure full compliance with Branch Technical Position EICSB 12 and satisfy Section 4.15 of IEEE std 279-1971. 16

Completion of Protective Action

57 The Reactor Shutdown Systems are designed so that, once initiated, a
protective action at the system level must go to completion. Return to
normal operation requires manual reset by the operator because the Primary RSS
57 scram breakers or Secondary scram latch circuitry must be manually closed
following trip. Trip signals must be cleared prior to closure of scram breakers.

Manual Initiation

The Plant Protection System includes means for manual initiation of each protective action at the system level with no single failure preventing initiation of the protective action. Manual initiation depends upon the operation of a minimum of equipment because the manual trip directly operates the scram breakers, solenoid scram valve power supply, or equivalent for Shutdown Heat Removal and Containment Isolation System.

Access

Administrative control of access to all setpoint adjustments, module
calibration adjustments, test points and the means for establishing a bypass
permissive condition is provided by locking cabinets and other access design
57 features of the control room and the equipment racks.

Information Read-Out

57 The trip or reset status of all PPS comparators,
logic channels, and power interrupting devices (scram breakers, primary
coolant pump breakers, etc.) is displayed on the Main Control Board.

Annunciator for PPS Alarm Trips

A visual and audible indication of all alarm conditions within the
PPS will be provided in the control room. These alarm conditions include any
tripped PPS comparators in the Primary RSS, Secondary RRS, Containment Iso-
lation System and Shutdown Heat Removal System. The Plant Data Handling and
Display system alerts the operator to significant deviations between redundant
57 RSS analog instrumentation used to monitor a reactor or plant parameter for the RSS.

Control and Protection System Interaction

The Plant Protection System and the Plant Control System have been designed to assure stable reactor plant operation and to protect the reactor plant in the event of worst case postulated Plant Control System failures. The Plant Protection System is designed to protect the plant regardless of control system action or lack of action. Isolation devices will be used between protection and control functions. Where this is done, all equipment common to both the protection and control function is classified as part of the Plant Protection System. Equipment sharing between protection and control is minimized. Where practical, separate equipment (sensors, signal conditioning, cabling penetrations, raceways, cabinets, monitoring etc.) is provided. The sharing of components does not lead to a situation where a single event both initiates an incident through Plant Control System malfunction and prevents the appropriate Plant Protection System.

Periodic Testing

The Plant Protection System is designed to permit periodic testing
of its functioning including actuation devices during reactor operation. In
57 the Primary RSS, a single instrument channel is tested by insert-
ing a test signal at the sensor transmitter and verifying it at the compara-
tor output. A logic train is tested by inserting a very short test signal
in 2 comparator inputs and verifying that the voltage on the scram breaker
trip coils decrease. Because of the time response of the undervoltage relay
coils of the scram breakers and very short duration of the test signal, the
57 reactor does not trip. In the Secondary RSS, an instrument

channel can be tested from sensor to scram actuator by inserting a single test signal because of the general coincidence configuration of the 3 redundant channels. The primary and secondary rod actuators cannot be tested during reactor operation since dropping a single control rod will initiate a reactor scram. Scram actuators and control rod drop will be tested and maintained when the plant is shutdown (See Section 7.1-2). Whenever the ability of a protective channel to respond to an accident signal is bypassed such as for testing or maintenance, the channel being tested is placed in the tripped state and its tripped condition is automatically indicated in the control room.

Failure Modes and Effects Analysis

A Failure Modes and Effects Analysis (FMEA) has been conducted to identify, analyze and document the possible failure modes within the Reactor Shutdown System and the effects of such failures on system
41 40 performance (see Appendix C, Supplement 1). Components of the RSS
57 analyzed are:

- Reactor Vessel Sodium Level Input
- PPS Sodium Flow Input
- 40 Pump Electric Power Sensor
- Compensated Ion Chamber Nuclear Input
- Fission Chamber Nuclear Input
- Primary Loop Inlet Plenum Pressure Input
- Sodium Pump Speed (Primary and Intermediate)
- Steam Mass Flow Rate Input
- Feedwater Mass Flow Rate Input
- Steam Drum Level Input
- Primary Comparator
- Secondary Comparator
- Primary Logic Train
- Secondary Logic Train
- Primary Calculational Unit
- Secondary Calculational Unit

- Scram Actuator Logic
- Heat Transport System (HTS) Shutdown Logic
- Control Rod Drive Mechanism (CRDM) Power Train
- 57 PPS Isolation Buffer

Figures 7.2-3 and 7.2-4 provide assistance in locating the above
57 system level components within the overall RSS.

The probability of occurrence of each failure mode is listed
57 in the tables of Appendix C. Supplement 1, in the Probability Column.
41 The effects of each potential failure mode have also been categorized in the
tables in the Criticality Column. Even though the failure of an individual
element may result in the inability to initiate channel trip, the pro-
vision of redundant independent instrument channels and logic trains assures
that single random failures cannot cause loss of either the Primary or
57 Secondary RSS thereby meeting the design requirements of RDT C16-1T
and IEEE 279-1971. The high reliability of components, redundant configuration,
provision for on-line monitoring and on-line periodic testing further
assure that random failures will not accumulate to the point that trip
57 initiation by either Primary or Secondary RSS is prevented. All
41 failure effects are therefore categorized as not causing any degradation
or failure of a system safety function. The majority of the identified failure
modes can be eliminated from consideration based on their low probability
of occurrence and the insignificance of their criticality. They are
included in the FMEA, however, to document their consideration.

TABLE 7.2-1

PLANT PROTECTION SYSTEM PROTECTIVE FUNCTIONS

Primary Reactor Shutdown System

• Flux-Delayed Flux (Positive and Negative)

57 • Flux-Pressure

• High Flux

57 • Primary to Intermediate Speed Mismatch

41 • Pump Electrics

• Reactor Vessel Level

• Steam-Feedwater Flow Mismatch

• IHX Primary Outlet Temperature

Secondary Reactor Shutdown System

57 • Modified Nuclear Rate (Positive and Negative)

• Flux-Total Flow

• Startup Nuclear

• Primary to Intermediate Flow Ratio

• Steam Drum Level

• Evaporator Outlet Sodium Temperature

47

• Sodium Water Reaction

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TABLE 7.2-2

PPS DESIGN BASIS FAULT EVENTS

	Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
	I. Anticipated Faults		
	A. Reactivity Disturbances[1]		
53	Positive Ramps $\leq$5¢/sec and Steps $\leq$10¢		
	Startup	Flux-Delayed Flux or Flux-Pressure	Startup Nuclear
	5-40% Power	Flux-Delayed Flux or Flux-Pressure	Modified Nuclear Rate or Flux-Total Flow
57	40-100% Power	Flux-Pressure	Flux-Total Flow
	Full Power	High Flux	Flux-Total Flow
	Negative Ramps and Steps	Flux-Delayed Flux	Modified Nuclear Rate
	B. Sodium Flow Disturbances		
57	Coastdown of a Single Primary or Intermediate Pump	Primary-Intermediate Speed Mismatch	Primary-Intermediate Flow Ratio
	Loss of 1 HTS Loop	Pump Electrics	Primary-Intermediate Flow Ratio
	Loss of 2 HTS Loops	Pump Electrics	Primary-Intermediate Flow Ratio
41	Loss of 3 HTS Loops	Pump Electrics	Flux-Total Flow

TABLE 7.2-2 (Continued)

Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
57 C. Steam Side Disturbances		
Evaporator Module Isolation Valve Closure	IHX Primary Outlet Temperature	Evaporator Outlet Na Temperature
Superheater Module Isolation Valve Closure	Steam-Feedwater Flow Mismatch	Evaporator Outlet Na Temperature
Water Side Isolation and Dump of Single Evaporator	IHX Primary Outlet Temperature	Evaporator Outlet Na Temperature
Water Side Isolation and Dump of Single Superheater	Steam-Feedwater Flow Mismatch	Evaporator Outlet Na Temperature
Water Side Isolation and Dump of Both Evaporators and Superheater	Steam-Feedwater Flow Mismatch	Evaporator Outlet Na Temperature
Loss of Normal Feedwater	Steam-Feedwater Flow Mismatch	Steam Drum Level
47 Turbine Trip with Reactor Trip (Loss of Main Condenser or Similar Problem)	Steam-Feedwater Flow Mismatch	Steam Drum Level
Inadvertent Opening of Evaporator Outlet Safety Valve	Steam-Feedwater Flow Mismatch	Steam Drum Level
Inadvertent Opening of Superheater Outlet Safety Valve	Steam-Feedwater Flow Mismatch	Steam Drum Level
Inadvertent Opening of Evaporator Inlet Dump Valve	IHX Primary Outlet Temperature	Evaporator Outlet Na Temperature

TABLE 7.2-2 (Continued)

	Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
57			
	II. Unlikely Faults		
	A. Reactivity Disturbances[2]		
53	Positive Ramps ≤35¢/sec and Steps ≤60¢		
57	Startup	Flux-Delayed Flux or Flux-Pressure	Startup Nuclear
57	5-40% Power	Flux-Delayed Flux or Flux-Pressure	Modified Nuclear Rate or Flux-Total Flow
	40-100% Power	Flux-Pressure	Flux-Total Flow
	Full Power	High Flux	Flux-Total Flow
	B. Sodium Flow Disturbances		
57	Primary Pump Seizure	Primary-Intermediate Speed Mismatch	Primary-Intermediate Flow Ratio
57	Intermediate Pump Seizure	Primary-Intermediate Speed Mismatch	Primary-Intermediate Flow Ratio
	C. Steam Side Disturbances[3]		
	Steam Line Break	Steam-Feedwater Flow Mismatch	Evaporator Outlet Na Temperature
	Recirculation Line Break	Steam-Feedwater Flow Mismatch	Steam Drum Level
	Feedwater Line Break	Steam-Feedwater Flow Mismatch	Steam Drum Level

TABLE 7.2-2 (Continued)

57 Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
Failure of Steam Dump System	Steam-Feedwater Flow Mismatch	Steam Drum Level
Sodium Water Reaction in Steam Generator(3)	Steam-Feedwater Flow Mismatch	Sodium-Water Reaction
III. Extremely Unlikely		
A. Reactivity Disturbances		
53 Positive Ramps ≤ $2.0/sec		
Startup	Flux-Delayed Flux	Startup Nuclear
5-40% Power	Flux-Delayed Flux or Flux-Pressure	Modified Nuclear Rate or Flux-Total Flow
57 40-100% Power	Flux-Pressure	Flux-Total Flow
Full Power	High Flux	Flux-Total Flow

53 (1) The maximum anticipated reactivity fault results from a single failure of the control system with a maximum insertion rate of approximately 4.1 cents per second.

53 (2) The maximum unlikely reactivity faults result from multiple control system failures leading to withdrawal of six rods at normal speed or one rod at the maximum mechanical speed.

(3) The PPS is required to terminate the results of these extremely unlikely events within the umbrella transient specified as emergency for the design o, the major components.

TABLE 7.2-3

ESSENTIAL PERFORMANCE REQUIREMENTS FOR PPS INSTRUMENTATION

	Plant Parameter	Accuracy (% of span)	Response Time (msec)
57			
	Neutron Flux		
	Primary	±1.0	<10
	Secondary	±1.0	<10
	Reactor Inlet Plenum Pressure	±2.0	<150
	Sodium HTS Pump Speeds	±2.0	<20
	Sodium HTS Flow	±10.0	<500
	Reactor Vessel Sodium Level	±5.0	<500
	Undervoltage Relay	±3.0	<200
	Steam Flow	±2.0	<500
	Feedwater Flow	±2.5	<500
	Evaporator Outlet Sodium Temperature	±2.0	<5000
57	Steam Drum Level	±1.0	<1000
	IHX Primary Outlet Temperature	±2.0	<5000

47

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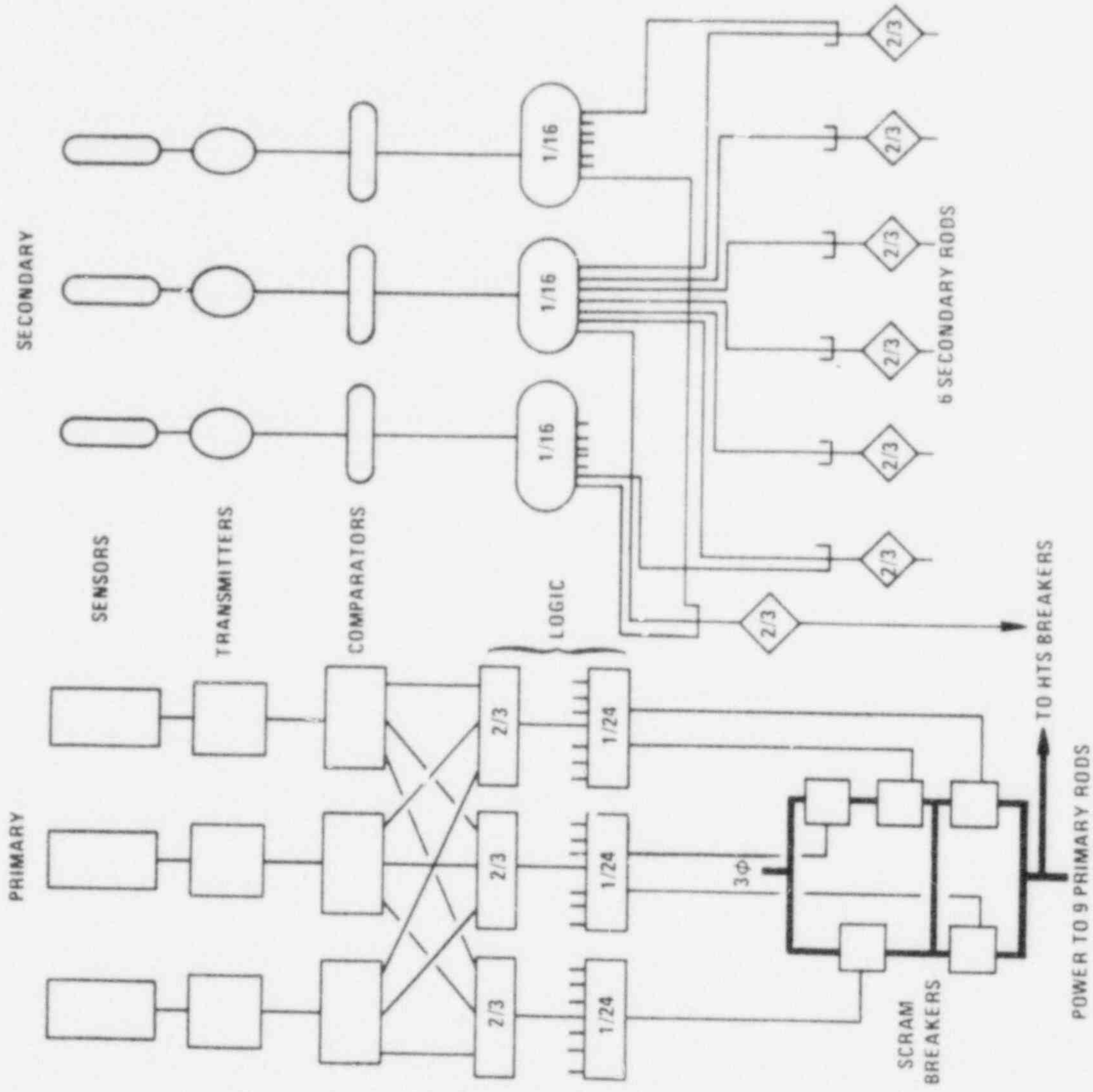

Figure 7.2-1. Reactor Shutdown System

4558-7

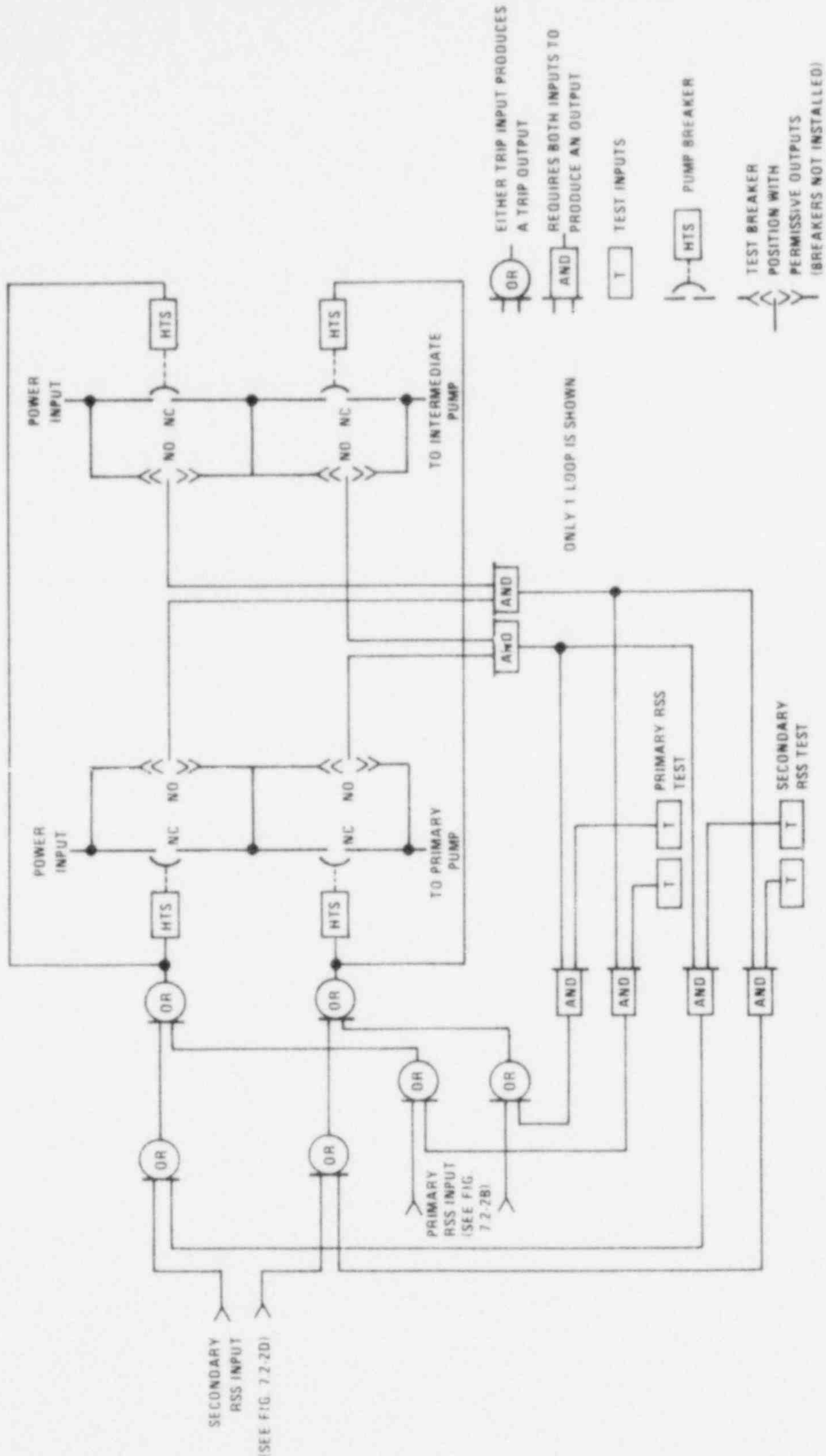

Figure 7.2-2. HTS Pump Breaker Logic Diagram

4558-8

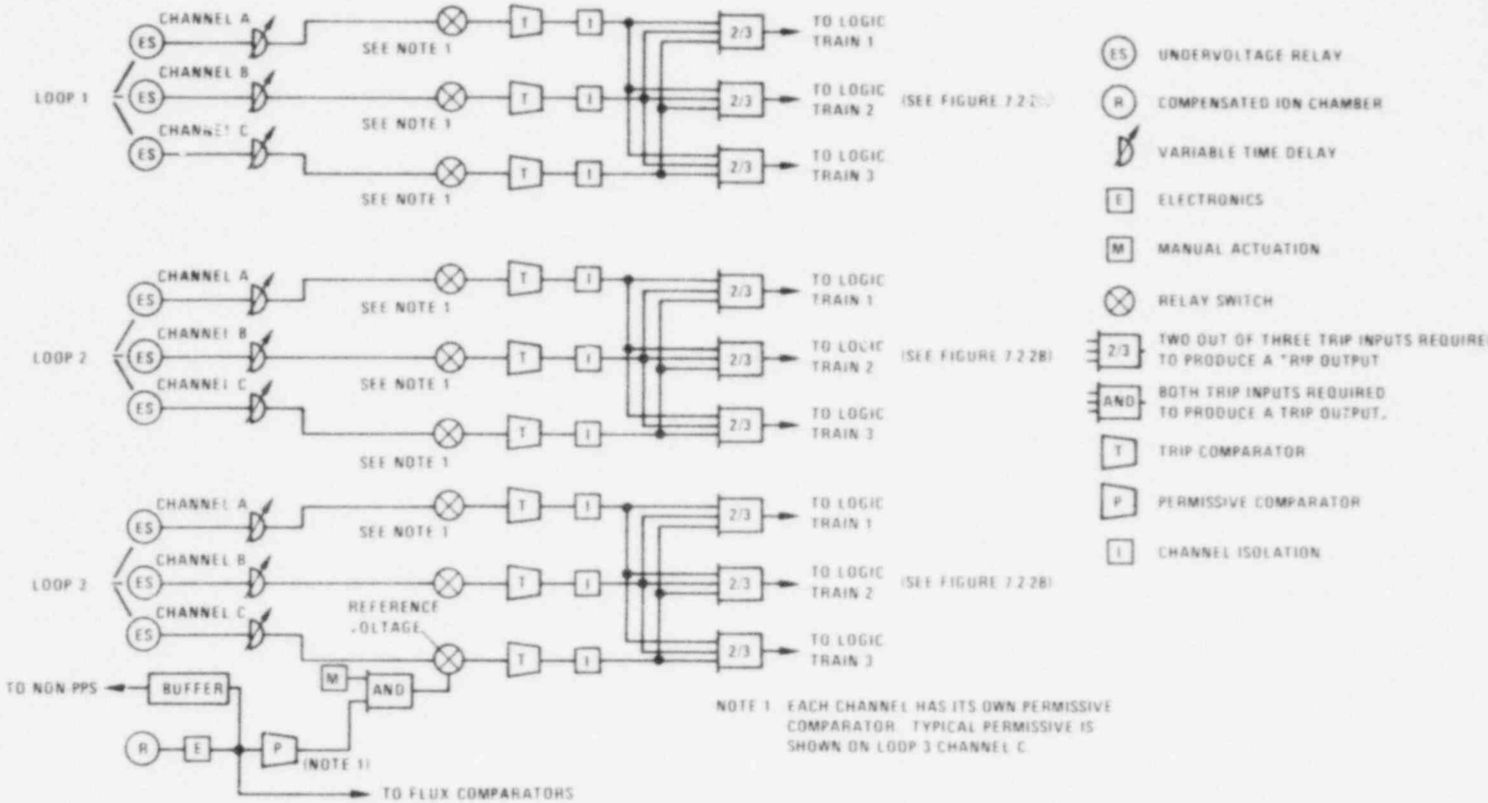

Figure 7.2-2A. Typical Primary PPS Instrument Channel Logic Diagram (Pump Electrics Subsystem Shown)

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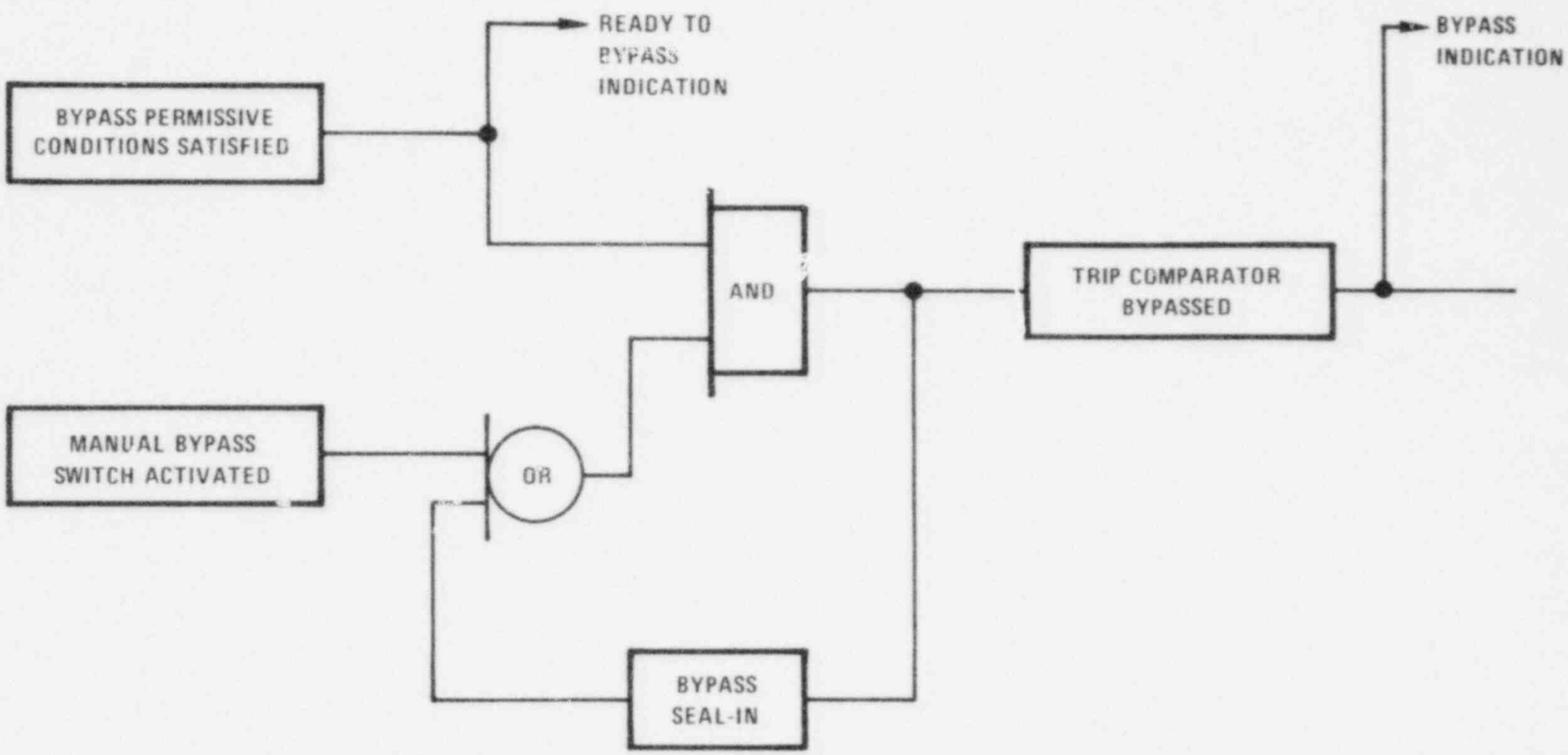

