June 24, 1997

LICENSEE: NORTHEAST NUCLEAR ENERGY COMPANY (NNECO)

FACILITY: Millstone Nuclear Power Station Unit 2

SUBJECT: SUMMARY OF THE JUNE 11, 1997, MEETING WITH PARSONS POWER GROUP TO DISCUSS THE PROCESS TO BE USED DURING THE TIER 2 ACCIDENT ANALYSIS REVIEW OF THE ICAVP FOR MILLSTONE UNIT 2

On June 11, 1997, the Special Project Office (SPO) staff of the Office of Nuclear Reactor Regulation (NRR) participated in a publicly observed meeting with Parsons Power Group (Parsons) representatives. The purpose of this meeting was to discuss the process to be used by Parsons to implement the Tier 2 accident analysis review of the Independent Corrective Action Verification Program (ICAVP) at Millstone Unit 2. During this meeting, Parsons used its preliminary review of the Main Steam Line Break accident as described in Chapter 14 of Unit 2's Final Safety Analysis Report (FSAR) to facilitate discussions regarding its processes for the Tier 2 review. Also discussed during the meeting were the depth of the system review for the systems directly required to mitigate the analyzed accident, the review required of supporting or interfacing systems, and the information (critical characteristics) for each of the systems that requires NRC review and approval prior to verification by Parsons.

Enclosure 1 provides a list of the attendees at the meeting. Enclosure 2 provides the handout used by Parsons as the outline for discussions during the meeting with the NRC. Enclosure 3 provides information used during the meeting to discuss and demonstrate the process Parsons proposed to use to conduct the Tier 2 review from its preliminary analysis of the Main Steam Line Break accident . Enclosure 4 provides examples of the information (critical characteristics) that Parsons would typically propose to the NRC for review and approval. During the meeting, the NRC indicated that information provided in section 2.3.5 of Enclosure 4 was the type of information that would be necessary for the NRC to receive from Parsons to support its review and approval of the critical characteristics for each of the systems involved in mitigating the consequences of the analyzed accidents. Also, the NRC staff indicated that the processes proposed by Parsons for the ICAVP Tier 2 review at Unit 2 appeared reasonable, but will be subject to further review and approval with the Unit 2 ICAVP audit plan provided by Parsons.

Original Signed by:

John A. Nakoski, ICAVP Program Coordinator
ICAVP Oversight Branch
Special Projects Office
Office of Nuclear Reactor Regulation

Dockets No. 50-336
Enclosures: As Stated (4)
cc w/att: See next page

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SReynolds	RPerch	DBeaulieu, RI

E-MAIL (w/encl 1 only)

SCollins/FMiraglia	PMcKee	WDean (WMD)
RZimmerman	DMcDonald	DRoss (e-mail to SAM)
WTravers	LBerry	

LIST OF ATTENDEES
June 11, 1997

NAME	ORGANIZATION	POSITION
Eugene Imbro	NRC	Deputy Director, ICAVP Oversight, SPO, NRR
Steve Reynolds	NRC	Chief, ICAVP Oversight Branch, SPO, NRR
John Nakoski	NRC	ICAVP Program Coordinator, SPO, NRR
Peter Koltay	NRC	Unit 3 Team Leader, SPO, NRR
Daniel L. Curry	Parsons Power	Project Director
Eric A. Blocher	Parsons Power	Deputy Project Director
John F. Hilbish	Parsons Power	Regulatory Review Group Manager
Wayne L. Dobson	Parsons Power	Process Model & Operational Analysis Manager
Randy Faust	Parsons Power	Accident Mitigation System Reviewer
Rich Glaviano	Parsons Power	Accident Mitigation System Review Lead
Mike Akins	Parsons Power	Accident Mitigation System Review Lead
John Ioannidi	Parsons Power	System Review Manager
Bruce Deist	Parsons Power	System Review Group
Paul Shipper	Parsons Power	Accident Mitigation System Reviewer
Abdul M. Ahmed	Parsons Power	Accident Mitigation System Reviewer
Juan M. Cajigas	Parsons Power	Accident Mitigation System Reviewer
Gordon Chen	Parsons Power	Accident Mitigation System Reviewer
R. Wayne Choromanski	Parsons Power	Accident Mitigation System Reviewer
William E. Meek	Parsons Power	Advisory Panel Member
Ed House	BWG, Inc.	

Enclosure 1

Accident Mitigation Systems Review

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Enclosure 2

AGENDA

- Introductions
- Logistics
- Steam Line Break Analysis
- System Boundary Diagrams
- Database
- Summary

Steam Line Break Analysis

- Chapter 14 Review method
- Initial Results

System Boundary Diagrams

- Source of Data
- Contents
- Relationship to Analysis

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Database

- Data Entry
- Fields
- Current Data
- Initial results
- Sorts

System Interface Boundaries

- How far?
- How deep?
- Linked to parameter only.

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Alignment Items

- Documents - Data at start of project
- System Interface Boundaries
- Timing for CDC for Validation to NRC

Summary

- Current Information Status
- What happens next.
- Questions and Answers

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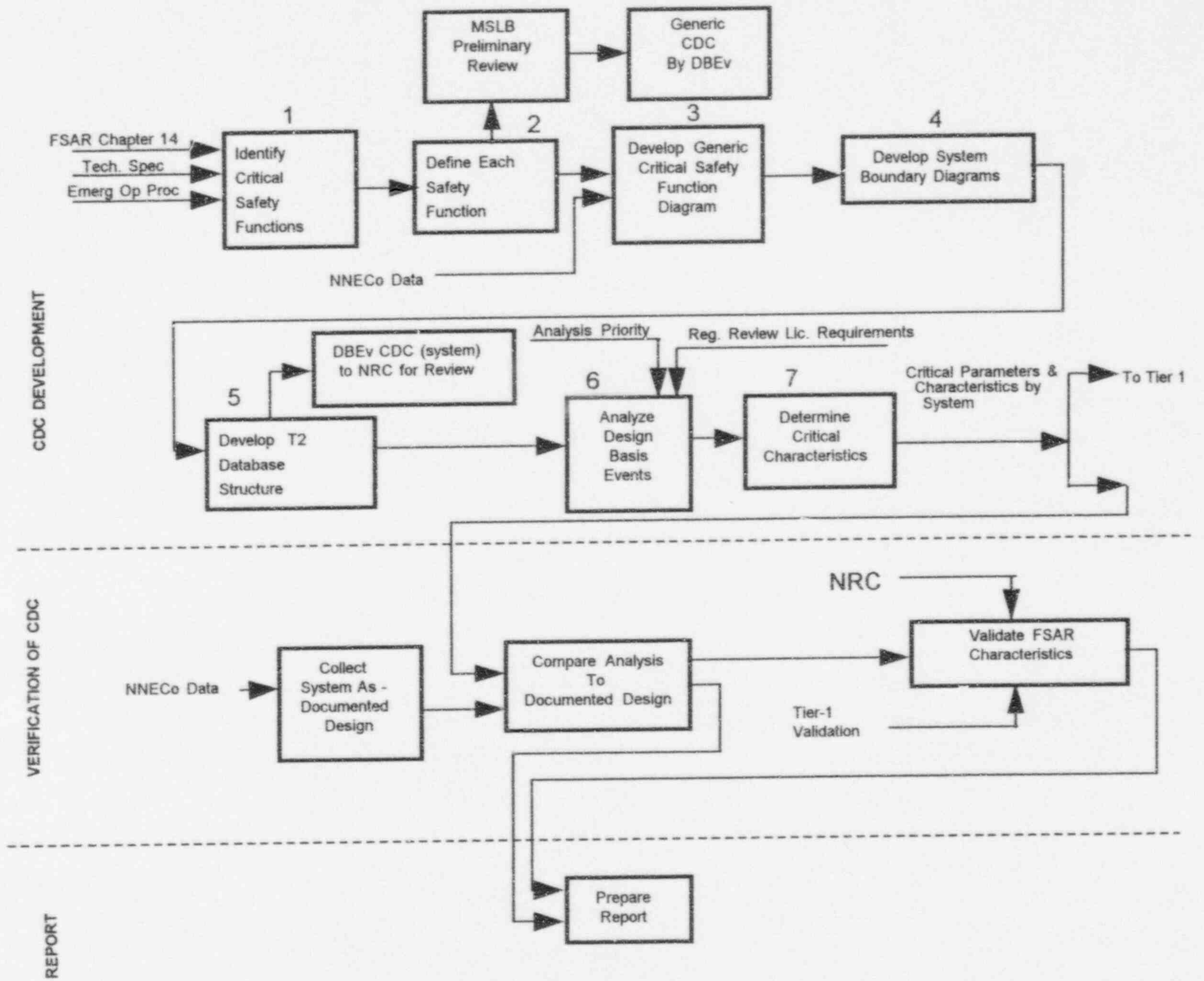
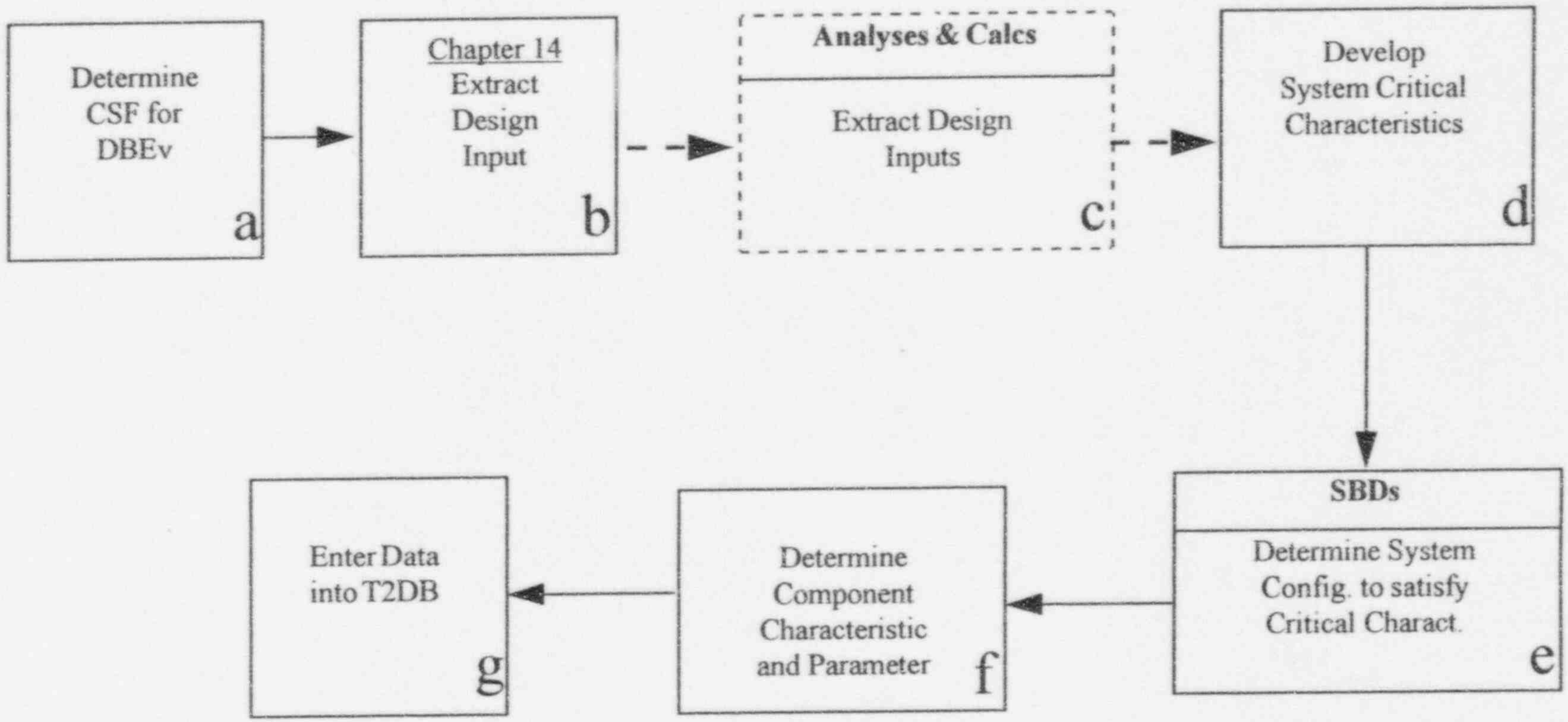

