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SUBJECT: Forwards marked-up Unit 2 Tech Spec pages, used to prepare draft Tech Specs for Unit 3. Advises that present Unit 3 schedule for fuel loading necessitates approval of Tech Specs by 870301.

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Tech Specs
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Arizona Nuclear Power Project
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November 26, 1986
ANPP-39135-JGH/JRP/98.05

Director of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Project Director
PWR Project Directorate #7
Divison of Pressurized Water Reactor Licensing - B
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 3
Docket No. STN 50-530
Draft Technical Specifications
File: 86-F-005-419.05; 86-D-056-026

Dear Mr. Knighton:

Enclosed are marked up PVNGS Unit 2 Technical Specification pages with justification, used to prepare the draft Unit 3 Technical Specifications. The changes presented herewith represent typo's, editorial comments, unit specific and technical changes to the Unit 2 Technical Specifications. The Unit 2 Technical Specifications that have not changed will be used as the basis for the remaining Unit 3 draft Technical Specifications as previously discussed with your staff.

Several of the changes presented here have been or will be submitted to the staff for review and approval on the Unit 1 and 2 Dockets. Where the Unit 1 and 2 Technical Specifications differ from the approved Unit 3 version, ANPP will, at a later date, request an amendment to have the changes made in Unit 3, incorporated into Unit 1 and 2 Technical Specifications for consistency.

Please be advised that the present Unit 3 schedule for fuel loading necessitates approval of the Technical Specifications by March 1, 1987. If you have any questions or concerns please contact Mr. Richard A. Bernier at (602) 943-7200 ext. 4295.

Very truly yours,

JG Haynes

J. G. Haynes
Vice President
Nuclear Production

JGH/JRP/ls
Attachements

cc: O. M. DeMichele
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The following proposed changes consist of typo's, editorial remarks and section/page renumbering.

1. Page v; delete Fire Detection Instrumentation 3/4 3-61

2. page v; renumber 3/4 3-69 to 3/4 3-61.

3. Page v; renumber 3/4 3-71 to 3/4 3-63.

4. Page viii; delete Section 3/4.7.11 Fire Suppression Systems

5. Page viii; delete Section 3/4.7.12 Fire Rated Assemblies

6. Page viii; renumber 3/4.7.13 to 3/4.7.11 and 3/4 7-45 to 3/4 7-29.

7. Page viii; renumber 3/4.7.14 to 3/4.7.12 and 3/4 7-46 to 3/4 7-30.

8. Page xiii; delete Section 3/4 7.11 Fire Suppression Systems.

9. Page xiii; delete Section 3/4.7.12 Fire Rated Assemblies.

10. Page xiii; renumber 3/4.7.13 to 3/4.7.11 and B3/4 7-8 to B3/4 7-7.

11. Page xiii; renumber 3/4.7.14 to 3/4.7.12 and B3/4 7-8 to B3/4 7-7.

12. Page xxi; delete Table 3.3-11.

13. Page xxi; renumber Table 3.3-12 to 3.3-11 and page 3/4 3-70 to 3/4 3-62.

14. Page xxi; renumber Table 3.3-13 to 3.3-12 and page 3/4 3-72 to 3/4 3-64.

15. Page xxi change page 3/4 3-77 to 3/4 3-69.

16. Page xxii delete Table 3.7-3.

17. Page xxii delete Table 3.7-4.

18. Page xxii delete Table 3.7-5.

19. Page 2-1; specification renumbering.

20. Page 2-2; editorial, change in nomenclature.

21. Page 3/4 3-69; section, table and page renumbering.

22. Page 3/4 3-70; table and page renumbering.

23. Page 3/4 3-71; table, section and page renumbering.

24. Page 3/4 3-72; table and page renumbering.

25. Page 3/4 3-73; table and page renumbering.

26. Page 3/4 3-74; table and page renumbering.

27. Page 3/4 3-75; table and page renumbering.

28. Page 3/4 3-76; table and page renumbering.

29. Page 3/4 3-77; page renumbering.

30. Page 3/4 3-78; page renumbering.

31. Page 3/4 3-79; page renumbering.

32. Page 3/4 3-80; page renumbering.

33. Page 3/4 7-45; section and page renumbering.

34. Page 3/4 7-46; section and page renumbering.

35. Page 3/4 8-22; typo.

36. Page 3/4 8-24; typo.

37. Page 3/4 3-2; editorial.

38. Page 6-14; editorial.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

 X

BASES

LIST OF TABLES

LIST OF TABLES

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The calculated DNBR of the reactor core shall be maintained greater than or equal to 1.231.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the calculated DNBR of the reactor has decreased to less than 1.231, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.~~1~~