Figure 7.2-2AA. RSS Bypass Function Block Diagram

4558-12

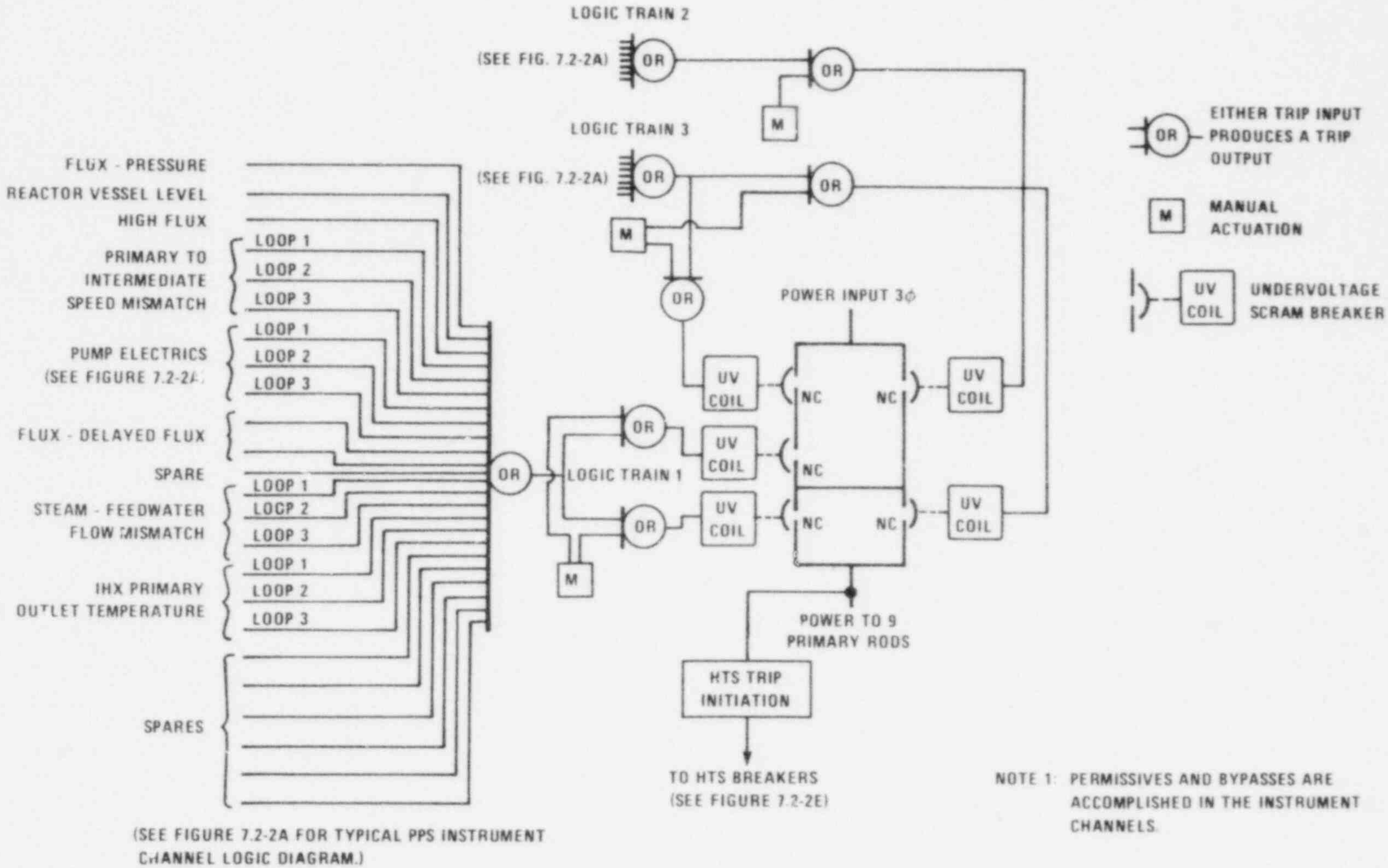

Figure 7.2-2B. Primary RSS Final Actuation Logic Diagram

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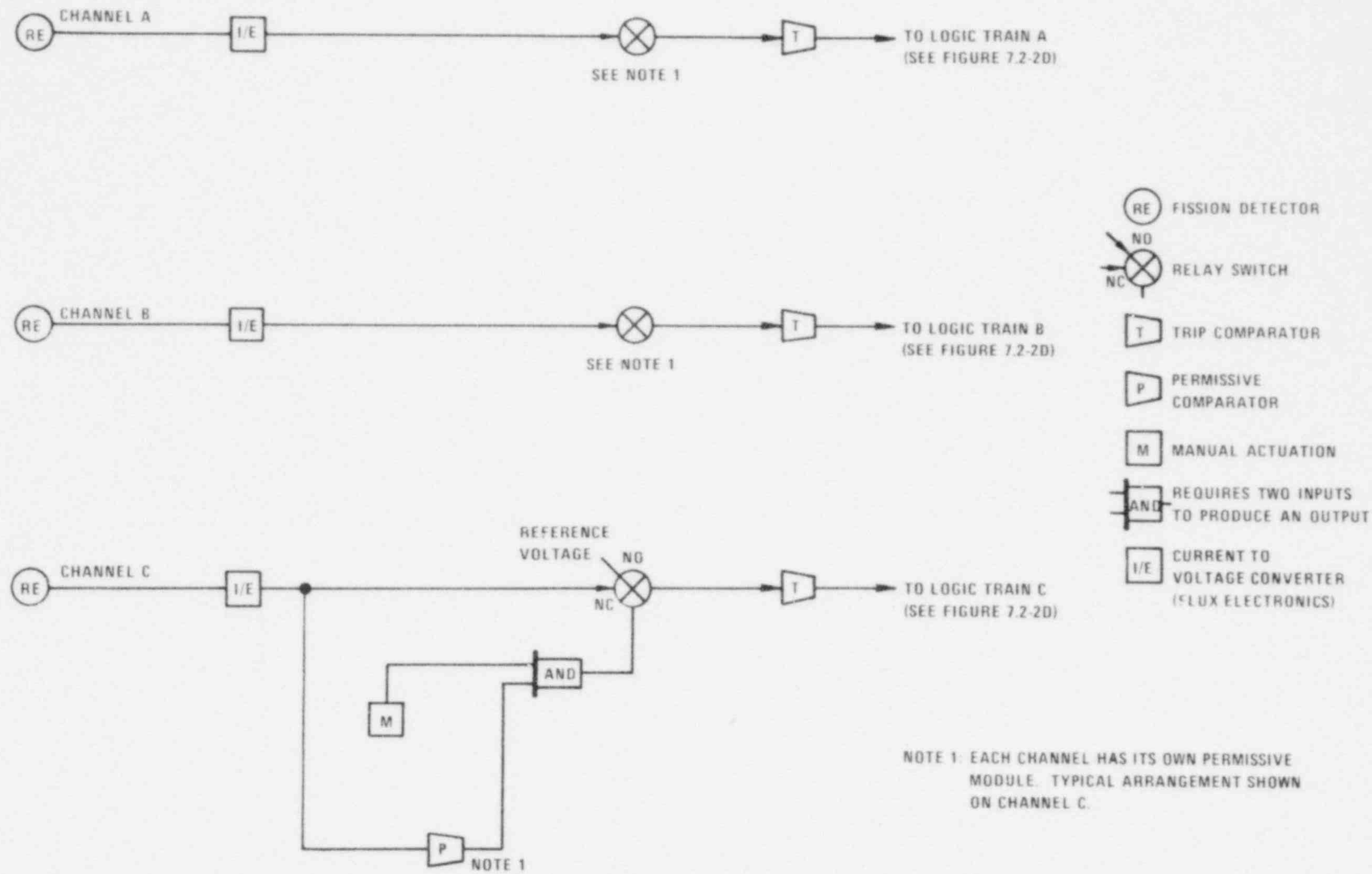

Figure 7.2-2C. Typical Secondary P.P.S. Instrument Channel Logic Diagram. (Startup Nuclear Shown).

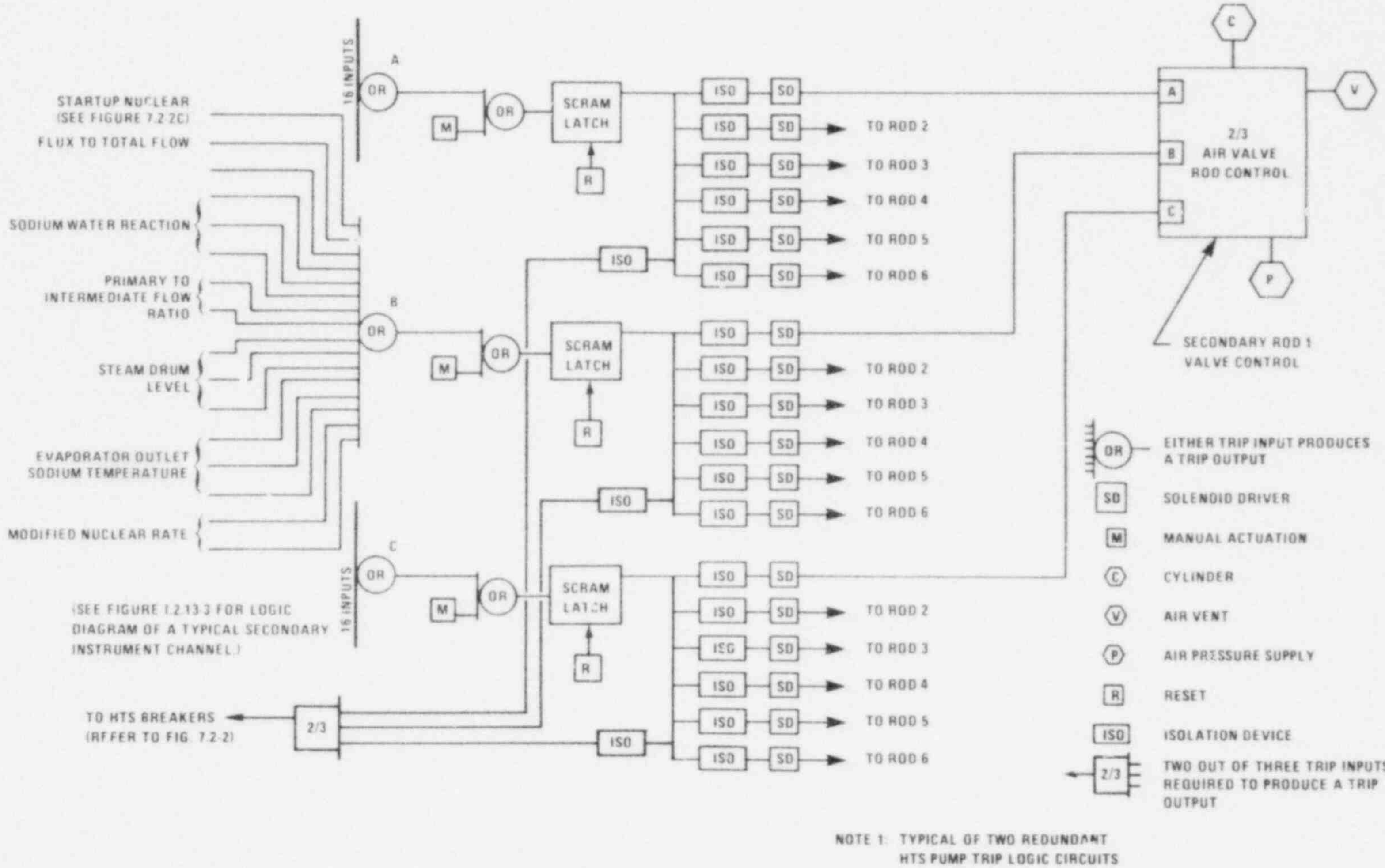

Figure 7.2-2D. Secondary RSS Final Actuation Logic Diagram

4558-2

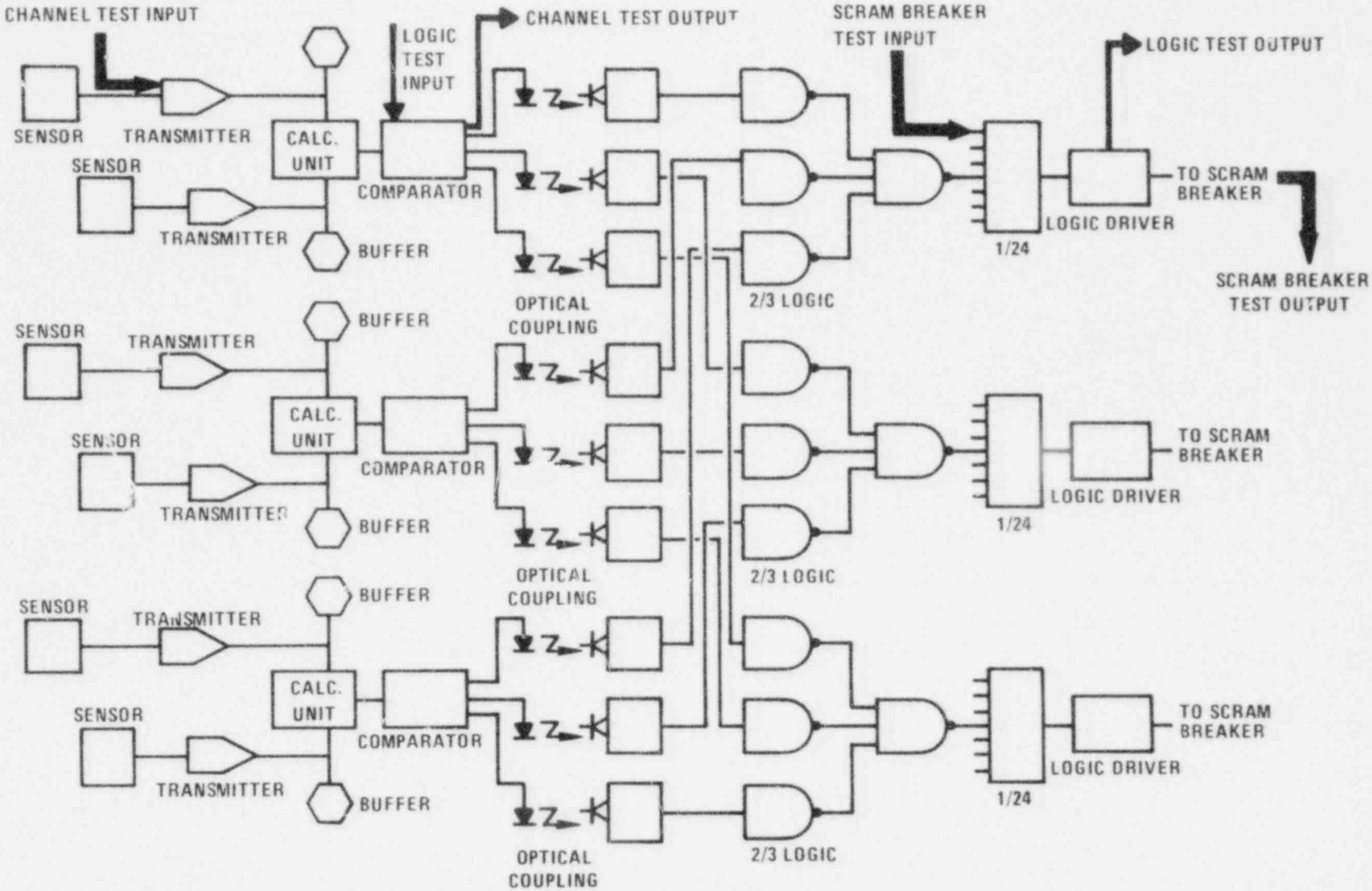

Figure 7.2-3. Typical Primary RSS Subsystem

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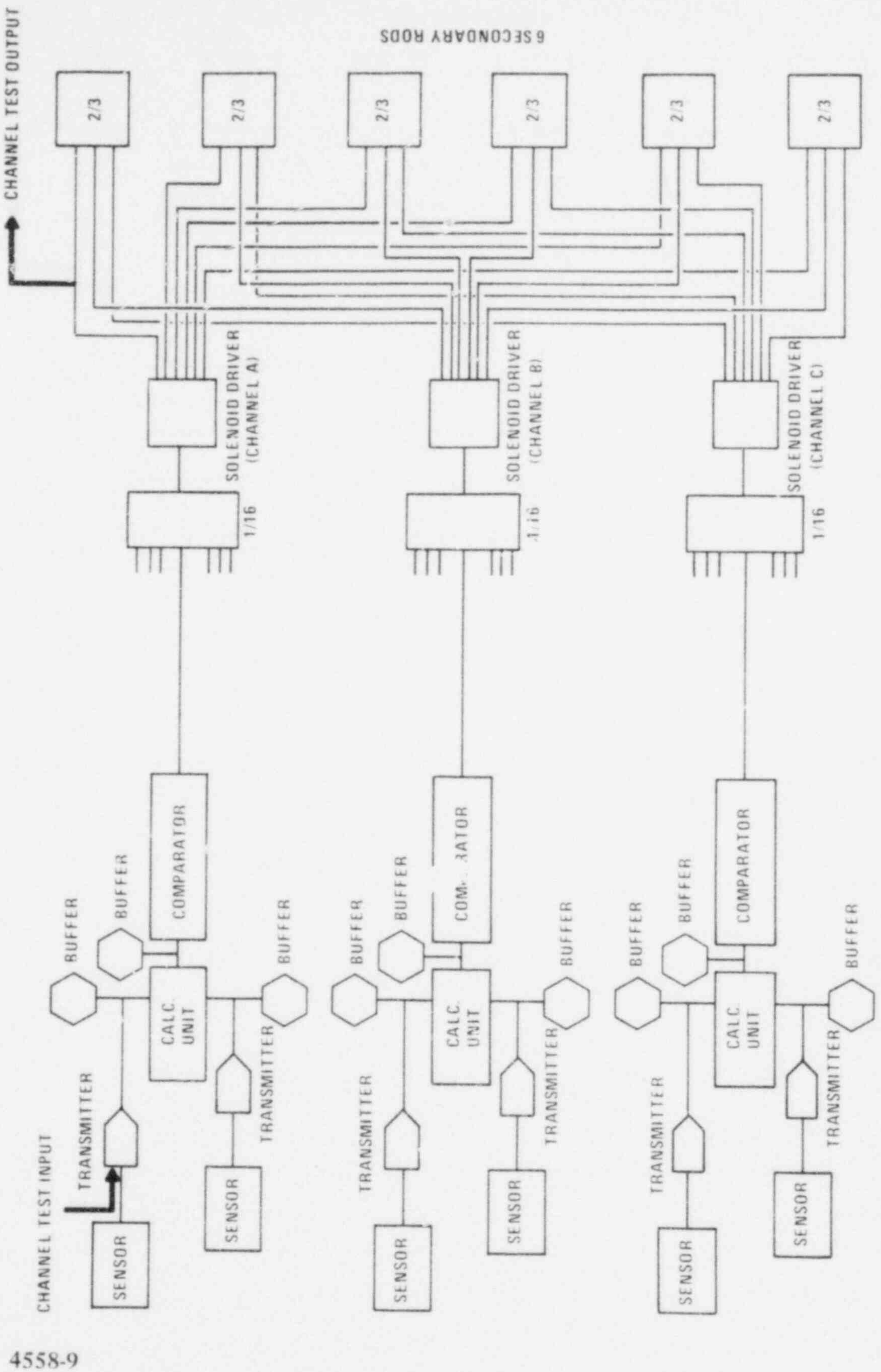


Figure 7.2-4. Typical Secondary RSS Subsystem

4558-9

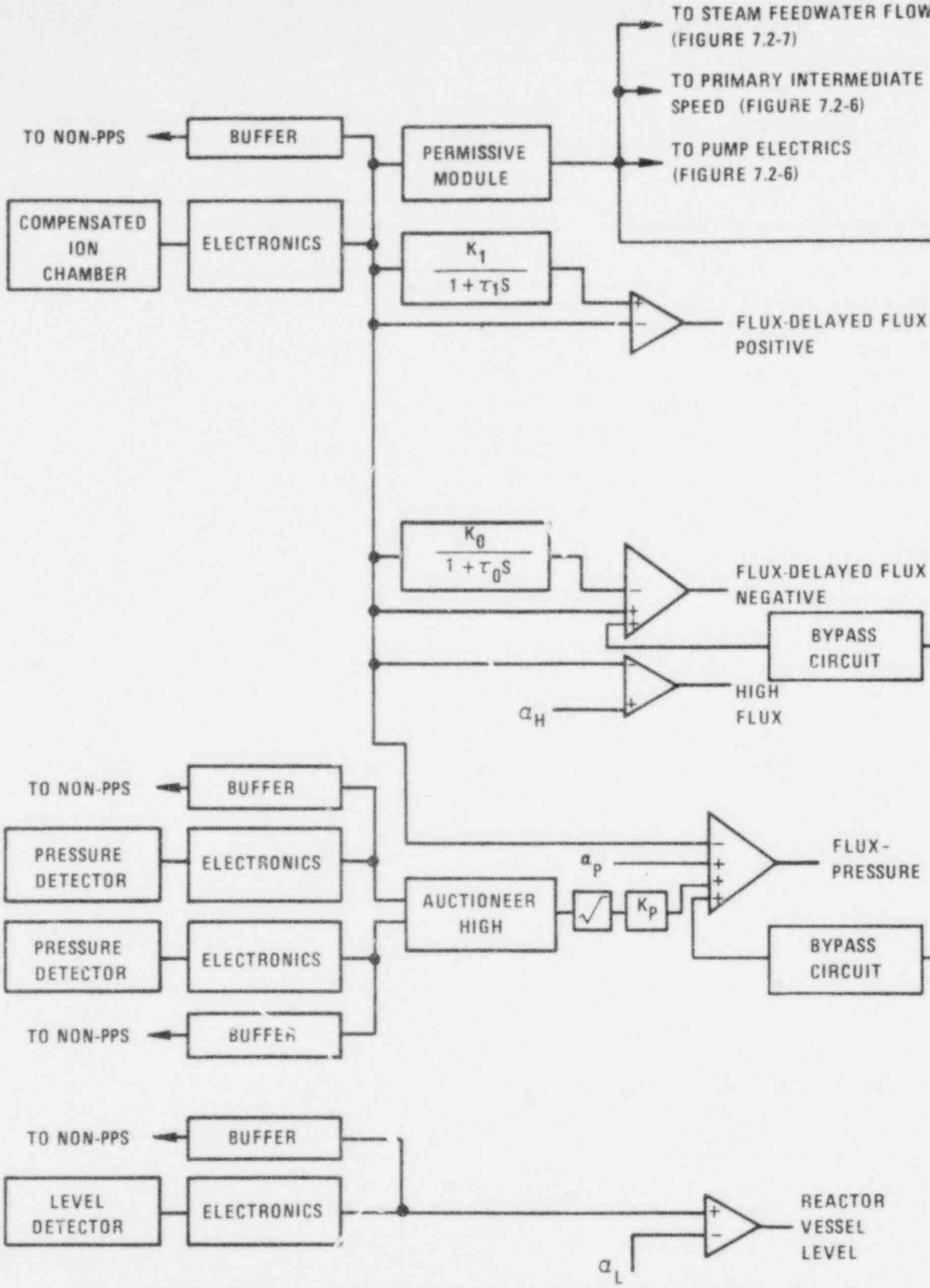


Figure 7.2-5. Functional Block Diagrams of the Flux-Delayed Flux, High Flux, Flux-Pressure, and Reactor Vessel Level Protective Subsystems, One Channel of Three Shown

4558-3

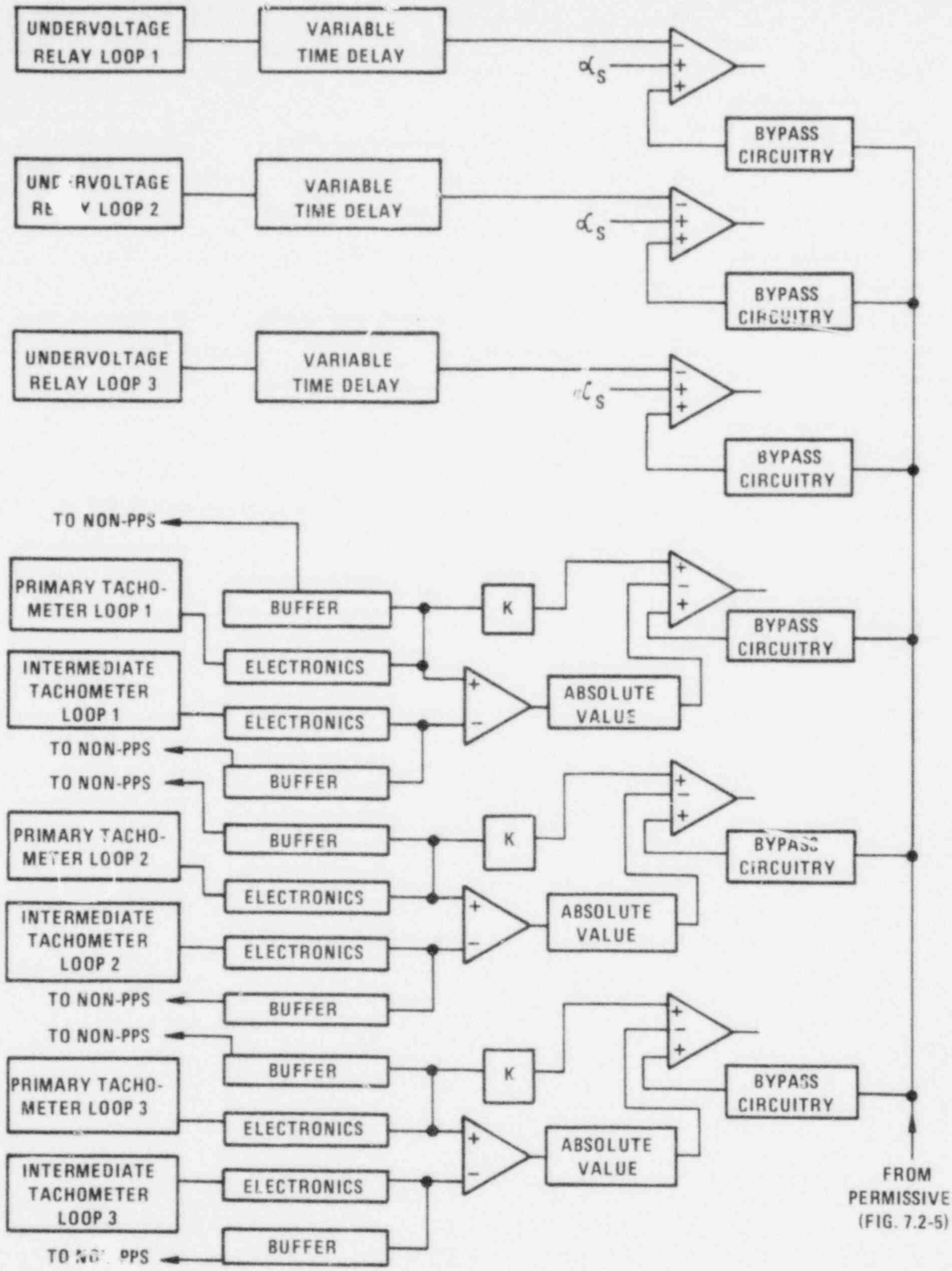


Figure 7.2-6. Functional Block Diagrams Of The Pump Electrics And Primary To Intermediate Speed Ratio Protective Subsystems. One Channel Of Three Is Shown

4558-6

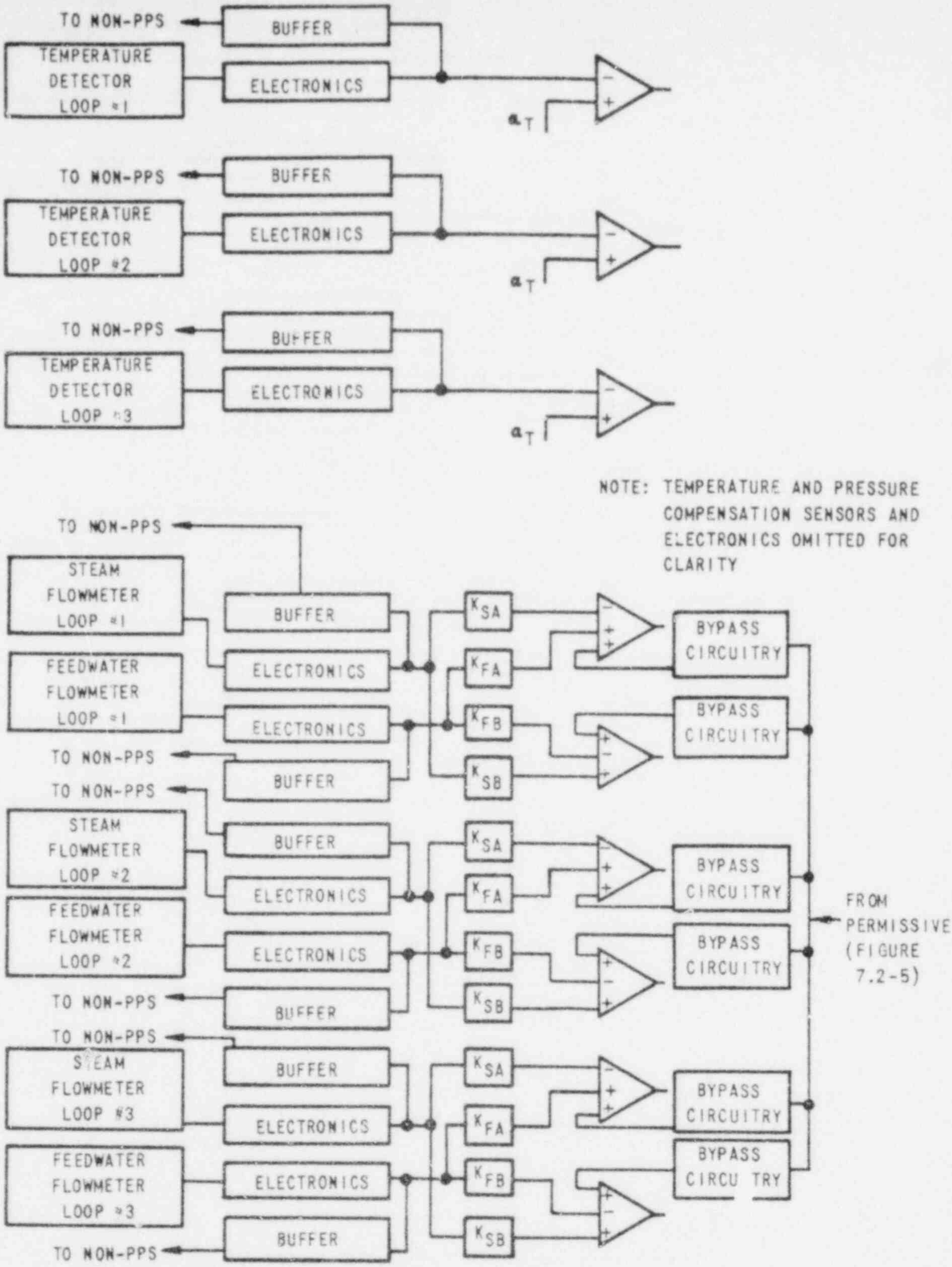


Figure 7.2-7. Functional Block Diagrams of the IHX Primary Outlet Temperature and Steam to Feedwater Flow Mismatch Protective Subsystem, One Channel of Three is Shown