MSLB
Preliminary
Review
Generic
CDC
By DBEv
1
2
3
4
FSAR Chapter 14
Tech. Spec
Emerg Op Proc
Identify
Critical
Safety
Functions
Define Each
Safety
Function
Develop Generic
Critical Safety
Function
Diagram
Develop System
Boundary Diagrams
NNECo Data
CDC DEVELOPMENT
Analysis Priority
Reg. Review Lic. Requirements
DBEv CDC (system)
to NRC for Review
5
6
7
Critical Parameters &
Characteristics by
System
To Tier 1
Develop T2
Database
Structure
Analyze
Design
Basis
Events
Determine
Critical
Characteristics
NRC
VERIFICATION OF CDC
NNECo Data
Collect
System As -
Documented
Design
Compare Analysis
To
Documented Design
Validate FSAR
Characteristics
Tier-1
Validation
REPORT
Prepare
Report

Review of Main Steam Line Break at HZP with Offsite Power Available

The attached information was developed from review of the Main Steam Line Break (MSLB) Design Basis Event (DBEv) for the Millstone-2 Plant. This information was assembled to demonstrate the process that will be used to determine the critical system and component level characteristics resulting from the Tier-2 Accident Mitigation Systems Review. The information was extracted primarily from Chapter 14 of the Millstone-2 FSAR and from the Millstone-2 Operations Critical Drawings. The specific analyses supporting the MSLB DBEv, along with plant operating procedures for the involved systems, were not available at the time this information was prepared. The information presented herein may change as additional design and operating information is obtained from the supporting analyses and plant procedures.


Determine
CSF for
DBEv
a
Chapter 14
Extract
Design
Input
b
Analyses & Calcs
Extract Design
Inputs
c
Develop
System Critical
Characteristics
d
SBDs
Determine System
Config. to satisfy
Critical Charact.
e
Determine
Component
Characteristic
and Parameter
f
Enter Data
into T2DB
g

(2)

Main Steam Line Break at HZP with Offsite Power Available

Event Description

The Main Steam Line Break (MSLB) event at Millstone-2 is a double-ended guillotine break inside containment between the steam generator and the flow restrictors.

Initial Conditions

Plant is critical at Hot Zero Power (HZP)

Main Steam Isolation Valves (MSIV's) are open

Steam generators are being fed by the AFW system using the motor driven pumps. The AFW control valves are in a fixed position to provide flow sufficient to remove Reactor Coolant Pump heat.

Offsite power is maintained throughout the event.

Once HPSI and one charging pump are assumed available.

The most reactive control rod is assumed to be stuck in its fully withdrawn position.

TABLE 14.1.5-6

STEAM LINE BREAK ANALYSIS SUMMARY

Initial Power Level	Offsite Power Available	Maximum Post Scram Return to Power (MWt)	MDNBR	Maximum LHGR (kW/ft)
HZP	Yes	686	2.40	< 21.0
HZP	No	294	1.18	16.5
HFP	Yes	394	3.00	17.1
HFP	No	147	4.60	5.7

5/90

(4)

TABLE 14.1.5-7

STEAM LINE BREAK SEQUENCE OF EVENTS--HOT ZERO POWER--POWER AVAILABLE

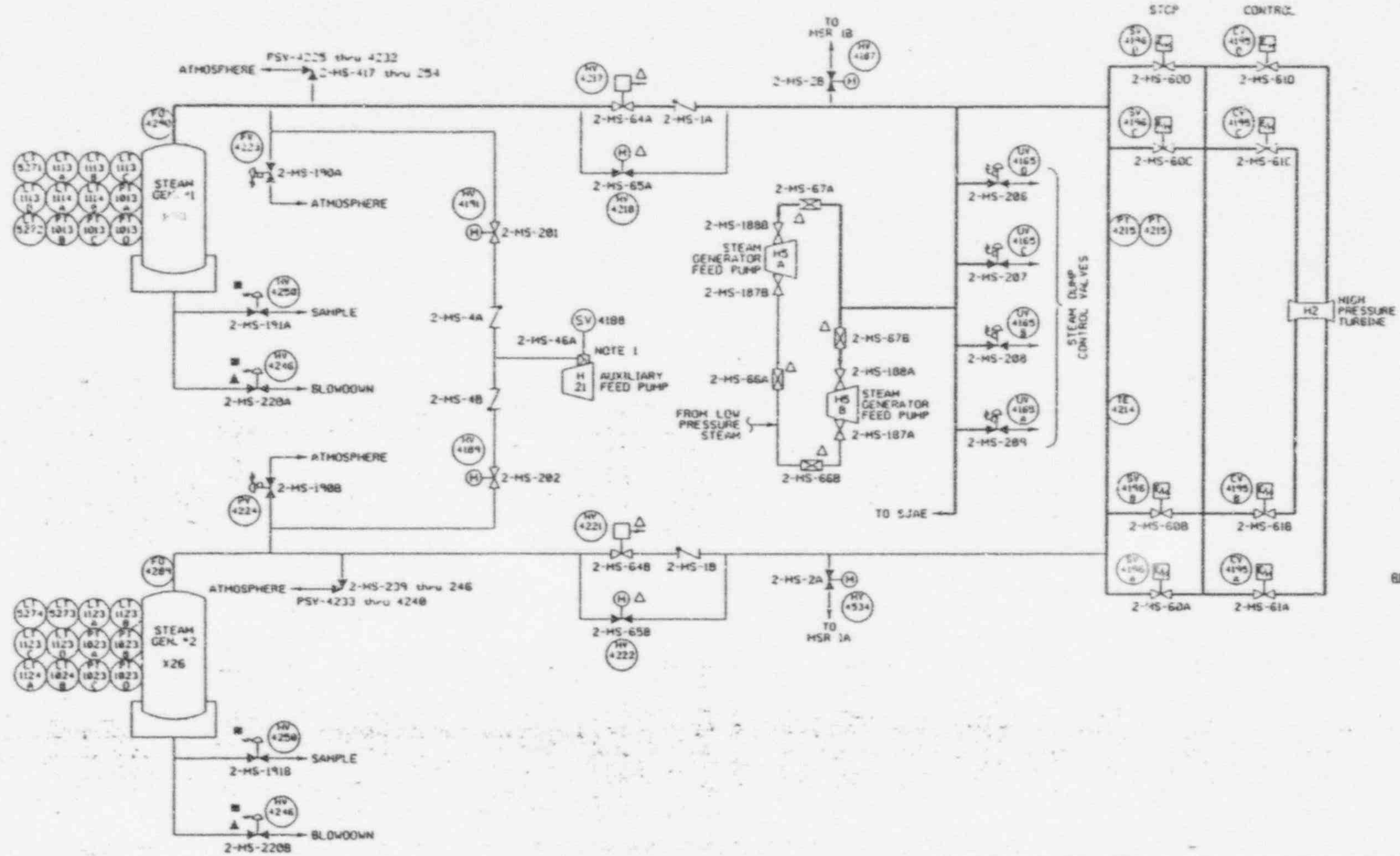
Time*	Event
0.	Reactor at hot zero power.
0.+	Double-ended guillotine break located between affected steam generator and the flow restrictors.
3.6	Main steam isolation valve closure signal generated by low steam generator pressure.
10.5	Main steam line isolation valves stop blowdown from intact steam generator 6.9 seconds after low steam generator pressure signal.
15.2	Safety injection signal generated by low primary coolant pressure.
32.	Reactor becomes critical.
45.2	HPSI and charging pumps actuated.
153.	Thermal power reaches maximum level at 25% of rated power.
153.	First boron has passed through core.
180.	Auxiliary feedwater initiated to affected steam generator.
600.	Auxiliary feedwater isolated manually.
600.+	Primary system temperature increase due to steam generator dryout and additional boron injection will terminate power excursion.

5/10

*Time after break, seconds

5

BB



LEGEND:

- □ - RPS SIGNAL
- △ - MAIN STEAM ISOLATION TRIP
- - HIGH PRESSURE STOP VALVE
- ■ - CIAS CLOSE
- ▲ - HIGH RAD CLOSE
- - LOW PRESSURE STOP VALVE
- —— - TRAIN "A" OR COMMON FLOW PATH

NOTES:

1. VALVE OPENS ON DEMAND SIGNAL.

REFERENCES:

DWG. 25203-26002, SH.1, REV. 36
DWG. 25203-26002, SH.2, REV. 21
DWG. 25203-26002, SH.3, REV. 23
DWG. 25203-26002, SH.4, REV. 16
DWG. 25203-26002, SH.5, REV. 6

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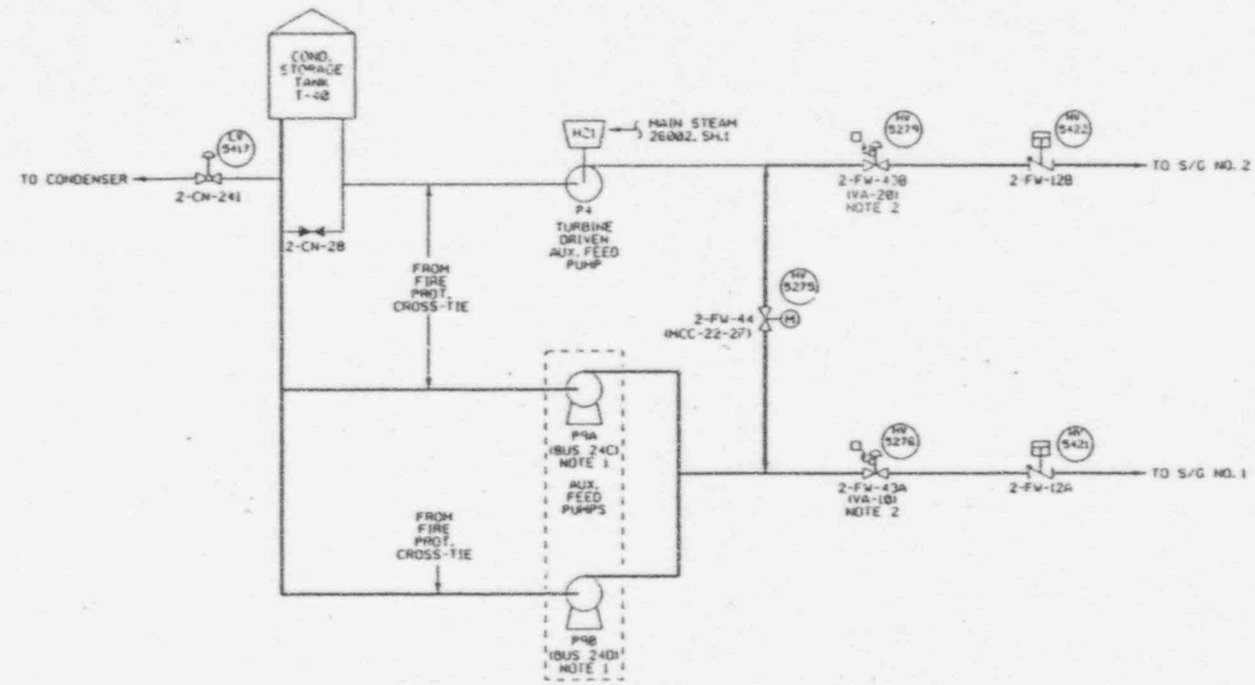
MAIN STEAM

REV	DATE	DESCRIPTION	DR	LD	P.E.

PPGI-File:
PPGI-Date:

ARCH HVAC I&C ELECT MECH CIVIL STRUC



LEGEND:
□ - AUXILIARY FEED INITIAL SIGNAL OPEN
—— - TRAIN 'A' OR COMMON FLOW PATH

NOTES:
1. DENOTES 4.16KV PANEL.
2. DENOTES PANEL NUMBER.