6678-3

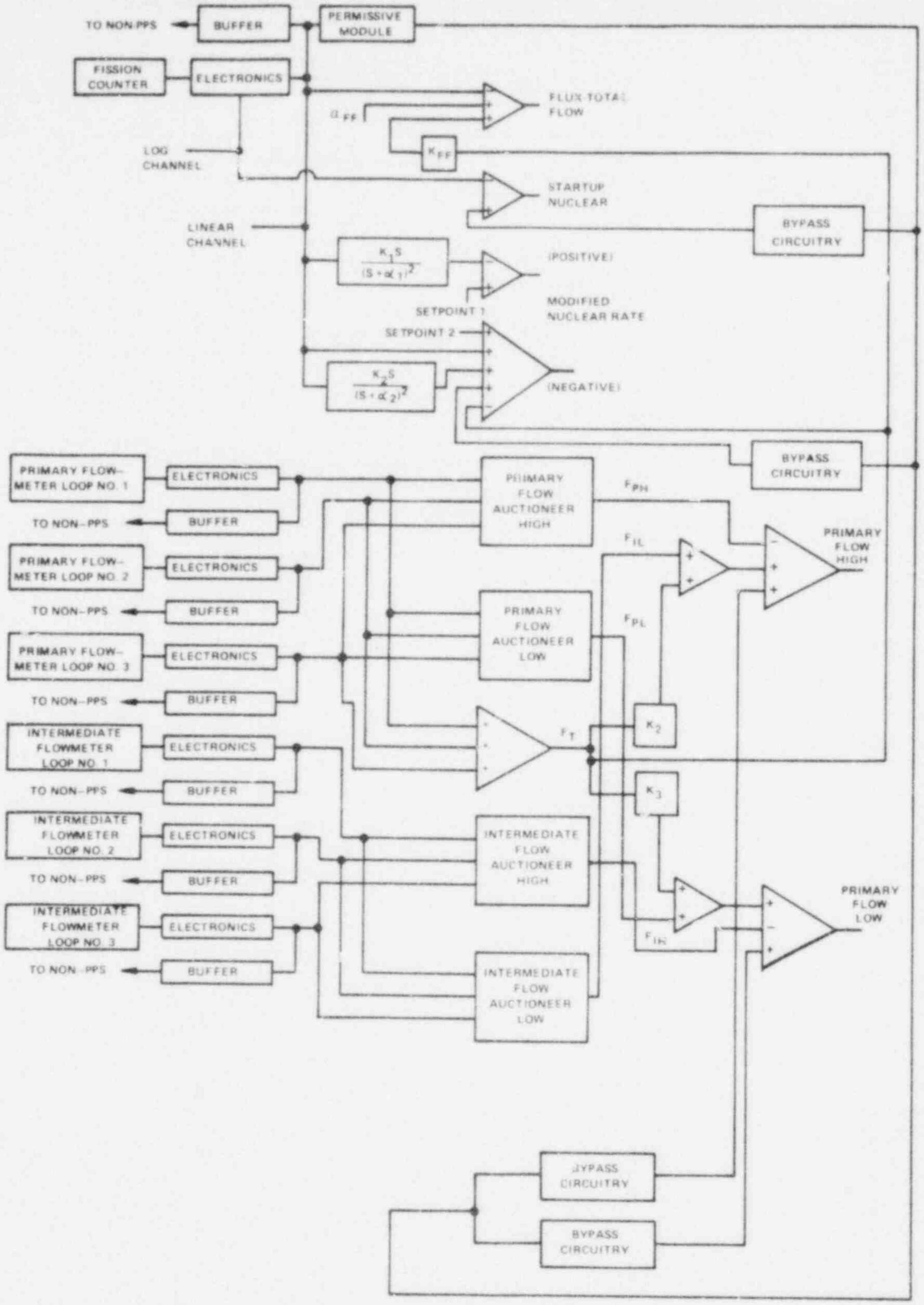


Figure 7.2-8. Functional Block Diagram of the Flux-Total Flow, Startup Nuclear, Modified Nuclear Rate, and Primary to Intermediate Flow Rate Protective Subsystems. One Channel of Three is Shown

4558-1

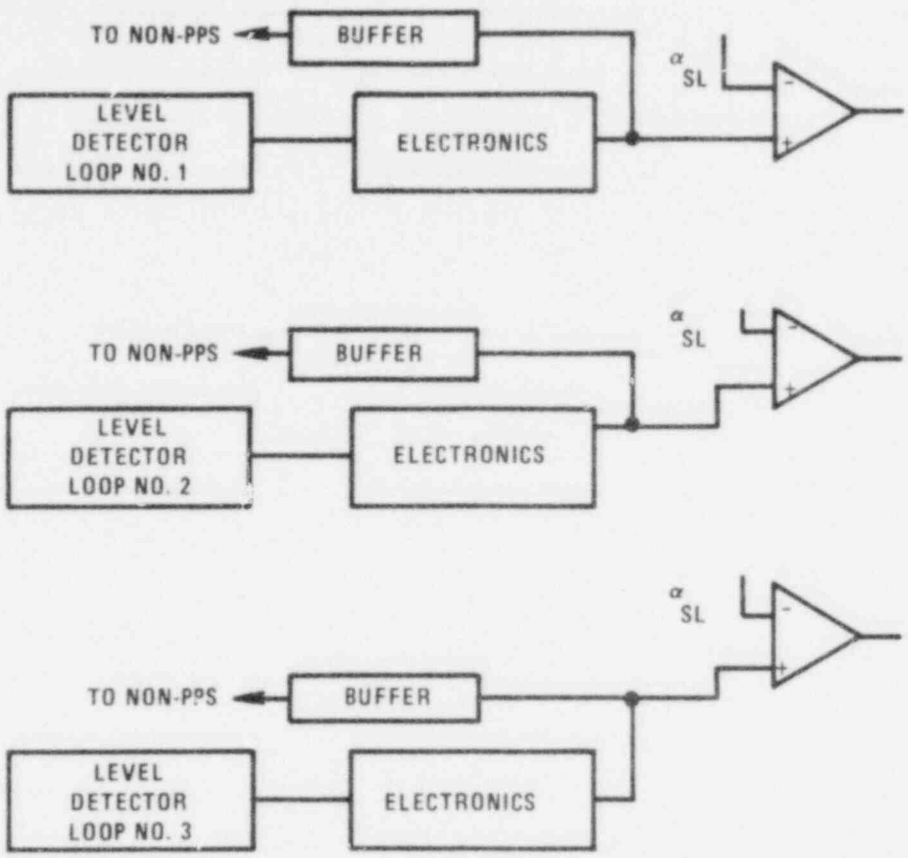
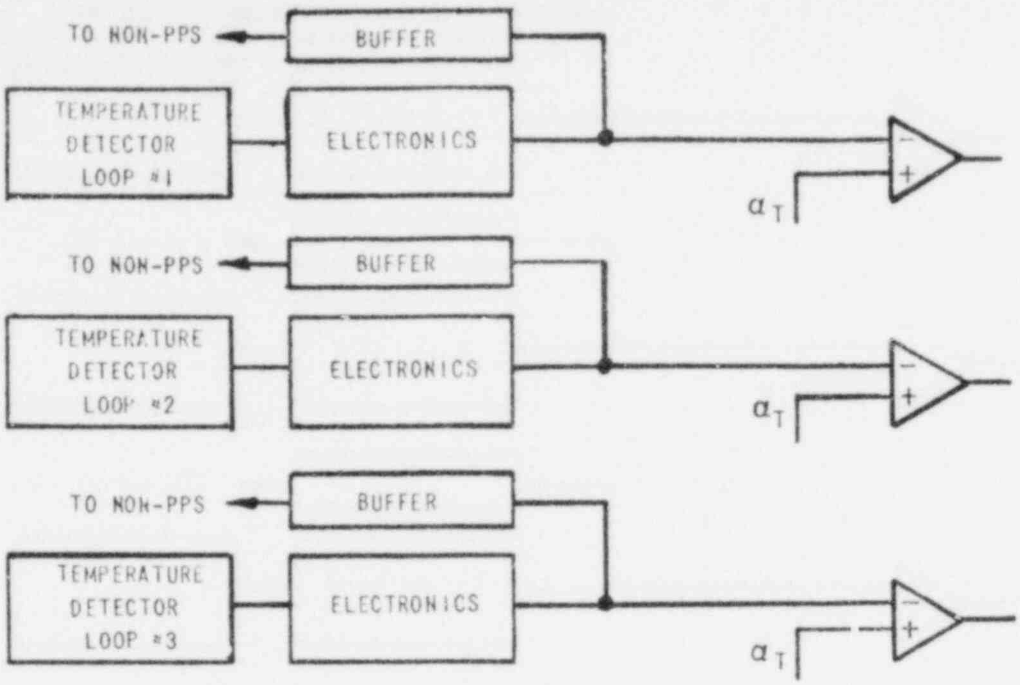
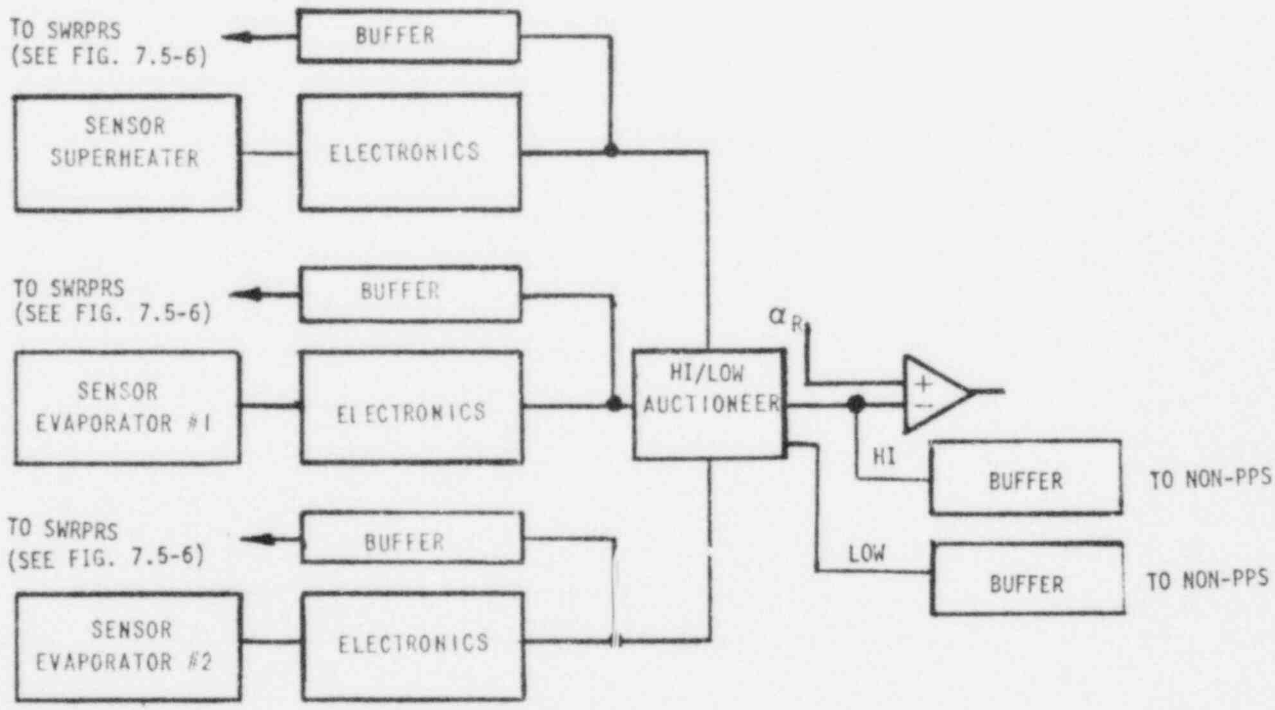


Figure 7.2-9. Functional Block Diagrams of the Steam Drum Level Subsystem. One Channel of Three is Shown.

4558-5



EVAPORATOR OUTLET TEMPERATURE



SODIUM WATER REACTION PROTECTIVE SUBSYSTEM

Figure 7.2-10 Functional Block Diagrams of the Evaporator Outlet Sodium Temperature and Sodium Water Reaction Protective Subsystems, One Channel of Three is Shown

6678-4

7.3 ENGINEERED SAFETY FEATURE INSTRUMENTATION AND CONTROL

The initiation of containment isolation is the only engineered safety feature identified which requires description in this section. Other engineered safety features are identified in Section 6.1. Table 6.1-1 lists these features and the sections in which they are discussed. The valves and associated characteristics are described in Section 6.2.4. Accident analyses are presented in Section 15 for the postulated events requiring containment isolation. The material below describes the design basis for the system, the implementation of the instrumentation and controls, and the evaluation of the capability of the system with respect to the General Design Criteria and applicable IEEE and RDT Standards and Regulatory Guides.

7.3.1 Containment Isolation System

7.3.1.1 System Description

57 The Containment Isolation System (CIS) is comprised of redundant instrumentation which senses the need for closure of certain valves in lines directly connected to the containment atmosphere, logic to initiate closure of the valves, manual initiation equipment, and the valves which are described in Section 6.2.4. Figure 7.3-1 shows a block diagram of the system. The Containment Isolation System is designed for automatic activation of these valves in lines directly connected to the containment atmosphere and valves which require closure in less than 10 minutes to remain within limits. Where closure time is not required in less than 10 minutes, manual actuation is provided. Radiation sensors are provided in two areas: the exhaust duct of the containment ventilation and the head access area. Three independent, redundant measurements are provided at each location. The sensor output is conditioned by the electronics and transmitted to the comparators where it is compared with a setpoint. If the signal is greater than the setpoint, a comparator trip is initiated. The logic for the automatic containment isolation is functionally identical to that used for the Primary Reactor Shutdown System. The comparator output is optically
44 coupled to the logic. Within the logic, the two comparator outputs (one from the head access area, the other from the exhaust duct of the containment ventilation) are combined to feed 2/3 coincidence modules. Within the 2/3 modules, the three independent channels are combined. Any two in the tripped state results in closure of the isolation valves. One of the logic trains drives the in-containment automatic valves. The other logic train drives the ex-containment automatic valves. Figure 7.3-2 is a
44 Logic Diagram of the system. 1

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Each detector has a check source which is used for test. In addition to the comparators and electronics, buffers and power supplies are provided for each channel. No permissives or bypasses are provided.

The provisions for on-line testing of electrical and mechanical equipment are included in the design. The test source is used to test the
44 instrument channel through the comparator. Signals are inserted prior to the 2/3 module to test the remainder of the logic and the closure of
57 the automatic valves. Since closure of the valves does not require reactor shutdown, this test can be performed during power operation.

57 Channel output meters are included to provide the operators with early indication of anomalous instrumentation performance. This equipment is not safety related.

There are no interlocks included in the design nor is there a necessity for sequencing the closure of the valves.

Manual initiation capability is provided locally at the CIS breaker cabinets and within the control room on the main control board to close all CIS
57 valves. Capability to close the CIS valves is provided if loss of access to control room is assumed.

For all containment isolation valves, position indicating lights are provided in the control room.

7.3.1.2 Design Basis Information

The CIS initiates and completes closure of the identified isolation valves to prevent the results of the faults identified in Table 7.3-1 from exceeding the specified limits. Note that these limits apply to the CIS when
57 the containment hatch is closed. Further, the design basis for the CIS is to provide appropriate design margin for postulated events and to assure that the radiological consequences of such events are within the guideline values of 10CFR100.

7.3.1.2.1 Containment Isolation System Subsystems

Containment Exhaust Radiation

The Containment Exhaust Radiation Subsystem initiates automatic containment isolation valve closure if the radiation level in the contain-
42 ment exhaust exceeds preset limits. This subsystem assures that events releasing activity within the containment do not result in exceeding the limits (10CFR20 or 10CFR100 as appropriate) for exposures in unrestricted areas. As shown in Figure 7.3-1, the subsystem includes 3 radiation sensors located in the conta[illegible]haust whose output is compared to a fixed setpoint. The subsyste[illegible]r bypassed. Worst case values for time response and repeatibilities will be used in the final analysis of the performance of this subsystem.

Head Access Area Radiation

The Head Access Area Radiation Subsystem initiates closure of the containment isolation valves in the event of large radiation releases in the head access area. Three radiation sensors are located in the head access area to provide early initiation and closure of the isolation valves to assure that releases from design basis events do not exceed the guideline values of 10CFR100.

7.3.1.2.2 Essential Performance Requirements

To implement the required isolation function within the specified limits, the CIS must meet the functional requirements specified below:

The closure time requirement for the inlet and exhaust isolation valves is 4 seconds with a three second or less detection time in the heating and ventilating system. A 10 second transport time from sensing point to the valve exists (see Section 15.1.1). The 3 seconds includes
43 sensor time response, comparator and logic time delays. 30

The CIS is designed to meet these requirements for the
57 environmental conditions described in Section 7.2.1.

7.3.2 Analysis

The design of the CIS provides the necessary functional performance and design features to meet the requirements of the appropriate standards specified in 7.1.2 as described below.

7.3.2.1 Functional Performance

The analyses in Sections 15.5 and 15.6 shows the results of the postulated fault conditions. These analyses assumed a closed containment where the events occurred with the containment hatch closed. For the limiting event, primary drain tank fire during maintenance, scoping analyses have been performed to determine the required closure time of the containment isolation valves. For the primary drain tank fire, closure within 20 minutes is adequate. Further, analyses to determine the required closure time under postulated accident conditions have been performed and are discussed in Section 15.1.1. These analyses are used to determine the available design margin. The results of this assumed condition do not exceed the guideline values of 10CFR100 if the main exhaust and inlet valves are closed within 4 seconds assuming the normal air transport time from the detector to the
57 valve is 10 seconds or more, a 14,000 cfm normal ventilation rate.

57 Since the automatic CIS is designed to isolate within the above time response requirements, all of the design basis conditions are terminated within the necessary limits for the present design concept.

7.3.2.2 Design Features

57 The CIS instrumentation, controls and actuators are designed to meet the requirements of IEEE-279-1971 and RDT Standard C16-1T, Dec. 1969. The analyses of compliance with these are summarized below.

Single Failure

No single failure within the CIS nor removal from service of any component or channel will prevent protective action when required. There are three independent instrument channels for each necessary measurement,
44 two independent 2/3 logic trains, and two independent actuators provided (as
57 shown in Figure 7.3-1).

Bypasses

No bypasses are provided.

Multiple Setpoints

Multiple setpoints are not required.

Completion of Protective Action

The automatic CIS is designed so that, once initiated, protective action at the system level must go to completion. Return to normal operation requires
57 manual reset of the CIS breakers by the operator.

Manual Initiation

The CIS includes means for manual initiation of containment isolation at the system level. No single failure will prevent manual
57 initiation of the containment isolation action.

Control and Protection Interaction

There are no shared components between the control system and the CIS.

The provisions for access, information read-out, annunciation of trips, and periodic testing are as specified for the Reactor Shutdown System in Section 7.2.2.

TABLE 7.3-1

CONTAINMENT ISOLATION SYSTEM DESIGN BASIS

Event	Applicable Federal Regulation	Limit
Anticipated Fault	10CFR20 § 105	<2 millirem in any one hour
No examples of anticipated faults which lead to release of activity have been identified.		<100 millirem in any one week
Unlikely Fault	10CFR20 § 403b	<5 rem in any two hours
No examples are presently identified for the automatic containment isolation system design basis.		
Extremely Unlikely Faults & Design Margin*	10CFR100	<25 rem in any two hours
Examples include major sodium fires		<300 rem iodine doses in the thyroid in any two hours
		<75 rem to the lung
44		<150 rem to the bone

*The design basis for the CIS includes limiting the results of postulated accidents within the guideline values of 10CFR100. See Section 15.1.1.

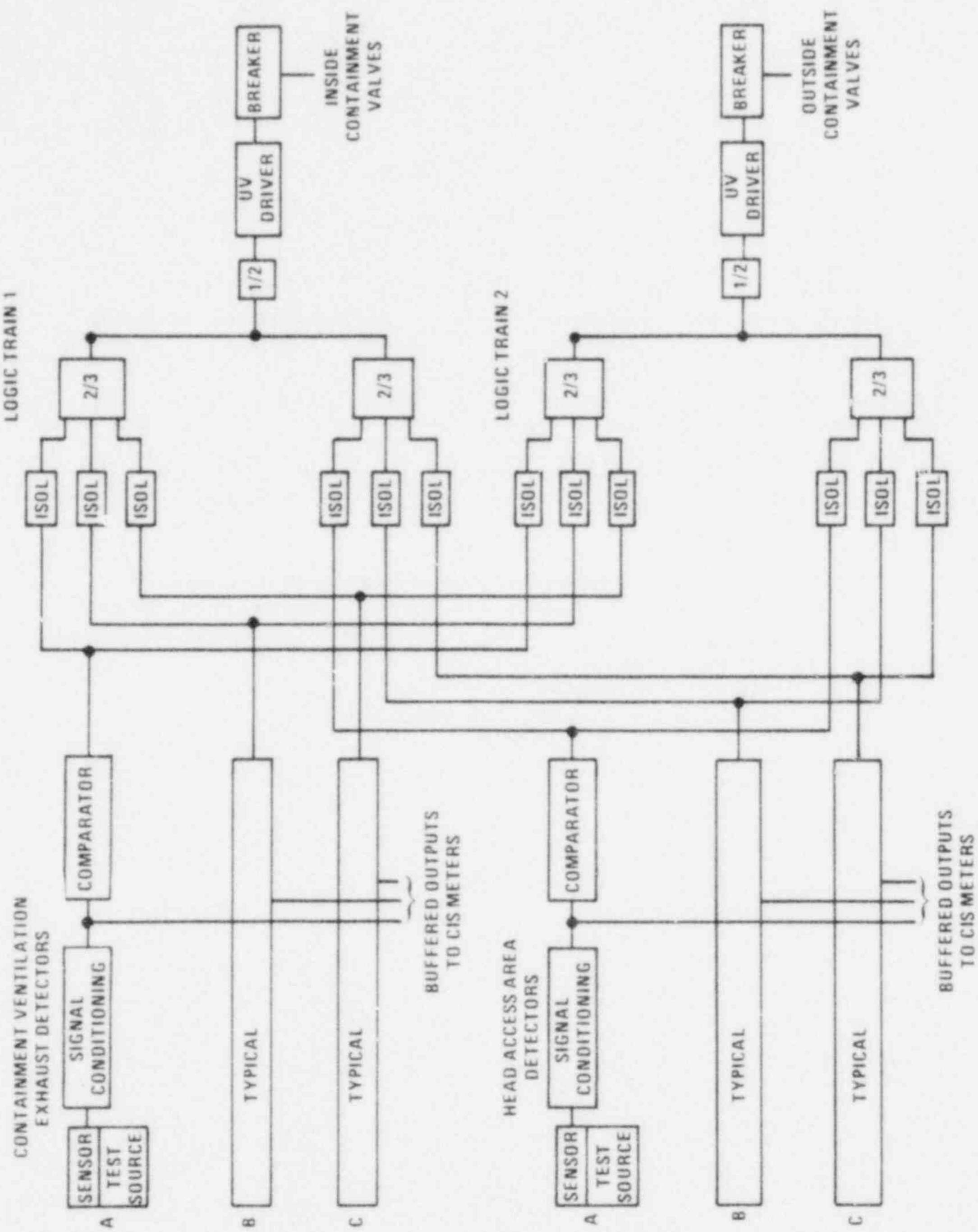

Figure 7.3-1. Containment Isolation System Block Diagram

4518-4

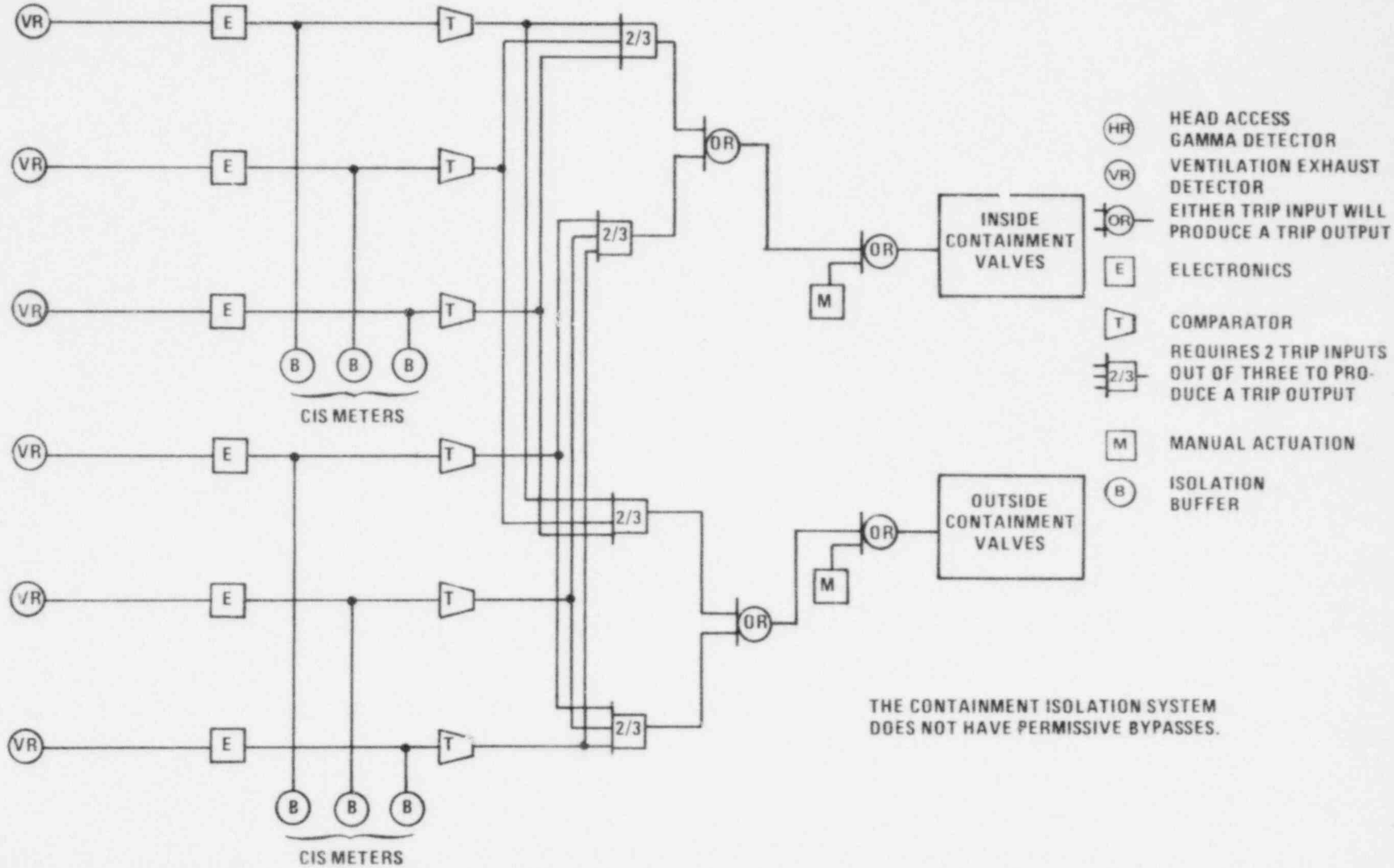

Figure 7.3-2. Containment Isolation System Logic Diagram

7.4.3 Remote Shutdown System

A Remote Shutdown System is provided. It consists of the following provisions:

Remote local control of:

- o Steam Generator Auxiliary Heat Removal System (SGAHRS)
- o Diesel Generator System
- o EVST Cooling Systems

Remote Shutdown Panel Instrumentation:

- o Primary and Intermediate Hot Leg Temperatures
- o Primary and Intermediate Cold Leg Temperatures
- o Primary and Intermediate Pump Speeds
- o Steam Drum Pressure, Temperature and Level
- o Steam Flow
- o Reactor Sodium Level
- o Diesel Generator Power Output, Current, Voltage Frequency, RPM, Fuel Oil Level

The Remote Shutdown System also includes the following features:

- o The SGAHRS and Diesel Generator System are provided with a local transfer switch which transfers controls from the control room to the local stations
- o An isolation switch is provided at the local station which transfers signals and isolates power from the control room for all necessary non redundant instrumentation.
- o Transfer of instrumentation or control to the local control station will not be possible until the local transfer switch is placed in the "local operating" position, resulting in actuating an alarm in the control room.
- o An independent soundpowered communications network is provided between the local safe shutdown control stations
- o All PPS signals necessary to remote shutdown control are isolated from the control room.

The need for remote shutdown (i.e., control room evacuation) is assumed not to occur simultaneously with the recovery from any other condition except the loss of off-site power. The Remote Shutdown Panel is
57 located in cell 271 of the SGB, adjacent to the SGAHRS control panels.

7.5 INSTRUMENTATION AND MONITORING SYSTEM

The instrumentation and monitoring systems included in this Section are the Flux Monitoring System, the Heat Transport Instrumentation System, the Reactor and Vessel Instrumentation System, the Fuel Failure Monitoring System, the Leak Detection Instrumentation System, and the Sodium-Water Reactor Pressure Relief System. Table 7.5-1 lists the measured parameters and instrumentation provided by these systems. The instrumentation which is safety related as defined in Section 3.2.1 is identified with an asterisk in column 2 of Table 7.5-1. Instrumentation and monitoring for TLTM parameters not included in the design basis are also discussed. These include containment hydrogen monitoring and containment vessel temperature and pressure monitoring. 25

7.5.1 Flux Monitoring System

The objective of the Flux Monitoring System (FMS) is to provide indications and electrical signals proportional to reactor power for reactor plant control and protection. The FMS meets its objective by means of neutron measuring instrumentation comprised of sensors and signal conditioning equipment which provide indications and signals for conditions of reactor shutdown, startup and full power operation.

Neutron sensors located around the periphery of the reactor guard vessel sense thermalized reactor leakage flux which is proportional to the reactor flux and thus to reactor power. Signals from the sensors are conditioned and then used to do the following:

- Determine the flux status of the reactor from shutdown through startup and all power levels.
- Provide signals to the Plant Protection System (PPS) to initiate reactor protective trips.
- Provide signals to the Plant Control System (PCS) for reactor and plant control.
- Provide neutron flux information for display, annunciation and recording.

A block diagram of the FM System is provided in Figure 7.5-1.

7.5.1.1 Design Description

The Neutron Flux Monitoring System provides three ranges of instrumentation: Source Range, Wide Range and Power Range. Each range of instrumentation is provided in three identical channels comprised of a detector, preamplifier (source and wide ranges), junction box (power range) and signal conditioning equipment. The Flux Monitoring System measures neutron flux proportional to reactor power over a span or more than ten decades from shutdown to above full power and provides indications and electrical outputs for plant protection, plant control, data handling and display, recording, and annunciation.

The flux detectors will be within the reactor cavity and contained in thimbles positioned approximately 120° apart on the periphery of the guard vessel. Each detector will be separately installed in a dry thimble which extends from the head compartment floor to below the reactor midplane to
57 position the detectors with the center of their active volumes
at or near the core midplane. The thimbles will be surrounded by neutron thermalizing blocks and provided with gamma shielding at the detector positions to provide suitable gamma and thermal neutron environments for the detectors.

Preamplifiers and junction boxes for the system will be located in the head access area.

The signal processing equipment will be located in the control building and will be contained in equipment racks. Within these racks, there will be drawers containing the electronic signal conditioning equipment packaged such that each channel or range within a channel (in the case of the wide range channels) will occupy one drawer. The drawers will be mounted on slides and will be capable of being withdrawn for alignment and maintenance. The front of each drawer will have meters, switches and adjustments for alignment, tests and monitoring of the channel. The arrangement of drawers in the racks will satisfy the separation requirements for redundant PPS channels. (See Section 7.1.2 and 7.2.2)

Remote meters and switches will be provided to the reactor plant operator to permit him to read the following parameters for each channel: Source Range Channel logarithmic level and rate, logarithmic percent power and rate for Wide Range counting and Mean Square Voltage (MSV) channels, linear percent power for Wide Range and Power Range. Linear and Logarithmic Source Range level will be provided on the IVTM control console. Audio Source Range count rate will be provided in the control room and at the IVTM control
57 console. Logarithmic source range count rate will be provided in the Refueling Communication Center.

The Flux Monitoring System detectors, cabling and signal conditioning equipment will be installed so as to preserve the separation requirements for redundant PPS channels (see Section 7.1.2 and 7.2.2).

Figure 7.5-2 presents the flux level coverage of the FMS instrumentation ranges based on the neutron flux at the detectors. The instrumentation ranges are shown to overlap so as to provide continuous indication from shutdown to more than full power.

7.5.1.1.1 Source Range

The design bases functions and operational requirements as stated in Section 4.3.2.1.5 are accomplished by incorporation of the design features described in the following paragraphs.

Each of the three Source Range channels will use a high sensitivity, dual
57 section BF_3 filled proportional counter assembly to sense thermal neutron flux
54 over the range of approximately 0.1 to 10^4 neutrons per cm^2 per second (nv).
This corresponds to a range from the fully shutdown fresh core conditions 1

to reactor low power (a few Kwt) operation. The sensitivity of the
54 BF_3 counters will be maintained at a minimum of 40 cps/thermal equivalent
57 nv at a gamma background count rate of approximately 1 cps. Shielding will be provided to limit the gamma dose rate at the detectors due to prompt gammas from the core and from decay of activation isotopes in the sodium coolant and structural materials to less than 100R/hr. This will apply over the operating range of the channel at all times when operation is required. The BF_3 counters and associated cables will be designed to operate under normal environmental conditions of 170°F maximum and atmospheric pressure and under emergency conditions of 260°F maximum and 12 psig during con-
57 tainment testing. The design life goal for the counters is 3 full power years based on a total fluence of 10^{18} nvt. In order to achieve this lifetime without retracting the counters, the operating voltage will be removed and the anode of the counter shorted to ground through an appropriate resistance when the flux level is above the operating range of the channel.

The output pulses from the detector assembly will be amplified by a dual section pre-amplifier mounted in the head access area and routed to the signal conditioning equipment in the control room racks. This equipment will consist of dual amplifiers with pulse shaping networks to minimize gamma pulse pile-up followed by discriminators to reject amplifier noise and gamma pulses.
57 The summed output of the discriminators will drive a count rate circuit followed by a scaling amplifier which will produce an analog output signal proportional to the logarithm of the input pulse rate. The analog signal will be displayed on a log scale countrate level meter calibrated from 1 to 10^6 cps. The accuracy and repeatability of the channel will be $\pm$ 3% and $\pm$ 1%, respectively, of linear equivalent full scale under the 17
worst case environmental conditions of temperature, pressure and input power fluctuations to be encountered while channel operation is required. The response time of the count rate circuits will vary with count rate, being on the order of 30 seconds at the lowest expected average count rate of approximately 4 cps which will occur during refueling beginning of core life conditions. The 1σ statistical counting error will be approximately $\pm$7%
54 at the 4 cps rate. The response time will decrease with increasing count rate to approximately 0.1 seconds at a count rate of 3×10^5 cps produced
57 at a few kilowatts. The analog output signal of the scaling amplifier also drives a differentiating circuit to produce a rate of change of level signal which is displayed on a linear scale meter from -1 to 0 to +3 decades per minute. Power supplies mounted in the equipment 17
drawers will provide detector excitation high voltage and low voltage instrument power. Individual scaler timers will be provided in each channel which will be driven from a buffered output of the discriminator which precedes the countrate circuit. The scaler timers will be programmable to count for a short preset time, stop, transfer the accumulated count to temporary storage for PDH&DS readout, immediately restart the count and repeat the
57 sequence to provide an accurate record of the count rate trace to implement the inverse kinetics rod drop technique used to establish control rod worth. The scaler-timers will also provide for counting of signal pulses for longer time
57 periods to accurately determine the system calibration constant which relates subcritical reactivity to count rate. 17

A visual/audio linear count rate circuit will be provided for operations at shutdown. This circuit will be provided with a switch 1

through which it can be connected to the summed discriminator output of any of
57 the three Source Range Channels, one at a time. The visual portion provides expanded scale indication of the flux level. The audio portion provides tone bursts whose rate changes (increase and decrease) with increasing and decreasing flux level.

Each channel will be provided with built-in counting level and rate of change calibration circuits for channel alignment and pre-operational testing to assure that the instrumentation circuitry is functioning properly.

The Source Range Channels are used to monitor the shutdown and startup flux only, no signals are provided to the PPS. Bistable comparators in each channel will activate individual annunciators in the control room to provide alarms if the specified minimum shutdown reactivity is exceeded during re-fueling or if detector excitation or instrument power is lost in any channel. To improve the operational effectiveness of the source range shutdown reactivity monitoring function, the reactivity alarm is inhibited during core assembly movement which could cause an erroneous alarm. The inhibit circuit is continuously monitored for proper operation and any malfunction of the inhibit circuit will activate a separate control room annunciator. An alarm will also be provided in the control room via the PDH&DS if any one channel deviates from the other two by a preset amount. The shutdown margin alarm bistables and the PDH&DS channel deviation alarm will operate off of the buffered outputs of the log count rate scaling amplifiers. The loss of detector excitation and instrument power alarm bistables will operate off of H.V. sense signals developed by a resistor divider network at the preamplifier H.V. outputs to the detector assemblies and instrument power supply output voltages, respectively.

The operation of the source range channels will be under manual control of the reactor operators. These channels will be in operation continuously during reactor refueling and other shutdown conditions. During reactor startup, the source range channels will be used to monitor the core flux level until a predetermined overlap between the source range channels and wide range log count rate channels is obtained. The high voltage will then be removed from the source range detectors by the operator. He will do this by actuating a momentary contact pushbutton switch on the main control panel. When actuated, this switch will remotely interrupt the input power to the detector high voltage supplies, and ground the detector anodes through an appropriate resistor. The relay control circuit established by the momentary contact of this switch will also illuminate a green indicator light located adjacent to the switch. Upon reactor shutdown, the operator will interrupt the relay control circuit by actuating a second momentary contact pushbutton switch when the wide range log counting channels indications fall to a pre-determined level within the source range channels operating range. This action will remove the ground from the detector anodes, restore the detector excitation voltage, extinguish the green indicator light and illuminate a red indicator light. Inadvertent removal of, or failure to restore the operating voltage when needed will be prevented by procedural control, utilizing the separate on/off switches and with monitoring through the color of the illuminated indi-
57 cator light.

Output of the detector will be processed in the signal conditioning equipment to provide linear indication of percent power and linear output signals for plant protection, plant control, data logging and annunciators. This instrumentation operates over the same flux span as the direct current circuitry of the wide range instrumentation to add redundancy and diversity to
57 the power range measurements.

The output of this instrument will be linear to at least 140 percent power and will have no foldover to as high a power level as required by the worst case power overshoot for which protection must be provided.

Built-in test circuits and controls will be provided to permit testing and aligning the equipment during plant operation and plant shutdown.

7.5.1.2 Design Analysis

The Flux Monitoring System will be a functional subsystem of the Plant Protection System and will meet the safety related channel performance and reliability requirements of the CRBRP General Design Criteria, RDT
27 Standard C16-IT, Dec. 1969, IEEE Standard 279-1971, applicable AEC Regulatory Guides and other appropriate criteria and standards by complying with the applicable design requirements delineated in Section 7.1.2.

The FMS meets CRBRP General Design Criterion 21, which is applicable to instrumentation for normal and accident conditions, as follows:

- The shutdown flux level will be monitored at all times while fuel is in the core so as to provide safe operational control of the reactor during low power, normal shutdown, refueling and shutdown maintenance operations.

- The reactor flux will be continuously monitored during operation from shutdown to full power operation (i.e., overlap will exist between cascaded channels so that all power levels can be monitored without a gap in range).

- Reactor power operations will be continuously monitored with linear response to power up to at least 140% full power. Significant positive response will be provided to as high a power level as required by the worst case power overshoot for which protection must be provided. This positive response will be provided for as long as is required to seal in the scram trip.

57
- The FMS instrument response times will meet the requirements of the Reactor Shutdown and Plant Control Systems.

- Indication of reactor power level and rate of change of power level will be provided to the operator. One set of meters and a selector switch will be provided for each range of instrumentation permitting the operator to select one channel at a time to be displayed on the related meters. Five power level meters, five selector switches and three rate of change of power meters will be provided for the operator.

- The source range level will be indicated in logarithmic counts per second and rate of change of level in decades per minute. Linear count rate will be provided at shutdown at the refueling console and at the FMS system panels in the control room. Audible count
57 rate indication will be provided in the control room and in containment at the refueling console.

- The wide ranges will be indicated as follows:

  Counting channels - Logarithmic percent power level and decades per minute rate of change.

  MSV channels - Logarithmic percent power level and decades per minute rate of change.

  DC channels - Linear percent power level.

- The power range will be indicated in linear percent power level.

Preliminary Failure Mode and Effect Analysis results applicable to the FMS have been determined in an analysis of possible failure modes and their effects on the Reactor Shutdown System performance and are presented in Tables C.S. 1-4 and C.S. 1-5.

41

7.5.2 Heat Transport Instrumentation System

7.5.2.1 Description

The Heat Transport Instrumentation System provides sensors, associated signal conditioning equipment and controls other than Plant Control, for the Primary Heat Transport, the Intermediate Heat Transport and the Steam Generator. The signals from the sensors are conditioned and then supplied to the Reactor Shutdown System logic, the Plant Control System, the Data Handling and Display System, and the Plant Annunciator System as appropriate. The location of the Heat Transport Instrumentation is provided in Figures 5.1-2 and 5.1-4 (P&ID's).

7.5.2.1.1 Primary and Intermediate Sodium Loops

Reactor Inlet Pressure

The measurement is made by pressure elements installed in the cold leg of the primary loop piping just before it enters the reactor vessel. NaK filled capillaries from the pressure elements are connected to pressure transducers which develop electrical signals proportional to the pressure. These pressure transducers provide a secondary boundary if the bellows in the pressure elements should fail.

Each pressure transducer consists of a diaphragm which moves in response to pressure changes in the NaK filled capillary, a strain gage which converts diaphragm motion to resistance change, and a bridge and amplifier to convert strain gage resistance change to a standard signal.

Since pressure element replacement requires plant shutdown, two pressure elements per loop are provided.

The signals from the six, two per loop, pressure measurements are transmitted to the control room in three separate isolated PPS channels for use in the Reactor Shutdown System logic. The Reactor Shutdown System supplies buffered signals to the DH & DS.

Primary and Intermediate Loop Flow

The flow measurements are made by a permanent magnet flowmeter located in the cold leg of each primary and intermediate loop (except
49 for intermediate loop 2, which has the PM flowmeter in the hot leg). The magnet assembly is in the shape of an inverted "U" which is suspended around the pipe. The magnet assembly is mounted rigidly to the building structure and is physically separated from the pipe.

Type K thermocouples are installed in the magnet structures to monitor the magnet temperatures. This permits temperature corrections to the flowmeter calibration. The signals from these thermocouples are routed to a local panel. Provisions will be made to permit periodic monitoring of the magnetic flux of the flowmeters without disassembly or entrance into HTS cell. This is also accomplished at the local panel.

Four independent pairs of 3/8" (approx.) electrodes are attached to the pipe. The electrodes are of the same composition as the pipe so that thermal potentials are not developed. Three pairs of electrodes are connected to the conditioning equipment. The fourth pair is available for a portable measuring instrument or as an installed spare.

Flexible mica, polyimide and fiberglass insulated cables in separate conduits to meet PPS separation requirements are used to bring the four signals from each flowmeter assembly out of the Heat Transport System cell. The signals are then routed to signal conditioning equipment.

From the signal conditioning equipment, the signals are sent to the control room for the Reactor Shutdown System logic which in turn supplies buffered signals to the PCS and the DH & DS.

IHX Primary Outlet Temperature

The IHX primary outlet temperature measurement is made by three Chromel/Alumel thermocouples per loop installed in thermowells in the elevated section of the HTS cold leg piping nearest the IHX primary outlet. The thermocouples are 1/8" insulated junction swaged to 1/16" at the tip to

7.5.3 Reactor and Vessel Instrumentation

7.5.3.1 Description

The Reactor and Vessel Instrumentation System includes all in-vessel temperature, sodium level and vibration sensors for instrumenting the reactor parameters required for the Reactor Shutdown System, PCS, surveillance and design verification. It also includes signal conditioning equipment needed to make the sensor signal usable in the systems receiving the signal.

Table 7.5-2 shows the in-vessel instruments provided, their location, their quantity and purpose.

7.5.3.1.1 Sodium Level

A total of five sodium level sensors are provided. All of these sensors are mounted in wells to provide the physical barrier maintaining the integrity of the primary loop closed system. The sensors are induction type probes continuously sensitive over their entire length. Four of the units, located approximately equally spaced on the top of the reactor, are short with a sensing range of from 6 inches above the operating level to 24 inches below. Three of these provide the level signals to the three Reactor shutdown system logic and are thus isolated from each other and from non-PPS equipment. The fourth is an installed spare unit providing a means of maintaining the three operating channels without a shutdown in the event of failure of one of them. The fifth level sensor is located close to one of the short units but provides a measuring range from 6 inches above the operating level to below the minimum safe sodium level. It has approximately fifteen feet of sensing length. The signal is supplied to the PCS for control room indication and is monitored at all times, including refueling.

7.5.3.1.2 Temperature

All in-vessel temperatures are sensed by chromel alumel, ungrounded, stainless steel sheathed thermocouples. Thirty wells are provided for thermocouples located in the sodium at the exit from the core. These thermocouples provide signals to the PCS. Additional wells are provided at the core exit (275), core periphery (2), and on parts of the upper internal structure (6) for thermocouples providing signals to the PDH & DS for surveillance and design verification.

7.5.3.1.3 Non-replaceable Instruments

Within the reactor vessel, four biaxial accelerometers are mounted on the upper internal structure so that they cannot be replaced. These sensors are not required to function beyond the first six months of operation although they are required to physically withstand the sodium environment for the life of the reactor. The signals provided by these sensors provide design verification information and the location of the sensors and their leads will not affect the safety of the plant.

7.5.3.2 Analysis

The in-vessel sensor mounting is designed to be operable during the combined stresses imposed by the reactor coolant velocity, vibration, pressure, environment, temperature, thermal shock, and radiation in order to provide operational lifetime that will not significantly affect the reactor availability. The sensors and their lead-out conductors are sufficiently rugged so that they will not be damaged during refueling and maintenance. Thermocouples used for measuring the sodium temperature are mounted to avoid close proximity to the structures so that the temperatures sensed will be that of the coolant and not be influenced by the structures.

57 The sodium level instruments, which are part of the Reactor Shutdown System, will comply with the PPS Design Requirements (see Sections 7.1.2 and 7.2.1). The design analysis for the Reactor Shutdown System applies (Section 7.2.2) and a Failure Mode and Effects analysis is performed as shown in Table C.S.1-1.

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7.5.4 Fuel Failure Monitoring System

The Fuel Failure Monitoring (FFM) System provides:

34 1. Equipment to detect occurrance of fuel or blanket cladding failure.

43

2. Equipment to locate failed fuel assemblies in the reactor core, and, to the extent practicable, failed radial blanket assemblies;

3. Equipment to characterize the failed pins as to burn up and other information, to permit correlation with core and blanket history, thus enhancing location capabilities, in particular for radial blanket.

34 4. Equipment to detect fuel or blanket failures involving exposure to and egress of fissile or fertile material to the sodium coolant.

The FFM System is comprised of several independently functioning parts, each providing information to the plant operations staff. This system does not provide control loop or reactor trip signals, but does supply information for surveillance, display and alarm purposes.

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57 • Pressure is monitored in the Reaction Products Separator Tank to alert the operator to off-normal conditions in the Reaction Products Vent System.

57 7.5.6.1.6 Sodium Dump Tank Instrumentation

A sodium dump tank is provided in each loop for drainage of sodium from the IHTS. Dump tank level is measured by inductive type level probes. Each tank is provided with two probes to meet the necessary range requirements. Each probe is connected to an excitation-conditioning module that provides local indication of sodium levels.

The conditioned signal from the wide-range probe is supplied to a trip unit, which provides a high-level alarm signal to the PAS. All level measurement channels provide inputs to the PDH&DS.

Surface thermocouples, which are part of the sodium dump tank trace heating system, are available for display as an indication of level and temperature. A pressure switch actuates an alarm and a rupture disc bursts to vent to atmosphere in case of excessive pressure buildup in the sodium dump tank.

57 7.5.6.1.7 Water Dump Tank Instrumentation

A water dump tank is provided in each loop to accept and store the water from the evaporators when rapid depressurization is required. Measured parameters for the water dump tank are level, pressure, and temperature.

Dump tank level is measured by a differential pressure transmitter that senses the difference in pressure caused by the variable height of water in the dump tank. The differential pressure signal is supplied to a local indicator, the PDH&DS, and a high-level alarm circuit that provides a signal to the PAS.

Pressure is measured on an appendage off the dump tank by a conventional pressure sensor-transmitter. The pressure signal is used for local indication, for the PDH&DS, and for high-pressure alarm to the PAS.

Dump tank temperature is monitored by chromel-alumel thermocouples attached to the surface of the tank. The signals are supplied to the multipoint temperature indicator.

40

7.5.6.2 Design Analysis

Because of the large increase in pressure from the formation of reaction products during a large sodium-water reaction, rupture disc operation is necessary to prevent excessive pressure surges in the Intermediate Heat Transport System and possible primary boundary rupture at the Intermediate Heat Exchanger. Reaction products vent line sensors are part of the Reactor Shutdown System and as such meet the requirements of the Plant Protection System (see Section 7.1.2 and 7.2.2). The initiation of isolation and dump of the water side of the steam generators, trip of the recirculation pumps and inerting of the steam generators normally follows after rupture disc operation in a large sodium-water reaction and is desirable from the operational standpoint of minimizing the time to recover from the incident. However, the initiation of these actions after rupture disc operation is not necessary from a safety standpoint to assure protection of the core or the safety of the public.

All SWRPRS equipment associated with isolation or dump of evaporator or superheater modules is designed to assure that no credible single event can disable more than one of the three redundant decay heat removal paths. All electronic equipment is designed to withstand the Safe Shutdown Earthquake. The mechanical equipment associated with the 3 steam generators is physically separated in 3 different steam generators bays. Electrical and pneumatic supplies are arranged such that a single failure disables at most one decay heat removal path. Where practicable, the preferred failure position for equipment is in a direction to assure the safe operation of the SGAHRS.

SWRPRS equipment whose failure could cause loss of decay heat removal capability of the SGAHRS is safety related. Any credible single failure in the SWRPRS can lead to the failure of at most one of the three decay heat removal loops. Since the three decay heat removal loops are redundant and independent, the SGAHRS will meet the single failure criterion and the adequacy of the decay heat removal system following a credible single failure in the SWRPRS is assured.

7.5.7 Containment Hydrogen Monitoring

The objective of Containment Hydrogen Monitoring is to provide indication in the Control Room of the hydrogen concentration in the upper levels of containment.

25 7.5.7.1 Design Description

The hydrogen instrumentation consists of two fully redundant and independent analyzer channels. The containment atmosphere is sampled through an entry filter located near the top of the RCB. The air samples are pumped to the analyzer which is located in the SGB and operates on the principle of thermal conductivity. From there signals go to the Control Room where the hydrogen concentration readout is provided. This instrument is also required to perform functions for events which lie beyond the design basis for the plant. This instrument is further discussed in
57 this capacity in Sections 2.1 and 3.3 of Reference 10b of PSAR Section 1.6.

7.5.8 Containment Vessel Temperature Monitoring

The objective of Containment Vessel Temperature Monitoring is to provide indication in the Control Room of the containment vessel
25 temperature.

7.5.8.1 Design Description

The temperature instrumentation consists of two fully redundant and independent channels. Each channel consists of eight thermocouples mounted at various locations on the inside of the containment wall, with each thermocouple providing a signal to conditioning instrumentation in the SGB. The instrumentation sends a signal to the Control Room where individual readout is provided. This instrument is also required to perform functions for events which lie beyond the design basis for the Plant. This instrument is further discussed in this capacity in Sections 2.1 and 2.2 of
57 Reference 10b of PSAR Section 1.6.

7.5.9 Containment Pressure Monitoring

The objective of the Containment Pressure Monitoring System is to provide indication in the Control Room of the pressure inside the containment above the operating floor.

7.5.9.1 Design Description

The pressure instrumentation consists of a pressure detector inside the containment vessel. Signals will be provided to the display
25 and alarm panel in the Control Room so that continuous readout will be provided to the plant operator. This instrument is also required to perform functions for events which lie beyond the design basis for the plant. This instrument is further discussed in this capacity in Sections
57 2.1 and 2.2 of Reference 10b of PSAR Section 1.6.

7.5.10 Containment Atmosphere Temperature

The objective of the Containment Atmosphere Temperature Monitoring System is to provide indication in the Control Room of the atmosphere temperature inside the containment building.

7.5.10.1 Design Description

The temperature instrumentation consists of two fully redundant and independent channels. Each channel consists of two thermocouples mounted on the RCB dome, with each thermocouple providing a signal to conditioning instrumentation in the SGB. The instrumentation sends a signal to the Control Room where individual readout is provided. This instrument is also required to perform functions for events which lie beyond the design basis for the plant. This instrument is further discussed in this capacity in Section 2.1 and 2.2 of Reference 10b of
57 PSAR Section 1.6.

57 7.5.11 Post Accident Monitoring

Table 7.5-4 provides a listing of those parameters which are
monitored to assure the plant is maintained in a safe shutdown
status. Equipment to condition, display, and record the instrument
50 signals is provided in the Control Room. The instruments which
serve the Post Accident Monitoring function are included in those discussed
49 in Sections 7.4.1, 7.5.2, 7.5.3, 7.5.8, 7.5.9, and 7.6.3. The functions
of these instruments corresponding to the parameter of Table 7.5-4
50 are described in the following paragraphs.

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The reactor sodium level is monitored to enable the operator to determine whether manual action is necessary to mitigate conditions which may cause low Reactor Vessel sodium levels. The operator may also use the reactor sodium level to determine conditions in the primary sodium systems such as the volume of primary coolant leakage.

The IHX (Intermediate Heat Exchanger) inlet and outlets temperatures are monitored to verify the heat transfer from PHTS to the IHTS. These monitors allow the operator to take manual actions necessary to assure decay heat removal from the reactor is maintained within design limits.

The DHRS cold leg temperature is monitored to assess the systems' decay heat removal performance so the operator may take manual actions necessary to achieve or maintain core temperature at a safe level.

Reactor Containment Building pressure and temperature monitors are provided to follow pressure and temperature changes due to an accident, and provide confidence that the accident consequences are within the capability of the Containment Vessel. Also, these monitors can be used to detect conditions beyond the design basis as discussed
57 in Reference 10b of Section 1.6.

The PWST (Protected Water Storage Tank) water level is monitored so the operator is assured of the adequacy of that water supply to remove reactor decay heat for an extended period of time. In addition, the level monitor provides indication of the long term need to refill the PWST or draw auxiliary feedwater from an alternate source.

The Auxiliary Feedwater Flow is monitored to inform the operator of normal, abnormal or inadequate flow. He can take manual actions to provide adequate flow and thus maintain adequate decay heat removal through the steam generator system.

Steam drum level and pressure are monitored to identify an accident and to allow the operator to take manual actions to initiate and control systems required to achieve and maintain decay heat removal via the Steam Generator System.

EVST sodium hot leg temperature is monitored to enable the operator to make manual actions during normal, transient, and accident conditions which are necessary to prevent or mitigate the exceeding of
50 Ex-Vessel Storage Tank and stored fuel assembly design limits.