REFERENCES:
25203-26002, SH.1, REV. 36
25203-26005, SH.2, REV. 38
25203-26005, SH.3, REV. 28

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AUXILIARY FEEDWATER
HOT ZERO POWER ALIGNMENT

BB

PPGI-FILE: 526005auxwater.dgn
PPGI-DATE: 09-Jun-97 14:23

6/9/97

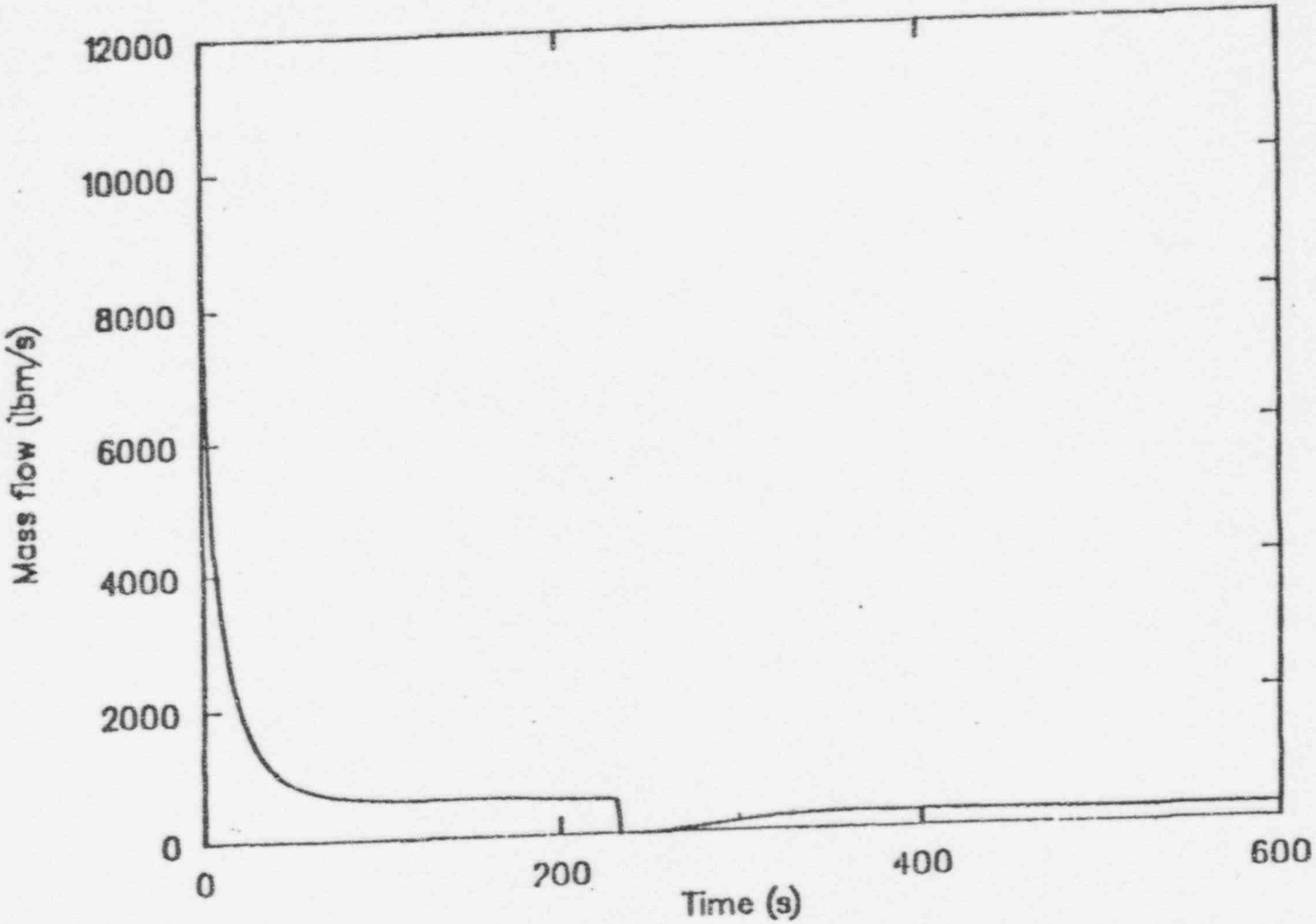


FIGURE 14.1.5-2 AFFECTED STEAM GENERATOR BREAK FLOW VS. TIME – HOT ZERO POWER WITH OFFSITE POWER

(8)

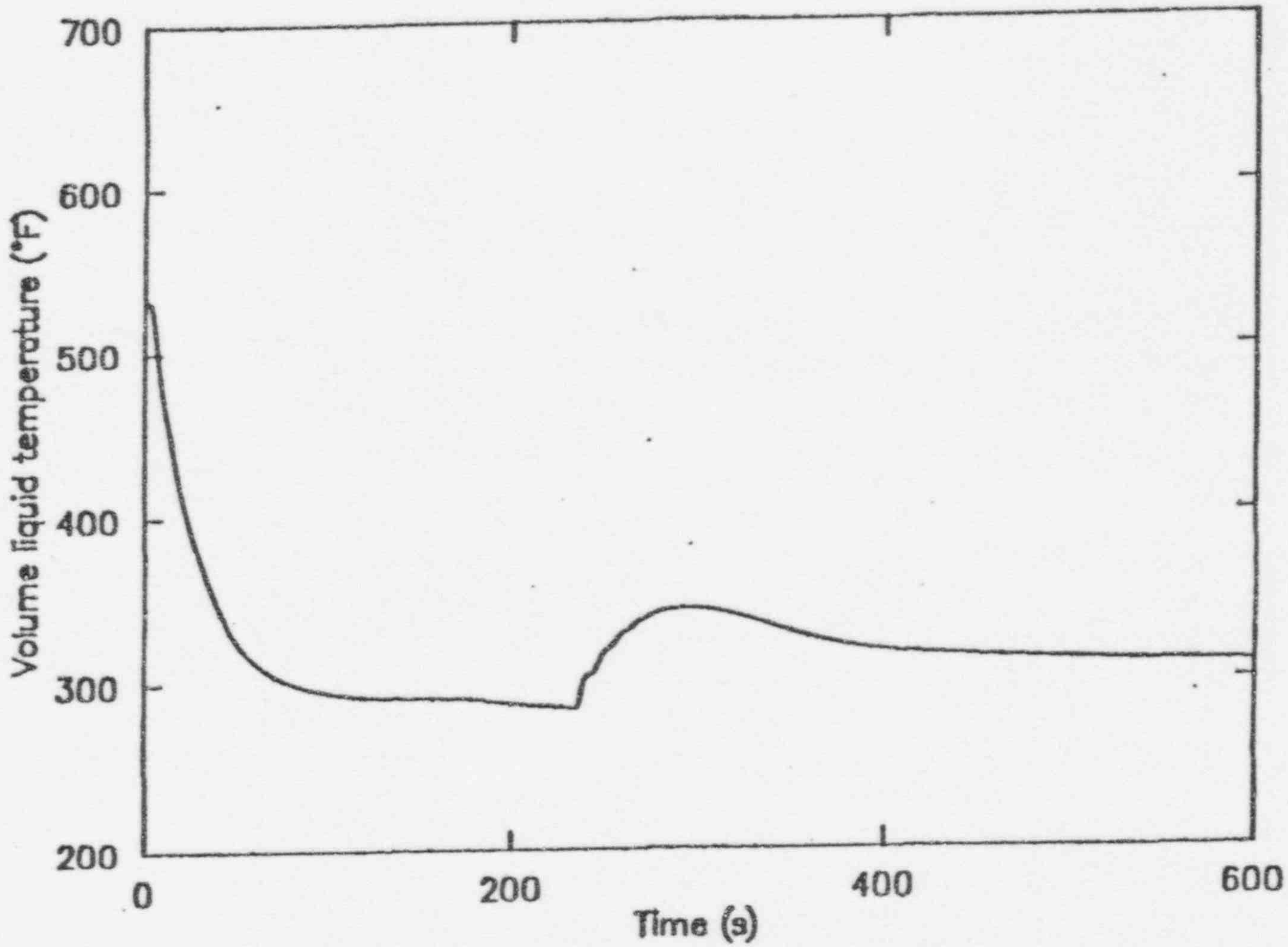


FIGURE 14.1.5-3 AFFECTED CORE SECTOR INLET TEMPERATURE VS. TIME – HOT ZERO POWER WITH OFFSITE POWER

9

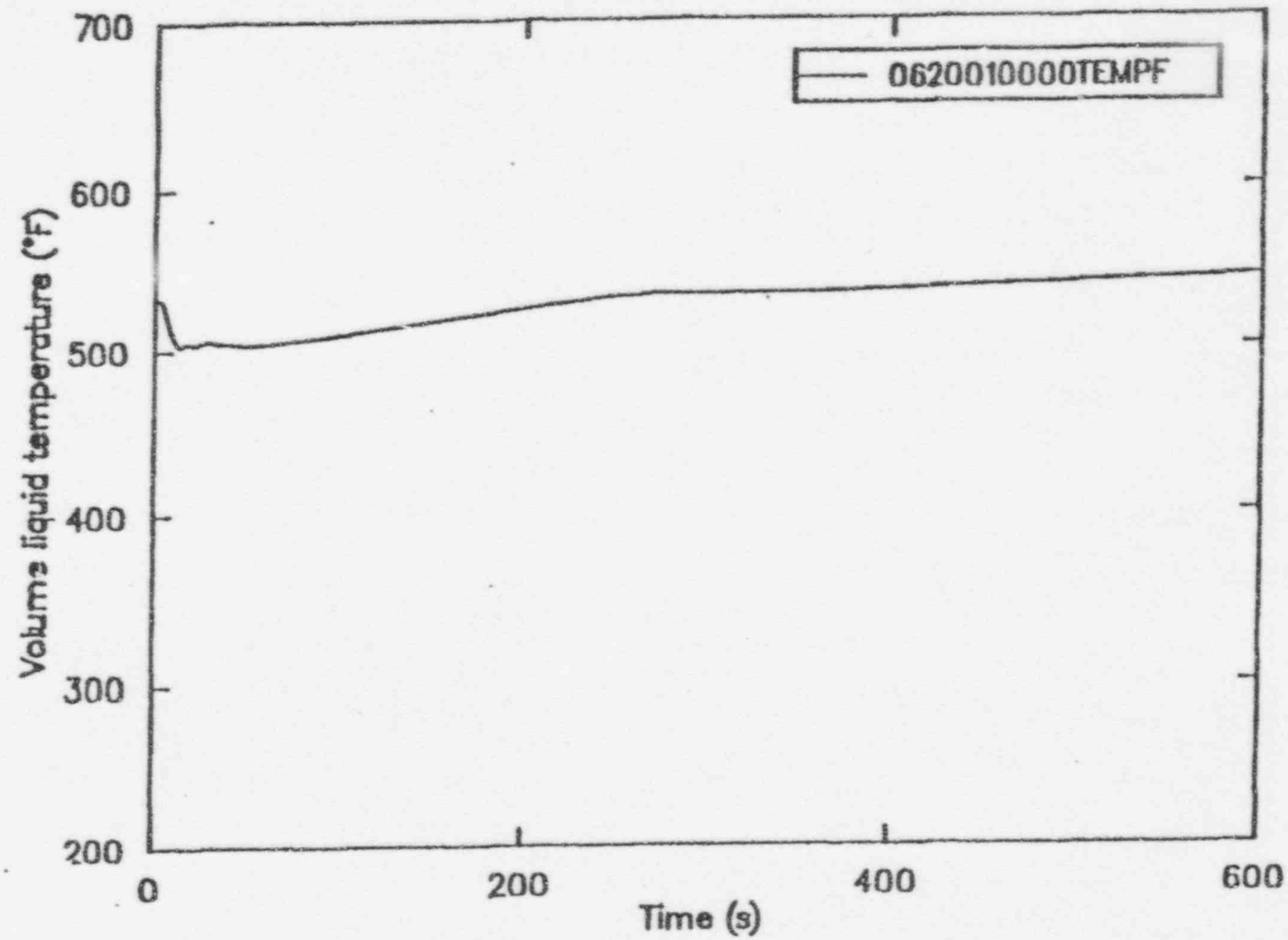


FIGURE 14.1.5-4 INTACT CORE SECTOR INLET TEMPERATURE VS. TIME - HOT ZERO POWER WITH OFFSITE POWER

(10)

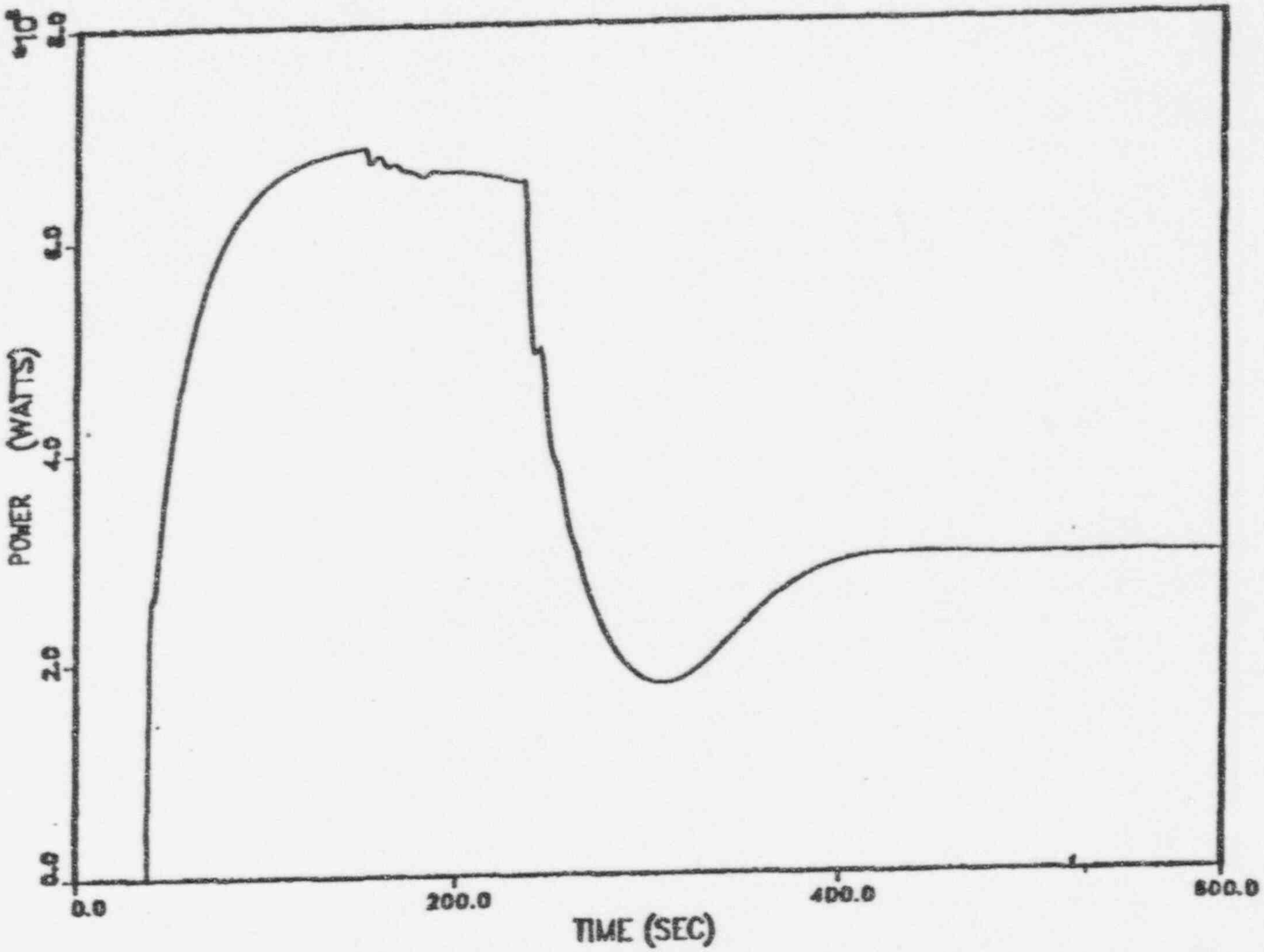


FIGURE 14.1.5-7 CORE POWER VS. TIME - HOT ZERO POWER WITH OFFSITE POWER

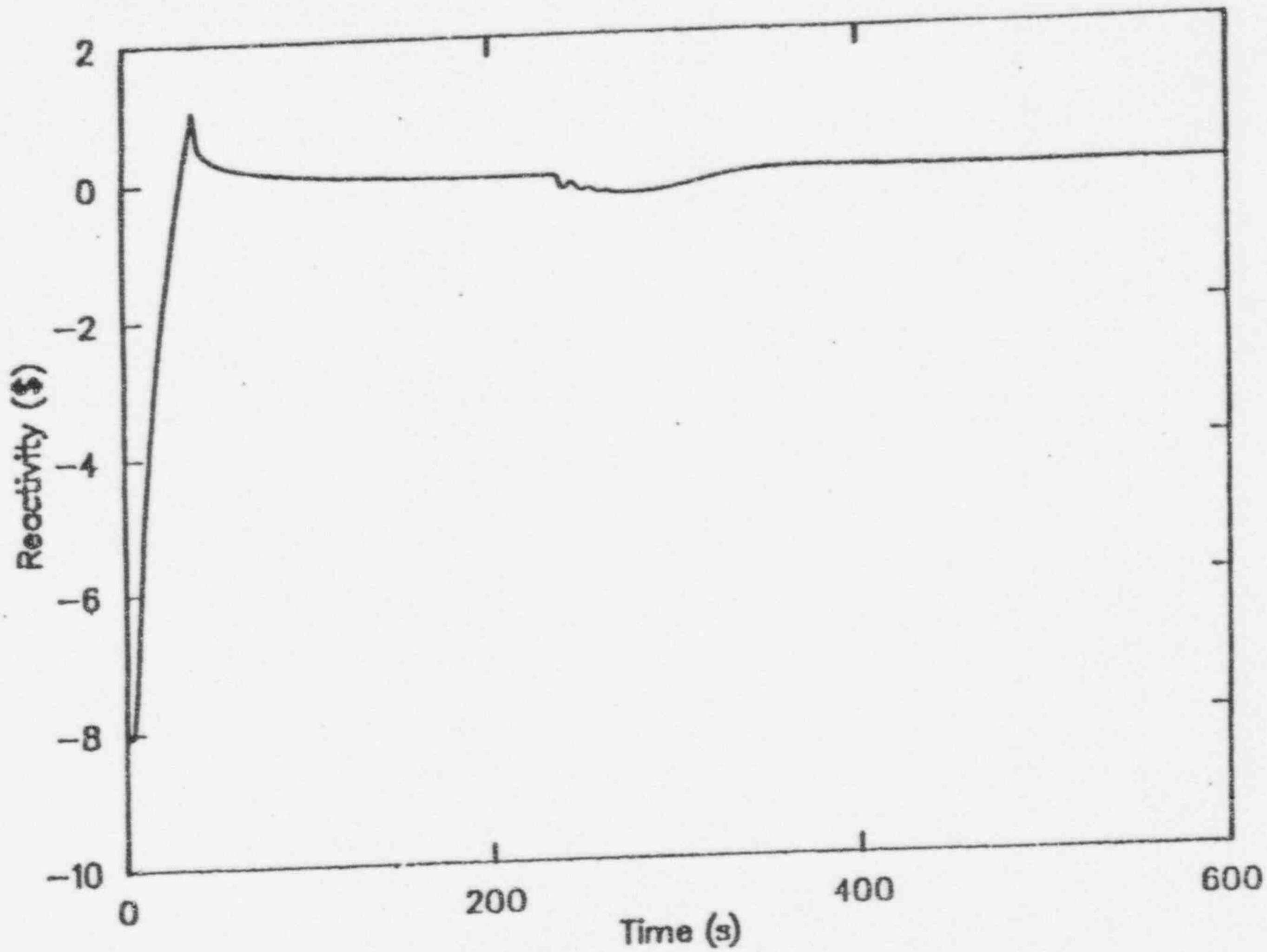


FIGURE 14.1.5−6 ADVANCED NUCLEAR FUELS − RELAP CALCULATED CORE REACTIVITY VS. TIME − HOT ZERO POWER WITH OFFSITE POWER

(21)