TABLE 7.5-1

INSTRUMENTATION SYSTEM FUNCTIONS AND SUMMARY

	System	Measured Parameters	Instrument	Measurement Location	Purpose
57 52	Flux Monitoring	Source Range	BF_3	Thimbles on periphery of guard vessel	Determines or Provides: 1. Flux status at shutdown, startup and power levels
		Wide Range*	Fission Chambers	Thimbles on periphery of guard vessel	2. Signals to PPS logic (except source range)
57		Power Range*	B-10, Compensated Ion Chamber	Thimbles on periphery of guard vessel	3. Signals for reactor and plant control (D.C. linear power ranges) 4. Signals for display, annunciation and recording
50	Heat Transport Primary/ Intermediate Loops	Reactor Inlet Pressure*	Pressure Element	Cold leg primary loop	PPS and display PHTS performances
50		Primary and Intermediate Flow*	PM Flowmeter	Cold leg of primary and intermediate loops (hot leg in intermediate loop 2)	PPS, Plant Control and Display, PHTS performance
		IHX Primary Outlet Temperature*	Thermocouple	Cold leg piping nearest to IHX primary outlet	Plant Control System (PCS), PPS, and Display
50		Primary and Intermediate Hot and Cold Leg Temperature	Resistance Temperature (RTD)	Primary and Intermediate hot and cold leg	Surveillance, display and use to calorimetrically calibrate PM flowmeters
50		Primary and Intermediate Pump Disharge Pressure	Pressure Elements	Drainline from discharge piping of the loops's sodium pump	Surveillance, display and monitor differential pressure between primary & intermediate loops PHTS performance
50		Intermediate IHX Outlet Pressure	Pressure Elements	Intermediate between IHX & Superheater	Surveillance, display & monitor differential pressure between intermediate loops

TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measurement Location	Purpose
Heat Transport Primary/ Intermediate Loops (cont'd)	Intermediate Pump Inlet Pressure	Pressure Elements	Pipes between evaporator and pump inlet	Display pump performance
	Intermediate Expansion Tank Sodium Level	Level Probe	Intermediate expansion tank	Display-intermediate loop sodium inventory and alarm
	Evaporator Sodium Outlet Temperature*	Thermocouple	Downstream where the two evaporator outlets join into the header	PPS and display
Sodium Pumps	Sodium Level	Level Probe	Pump Tank	Display-used for sodium inventory and pump protection (alarm)
	Primary/Intermediate Pump Speed*	Tachometer	Main Shaft of each pump	PPS, display, pump speed control, performance
	Pony Motor Running	Speed Switch	Pumps	Display, performance
	Diagnostic Instrumentation	Various	Pumps	Display, pump performance
Steam Generator	Sodium Flow	Venturi	Superheater sodium outlet (1 loop)	Display & superheater & evaporator performance
	Sodium Temperature	Thermocouple	Superheater evaporator outlet (3 loops)	Display & steam generator performance evaluation
	Sodium Pressure	Pressure Element	1 loop-superheater inlet, outlet (both legs) and evaporator outlet (one leg)	Display & steam generator performance evaluation
	Feedwater Flow*	Venturi	Inlet line to steam drum (feedwater)	PPS, display & steam generator performance evaluation
	Superheat Steam Flow*	Venturi	Outlet of each superheater (steam)	PPS, display & steam generator performance evaluation
	Steam Drum Drain Flow	Orifice	Steam Drum Drain line for each steamdrum	Performance evaluation
	Evaporator Inlet Flow	Venturi	Inlet to one evaporator (1 loop)	Performance evaluation

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TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measured Location	Purpose
Steam Generator (cont'd)	Feedwater Temp.*	RTD	Steam drum inlet (feedwater)	PPS, display & steam generator performance evaluation
	Recirc. Water Temp.	Thermocouple	Recirculation pump discharge header	Performance evaluation
	Steam Temp.	Thermocouple	Outlet header from steam drum (steam)	Performance evaluation
	Superheat Steam Temp.*	RTD	Superheater outlet line (steam)	PPS and Display
	Evap. & Superheater Inlet-Outlet Temp.	RTD	Inlet & outlet nozzles for 1 evaporator & superheater (1 loop)	Display & performance evaluation
	Blowdown Temp.	Thermocouple	Blowdown line for each steam drum	Performance evaluation
	Feedwater Pressure	Pressure element	Inlet line to each steam drum	Display and steam generator performance
	Steam Drum Pressure*	Pressure element	Appendage from steam drum	PPS and display
	Recirc. Pump Outlet & Inlet pressure.	Pressure element	Appendage from recirculation pump discharge and suction header.	Performance evaluation
	Superheat Steam* Pressure	Pressure element	Loop output steam line	PPS and display
	Evaporator and Superheater Inlet Pressure	Pressure element	Inlet nozzle for 1 evaporator and superheater (1 loop)	Performance evaluation
	Evaporator and Superheater Outlet Pressure	Pressure element	Outlet nozzle for evaporator and superheater in each loop	SWRPS and performance evaluation
	Steam Drum Level*	Differential Pressure Element	Differential pressure across steam drum	PPS and display

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TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measured Location	Purpose
51 Reactor and Vessel Instrumentation	Core Sodium Exit Temperature	Thermocouples	Selected fuel and blanket assemblies.	Display and control - core outlet temperature
	Core Peripheral Temperature	Thermocouples	Core periphery - 2 locations	Display - Design verification
	Upper Internals Temperature	Thermocouples	Parts of upper internal structure 6 locations	Display - design verification, predict stress on various components 40
	Sodium Level above Core*	Level Probe	Reactor vessel plenum	PPS and Control, Display 40
57 41	Upper Internals Movement	Vibration Element	4 Biaxial on parts of appropriate structure	Display - measure vibrations induced by sodium flow
Fuel Failure Monitoring	Cover Gas Gamma Activity	Gamma Spectrometer	Sampling in RSB	Detect each instance of fuel clad failure and characterize failure
	Delayed Neutron Monitoring	BF_3 Counter	Shielded moderator assembly adjacent to each of the PHTS hot leg pipes	Detect fuel in PHTS
	Tag Gas Isotopic Composition	Mass Spectrometer	Gas tag sampling traps in RSB	Locate failed fuel
Leak Detection	Liquid Metal to Gas Leaks	Contact detectors cable detectors aerosol monitors	In various locations in sodium circuits	Identify location of liquid metal to gas leaks for continuous surveillance of liquid metal systems boundaries 28

TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measurement Location	Purpose
Leak Detection (cont'd)	Intermediate to Primary Leak	Level probe	IHTS expansion tank	Detect leak in IHX
	Steam Generator Leaks	Hydrogen and Oxygen detectors	Sodium exiting either or both superheater outlets	
	Steam Generator Leaks	Hydrogen and Oxygen detectors	Sodium filled vent line from superheater	Detect small water-sodium and steam-sodium leak in steam generator, identify leaking module, provide signal for operator action
	Steam Generator Leaks	Hydrogen and Oxygen detectors	Sodium filled vent lines from either combined evaporator vents or evaporator A vent	
47	Steam Generator Leaks	Hydrogen and Oxygen Detectors	Sodium exiting combined evaporator outlets or evaporator A outlet.	
Sodium-Water Reaction Pressure Relief	Rupture discs Operation*	Sensors	Downstream of rupture discs-before reaction products separation tank	PPS and SWRPS initiation
43	Rupture discs Operation	Sensors	Downstream of rupture discs - before sodium dump tank	SWRPRS initiation
	Rupture discs integrity	Pressure Element	Gas space between rupture discs	Surveillance of discs
	SWRPS equipment Temperature	Thermocouples	Surface temperatures of reactor products separation tank, centrifugal separator, drain tank and hydrogen igniter	Surveillance 29
	Separation Tank Pressure	Pressure Elements	Reactor products separation tank	Surveillance
	Evaporator Water and sodium dump tanks level, pressure and temperature	Level, Pressure, Temperature Elements	Evaporator water and sodium dump tanks	Surveillance

*Safety Related.

TABLE 7.5-2

REACTOR AND VESSEL INSTRUMENTATION

	Instrument	Measured Parameter	Location	Purpose
	Thermocouple	Core Exit Sodium Temperature	One at each of 30 selected fuel and blanket assemblies	Control and surveillance - Core outlet temp.
57			275 additional locations at selected fuel and blanket assemblies.	Surveillance and Diagnostic - Distribution of temperature across the core
51	Thermocouple	Core Peripheral Temperature	Two spaced locations on the core periphery	Design Verification - Distribution temp. around the core
40		Upper Internals Temperature	Six on parts of the upper internal structure	Design Verification - Distribution of temp. to predict stress on various components
	Sodium Level Detector	Sodium Level above the core	Four short units distributed equally around periphery	Protection and Control - Measures the operating level of the sodium in the reactor
			One long unit near one of the four short ones	Monitoring - Measure the sodium level from operating level down to minimum safe level
41	Vibration Detector	Upper Internals Vibration	Four biaxial on parts of appropriate structure	Design Verification - Measure vibrations induced by sodium flow

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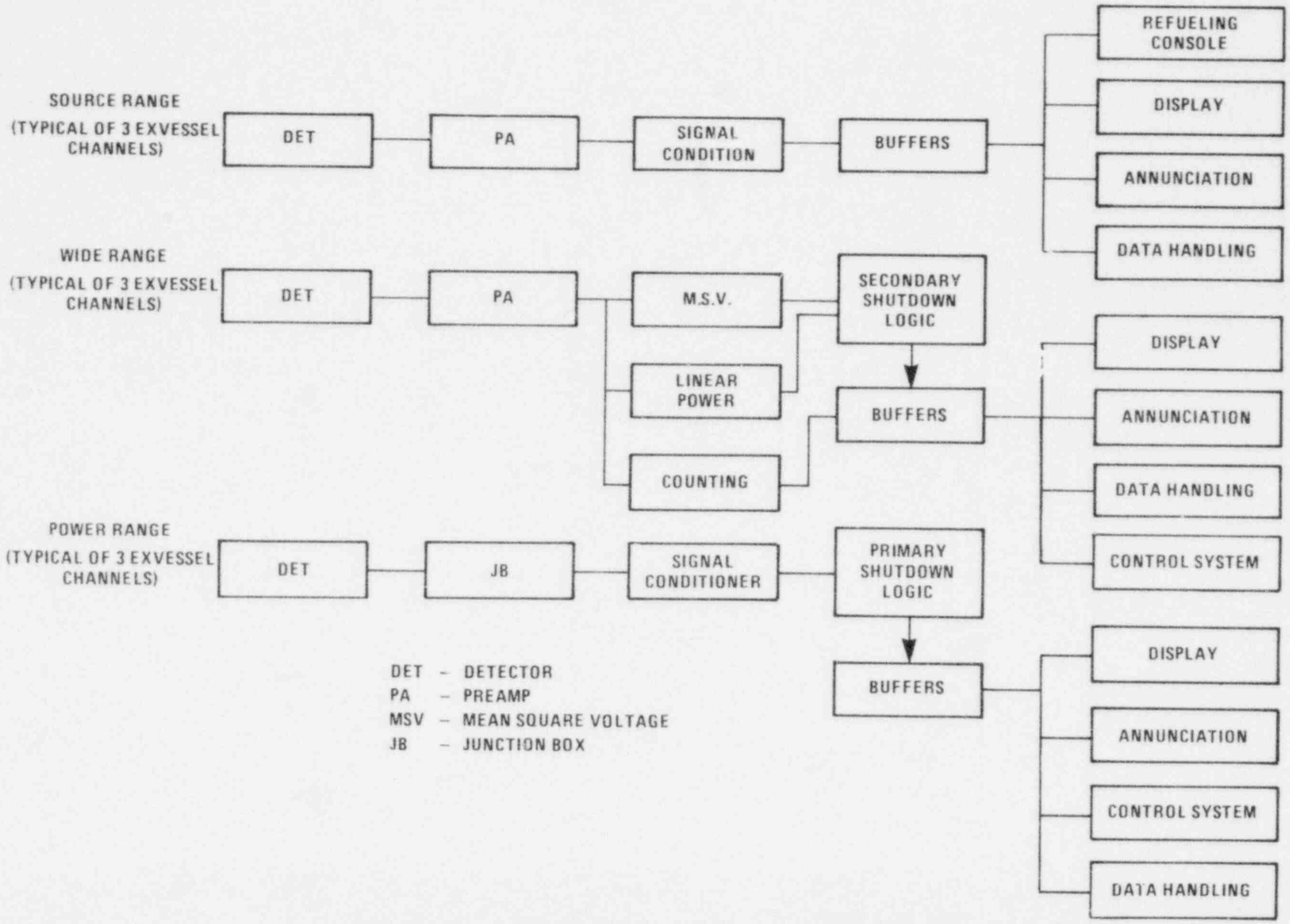

Figure 7.5-1. CRBRP Flux Monitoring System Block Diagram

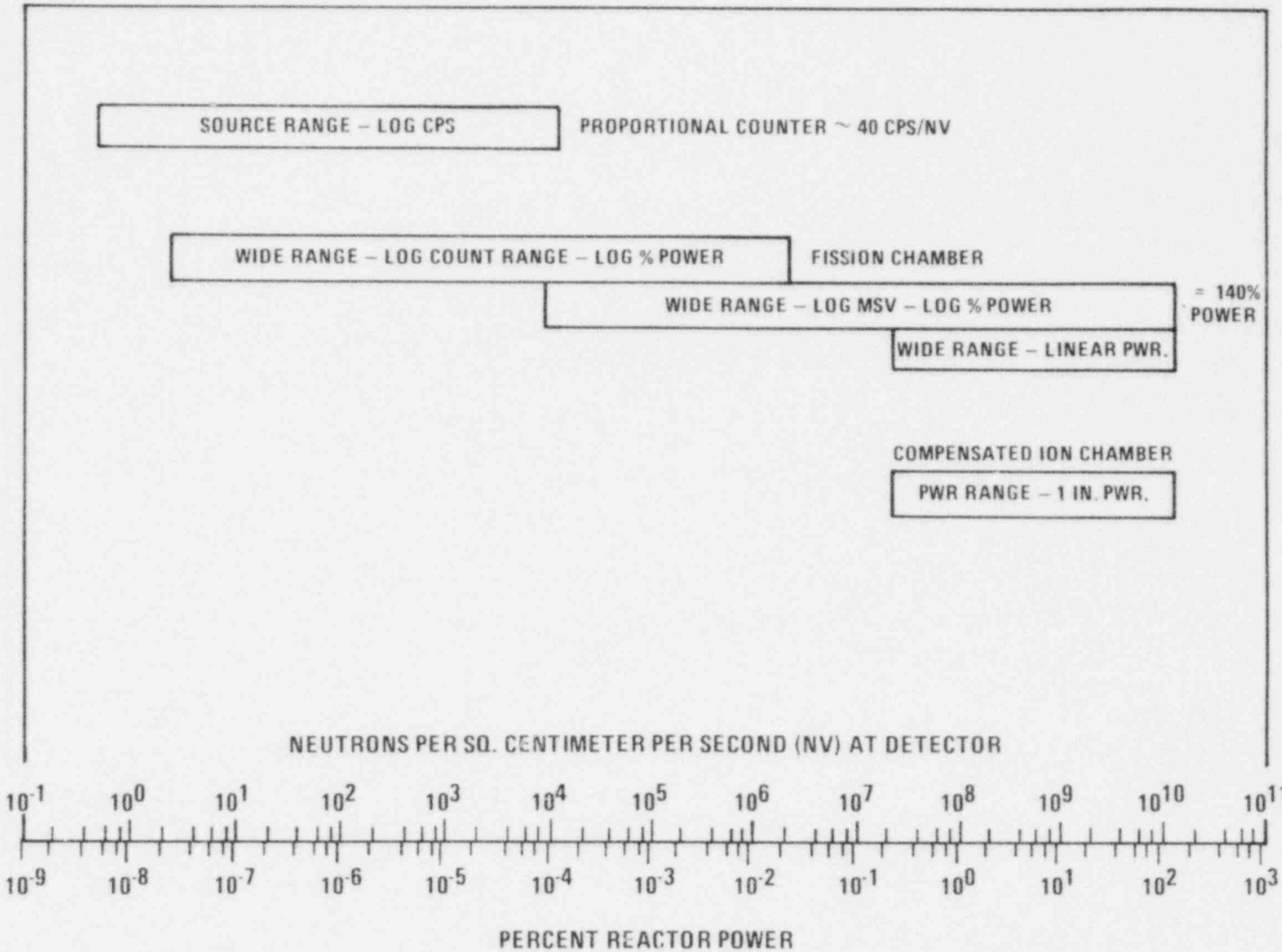

Figure 7.5-2. CRBRP Flux Monitoring System Instrument Range Coverage

7.7.1.3 Primary and Secondary CRDM (Control Rod Drive Mechanism) Controller and Rod Position Indication

The Primary Control Rod Drive Mechanism Control System transforms the bulk 3 phase power into the pulsed DC necessary to operate the Control Rod Drive Mechanism in response to input commands from the Reactor Control System. Interlocks and permissives are provided to prevent operating sequences of the control rods which would damage the equipment, and assure that the rods are maintained in the banked configuration required to maximize core performance. Rod Position Indication is provided redundantly for each rod to permit the operator to verify the reactivity status and operation of the control system. The Secondary CRDM Controller and Rod Position Indicators are described in Section 4.2.3.

7.7.1.3.1 Primary CRDM Control

The control rod drive mechanism is actuated by a 4 pole, 6 winding reluctance stepping motor. The mechanism lead screw has a thread pitch of 0.6 inch, and moves 0.025 inches for each pulse to the drive stator. A block diagram of the drive system is shown in Figure 7.7-4. Driving power is supplied from the site power system through redundant motor-generator sets, Reactor Shutdown System scram breakers, a 3 phase to 6 phase transformer, and banks of silicon controlled rectifiers (SCR's) in the individual controllers to the stator windings of the CRDM.

The primary rods are divided into 2 groups. One group of 3 startup rods responds only to single rod manual control and while operating during reactor operation is normally fully withdrawn. A group of 6 control rods respond to manual control or to an analog signal from the Reactor Control
57 System.

Rod speed demand limits are included in the reactor controller as well as rod speed limits in the individual Primary CRDM Controllers. Rod block interlocks are included in the "OUT" demand input as shown on Figure 7.7-5.

When the Reactor Shutdown System initiates a scram, the "Scram Breakers" open and interrupt the power to the Primary CRDM stator coils; the rotor collapses and disengages the rollers from the lead screw; and the CRDM drive train falls under the force of gravity and the scram assist spring to insert the control rods into the core. Failures within the sequence and controller units cannot prevent removal of the power required to hold the CRDM's in the withdrawn position. The components are described below. 19

Motor-Generator Set

57 Dual M-G sets provide the 3 phase power for CRDM operations. When a latch signal is received at the voltage control, the output voltage of a generator is increased. The M-G sets for the primary rods use a 200 Hp motor and a 150 Kw generator.

Mechanism loads are shared by the two M-G sets; however, either M-G set has the capacity to power the entire load of the primary system. Controls are provided to synchronize the two M-G sets. The motor-generator sets are designed to provide sufficient inertia and voltage control to prevent rods dropping in the event of power dips of 0.3 seconds or less.

Generator output circuit breakers provide the necessary electrical
57| protection for the generators and for system maintenance

<u>Power Supplies and Transformers</u>

57| Since the CRDM controllers use 6 phase AC power, one 3 phase to
6 phase transformer is provided for the primary rods. The transformer includes appropriate secondary side surge protection.

Each CRDM controller requires control power to operate the interface circuitry, programmer, gate drives, internal interlocks and display equipment. As shown on Figure 7.7-4, redundant AC power sources
57| energize redundant DC logic power supplies whose outputs are auctioreered.
This design prevents failure of a power supply from causing a rod to drop.

The power supplies are sized to provide sufficient capacity for all of the CRDM controllers in the primary group. Transformer isolation, including grounded Faraday shields, is used to prevent failures from propagating into the controller electronics.

CRDM Motor Controller

The CRDM Motor requires DC energization of coils in the proper sequence to develop the required setpoint motion. The sequence of coil energization for rod motion is in a two coil-three coil
57| sequence. Thus a forward step is produced each time
a leading coil is energized and also when a trailing coil is de-energized. To reverse the motion, the sequence is reversed.

The CRDM Controller uses six SCR's for each stator coil to half wave rectify the 6 phase AC input power and supply DC output to a stator coil. All six SCR's for a stator coil are turned on by one gate drive unit. The Controller incorporates the logic necessary to correctly sequence the gate drive units on and off, thereby sequencing the coils in appropriate order. Separate controllers are provided for each individual mechanism. Holds are provided when input or output logic errors are detected.

57| In Single Rod Control Mode, the input circuitry to each controller accepts
on-off inputs for IN, OUT, and HOLD commands and provides the sequencer with an IN pulse train, OUT pulse train, or HOLD DC output. The IN command steps a single rod down in the core at a predetermined rate. The OUT command steps a single rod up out of the core at a predetermined rate (not necessarily the same as the IN rate) and the HOLD command maintains the rod in its present position (no motion). The input circuitry also incorporates adjustable speed settings for the IN, OUT, and LATCH modes of CRDM operation and assures that an IN command takes precedence over an OUT command. In addition to the adjustable speed settings, the controller provides an independent speed limitation which has a separate clock and power supply from that used by the input circuitry. If the input circuitry called for a speed greater
57| than 10% above 9 inches per minute due to a postulated failure, the
speed limiter circuit will place the rod in the Hold Mode.

In any automatic control mode, or in Group Manual mode, the mechanism controllers are operated in sequence one step at a time to keep the rod bank in required alignment. The sequence rate and direction are determined respectively by analog and digital signals from the reactor control system. If the selector sequence rate is higher than a predetermined trip point, an overspeed detector will alarm and place the controllers
57 in HOLD. A functional block diagram of the control is shown in Figure 7.7-5.

Hold Bus

A Hold Bus Power Supply and transfer select circuitry are provided to allow any controller to be replaced without a plant shutdown. In the event of a controller failure, the mechanism controller in
57 question can be switched out and transferred to a Hold Bus. Power to the Hold Bus Power Supply is provided downstream from the scram breakers. This ensures that if a scram is initiated, a rod on the Hold Bus will also scram.

7.7.1.3.2 Rod Position Indication System

Two independent Rod Position Indicating Systems are provided for each control rod: An Absolute Position Indication System (ARPI) and a
57 Relative Position Indication System (RRPI). These systems assure that the plant operators can continuously determine the position of the control rods.

The ARPI provides a direct measurement of rod position at any time and, unlike the RRPI, does not require re-zeroing after a scram or temporary loss of power. The system is solid state, utilizing ultrasonics and magnetics to provide a D.C. output indicative of rod position.

The sensor for this system consists of a tube extending down from the top of the motor tube and into the inside diameter of the PCRDM lead screw. A nickel-cadmium wire is stretched axially through the tube. As the lead screw translates, the flux from a torroidal magnet mounted on top of the lead screw intersects the wire at a point indicative of the rod position. Electrical pulses sent down the wire generate magnetic fields which, when they intersect the flux of the lead screw magnet, causes a torsional strain creating a sonic pulse which travels from the point of flux intersection upward. The sonic pulse is detected at the top of the wire, and the time of propogation is measured electronically. This propogation time is converted to a D.C. signal which is analagous to rod position.

This signal is read out on the main control panel by rod top and rod
57 bottom indicator lights and a vertical bar graph indicator. It is also used to operate the rod out of alignment alarm and rod control interlocks.

The Relative Rod Position Indication System provides a digital rod posi-
57 tion indication on a CRT at the Main Control Board. Two pairs of magnetic coil pick-ups are mounted within each stator jacket above the stator and on opposite sides. A 6 pole magnetic section is attached to the mechanism rotor and rotates in the plane of the pick-up coils. Voltage pulses caused by the move-
57 ment of the poles in the proximity of the pick-up coils are sent to the PDH&DS. The resolution of the indicator is $\pm$0.1 inch. Unlike the Absolute Position Indication System, this system must be reset after each scram and in the event of a power failure reset after power is restored. The pulses are also
57 counted by an odometer type readout in the rod control equipment room.

7.7.1.4 Sodium Flow Control System

The Sodium Flow Control System consists of six controllers used to drive the three primary and three intermediate sodium pumps. Each controller consists of a cascade system with an inner loop using speed as the feedback signal and an outer loop based on a flow feedback signal. The flow control range is 30 to 100% of rated flow. The flow setpoints are generated either
41 manually or by the Supervisory Control.

Figure 7.7-7 is a block diagram of the flow/speed control loop which is typical of the six controllers in the system. The Speed Control System is an inner loop and used pump speed, which is sensed via a pump shaft mounted tachometer, as the feedback variable. The Speed Control System is limited internally by the torque limit circuit which sets both the accelerating and decelerating torque of the variable speed pump drive.

The demand to the Speed Controller is set by the FLOW/SPEED Mode Select Switch. In the Speed Mode, pump speed is set by a manually adjusted potentiometer; in the Flow Mode, pump speed is set by the Flow Controller.
44 The Flow Controller uses the filtered, median select signal of three available redundant flow meter buffered PPS outputs as the feedback signal. This signal, along with the flow demand, is used to generate the error signal which is compensated through the Control Compensation Network and then limited by the High Speed Limit Circuit prior to being used as the speed demand signal. The demand to the Flow Controller is set by the MAN/AUTO Select Switch. In the automatic mode, the demand comes from the supervisory control, while in the Manual Mode, the demand comes from a manually adjusted potentiometer on the control panel.
47

7.7.1.5 Steam Generator Feedwater Flow Control System

The Feedwater Control System provides for the controlled supply of feedwater to the Steam Generator System steam drums. Flow of water into the steam drums is automatically controlled to maintain water in the drum within specified limits. The range of water level is based upon the ability of the steam separators to function properly. Primary control is effected by modulating a Feedwater Flow Control Valve at the inlet to the steam drum. In addition, Feedwater pump speed is varied to maintain the pressure drop across the Feedwater Flow Control Valves at a value that does not change excessively with load. This reduces the operating requirements of the Flow Control Valve and improves feedwater pump efficiency. Control logic for the Feedwater Flow Control System is shown in Figure 7.7-1.

7.7.1.5.1 Feedwater Flow Control Valve Control System

The Feedwater Flow Control Valve is positioned via a conventional three-element cascade control system which consists of drum level and feedwater flow controllers.

The drum level controller compares measured drum level with a predetermined setpoint value. The resulting error signal is summed with a signal representing drum steam flow and constitutes the input (i.e., flow setpoint) to the feedwater flow controller. The output of the feedwater flow controller is the position demand signal to the Feedwater Flow Control Valve.

Instrumentation required by this control system is obtained as follows:

- Steam Drum Level - Water level is measured by a differential pressure transmitter which senses the difference between the pressure resulting from a constant reference column of water and the pressure resulting from the variable height of water in the steam drum. The measurement is density compensated.

- Steam Flow - Steam flow is sensed at a flow element in the outlet line from the superheater by a differential pressure transmitter. The differential pressure signal is compensated for temperature and pressure variations and linearized to provide a mass flow signal.

- Feedwater Flow - Feedwater flow is sensed at a flow element in the inlet line to the steam drum by a differential pressure transmitter. The differential pressure signal is corrected for temperature variations and linearized to provide a mass flow signal.