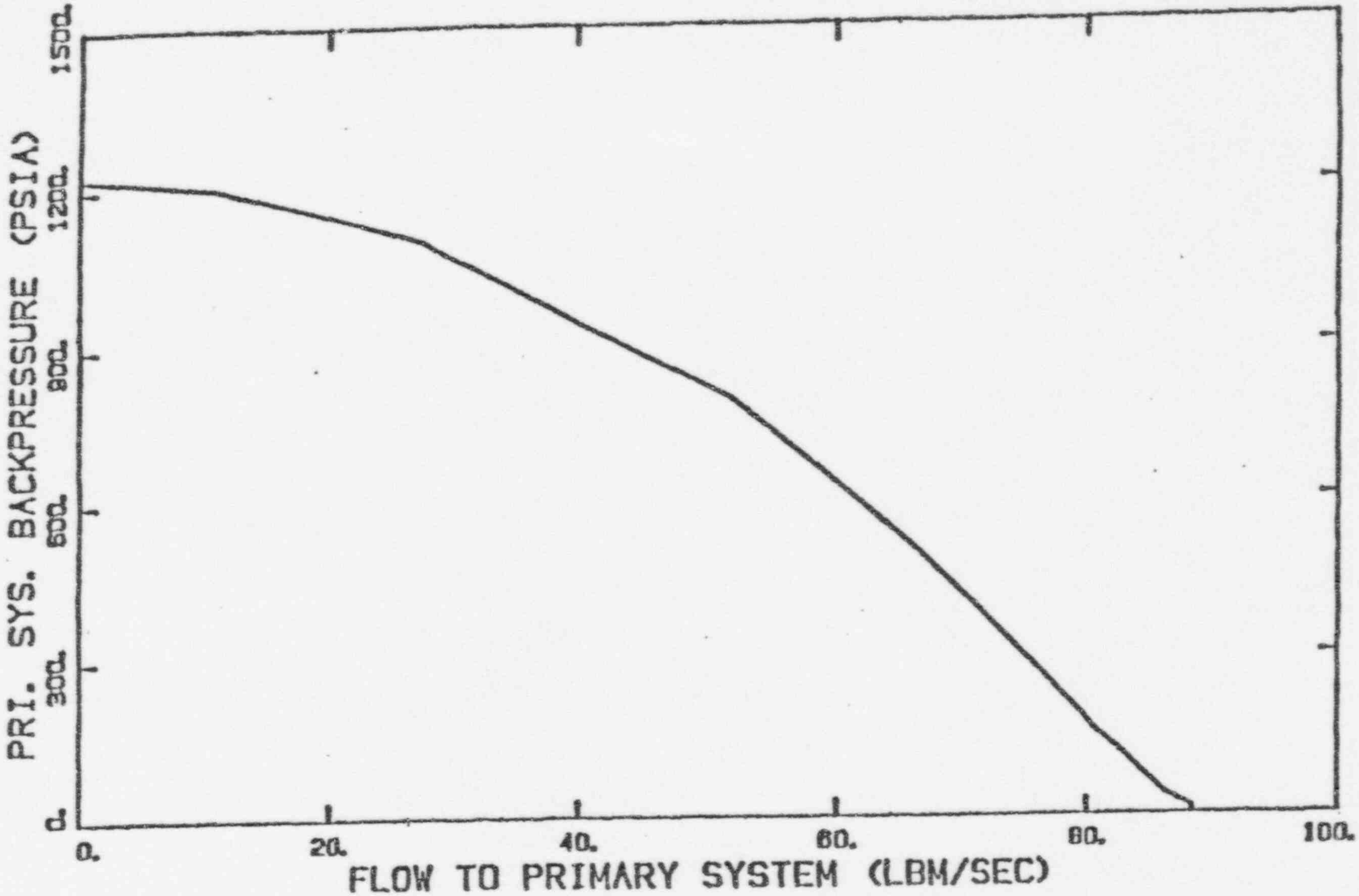


FIGURE 14.1.5-1 ONE PUMP HIGH PRESSURE SAFETY INJECTION SYSTEM DELIVERY VS. PRIMARY PRESSURE

(13)

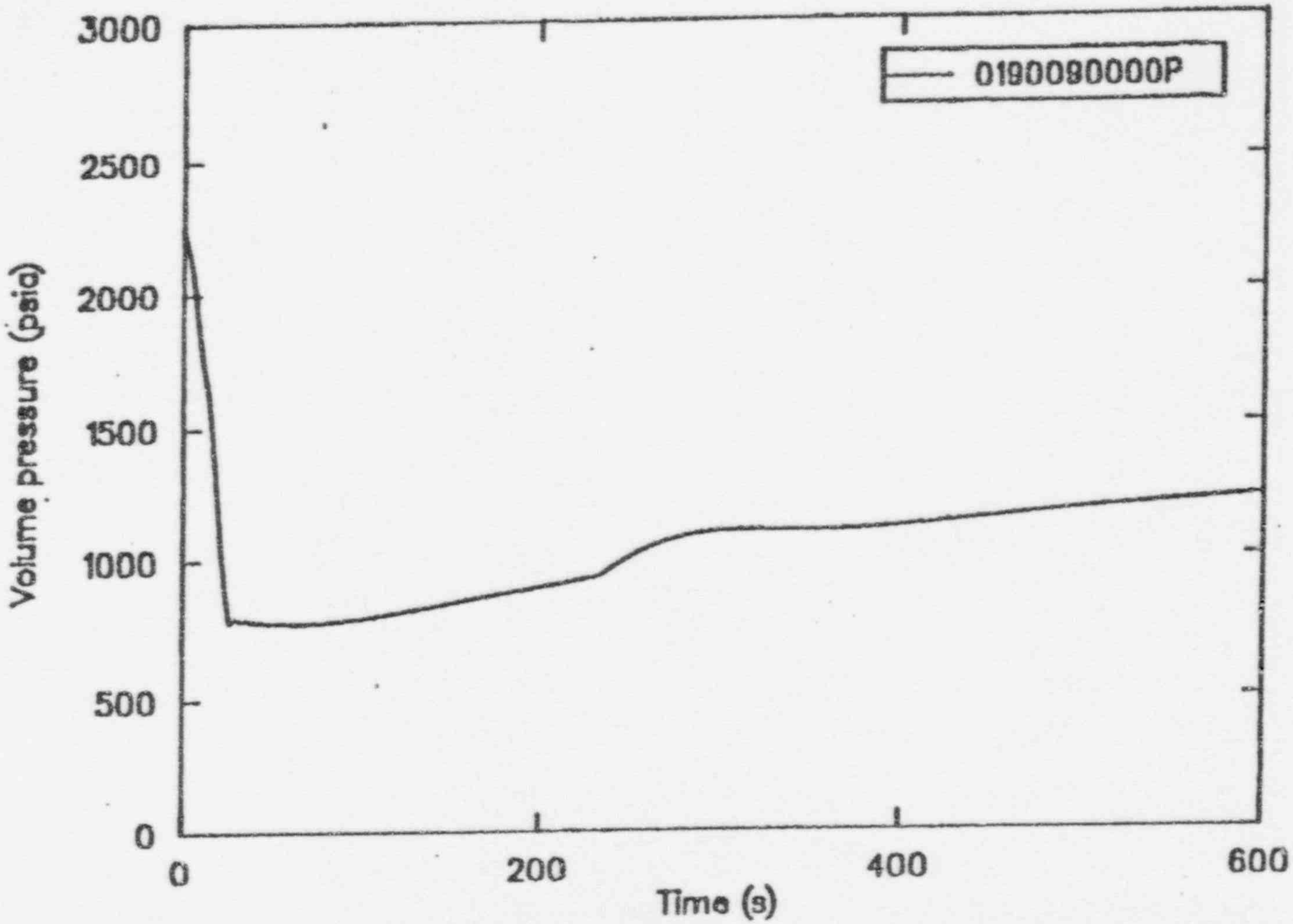
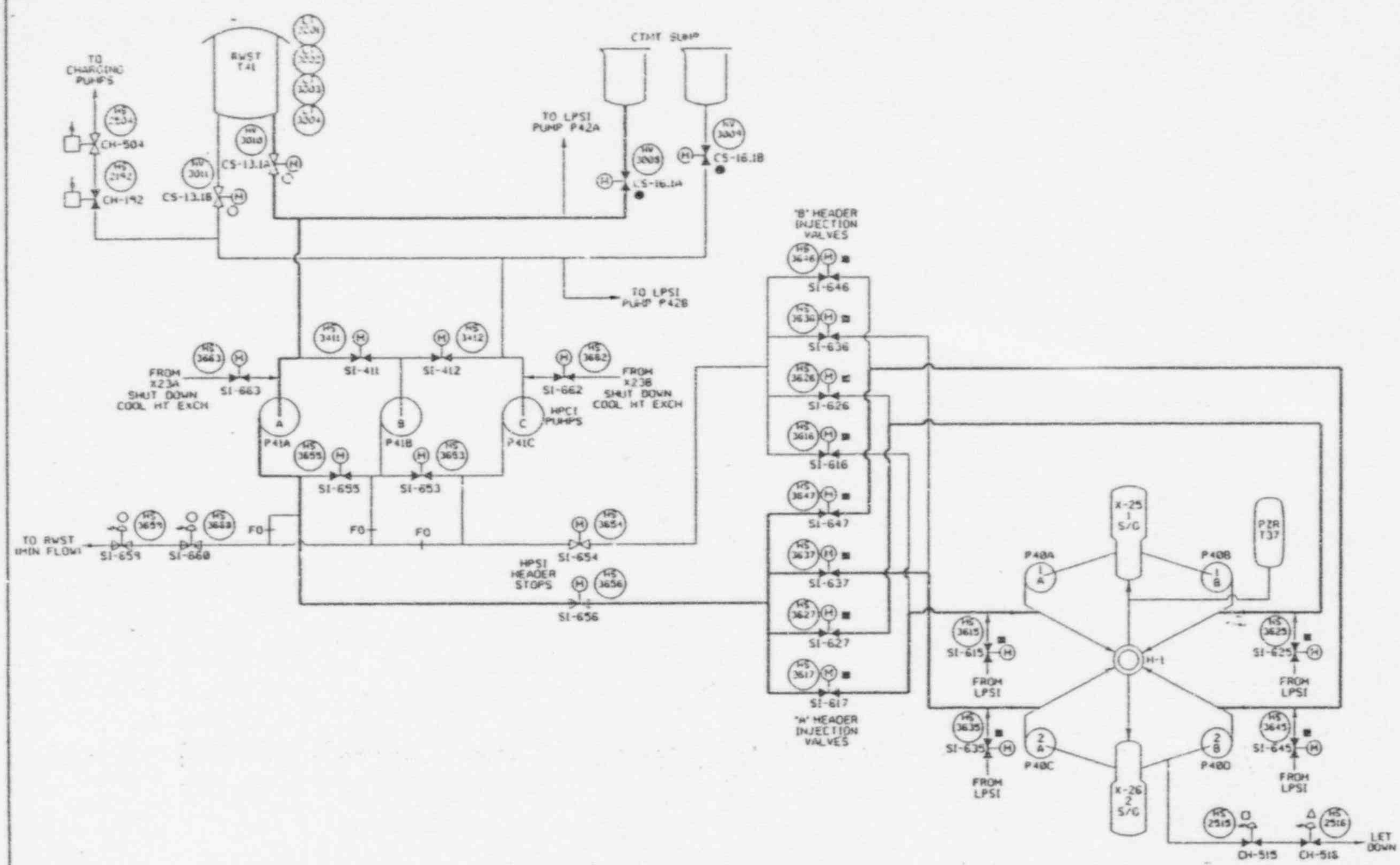


FIGURE 14.1.5-5 PRESSURIZER PRESSURE VS. TIME - HOT ZERO POWER WITH OFFSITE POWER



LEGEND:

- □ - CLOSES ON SIAS
- △ - CLOSES ON CIAS
- ○ - CLOSES ON SRAS
- ■ - OPENS ON SIAS
- ● - OPENS ON SRAS
- —— - TRAIN 'A' OR COMMON FLOW PATH

NOTES:

ALL COMPONENT ID'S PRECEEDED BY "2-" UNLESS OTHERWISE NOTED

REFERENCES:

25203-26015 SHT 2 REV 10
25203-26015 SHT 1 REV 13
25203-26017 SHT 3 REV 15
25203-26014 SHT 1 REV 17

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HPSI

PPCI-FILE: hpsi.dgn
PPCI-DATE: 5-Jun-97 03:50

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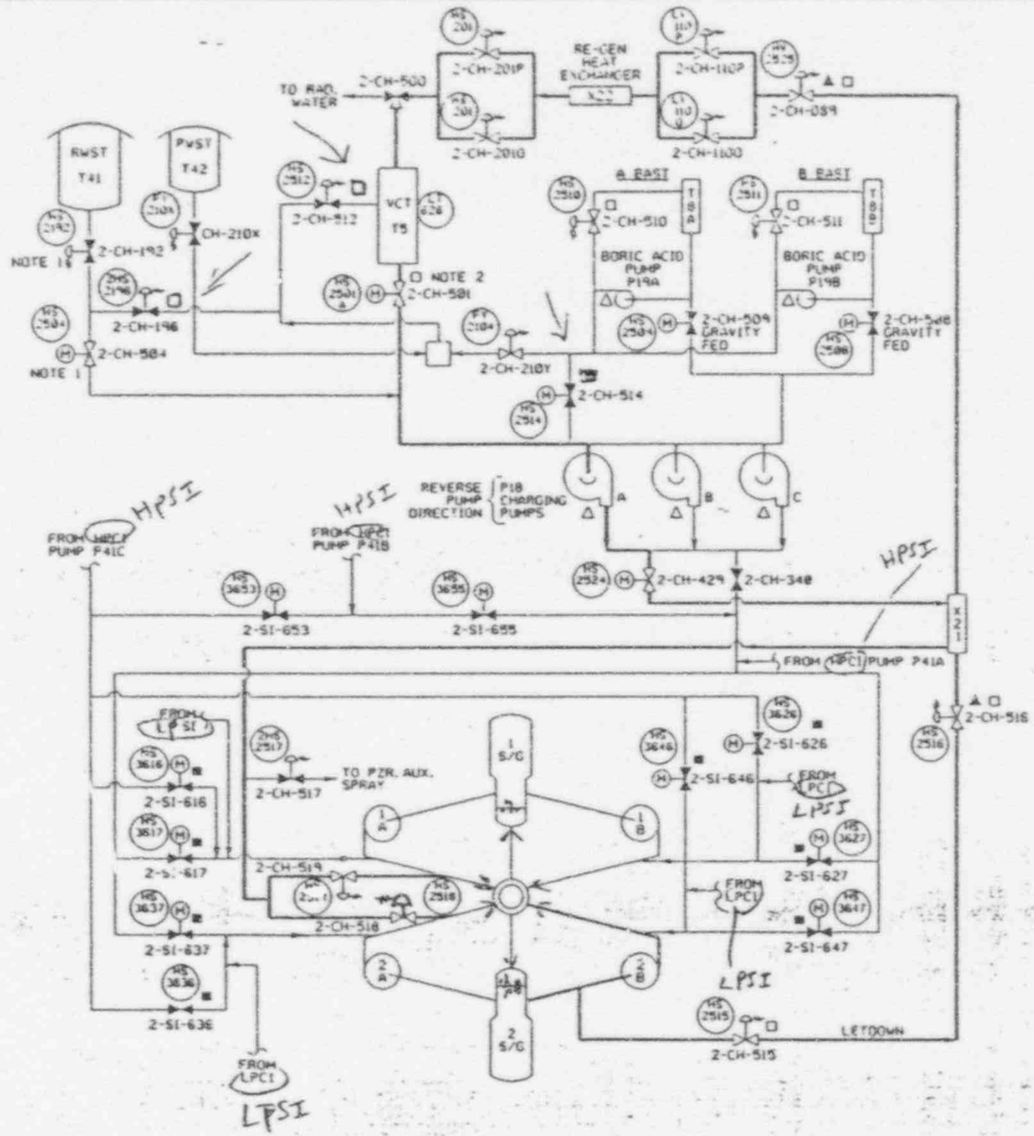

LEGEND:

- □ - CLOSES ON SIAS
- △ - STARTS ON SIAS
- ■ - OPENS ON SIAS
- ▲ - CLOSES ON CIAS
- ——— - TRAIN "A" OR COMMON FLOW PATH

NOTES:

1. VALVES OPEN ON LOW VCT LEVEL.
2. VALVE CLOSES ON LOW VCT LEVEL.

REFERENCES:

25203-26014.SH.1. REV.17
25203-26015.SH.2. REV.18
25203-26017.SH.1. REV.20
25203-26017.SH.3. REV.15
25203-26030.SH.1. REV.26

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PARSONS POWER GROUP INC.

C.V.C.S.

REV	DATE	DESCRIPTION	DA	LD	LDE

ARCH · HVAC · I & C · ELECT · MECH · CIVIL · STRUC

BB

FILE: 5c\905icvcs.dgn
DATE: 5-Jun-97 20:03

(16)

2.3 MAIN STEAM LINE BREAK (MSLB) ACCIDENT, RCS ANALYSIS.

2.3.1 ACCIDENT DESCRIPTION

This event is initiated by a rupture in the main steam piping upstream of the MSIVs which results in an uncontrolled steam release from the secondary system. The increase in energy removal through the secondary system results in severe ove cooling of the primary system. In the presence of a negative Moderator Temperature Coefficient (MTC), this cooldown causes a decrease in the shutdown margin (following reactor trip) such that a return to power might be possible following a steam line rupture assuming that the most reactive control rod is stuck in its fully withdrawn position. The MNPS-2 limiting MSLB from a safety standpoint is a Hot Zero Power (HZP) double-ended guillotine break inside containment between the steam generator and the flow restrictors. Regulatory requirements require that the plant be equipped with an emergency core cooling system (ECCS) that refills the vessel in a timely manner to satisfy the requirements of 10CFR50 Appendix A GDC 27, 28, 31, and 35 as well as appropriate sections of NUREGs 0694, 0718, and 0737. The MNPS-2 MSLB-RCS analysis is described in FSAR Section 14.1.5.1.