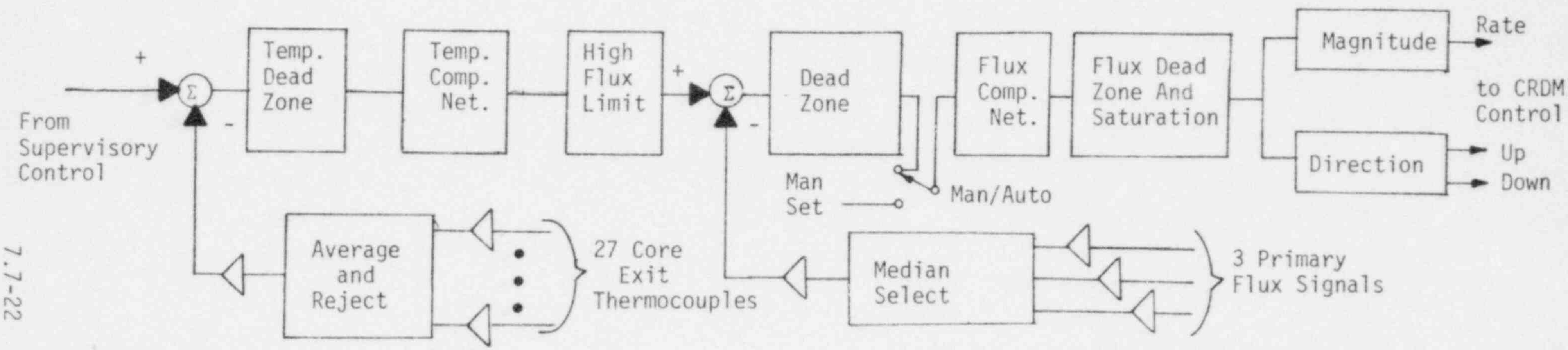

Figure 7.7-3 Block Diagram of Reactor Control System

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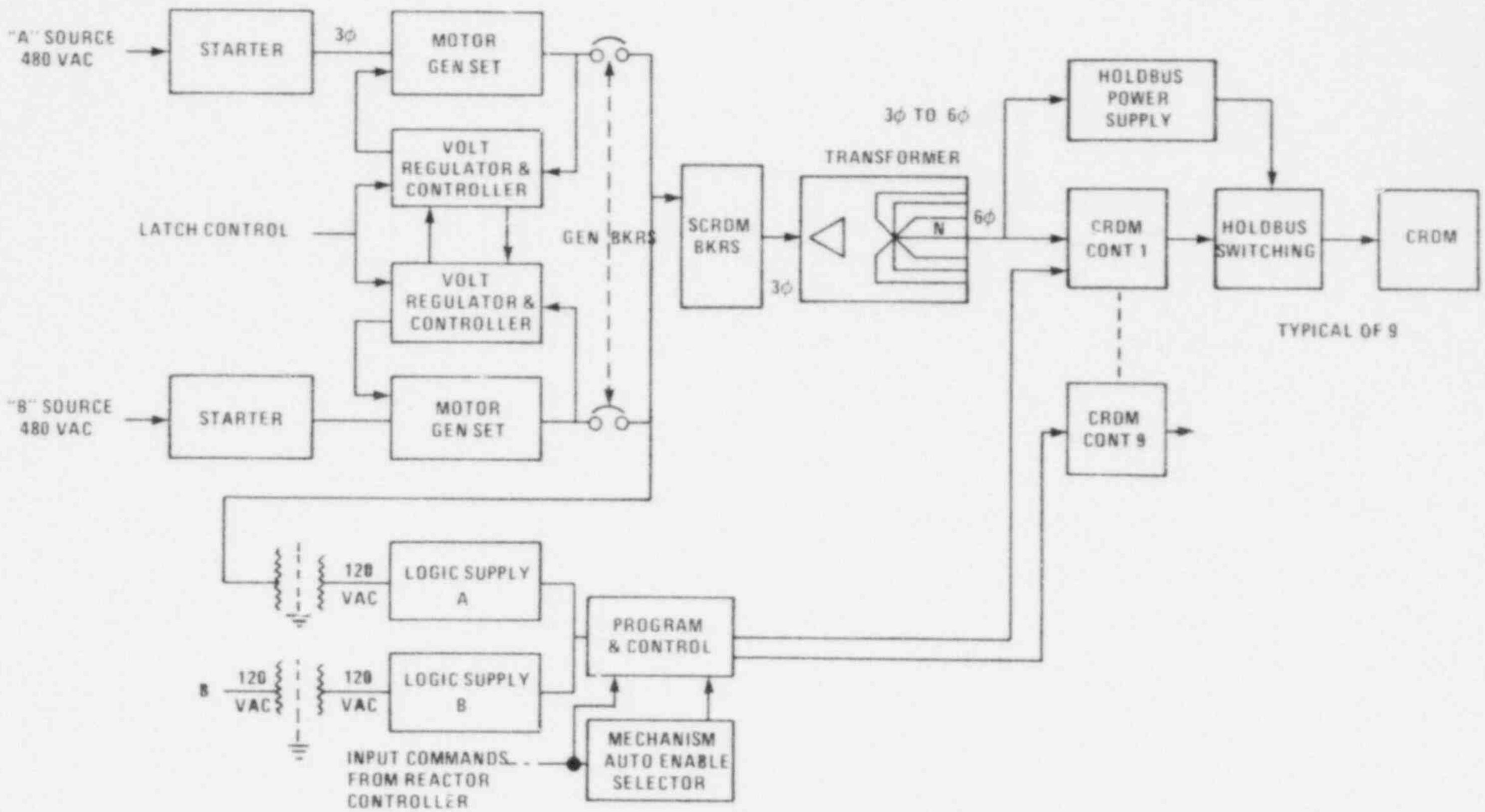

Figure 7.7-4. CRDM Controller and Power Train for Primary Rods

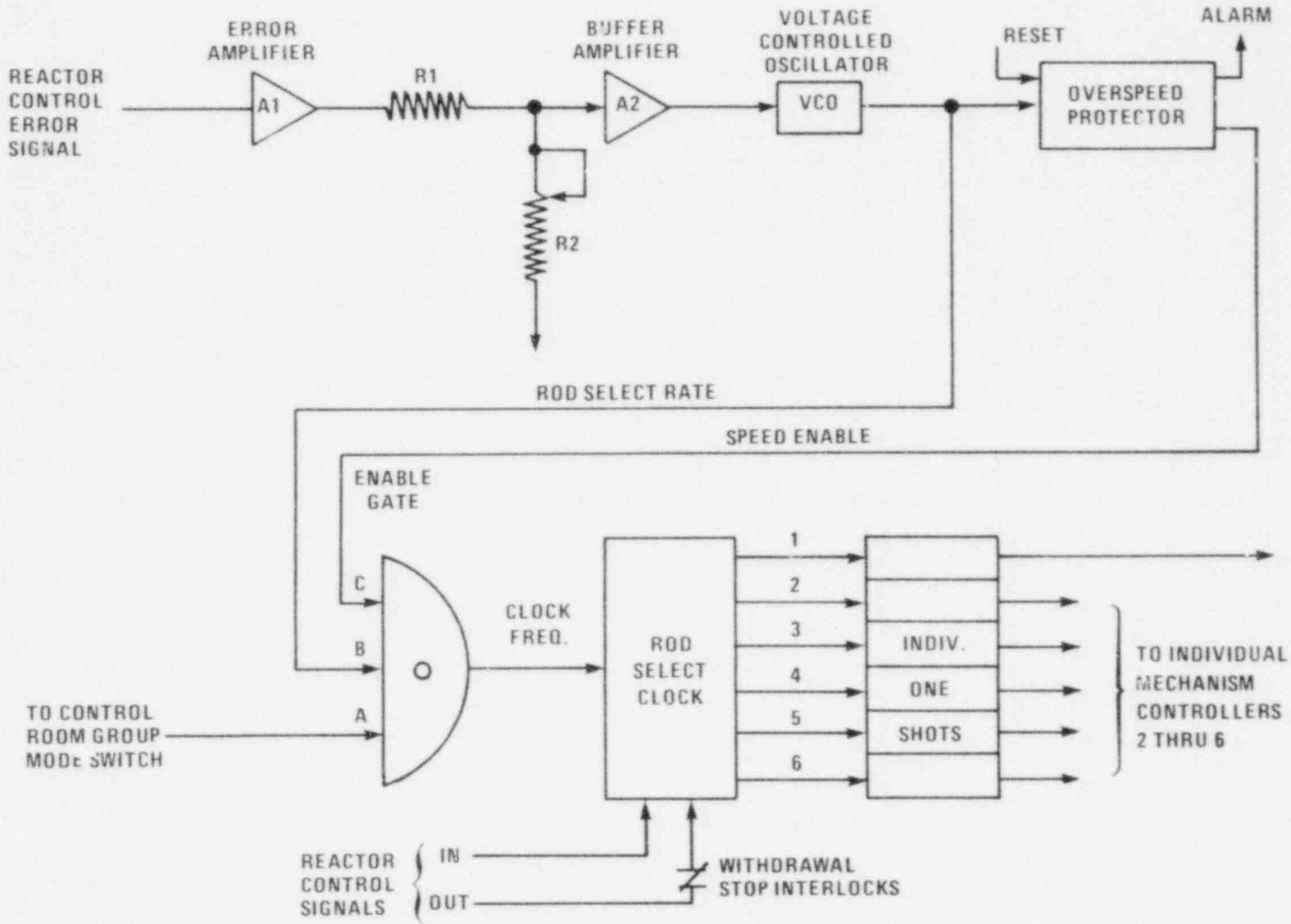

Figure 7.7-5. Block Diagram of Primary Rod Group Control

57 FIGURE 7.7-6 HAS BEEN DELETED

7.8 PLANT DATA HANDLING AND DISPLAY SYSTEM

7.8.1 Design Description

57 The Plant Data Handling and Display System PDH&DS supports plant
operations and performance by monitoring, limit checking, trending, and dis-
playing plant information. It supplements other monitoring and displaying
57 systems including Plant Annunciation and Plant Control. The PDH&DS performs
only diagnostic and informative functions and its operation is not a require-
ment for startup, operation, or shutdown of the plant. However, additional
57 requirements may be placed on the plant operations should the PDH&DS be
inoperative.

Specific functions of the PDH&DS include:

57 • Monitoring of plant variables and alerting the operator when selected variables exceed predetermined limits.

• Recording of the operating history of the plant.

• Performance parameter calculations including:

  plant and equipment calorimetries

57 plant protection system channel output monitoring

  shutdown margins

  control rod worth

57 core assembly exit temperatures

  reactivity calculations

  sodium inventory calculations

57 component efficiencies

• Providing Cathode Ray Tube (CRT) display units in the control room to present pertinent plant data for surveillance of plant protection and control systems, auxiliary systems and balance of plant.

57 • Supporting of refueling operations by providing capacity for long term core component inventory.

57 • Displaying group annunciation of measurements to reduce the number of trips by plant personnel to read local indicators.

• Providing a mechanism to forewarn the operator of potential harmful conditions. Examples of these include high bearing temperature, detection of small sodium leaks and radiation levels. If the condition deteriorates further, the operator will be warned by the annunciator system.

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57

- Providing pre and post trip information for review.

57 - Providing for acquisition of data for design verification of plant components.

57

7.8.2 Design Analysis

The PDH&DS is inherently designed for high system availability. The sensor measurements being monitored are divided into two groups according to their importance to plant operation. Significant information (Group 1) will have the capability to be processed, recorded, and displayed by more than one piece of equipment. The less important beneficial information (Group 2) may be processed, recorded and displayed by only one piece of equipment. When one piece of equipment fails, significant information is
57 not affected but the failure may limit the amount of beneficial information that can be processed, recorded and displayed.

The main processing part of this system is located adjacent to the control room in the computer room. Information generated by the PDH&DS is presented in the control room. The data acquisition components of the system are located near sensor local panels. The data acquisition components multiplex sensor signals to reduce the number of control room panels and associated cabling.

Although the PDH&DS is designed for high availability, it should be emphasized that the system performs no direct safety or protection function and is not essential for plant operation. Operating procedures of systems which normally use PDH&DS capabilities are written to allow operation
57 of the plant with manual data recording and calculations.

7.9 OPERATING CONTROL STATIONS

7.9.1 Design Basis

A control room is provided from which action can be taken to operate the nuclear power station safely under normal operation and off-normal conditions and to maintain it in a safe condition under postulated accident conditions. Adequate radiation protection is provided to permit access and occupancy under Extremely Unlikely Fault conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the postulated accident. The control room provides protection from substances such as sodium oxide which might be released to the local environment under Extremely Unlikely Fault conditions.

The basic criteria for inclusion of displays or controls in the control room shall be:

- The displays or controls necessary to support all normal plant operating conditions;
- The displays and controls necessary to respond to off-normal or casualty conditions which impact on power operations capability;
- The displays or controls necessary to prevent potential radiological hazards to offsite personnel;
- The displays necessary to the operator for detection of fire hazards; or
- The display and controls necessary to prevent potential damage to the plant.

The control boards are arranged functionally based on normal and off-normal operational considerations to minimize the number of operators required and to enhance the capability of the operational personnel to monitor and assure the safe status of the plant during all operations.

Remote control stations are provided outside of the control room to shut the plant down and maintain it in a safe condition assuming loss of control
57| room habitability. Access to the control room is controlled by card key to assure that only qualified personnel use the equipment provided to monitor and maneuver the reactor plant.

7.9.2 Control Room

7.9.2.1. General Description

The Control Room is located in the Control Building at approximately grade level. The Control Room proper occupies approximately a space of

57| 70' x 75'. The remainder of the floor is devoted to auxiliary services for the control room operating personnel. The Control Building provides the necessary structural and atmospheric protection to allow continued habit-
57| ability that collectively satisfy CRBRP General Design Criterion 17. These features are summarized in Section 3.A.3.

The indicators, annunciators, and controls included in the Control Room provide the capability to operate the plant through all normal operational sequences and to respond to off-normal or emergency conditions without continuously manned remote stations.

7.9.2.2. Control Room Arrangement

The control room is arranged to provide an effective interface between the plant and the operating personnel (refer to Figure 7.9-1). Frequently used safety related instrumentation and controls are located on the
57| Main Control Panel. This equipment is grouped by operational category to assure that determination of plant condition and action to correct the condition are in close proximity. Less frequently used equipment and certain electronic equipment for which access control is desired are located in the rear panel area.

7.9.2.3 Main Control Board Arrangement

As shown in Figure 7.9-1, an open U-shaped main control panel is provided. The main control panel and the area it encloses form the central operating area for the plant. Equipment on the main control panel is arranged functionally and according to the power generation flow path. From left to right, the main control panel sections are as follows: emergency systems, plant protection system and engineered safety features, plant control and primary heat transport systems, intermediate heat transport and steam generator systems, steam generator auxiliary systems, turbine system and generator and switchyard. The equipment is arranged with annunciators at the top and display, controls, and switches in functional groups on the vertical and sloping bench sections. The size and arrangement of
57| equipment is based on the following guides:

- Displays, annunciation, switches, and control necessary to operate the plant without continuously manned remote stations are located on the main control panel or displayed through the
57| Plant Digital Data Handling System and the cathode ray tube (CRT) displays.

- Graphic or mimic displays are provided where warranted to enhance the operator/plant control interface and minimize the chances for inappropriate operator action.

- Physical separation of redundant safety related instrumentation equipment is incorporated.

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- Physical, color, and geometric differentiation of displays and controls mounted on the board is provided to assure ease of recognition by operating personnel and minimize the chances for inappropriate actions.

- Arrangement and design of displays and controls is specified to provide arrays which permit determination of proper alignment at a glance, where practical.

- Modular design of switches, controls, and indicators is used to permit ease of maintenance and minimum interference with operation. The equipment included on the main control board is summarized below (refer to Figure 7.9-1 and Table 7.9-1).

The arrangement of the instrumentation and control devices on the main control panel is as follows:

Sections 1 & 2 - Emergency Systems

- Emergency Chilled Water
- Emergency Plant Service Water

Section 3 - Plant Protection System and Engineered Safety Features

- Reactor Shutdown
- Containment Isolation
- Steam Generator Auxiliary Heat Removal System Status
- Sodium Water Reaction Pressure Relief System Status
- Sodium Dump
- Control Room Heating, Ventilating, and Air Conditioning
- Containment Instrumentation
- Flux Monitoring

Section 4 - Plant Control and Primary Heat Transport Systems

- Primary and Secondary Manual Scram Switches
- Supervisory and Reactor Control
- Reactor Instrumentation
- Rod Control and Rod Position Indication
- 57 Primary Heat Transport

Section 5 - Intermediate Heat Transport and Steam Generator Systems

- o Intermediate Heat Transport
- o Steam Generator
- o Feedwater

Section 6 - Steam Generator Auxiliary Systems

- o Condensate
- o Auxiliary Feedwater
- o Protected Air Cooled Condenser

Section 7 - Turbine System

- o Turbine Control Panels
- o Turbine Instrumentation
- o Turbine Steam Bypass
- o Circulating Water
- o Secondary Plant Service Water
- o Secondary Manual Scram Switch

Sections 8 & 9 - Generator and Switchyard

- o Graphic Arrangement of High Voltage AC
- o AC Bus Circuit Breaker Control
- o Generator Syncroscope

The Following instrumentation and control panels, while not a part of the Main Control Panel, demand rapid operator response and have been arranged to permit operator scanning from the Main Control Panel:

- o Failed Fuel Monitoring
- o Sodium Leak Detection
- o Sodium Fire Detection
- 57 o Non-Sodium Fire Detection

- Control Building Fire Detection
- Emergency Diesel Generators
- Switchyard and Station Electrical Distribution
- Direct Heat Removal Service

The layout of Section 3 of the main control panel is designed to minimize the time required for the operator to evaluate system performance under accident conditions. Deviations from predetermined conditions are alarmed and the status of automatic safety systems is alarmed and/or indicated so that corrective action may be taken by the operator.

The control room also includes control and instrumentation equipment that is used infrequently or for which controlled access is desirable. Included in this control room back panel area are power distribution, chilled water, containment instrumentation, recirculating gas, heat transport, steam generator, heat ventilation and air conditioning, annunciator electronics, turbine, balance of plant, plant control, plant data handling and display system multiplexers, flux monitoring, radiation monitoring, reactor shutdown
57 and containment isolation panels.

7.9.2.4 Main Control Panel Design

57 The Main Control Panel is an open U-shaped, stand up vertical panel as shown in Figures 7.9-1 (plan view) and 7.9-2 (side view). There are 3 significant features of the control board mechanical design: seismic capability; separation of redundant safety related equipment and wiring; and modular construction of switch, indicator and control equipment.

57 Since the Main Control Panel includes safety related equipment, the
52 sections including this equipment are designed to Seismic Category I. Structures, wiring, wireways, and connectors are designed and installed to ensure that safety related equipment on the control panel remains operational during
57 and after the SSE. The Main Control Panel is constructed of heavy gauge steel within appropriate supports to provide the requisite stiffness.

Within the boundaries of the Main Control Panel Sections, modules are arranged according to control functions. Fire retardant wire is used.
57 Modular train wiring is formed into wire bundles and carried to metal wire ways (gutters). Gutters are run into metal vertical wireways (risers). The risers are the interface between external wire trays feeding the panel and
57 Main Control Panel wiring. Risers are arranged to maintain the separated routing of the external wire trays. (See Figures 7.9-3 and 7.9-4).

Mutually redundant safety train wiring is routed so as to maintain a minimum of six inches air separation between wires associated with different trains. Where such air separation is not available, mechanical barriers are provided in lieu of air space.

57

7.9.3 Local Control Stations

Local control panels are provided for systems and components which do not require full time operator attendance and are not used on a continuous basis. In these cases, however, appropriate alarms are activated in the Control Room to alert the operator of an equipment malfunction or approach to an off-
57 normal condition.

7.9.4 Communications

Communications are provided between the Control Room and all operating or manned areas of the plant. In addition to public address and interplant commu-
41 nications and the private automatic exchange (used for in-plant and external communications) a sound powered maintenance communication jacking system is provided. Redundant and separate methods of communication between the control room
57 and other TVA generating plants is also provided.

7.9.5 Design Evaluation

Safe and continuous occupancy of the Control Room during normal and off-normal conditions is provided for in the design of the Control Building. The probability of the Control Room becoming uninhabitable due to fire or other cause is considered extremely remote. However, in the event the Control Room must be vacated temporarily, the reactor plant can be brought to and maintained in a safe shutdown condition for an extended period of time from local stations outside the control room. These local stations have been discussed
57 in Section 7.4.3 as the Remote Shutdown System.

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57 PAGES 7.9-7 and 7.9-7a

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Table 7.9-1

CONTROL ROOM ARRANGEMENT

	Item	
41	1.	Reactor Support Systems
	2.	Reactor Support Systems
	3.	Engineered Safety Systems
	4.	Reactor Primary Heat Transport Systems
41	5.	Intermediate Sodium Heat Transport Systems
	6.	Steam Generator and Associated Systems
	7.	Turbine Systems
	8.	Generator Systems
	9.	Synchronization and Main Unit Control
41	10.	Switch Yard and Station Electrical Distribution
	11.	Emergency Diesel Generator Panels
41	12.	Reactor Operator
	13.	Sodium Leak Detection Panel
	14.	Desk
	15.	Chair
	16.	Seismic Instrument Panel
	17.	Flux Monitoring
	18.	Process Radiation Monitoring
	19.	Area Radiation Monitoring
	20.	Weather Station Space
	21.	Phones (Station Communications)
	22.	HVAC System Recirculating Air Status Cabinet
	23.	HVAC Control Rack
41	24.	Generator and Transformer Protection

Table 7.9-1 (Continued)

Item	
25.	Turbine/Generator Supervisory Panel
26.	Turbine (Electro Hydraulic Control) Equipment Bays
27.	BOP Auxiliaries
57 28.	Non-Sodium Fire Protection Rack
29.	Sodium Fire Protection Zone Indicating Panel
30.	Heat Removal and Conditioning Logic Rack
31	Steam Generator Logic Rack
32.	Chilled Water Control Cabinet
6 33.	Auxiliary Liquid Metal Control Cabinet
34.	Remote Annunciator Cabinets
6 35.	Steam Plant Conditioning Rack
36.	Steam Plant Logic Rack
57 37.	Not Used
38.	Termination Racks
39.	PPS Containment Isolation Instrumentation Racks
57 40.	Not Used
41.	Primary PPS Buffers
42.	Primary PPS Termination Cabinet
43.	Primary PPS Comparator Panels
44.	Primary PPS Logic Racks
45.	Primary PPS Isolation Racks
46.	Secondary PPS Buffers
47.	Secondary PPS Termination Cabinets
48.	Secondary PPS Comparator Panels

41

Table 7.9-1 (Continued)

Item	
49.	Secondary PPS Logic Racks
50.	PPS Monitor Rack
51.	Not Used
57 52.	Not Used
53.	Desk Top Radio
54.	Reactor Control Rack
55.	Plant Supervisory Rack
56.	Sodium Flow Logic/Control Interface Rack
57.	Plant Switching Logic Racks
58.	Failed Fuel Readout Panel
59.	Computer Typewriters
60.	Computer Line Printer/Roller
61.	Cathode Ray Tube Display & Keyboard
62.	Cathode Ray Tube Display & Keyboard
63.	Cathode Ray Tube Display & Keyboard
64.	Cathode Ray Tube Display & Keyboard
65.	Cathode Ray Tube Display & Keyboard
66.	PDH & [illegible] Remote Data Acquisition Terminal
67.	Recirculation Gas Control Cabinet
68.	Not Used
57 69.	Not Used
70.	Security Surveillance Station
41 71.	Containment Instrumentation

TABLE 7.9-1 (Continued)

Item	
72.	Plant Control System Switching Logic
73.	Auxiliary Equipment Isolation Logic
74.	Portable Radio Communications
75.	Secondary Rod Control Cabinet
76.	Heat Removal & Conditioning Logic Rack
77.	PDH&DS Remote Data Acq. Term
78.	Remote Annunciator Cabinets
79.	Not Used
80.	CRT Display and Keyboard
81.	Not Used
82.	Building Fire Protection Panel
83.	Plant Security
84.	Patch Panel
57 85.	Containment Instrumentation

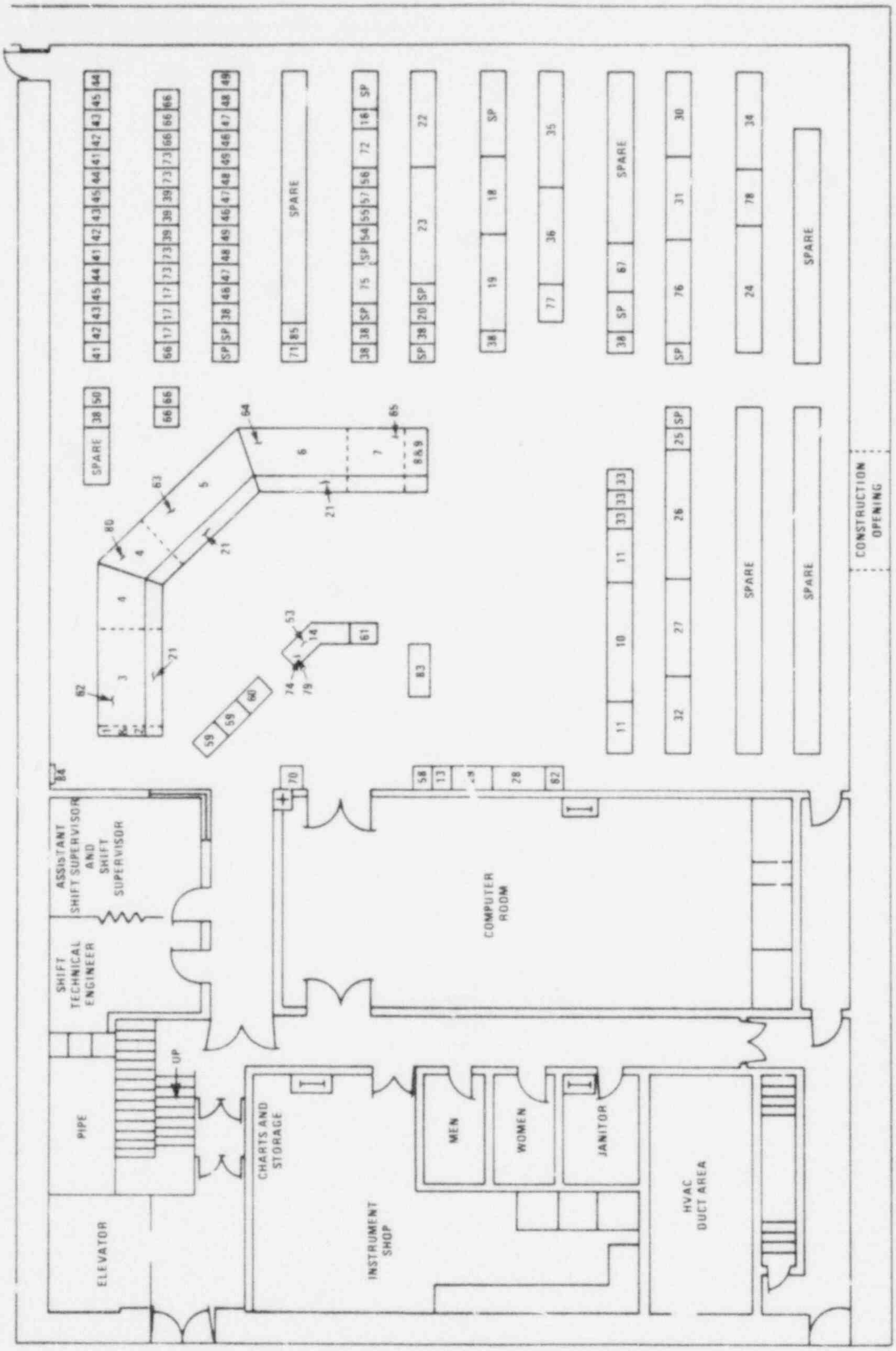

Figure 7.9-1. Control Room Layout

1489-2

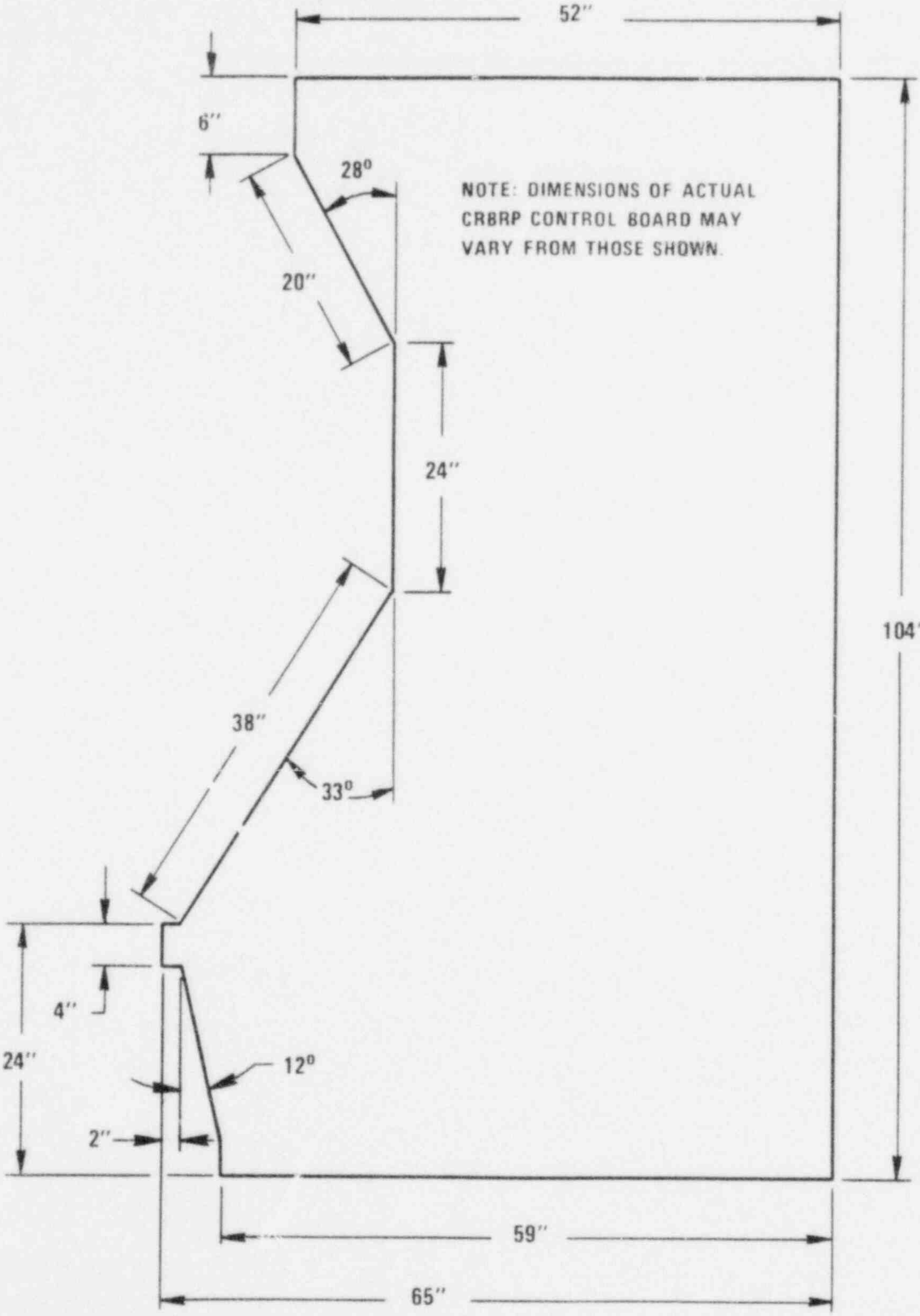

Figure 7.9-2. Typical Control Board (Side View)

11.6 OFFSITE RADIOLOGICAL MONITORING PROGRAM

The preoperational environmental monitoring program has the objective of establishing a baseline of data on the distribution of background radioactivity in the environment near the plant site. With this background information, it will then be possible to determine any statistically significant changes in the radioactivity levels. The preoperational environmental radiological monitoring program will be initiated approximately two years prior to plant operation. The program outlined herein is based on current regulatory guidelines and
57 monitoring philosophy. At such time that any monitoring program is implemented, it may be revised to reflect changes in regulatory requirements and monitoring philosophy.