2.3.2 DESIGN BASIS

The MNPS-2 MSLB-RCS analysis is based on the following primary assumptions:

a. Most reactive control rod stuck in its fully withdrawn position.
b. Rated power mode is bounding for all full power modes and Mode 2 is bounding for HZP.
c. Single failure criteria for offsite power case is loss of one HPSI pump.
d. Single failure criteria for LOOP case is loss of one diesel generator.
e. Safety injection actuation signal (SIAS) actuated by low pressurizer pressure.
f. Secondary isolation signal actuated by low steam pressure.

2.3.3 SYSTEM INTERFACE

The following systems interface during the postulated MSLB-RCS recovery analysis:

a. Safety Injection System
b. Shutdown Cooling System
c. Emergency Power System
d. Auxiliary Feedwater System
e. Main Feedwater System (hot full power case)

2.3.4 SUMMARY OF DESIGN INPUTS

The following design inputs and assumptions have been identified during the FSAR review of the MSLB-RCS analyses. Additional inputs/assumptions and/or revisions will be developed upon review of the corresponding analysis calculation packages.

2.3.4.1 Control Rods

Most reactive control rod to be stuck in its fully withdrawn position.
Reference: FSAR Section 14.1.5.2

2.3.4.2 Power Mode - Full Power

Rated power mode bounding for all HFP modes.
Reference: FSAR Section 14.1.5.4

2.3.4.3 Power Mode - Zero Power

Mode 2 bounding for all HZP modes. Four RCPs assumed to operate to maximize initial loop flow.
Reference: FSAR Section 14.1.5.4

2.3.4.4 Single Failure - Offsite Power Available

One HPSI pump available.
Reference: FSAR Section 14.1.5.4 and 14.1.5.5.1.3

2.3.4.5 Single Failure - LOOP

Lose one DG with consequential loss of one HPSI pump and one charging pump.
Reference: FSAR Section 14.1.5.5 and 14.1.5.5.1.3

2.3.4.6 Limiting Break

Double-ended guillotine break inside containment between SG and flow restrictors.
A(affected SG) = 6.31 ft^2 and A(intact SG) = 2.35 ft^2
Reference: FSAR Section 14.1.5.5.1.1, Table 14.1.5-3

2.3.5 MSLB ANALYSIS - HZP WITH OFFSITE POWER AVAILABLE

Reactivity Control

System Requirement - Insert control rods within 3.9 seconds of reaching reactor trip setpoint.

Reactor Trip Delay - 3.9 seconds
3.0 sec. insertion time plus 0.9 sec instrument delay. Setpoint on low steam pressure or low pressurizer pressure.
Reference: FSAR Section 14.1.5.5.1.5, Table 14.1.5-4

Additional info:

What event starts the 3.9 second clock?
Does the 3.0 seconds include the CRD breaker time?
Is the rod insertion requirement to 0% or some other position?

Validation: Plant surveillance data for RPS testing, CRD breaker testing, and control rod drop times.

System Requirement - Inject boron to reach the core by 153 seconds.

Boron Injection - Assumed from (1) HPSI pump and (1) Charging pump, both taking suction from the RWST.
Reference: FSAR Section 14.1.5.5.1.2, Table 14.1.5-3

Additional info:

Verify suction source for charging pumps at the beginning of the event. Is it the RWST or the VCT?
Is the VCT a conservative source? If the VCT, then how long to switch over to the RWST?
Clarify the meaning of the comment in section 14.1.5.5.1.3 that describes "crediting charging" as not invalidating the conclusions of this analysis.

Validation: Charging system - System lineup and pump capacity from plant surveillance test. HPSI system - System lineup and pump capacity from plant surveillance test.

RWST Boron Concentration - 1720 PPM.
Reference: FSAR Table 14.1.5-3

Validation: TS value is 1720 PPM per TS 3.1.2.8. No further validation required.

RCS Heat Removal

System requirement - close MSIV on intact main steam line within 10.5 seconds to limit cooldown from non-affected steam generator blowdown.

Low Steam Line Pressure Trip Signal - FSAR indicates "analysis setpoint" of 500 psia, "uncertainty" of -22 psia for a "value" of 478 psia.
Reference: FSAR Section 14.1.5.5.1.5, Table 14.1.5-4

Additional info: Clarify setpoint and basis from the analysis.
Data to establish signal development time.

Validation: TS allowable is ≥492.5 psia. , TS Table 3.3-4 Need to validate with TS and surveillance requirements once analysis value can be determined. Validate signal development time using time response test data.

MSIV Closure Delay - 6.9 seconds. Time from trip setpoint to full valve closure. Setpoint on low steam pressure.
Reference: FSAR Section 14.1.5.5.1.5, Table 14.1.5-4

Validation: TRM value is ≤ 6.9 sec., TRM Table 3.3-5. Further validation required?

System Requirement - Limit AFW flow to affected steam generator for first 180 seconds of event.

AFW Flow - Initialized to match RCP heat. Then allowed to increase based on a fixed CV setting. Flow increased to pump runout flow, 229.5 lbm/sec, at 180 sec.
Reference: FSAR Section 14.1.5.5.1.4, Table 14.1.5-3

Additional info:
Is valve position controlled by the steam generator level control system at HZP?
If so, does the control valve open to maintain steam generator level? If so, is this conservative?
Are both motor driven AFW pumps assumed to be operating as an initial condition?

Validation: For t<180 sec, calculate flow and compare with analysis value for "fixed CV setting." For t>180 sec, compare pump flow at runout with analysis value.

System requirement - Isolate AFW flow to affected steam generator at 600 seconds.

AFW flow control valve - Close valve from control room.
Reference: FSAR Table 14.1.5-7

Validation: AFW surveillance test.

AFW Temperature - Limiting AFW temperature assumed AFWTi = 32.1°F
Reference: FSAR Table 14.1.5-3

Validation: None required.

- RCS Pressure & Inventory Control

System requirement - initiate HPSI flow to core within 30 seconds per assumed HPSI pump head curve.

Low Pressurizer Pressure Trip Signal - FSAR indicates "analysis setpoint" of 1600 psia, "uncertainty" of -22 psia for a "value" of 1578 psia.
Reference: FSAR Section 14.1.5.5.1.5, Table 14.1.5-4

Additional info: Clarify setpoint and basis from the analysis.
Data to establish signal development time

Validation: Need to validate with TS and surveillance requirements once analysis value can be determined. TS allowable is ≥1592.5 psia., TS Table 3.3-4. Validate signal development time using time response test data.

HPSI Actuation Delay - 30 seconds. Time from trip setpoint to full pump speed. Setpoint on low pressurizer pressure.
Reference: FSAR Section 14.1.5.5.1.5, Table 14.1.5-4

Validation: Need to validate with plant surveillance requirements. Note that TRM value is ≤ 25 sec., TRM Table 3.3-5

HPSI Pump Performance - Provided as Figure 14.1.5-1, HPSI flow vs. RCS backpressure curve.
Reference: FSAR Section 14.1.5.5.1.2, Figure 14.1.5-1.

Validation: Compare to plant ISI procedure acceptance criteria.

System requirement - initiate charging flow to core within 40 seconds per assumed pump capacity.

Charging Pump Actuation Delay - 40 seconds. Time from trip setpoint to full pump speed. Setpoint on low pressurizer pressure.
Reference: FSAR Section 14.1.5.5.1.5, Table 14.1.5-4, TRM Table 3.3-5

Validation: Compare to plant ISI procedure acceptance criteria.

Charging Pump Performance - No data provided in FSAR.
Reference:

Additional info: Charging pump flow used in analysis

RWST Temperature -
Reference:

Additional info:
What RWST temperature was used for this analysis?
Is it important from an RCS Pressure/Inventory control perspective?

Validation: Compare analysis value to TS requirement.

2.3.6 MSLB ANALYSIS - HZP WITH LOOP

Additional review: diesel generator start and loading.

2.3.7 MSLB ANALYSIS - HOT FULL POWER WITH OFFSITE POWER AVAILABLE

FW Temperature - All FW heating ceases at time of the break. Limiting FW temperature assumed FWTi = 432.1°F
Reference: FSAR Table 14.1.5.3

FW Flow - Prior to FW flow termination, FW flow is a function of secondary system pressure. No FW flow vs. secondary system backpressure provided in FSAR.
Reference: FSAR Section 14.1.5.5.1.4

FW Flow - FW terminated at 30 seconds after the reactor trip per closure of FW regulator valves.
Reference: FSAR Section 14.1.5.5.1.4

Main FW Valve Closure Delay - 30 seconds. Time from trip setpoint to full valve closure. Setpoint on low steam pressure.
Reference: FSAR Section 14.1.5.5.1.5, Table 14.1.5-4
TRM value is ≤ 14 sec., TRM Table 3.3-5

2.3.8 MSLB ANALYSIS - HOT FULL POWER WITH LOOP

Additional review: diesel generator start and loading.

cc:

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