Evaluations after plant startup will be made on the basis of the baselines established in the preoperational program, considering geography and the time of the year where these factors are applicable, and by comparisons to control stations where the concentrations of station effluents are expected to be negligible. In those cases where statistically significant increase in the radioactivity level is seen in a particular sampling vector but not in the control station, meteorology and specific nuclide analysis will be used to identify the source of the increase.

The planned sampling frequencies will ensure that significant changes in the environmental radioactivity can be detected. The vectors which would first indicate increases in radioactivity are sampled most frequently. Those which are less effected by transient changes but show long-term accumulations are sampled less frequently. However, specific sampling dates are not crucial and adverse weather conditions or equipment failure may on occasion prevent collection of specific samples.

The capability of the environmental monitoring program to detect design-level releases from plant effluents is uncertain because of the small quantities which are expected to be released. The program will however provide the capability of detecting any significant buildup of radioactive material in the environment above and beyond that which is already present. Those vectors which are most sensitive to reconcentration of specific isotopes are sampled. If any increase in radioactivity levels is detected in these vectors, the program will be evaluated and broadened if deemed necessary.

From the data obtained from the radioanalytical and radiochemical analyses of the vectors sampled, dose estimates can be made for an individual or the population living near the plant site.

11.6.1 Expected Background

For a number of years measurements of background radiation have been made at various locations throughout the Tennessee Valley region. Environmental monitoring programs have been conducted in the vicinity of Oak Ridge, Watts Bar, and Chattanooga, Tennessee, and Decatur, Alabama. Over periods of not less than two years, the measurements made in these areas have indicated only very slight variations from location to location. The measurements obtained utilizing film badges or thermoluminescent dosimeters have revealed the following background radiation levels: Oak Ridge 78 mR/year, Chattanooga 71 mR/year, Watts Bar 68 mR/year, and Decatur 71 mR/year. It is estimated that the expected

57 background levels in the vicinity of the Clinch River Breeder Reactor Plant (CRBRP) will be between 60 and 90 mR/year. Measurements for the period 1977-1978 indicate the following yearly variations: Oak Ridge 73-83 mR, Chattanooga 68-74 mR, Watts Bar 68-69 mR, and Decatur 70-73 mR.

Measurements will be made in the immediate vicinity of the CRBRP site and will provide baseline data necessary for comparison of background radiation levels prior to and after startup of the plant.

11.6.2 Critical Pathways to Man

Although the amounts of radioactivity added to the environment from plant operations are small, critical exposure pathways to man have been identified in order to estimate the maximum dose to the individual and to establish the sampling requirements for the environmental radioactivity monitoring program. The six principal pathways which can result in radiation exposure to man are as follows:

a. External exposures and inhalation of gaseous releases.

b. Drinking water from the Clinch River and from wells in the immediate vicinity of the plant.

c. Swimming, boating, and fishing in the Clinch River.

d. Eating fish from the Clinch River.

e. Consuming animal flesh and other animal products which may be affected by plant operations.

f. Eating foods grown in areas adjacent to the plant site affected by plant releases.

The environmental monitoring program as outlined, provides sampling necessary to evaluate the dose received through the critical pathways in items a. through f. above. The following items indicate the samples collected in order to make the critical pathway-dose correlations:

57 a. Data from readings of the thermoluminescent dosimeters will be utilized to estimate external exposures and data from offsite air monitors will be used to estimate contributing internal exposures.

b. Analysis of water samples collected will be used to estimate the dose that might be received from drinking water from the Clinch River or from wells in the vicinity of the plant.

c. Analysis of water samples will also be used to estimate the dose an individual might receive while swimming, boating, or fishing on the lake in the vicinity of the plant.

57 d. Analysis of samples of river water, sediment, and fish will be correlated to estimate the dose that might be received by an individual who eats fish from the Clinch River.

e&f. Analysis of samples of air, particulate matter, soil, vegetation, food crops, and milk will be used to estimate the dose to the surrounding popu-
57 lation through the consumption of food or dairy products.

The environmental monitoring program to be conducted throughout operation of the plant provides the necessary means of evaluating the dose to man through critical exposure pathways.

Environmental concentrations of radioactivity due to plant releases to unrestricted areas may be so low as to be unmeasurable with present techniques. Therefore, methods to calculate the potential exposure to man have been derived for both gaseous and liquid releases.

11.6.2.1 <u>Doses from Gaseous Effluents</u>

The following doses to humans living in the vicinity of the CRBRP will be calculated for the releases of radioactive gases:

a. External beta- and gamma-air doses from airborne radioactivity

57 b. Total body and skin doses from direct radiation due to ground contamination

c. Internal doses from inhalation

d. Internal doses from ingestion

The basic assumptions and calculational methods that will be used in computing these doses are similar to that described in the appendix to Section 11.3.

57 Review of the data resulting from the offiste monitoring program and reevaluations of the adequacy of the dose models will verify that the actual doses received by individuals and the population as a whole remain within the applicable Federal Regulations and as low as reasonably achievable.

11.6.2.2 <u>Internal Doses from Liquid Effluents</u>

The following doses will be calculated for exposures to radionuclides routinely released in liquid effluents:

a. Internal doses from the ingestion of water

b. Internal doses from the consumption of fish

57 c. External and internal doses from water sports

d. External doses from shoreline activities

The basic assumptions and calculational methods that will be used in computing these doses are similar to that described in the appendix to Section 11.2.

The dose models that are employed will be reevaluated in light of the data resulting from the offsite monitoring program to ensure that all signi-

57 ficant pathways are included in the calculations and to verify that the actual doses received by individuals and the population as a whole remain within the
38 applicable Federal Regulations and as low as reasonably achievable.

11.6.3 Sampling Media, Locations, and Frequencies

The sampling media, the locations from which the samples are collected, and the frequency with which the samples are collected are presented in Table 11.6-1. Tentative sampling locations are shown in Figures 11.6-1 and
57 11.6-2. The final selection of sampling locations will be made approximately one year prior to implementation of the program. The media selected were chosen on two bases: First, those vectors which would readily indicate significant increases in radioactivity levels, and secondly, those vectors which would indicate long-term buildup of radioactivity. Consideration was also given to the pathways which would result in exposure to man, such as milk and food crops. Locations for sampling stations were chosen after considering meteorological factors and population density around the site. Frequencies for sampling the various vectors were established so that seasonal variations in radioactivity levels might be determined. In addition, samples are collected during the season in which the major growth occurs to ascertain radioactivity uptake by the vectors during their most susceptible period of growth.

11.6.4 Analytical Sensitivity

Samples will be collected routinely following established procedures so that uniformity in sampling methods will always be assured. The samples
57 will be transported to a laboratory facility for preparation and processing. All the radioanalytical and radiochemical analyses will be conducted in that laboratory. The following types of equipment will be utilized in performing the analyses: Pulse height analyzers with solid and well NaI detectors and Ge(Li) detectors; low background beta counters; liquid scintillation counters; GM detec-
57 tors; and internal proportional counters. Data will be coded and stored in computerized data base.

The detection capabilities for environmental sample analyses will be presented in the PSAR. The nominal lower limit of detection (LLD) for the various analytical techniques will be based on the method discussed in HASL-300
57 (ref. 1). The nominal LLD values are expected to approximate the values recommended in Regulatory Guide 4.2: However, the LLDs will vary depending on the
38 activities of the various components in the samples.

11.6.5 Data Analysis and Presentation

A quality control program has been established with the Tennessee Department of Public Health Radiological Laboratory. Samples of air particu-
57 lates, water, and milk will be collected and forwarded to this laboratory for analysis. The results will be exchanged for comparison to aid the laboratories in evaluating their analytical systems and minimizing errors in data production.

Data collection around the operating plant will be compared to data from control stations and from the preoperational program to identify the earliest possible indications of the accumulation or buildup of radionuclides in the environment. During the life of the plant, this accumulation should exist in
38 no more than trace amounts, with only minor impact on the environment.

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Reports describing the results of the environmental radiological monitoring activities will be submitted routinely as required by the Technical
38 Specifications. The reports will follow the format used in reporting environmental radiological data from TVA's other nuclear power facilities.

11.6.6 Program Statistical Sensitivity

As previously noted, because of the expected small quantities of radioactive material which may be released to the environment from the CRBRP, it is uncertain as to what extent the results from the environmental monitoring program will be used to estimate the probable radiation exposure to man. Only if the radioactive waste releases from the plant cause statistically measurable increses of radiation in the environment can dose correlations be made.

Results from the analysis of effluent samples, which contain higher concentrations of radionuclides than environmental samples, will be used in the TVA models similar to those given in the appendices to Sections 11.2 and 11.3 to estimate the possible exposure to man. Because of the conservative assumptions applied in these models, the estimated dose to the population should be higher than that actually received. However, TVA, even using the conservative assumptions, will control the releases of radioactive materials to the environ-
57 ment such that the releases remain within the applicable Federal Regulations and as low as reasonably achievable.

57 The statistical sensitivity of the monitoring during accident conditions will not be different from those during normal plant operation and is expected to detect concentrations well below 10CFR Part 20 limits.

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REFERENCES

57 1. HASL-300, HASL Procedure Manual, J. H. Harley, Ed., Rev. August 1974.

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TABLE 11.6-1

ENVIRONMENTAL RADIOLOGICAL SURVEILLANCE PROGRAM

	Criteria and Sampling Locations	Collection Frequency	Analysis/Counting
I. Atmospheric			
A. Air			
1. Particulate	Filter paper at 12-15 locations	Weekly[b]	Gross beta, gross alpha, (gamma scan monthly, Pu and U quarterly), (^{89}Sr, ^{90}Sr)[c]
2. Radioiodine	Charcoal filter at 12-15 locations	Weekly[b]	^{131}I
3. Moisture	Sampling at 3-8 locations	Weekly[b]	^{3}H
B. Fallout	Gummed acetate at 12-15 locations	Monthly	Gross beta, gross alpha
C. Rainwater	Rainwater collection trays at 12-15 locations	Monthly	Gross beta, gamma scan ^{89}Sr, ^{90}Sr, ^{3}H
II. Reservoir			
A. Water			
1. Municipal (Public supplies)	All public water supply intakes within 10 miles upstream and downstream of the plant	Monthly[a]	Gross beta, gross alpha, gamma scan, ^{3}H monthly, Pu quarterly
2. River	Continuous samples from 4-8 locations	Analyzed Monthly	Gross beta, gross alpha, gamma scan ^{3}H monthly, ^{89}Sr, ^{90}Sr, Pu, U quarterly

a. All public water supplies within 10 miles downstream of the plant will be collected automatically and analyzed monthly.

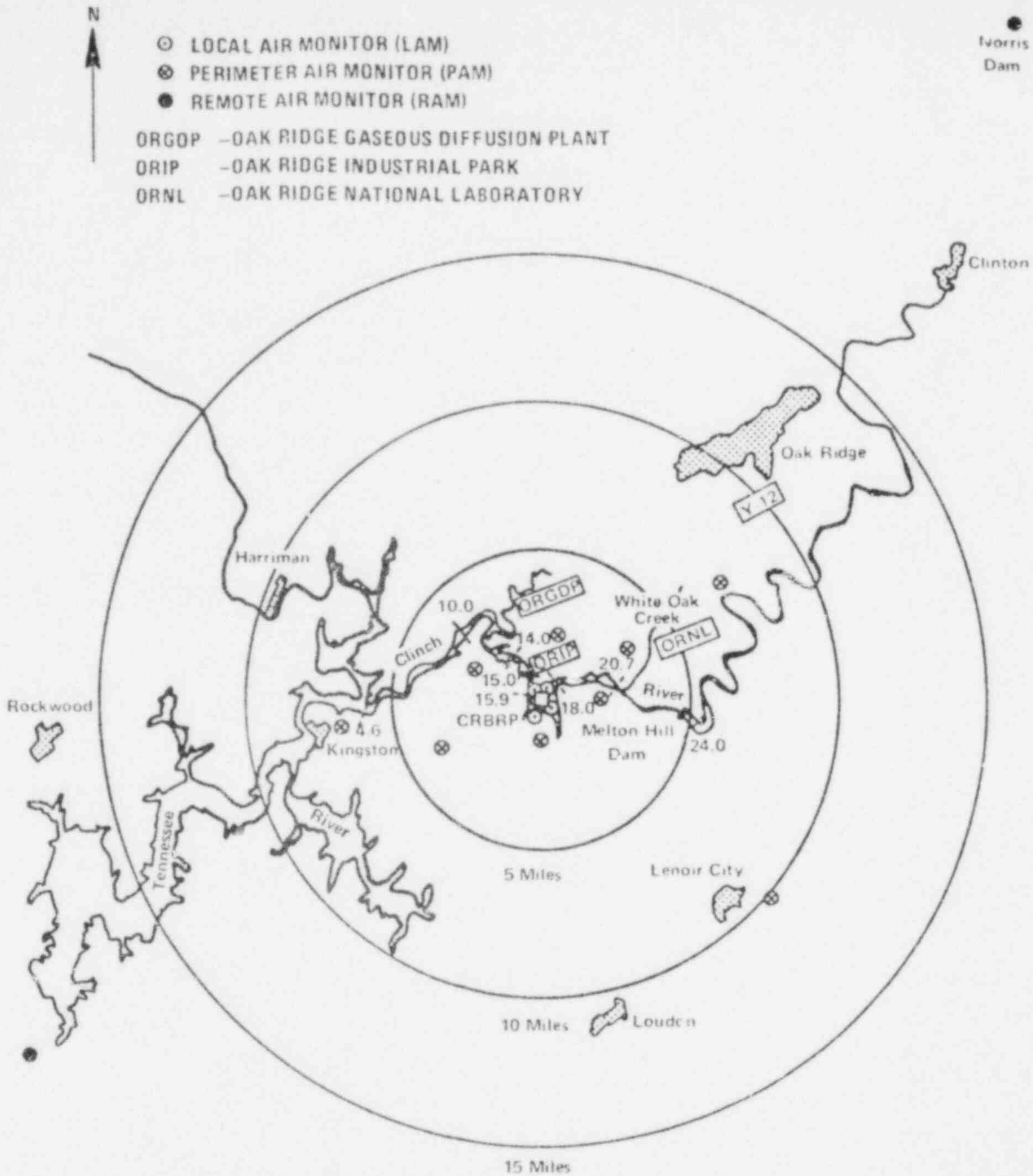
b. Continuous sampling.

c. Radiostrontium composite analyses if ^{137}Cs is found in the gamma scan.

TABLE 11.6-1 (Continued)

	Criteria and Sampling Locations	Collection Frequency	Analysis/Counting
B. Aquatic Biota			
1. Fish	Two locations	Semiannually	Gross beta, gross alpha, gamma scan ^{89}Sr, ^{90}Sr, Pu
2. Shellfish (if available)	Four to six locations	Semiannually	Gross beta, gross alpha, gamma scan (^{89}Sr, ^{90}Sr, Pu shells only)
C. Sediment	Four to six locations	Semiannually	Gross beta, gross alpha, gamma scan, ^{89}Sr, ^{90}Sr, Pu
III. Terrestrial			
A. Soil	Atmospheric monitoring locations	Annually	Gross beta, gross alpha, gamma scan, Pu, U
B. Vegetation			
1. Pasturage grass	Selected dairy farms within 10-mile radius of plant	Quarterly	Gross beta, gross alpha, gamma scan, ^{89}Sr, ^{90}Sr, Pu
2. Grass	Collected at atmospheric monitoring stations	Quarterly	Same as pasturage grass
3. Food crops	Within 10-mile radius of plant	Annually	Gross beta, gross alpha gamma scan, ^{89}Sr, ^{90}Sr, Pu
C. Milk	Selected dairy farms within 10-mile radius of plant	Monthly[d]	Gamma scan, ^{89}Sr, ^{90}Sr, ^{131}I
D. Well water	Selected farms within 5 miles of plant and 1-5 wells on site	Monthly	Gross beta, gross alpha, gamma scan monthly, Pu quarterly
E. Direct radiation	TLDs on site and at atmospheric monitors	Quarterly	Dose determination

d. After the plant begins operation milk samples will be taken at least biweekly for ^{131}I analysis when cows are on pasture.



NOTE: THE FOLLOWING SAMPLES ARE COLLECTED AT EACH MONITORING SITE:

AIR PARTICULATE	RAINWATER
RADIOIODINE	SOIL
HEAVY PARTICLE FALLOUT	VEGETATION

Figure 11.6-1. Atmospheric and Terrestrial Monitoring Network — Clinch River Breeder Reactor

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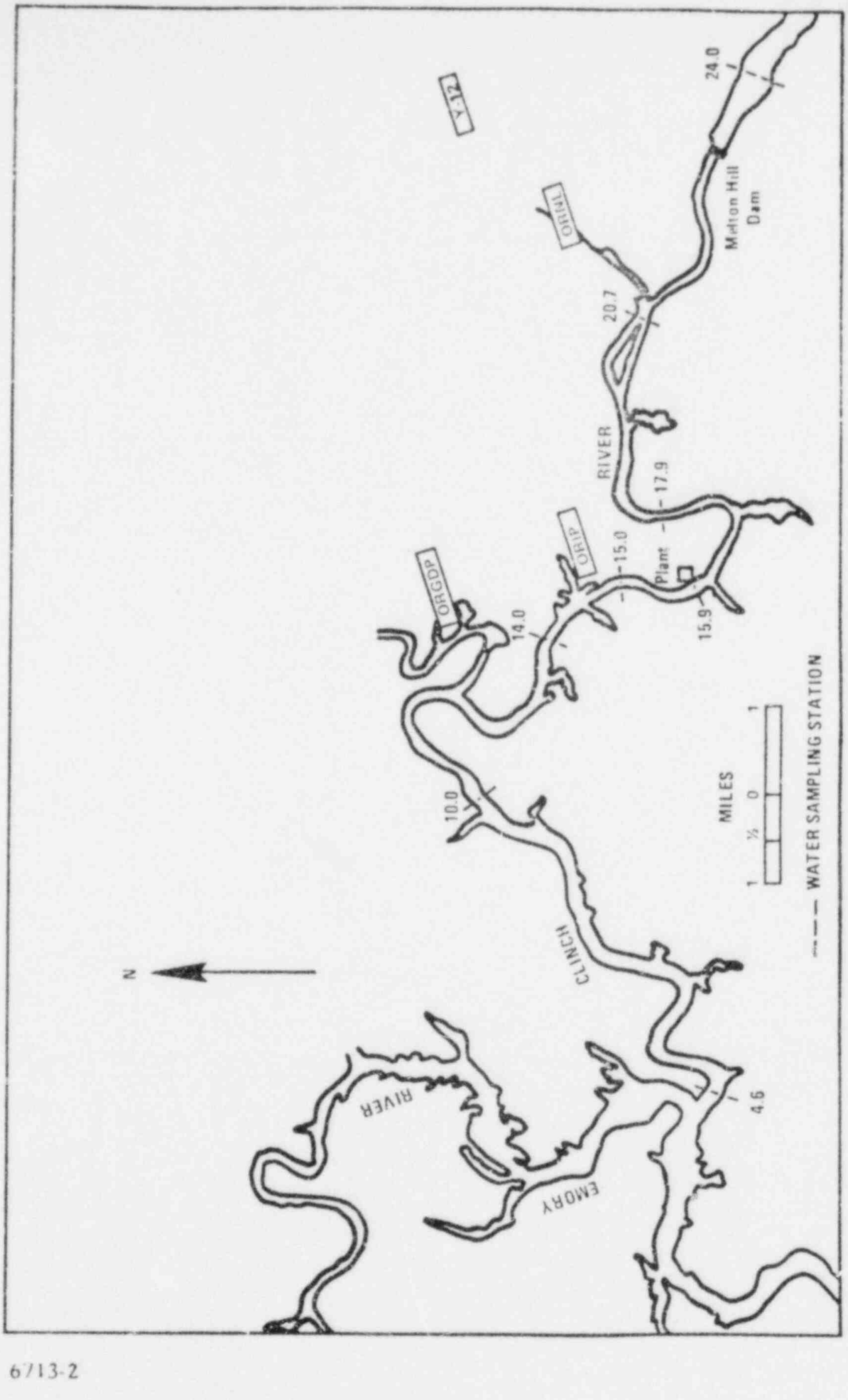

Figure 11.6-2. Reservoir Monitoring Network Clinch River Breeder Reactor

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12.3 HEALTH PHYSICS PROGRAM

12.3.1 Program Objectives

The health physics staff is a unit of the Radiological Hygiene Branch and is responsible for the health physics activities at the plant. It applies radiation standards and procedures; reviews proposed methods of plant operation; participates in development of plant documents; and assists in the plant training program, providing specialized training in radiation protection. During preoperational tests and after plant startup it provides health physics coverage for all operations including maintenance, fuel handling, waste disposal, and decontamination. It is responsible for personnel and inplant radiation monitoring, and maintains continuing records of personnel exposures, plant radiation, and contamination levels. Through implementation of the described program, plant personnel exposures will be maintained as low as is reasonably achievable (ALARA).

The health physics staff is under the administrative supervision of the Chief, Radiological Hygiene Branch, in the TVA Division
57 of Occupational Health and Safety.

The plant health physicist is responsible for direction of an adequate program of health surveillance for all plant operations involving potential radiation hazards. He keeps the plant superintendent informed at all times of radiation hazards and conditions related to potential exposure, contamination of plant and equipment, or contamination of site and environs. His duties include training and supervising health physics technicians; planning and scheduling monitoring and surveillance services; scheduling technicians to ensure around-the-clock shift coverage as required; maintaining current data files on radiation and contamination levels, personnel exposures, and work restrictions; and ensuring that operations are carried out within the provisions of developed radiological hygiene standards and procedures. He provides assistance and advice to the plant superintendent during radiological emergencies.

In addition, off-site staff from the Radiological Hygiene Branch is responsible for conducting a comprehensive environmental monitoring program prior to, during, and after plant startup. The Division
57 of Occupational Health and Safety also advises on potentially harmful factors in the working environment other than radiation, all in relation to identified and approved standards as given in the TVA Hazard Control Manual.

Health physics personnel that are assigned to the plant staff will meet the provisions of Regulatory Guides 1.8 and 8.10 or Regulatory Guide 1.8 and Section 4 of ANSI N18.1, whichever is applicable at the time of appointment. As TVA has a highly qualified health physics staff, there should be no problem in meeting the provisions of the applicable
49 guides.

TVA has established a formal program to ensure that occupational radiation exposures to employees are kept as low as is reasonably achievable (ALARA) which will be applied to CRBRP. The program consists of: (1) full management commitment to the overall objectives of ALARA; (2) issuance of specific administrative documents and procedures to the TVA design and operating groups that emphasize the importance of ALARA throughout the design, testing, startup, operation, and maintenance phases of TVA nuclear plants; (3) continual appraisal of radiological conditions in the operating nuclear plants by an on-site health physics staff, and (4) a 4-member corporate ALARA committee consisting of representatives from the TVA design, operations, and radiation protection groups, whose purpose is to review and appraise the effectiveness of the ALARA program on a plant-by-plant basis including CRBRP. This committee consists of key management and technical staff who have extensive backgrounds in inplant radiation control, including such areas as plant layout, shielding, personnel access control, ventilation, waste management, area and personnel monitoring, plant operations, and plant maintenance. The committee periodically evaluates TVA's overall ALARA program by assessing trends in occupational exposures or other radiation control problems, reviewing plant operating reports and radiation exposure profiles, and conducting onsite audits of each plant's ALARA efforts.

Specific responsibilities of the ALARA committee include the following:

(a) Determines that an effective ALARA program is established at each TVA nuclear power plant that appropriately integrates TVA management philosophy and NRC regulatory requirements;

(b) Determines that the ALARA program is implemented from initial planning through decommissioning of the plant;

(c) Reviews plant design features, operating procedures, and maintenance practices and audits the onsite radiation control program at least annually to assure that the objectives of the ALARA program are attained;

(d) Determines that information and data pertaining to radiation exposure of personnel from other operating LWR power plants are reflected in the design and operation of new TVA plants;

(e) Determines that experience gained during the operation of nuclear power plants relative to inplant radiation control is factored into revisions of operating procedures, where necessary, to assure that the procedures indeed do meet the objectives of the ALARA program;

(f) Determines that all maintenance activities are planned and accomplished in accordance with the objectives of the ALARA program; and

49

(g) Determines trends in the exposure of station personnel in order to permit actions to be taken to correct adverse trends.

Reports of the findings of the ALARA committee are promptly conveyed to top-level management staff along with appropriate recommendations for improvements in the design of new plants or corrections in operating plants.

12.3.2 Facilities and Equipment

The health physics facilities necessary to monitor and control the routine radiological condition of the plant is shown on the Plant Service Building General Arrangement drawing in Section 1.2. The focal point of control is the assembly area. All personnel entering or leaving the controlled (restricted) area must pass through this area. All other
52 doors are for emergency use only. The general entry control requirements will be coordinated with industrial security.

The control point is equipped with the following:

a) A health physics work station for routine counting and assignment of required equipment to workers entering the restricted area.

b) Mask-Protective clothes issue and storage area.

c) Male/female lockers, toilet facilities shower station

d) Mask cleaning station

e) Combined laboratory

f) Chemical storage room

g) Counting room

h) Hot instrument shop

The main personnel monitoring station is at the Reactor Service Building entrance from the Plant Service Building. This entrance/exit door will be equipped with all necessary monitoring equipment.

Contamination control, within the Plant Service Building control area, such as entering and exiting from counting room, hot instrument shop, mask cleaning area, laboratory area, radwaste area, will be accomplished by use of local survey equipment located at the accesses to these areas.

Contaminated laundry will be bagged, surveyed, and shipped
49 off-site for laundering.

For radiological purposes, all other areas, i.e., outside the RCB, RSB, Intermediate Bay and the cells in the PSB are uncontrolled (unrestricted).

Portable and laboratory equipment located in the health physics work station will allow the health physics personnel to measure dose rates and contamination levels throughout the plant in all routine and emergency situations. The portable health physics survey instrumentation is listed in Table 12.3-1 with the operational characteristics for each instrument. The fixed health physics laboratory counting systems are described in Table 12.3-2.

All potentially contaminated liquid drains in this area will be routed to the radwaste system and all potentially contaminated gaseous exhausts will be HEPA filtered.

Portable survey instrumentation will be checked and calibrated
57 routinely with standard radioactive sources by the TVA Branch Laboratory in Muscle Shoals, Alabama. Accurate records on the performance of each instrument during each calibration will be maintained at this laboratory. Calibration and maintenance procedures specific for each instrument are written and routinely used. Each laboratory counting system is checked at regular intervals with standard radioactive sources for proper counting efficiencies, background count rates, and high voltage settings by health physics personnel at the plant.

TVA will provide protective clothing for use in radiation areas. Clothing required for a particular instance shall be prescribed by the Health Physics Staff based upon the actual or potential radiological conditions. Protective clothing available for use are:

a. Coveralls

b. Lab coats

c. Gloves - plastic and/or latex in light and heavy weights

d. Gloves - cotton in heavy weights and light weights

e. Head covers - skull caps and hoods

f. Foot covers - shoe rubbers and plastic booties

g. Plastic suits

Tape will be provided so that openings in clothing and between
49 pieces can be sealed.

TABLE 13.3-4

PROJECTED MAXIMUM RESIDENT AND TRANSIENT POPULATION* IN EVACUATION SECTORS WITHIN 5 MILES OF CRBRP

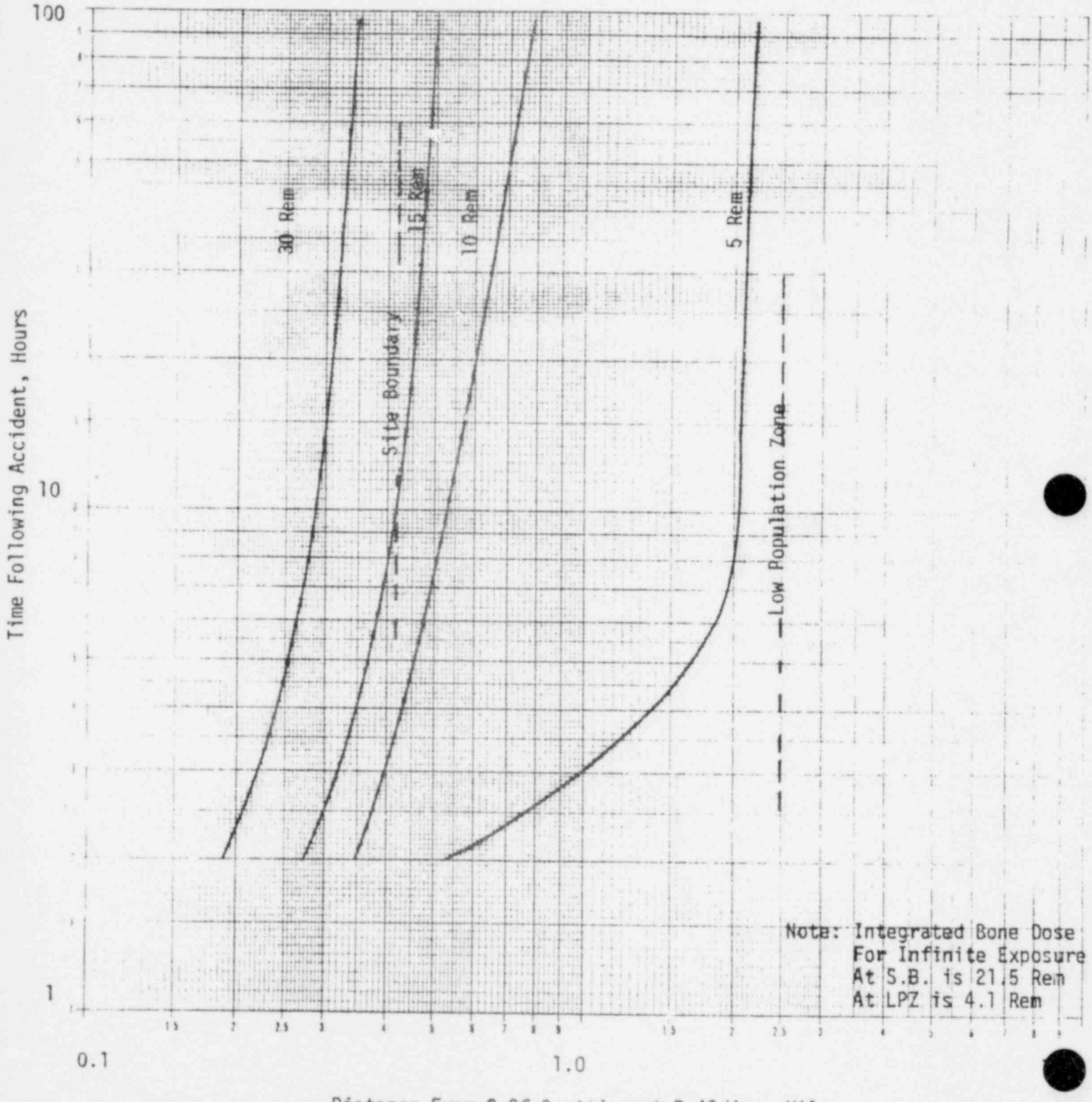
Sector+	1980	2010
A	6545	6520
B	7497	9162
C	885	885
D	960	1055
E	1365	1955
F	6220	6295

*Transient population is based on current available information

50 +See Figures 13.3-5 and 13.3-6

FIGURE 13.3-1

ELAPSED EXPOSURE TIME TO REACH SPECIFIC BONE DOSE VERSUS DOWNWIND DISTANCE (BASED ON SITE SUITABILITY SOURCE TERM)



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POOR ORIGINAL

FIGURE 13.3-2

ELAPSED EXPOSURE TIME TO REACH SPECIFIC LUNG DOSE VERSUS DOWNWIND DISTANCE (BASED ON SITE SUITABILITY SOURCE TERM)

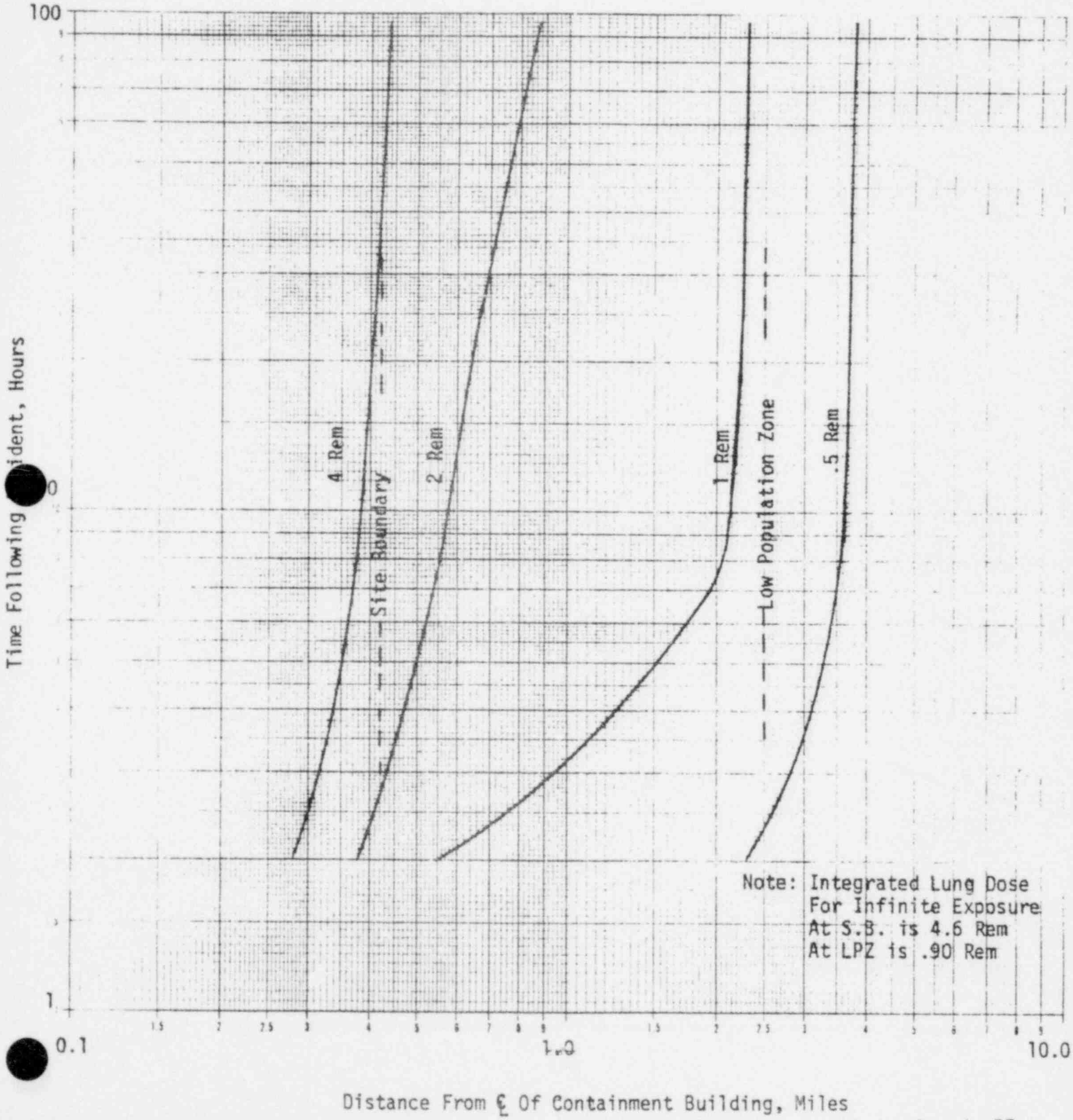

FIGURE 13.3-3

ELAPSED EXPOSURE TIME TO REACH SPECIFIC THYROID DOSE VERSUS DOWNWIND DISTANCE (BASED ON SITE SUITABILITY SOURCE TERM)

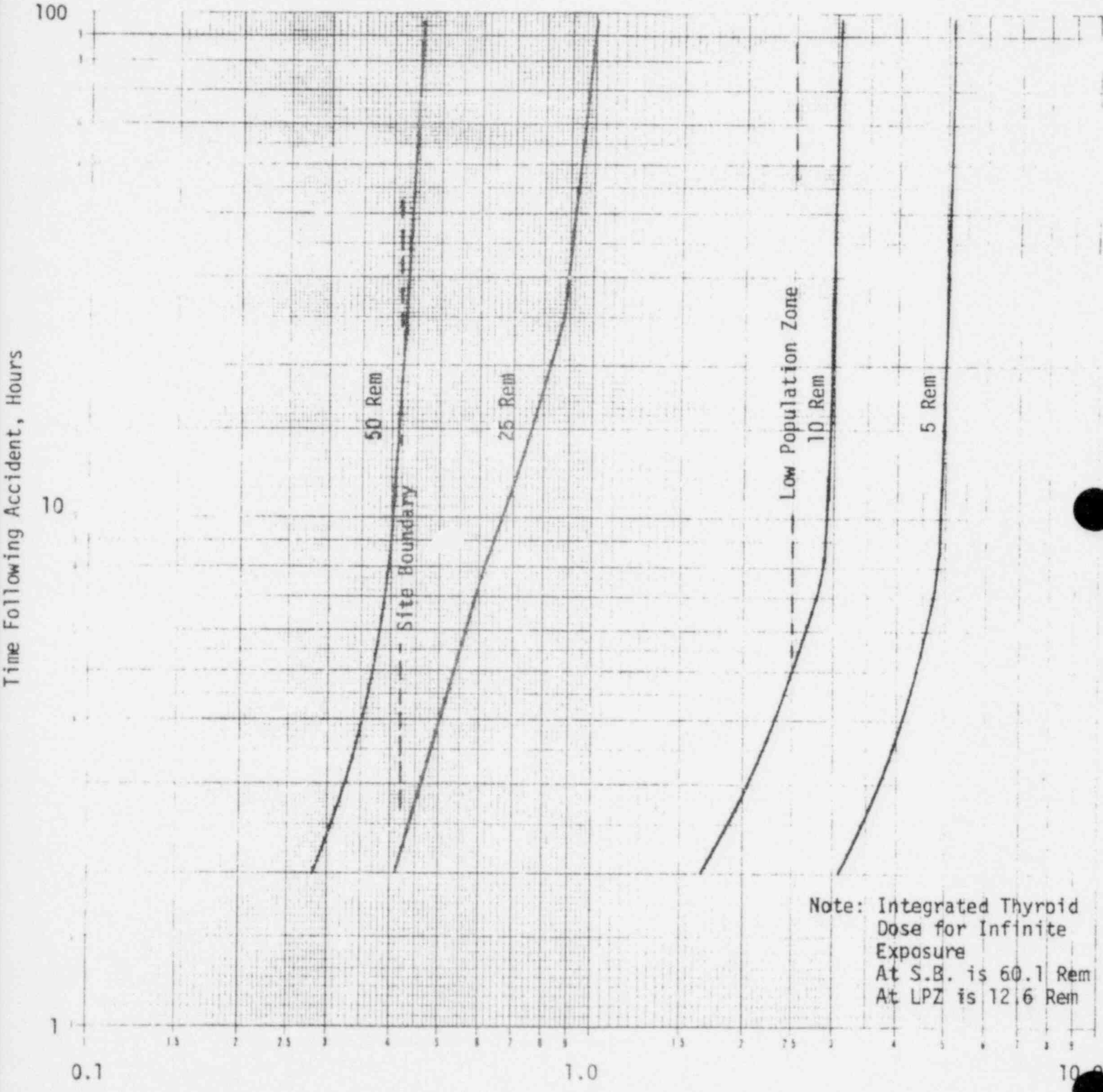

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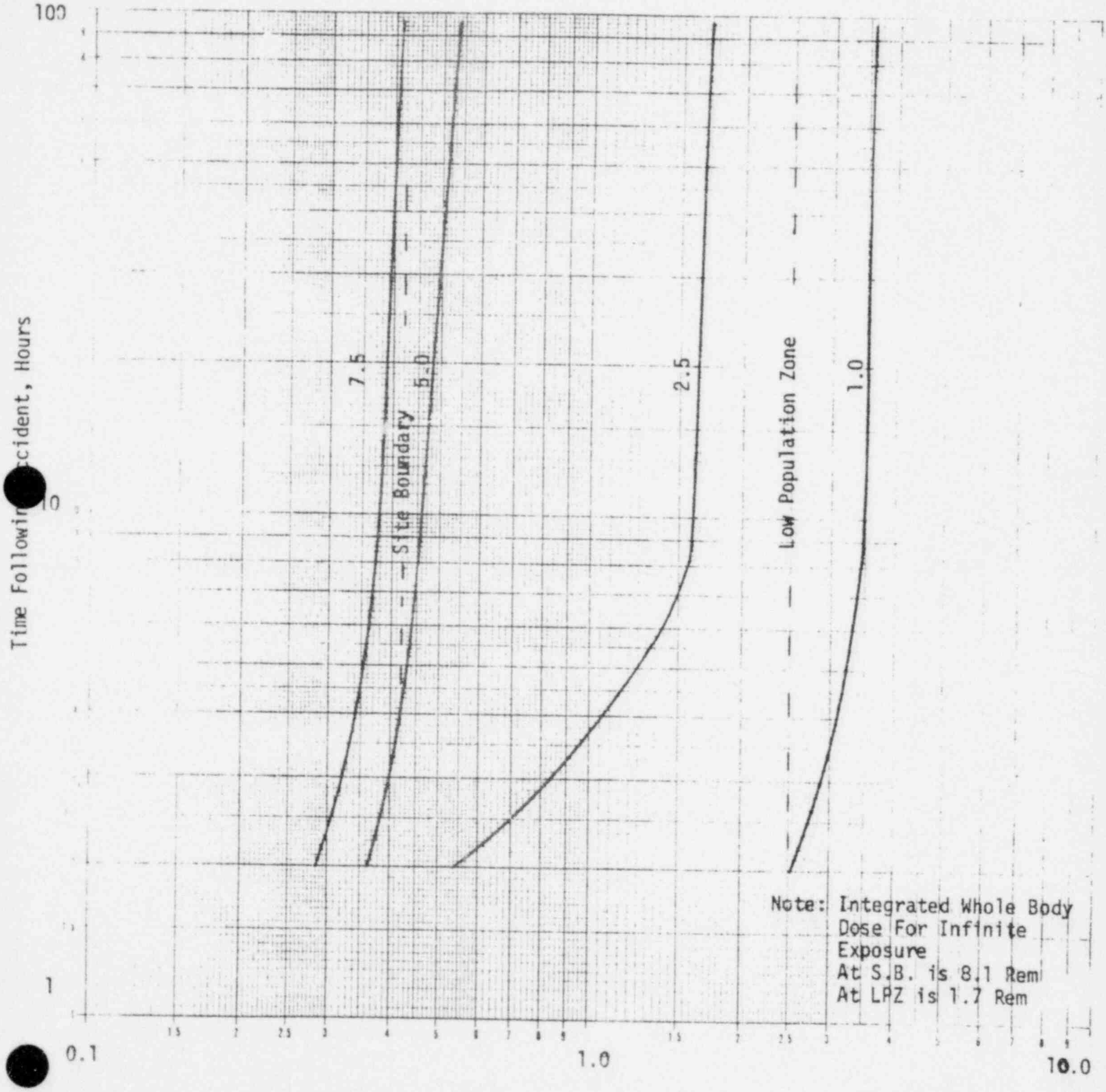
FIGURE 13.3-4

ELAPSED EXPOSURE TIME TO REACH SPECIFIC WHOLE BODY DOSE VERSUS DOWNWIND DISTANCE (BASED ON SITE SUITABILITY SOURCE TERM)



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40

15.A.1 INTRODUCTION

In accordance with Title 10 Code of Federal Regulations Part 50 (10CFR50), the CRBRP Project has submitted an Environmental Report (ER) and a Preliminary Safety Analysis Report (PSAR) to support an application
40 for a license to construct the CRBRP. These reports include an evaluation of a spectrum of postulated accidents. For each accident an analysis of the potential consequences to the health and safety of the public is presented. Consistent with the intent of Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Plants", the accident evaluations presented in the ER are based on realistic accident analyses and analytical assumptions. The evaluations presented in the PSAR are based on conservative accident analyses and analytical assumptions. The spectrum of accidents considered in Chapter 15 of the PSAR and Chapter 7 of the ER encompass Class 1 through Class 8 events. This spectrum constitutes the accidents included in the design base for the plant. Class 9 events are of such low probability that they can be excluded from the design bases.

29

In accordance with Title 10 Code of Federal Regulations Part 100 (10CFR100), a major fission product release from the core has been hypothesized for the purpose of determining the suitability of the selected site for the construction and operation of the CRBRP. In compliance with 10CFR100, the potential hazards resulting from this hypothesized release are not exceeded by those from any design basis accident analyzed in Chapter 15 of the PSAR. The radiological source term associated with this hypothetical release is specified in terms of percentages of fission products and fuel material released from the core to the Reactor Containment Building. The source term used for site suitability assessment is as follows:

100% Noble Gas Inventory
50% Halogen Inventory (25% Airborne)
1% Solid Fission Product Inventory
1% Plutonium

The applicant has utilized this source term in compliance with specific direction from the Nuclear Regulatory Commission (Ref. 1). However, while accepting this source term and committing to design features to assure acceptable consequences as a result of it, the applicant considers
40 this source term to be overly conservative.

The source term specified by NRC not only envelopes all design basis accidents considered in Chapter 15, but further envelopes a wide range of conservatively hypothesized core-related events. Evidence, both analytical and experimental, supports the Applicant's position that compliance with the requirements of 10CFR100 could be demonstrated with a less stringent source term.

40 The potential radiological consequences of the above source term are conservatively calculated and compared to the guideline values of 10CFR100, thus providing the basis for conducting an assessment of the 29 site suitability.

15.A.2 SITE SUITABILITY SOURCE TERM

15.A.2.1 Source Term

The source term is identified in terms of percentages of fission products and fuel material released from the core to the Reactor Containment Building. The source term is itemized in Table 15.A-1. The indicated percentages of these materials are assumed instantly released to and uniformly distributed in the RCB. For the halogens, 50% of the halogens released to the RCB are assumed to immediately plateout on surfaces (consistent with LWR practice), thus being removed from the airborne source term available for leakage from the RCB, with the net result that 25% of the initial halogen inventory is assumed airborne in the RCB.

The initial core fission product inventories are based on end-of-cycle equilibrium core conditions for power operation at 975 megawatts-
57 54 thermal.

The specific isotopes included in each fission product category, as identified in Table 15.A-1, are as follows:

Noble Gases:	Xe, Kr
Halogens:	Br, I
40 Solids:	All remaining fission products

The mass associated with the non-gaseous portion of the source term, initially airborne in the RCB, is shown in Table 15.A-4. The resultant initial airborne concentration is also provided.

57 The quantity of fuel (62.4 kg) included in the source term aerosol
51 analysis was selected to represent 1% of the total (core plus blanket)
plutonium-oxide mass plus 1% of the core uranium-oxide mass. The mass of the uranium-oxide in the blanket was not included in the aerosol analysis. This approach is conservative since including the uranium blanket mass would result in a much higher initial airborne concentration and subsequently more rapid aerosol depletion. Even though the uranium blanket mass has been excluded from the aerosol analysis, it has been conservatively assumed that the radioactive inventory of the blanket is included in the source term.

Table 15.A-5 presents the important input parameters to the HAA-3 code, used to compute the concentration-time behavior of the source term aerosol and resultant aerosol depletion factors. The time-dependent depletion factors computed by HAA-3 for the source term aerosol and used in the COMRADEX radiological analysis are itemized in Table 15.A-6.

15.A.2.3 Containment Modeling

A complete description of the reactor containment/confinement system and the engineered safeguards associated with it is presented in Chapter 6 of the PSAR.

For the radiological analysis, it is conservatively assumed that all leakage (except bypass) from the RCB to the annulus is directly to the intake of the filter system. This assumption neglects any credit for delay time in the annulus. The recirculation flow was assumed to mix in 50% of the annulus volume. Only one-half of the annulus volume is used to be consistent with the 50% mixing assumption specified in
40 Standard Review Plan Section 6.5.3.

Leakage of airborne radioactivity from the RCB was assumed to occur at the containment design leak rate, 0.1% Vol/Day for the duration of the evaluation. The RCB is designed to limit leakage to 0.1% Vol/Day at a containment overpressure of 10 psig. The use of the containment design leak rate (0.1% Vol/Day) for the duration of the site suitability source term evaluation is conservative, since assuming a constant 10 psig containment overpressure for the duration of the site suitability source term evaluation is conservative.

A portion of the leakage from the RCB may bypass the confinement annulus. Chapter 6 of the PSAR identifies the individual containment penetrations contributing to bypass leakage; the majority of the bypass leakage is associated with the containment airlocks. The containment/confinement system is being designed to achieve a bypass leakage value of less than 1% of the RCB design leak rate, i.e., 1% x 0.1% Vol/Day = 0.001% Vol/Day. Sixty percent of this bypass leakage escapes directly to the outside atmosphere and the remaining forty percent escapes to the Reactor Service Building (RSB). The treatment of leakage to the RSB depends upon the status of the railroad door in the RSB. When the railroad door is closed, the RSB atmosphere is maintained at a negative pressure with respect to the outside atmosphere. When the railroad door is open, maintenance of a negative pressure in the RSB is not assured.

If the RSB railroad door is open, both doors of the equipment hatch airlock are secured and the airlock atmosphere is vented to the containment/confinement Annulus Filtration System. In this mode, essentially all (96.4%) the bypass leakage from the RCB to the RSB (40% of total bypass) is vented from the equipment hatch airlock directly to the Annulus Filtration System, where it is subject to filtration and recirculation prior to release to the environment. The remainder of leakage into the RSB (3.6%) escapes directly to the atmosphere. When the railroad door is closed, the airlock vent to the Annulus Filtration System is closed and the airlock atmosphere is isolated from the containment/confinement annulus. In this mode, all bypass leakage from the RCB to the RSB (40% of total bypass) escapes directly to the RSB where it is subject to recirculation and filtration prior to release to the environment.

Airlock operation with the railroad door open (i.e., with the airlock atmosphere vented to the annulus) results in larger potential off-site exposures for the site suitability source term analysis than operation with the railroad door closed and the radiological consequences are therefore presented when the railroad door is assumed open. Confirmation that this does result in more limiting exposures is given below.

When the railroad door is closed and all bypass leakage to the airlock escapes to the RSB, this leakage is filtered prior to ultimate release to the environment. Considering the efficiencies (99% particulate and 95% iodine) of the RSB filters and the recirculation flow pattern (1700 cfm exhausted per 14300 cfm recirculated), the net filtration efficiency of the RSB system is greater than 99% for both particulates and halogens. Conse-
40 quently, non-gaseous releases (which are controlling with respect to off-site

TABLE 15.A-2

HEAVY METAL* MASS (KG) INVENTORY IN THE CRBRP (EOFC)

	Fuel	Inner Blanket[(a)]	Radial Blanket[(a)]	Lower Axial Blanket	Upper Axial Blanket
End-of-Fourth-Cycle					
Pu-239	1216.	206.8	285.6	34.9	21.2
Pu-240	273.5	8.0	11.3	0.9	0.3
Pu-241	32.7	---	---	---	---
Pu-242	5.2	---	---	---	---
U-235	5.4	11.6	21.3	3.8	4.0
U-238	3421.	7381.	12936.	2149.	2165.
Fission Products	414.2	55.2	55.7	4.4	2.4
Total Heavy Metal	5368.0	7662.6	13309.9	2193.0	2192.9

* Heavy metal excludes oxygen.

51 57 (a) Including axial extensions

TABLE 15.A-3

CRBRP TRANSURANIC INVENTORY (EOEC)

	Isotope	Half-Life	Mass (gms)	Curies
	Np237	2.14×10^{6} Y	3.38×10^{3}	2.38×10^{0}
	Np238	2.1 D	1.50×10^{0}	3.93×10^{5}
	Np239	2.35 D	4.08×10^{3}	9.48×10^{8}
	Am241	458 Y	7.33×10^{3}	2.51×10^{4}
	$Am242^{m}$	152 Y	1.62×10^{2}*	1.57×10^{3}*
	Am242	16 H	3.81×10^{0}	3.08×10^{6}
	Am243	7650 Y	2.17×10^{2}	4.[illegible]8 $\times 10^{1}$
	Am244	10 H	8.69×10^{-4}	[illegible].58 $\times 10^{4}$
	Cm242	163 D	6.23×10^{2}	2.06×10^{6}
	Cm243	32 Y	2.27×10^{1}	1.04×10^{3}
	Cm244	18.1 Y	8.70×10^{0}	7.05×10^{2}
	Cm245	9320 Y	1.45×10^{-1}	2.57×10^{-2}
	Cm246	5480 Y	2.19×10^{-3}	6.77×10^{-4}
	Cm247	1.67×10^{7} Y	1.92×10^{-5}	1.69×10^{-9}
	Cm248	4.7×10^{5} Y	1.78×10^{-7}	7.29×10^{-10}
57 40	Cf252	2.55 Y	4.15×10^{-16}	2.22×10^{-13}

51 *Estimated Value

TABLE 15.A-4

MASS OF SOURCE TERMS INITIALLY AIRBORNE IN RCB

	Isotope Class	Mass (kg)
	Noble Gases*	74.34
	Halogens**	1.59
	Solid Fission Product	5.55
	Fuel	62.45
51	Total Non-Gaseous	69.59
57	Initial RCB Concentration (μgm/cc)	0.68

*Mass of Noble Gases excluded from aerosol analysis.

**25% of EOEC Inventory.

40

TABLE 15.A-5

HAA-3 INPUT PARAMETERS USED FOR SOURCE TERM ANALYSIS

Parameter	
Initial Concentration, Particles/cc	1.34×10^{8}
Count Mean Particle Radius, μm	0.1
Geometric Mean Deviation, μm	2.0
Aerosol Material Density, gm/cc	10.55
Stokes Correction Factor, α	0.1
Gravitational Collision Efficiency, ϵ	1.0
RCB Volume, cm^3	1.02×10^{11}
RCB Leak Rate, fraction/sec	1.16×10^{-8}
57 Plating Constant, Δ	4×10^{-5}

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TABLE 15.A-6

AEROSOL DEPLETION FACTORS USED FOR SOURCE TERM

Time (Sec)	Depletion Factor (Fraction/Sec)
6.50-3	9.55-5
4.60-2	9.49-5
2.63-1	9.17-5
1.25+0	8.21-5
3.84+0	6.92-5
9.22+0	5.79-5
1.95+1	4.87-5
3.94+1	4.10-5
7.73+1	3.45-5
1.47+2	2.91-5
5.20+2	2.04-5
1.90+3	1.37-5
6.42+3	9.43-6
1.98+4	7.06-6
1.09+5	6.33-6
4.21+5	4.56-6
1.09+6	2.24-6
2.60+6	1.40-6

57

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TABLE 15.A-7

CONTAINMENT/CONFINEMENT PARAMETERS USED FOR SOURCE TERM ANALYSIS

RCB Leakage to Annulus (Direct to Annulus Filter Intake)	0.1% Volume/Day
Annulus Flow Rates	
Filtered Exhaust	3000 CFM
Filtered Recirculation	3500 CFM per 1000 CFM Exhausted
Time Delay from Source Term Release to Initiation of Annulus Filtration	No Delay
Time Delay from Source Term Release to Initiation of Annulus Recirculation	<10 Seconds
Total Bypass Leakage (1% of RCB Leakage)	0.001% Volume/Day
Bypass Leakage Direct to Environment (60% of Total Bypass)	0.0006% Volume/Day
Bypass Leakage to the RSB (40% of Total Bypass)	0.0004% Volume/Day
Sources of Bypass Leakage to the RSB	96.4% Personnel and Airlock Equipment 3.6% All other sources
Gamma Shielding	1.5" Steel (RCB) Plus 4' Concrete
Filter Efficiencies	
Iodine	95%
Particulate	99%
40 Noble Gases	0

TABLE 15.A-8

METEOROLOGICAL PARAMETERS USED FOR SITE SUITABILITY ASSESSMENT

Exclusion Boundary (0.42 Miles)	X/Q (sec/m^3)
0-2 Hours	3.12×10^{-3}*
Low Population Zone (2.5 Miles)	
0-2 Hours	8.55×10^{-4}*
2-8 Hours	2.85×10^{-4}
8-24 Hours	2.60×10^{-5}
1-4 Days	1.30×10^{-5}
4-30 Days	8.70×10^{-6}

40 * 0-2 Hour X/Q's based on single-hour 95% X/Q Values.

TABLE 15.A-9

OFF-SITE EXPOSURE SUMMARY

(Power Level = 975 Megawatts-Thermal)

Dose (Rem)

	Organ	10CFR100	2-Hour Site Boundary (0.42 Miles)	30-Day Low Population Zone (2.5 Miles)
	Bone	150*	7.2	4.1
	Lung	75*	1.6	.9
	Thyroid	300	23.1	12.6
57 51	Whole Body**	25	3.7	1.7

*Equivalent to 10CFR100 guideline values; see Reference 4.

**Includes inhalation, external gamma cloud, and direct gamma shine exposures.

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57 Pages 16.6-11 and 16.6-12

HAVE BEEN DELETED

Amendmend 57

LIST OF RESPONSES TO NRC QUESTIONS

There are no new NRC Questions in Amendment 57.

Question 7

Provide an overview of the methods used to evaluate the structural integrity of the fuel assembly including a description of all analytical methods used (e.g., PECT2) and all applicable data. The report should be in the form of a summary addressing all calculational limits (e.g., stress and deflection).

Response:

The information requested concerning the CRBRP Fuel Assembly was provided under separate cover in the following topical report:

> "CRBRP Fuel Assembly Structural Analysis in Support of the Final Design Review", CRBRP-ARD-0204

Additional information concerning the CRBRP Fuel Rod will be provided in a topical report at a later date. A Table of Contents for this report
57 was provided to the NRC in December, 1976.