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21 kW/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21 kW/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.~~1~~

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.~~1~~

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.~~1~~

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint ~~values~~ Limits shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.~~8~~ 7 The loose-part detection system shall be OPERABLE with all sensors specified in Table 3.3-~~12~~. -11.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.~~8~~ 7 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of:

a. a CHANNEL CHECK at least once per 24 hours,

b. a CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. a CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.3-~~12~~ 11

LOOSE PARTS SENSOR LOCATIONS

INSTRUMENT NO.	LOCATION
JSVNYE - 1	UPPER VESSEL A (STUD BOLTS)
JSVNYE - 2	UPPER VESSEL B (STUD BOLTS)
JSVNYE - 3	LOWER VESSEL A (INCORE NOZZLE)
JSVNYE - 4	LOWER VESSEL B (INCORE NOZZLE)
JSVNYE - 5	SG-1A (HOT LEG)
JSVNYE - 6	SG-1B (COLD LEG 1A)
JSVNYE - 7	SG-2A (HOT LEG)
JSVNYE - 8	SG-2B (COLD LEG 2A)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.~~9~~8 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-~~13~~12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-~~13~~12.

ACTION:

a. With a low range radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-~~13~~12. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.~~9~~8 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

TABLE 3.3-~~13~~12

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GASEOUS RADWASTE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release #RU-12	1	#	35
b. Flow Rate Monitor	1	#	36
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	39
b. Oxygen Monitor	2	**	39

TABLE 3.3-~~13~~ 12 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. CONDENSER EVACUATION SYSTEM			
A. Low Range Monitors			
a. Noble Gas Activity Monitor #RU-141	1	1, 2, 3***, 4***	37
b. Iodine Sampler	1	1, 2, 3***, 4***	40
c. Particulate Sampler	1	1, 2, 3***, 4***	40
d. Flow Rate Monitor	1	1, 2, 3***, 4***	36
e. Sampler Flow Rate Measuring Device	1	1, 2, 3***, 4***	36
B. High Range Monitors			
a. Noble Gas Activity Monitor #RU-142	1	1, 2, 3***, 4***	42
b. Iodine Sampler	1	1, 2, 3***, 4***	42
c. Particulate Sampler	1	1, 2, 3***, 4***	42
d. Sampler Flow Rate Measuring Device	1	1, 2, 3***, 4***	42
4. PLANT VENT SYSTEM			
A. Low Range Monitors			
a. Noble Gas Activity Monitor #RU-143	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36

TABLE 3.3-~~13~~ 12 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. PLANT VENT SYSTEM (Continued)			
B. High Range Monitors			
a. Noble Gas Activity Monitor #RU-144	1	*	42
b. Iodine Sampler	1	*	42
c. Particulate Sampler	1	*	42
d. Sampler Flow Rate Measuring Device	1	*	42
5. FUEL BUILDING VENTILATION SYSTEM			
A. Low Range Monitors			
a. Noble Gas Activity Monitor #RU-145	1	##	37,41
b. Iodine Sampler	1	##	40
c. Particulate Sampler	1	##	40
d. Flow Rate Monitor	1	##	36
e. Sampler Flow Rate Measuring Device	1	##	36
B. High Range Monitors			
a. Noble Gas Activity Monitor #RU-146	1	##	41,42
b. Iodine Sampler	1	##	42
c. Particulate Sampler	1	##	42
d. Sampler Flow Rate Measuring Device	1	##	42

TABLE 3.3-~~13~~ 12 (Continued)

TABLE NOTATION

* At all times.
** During GASEOUS RADWASTE SYSTEM operation.
*** Whenever the condenser air removal system is in operation, or whenever turbine glands are being supplied with steam from sources other than the auxiliary boiler(s).
\# During waste gas release.
\## In MODES 1, 2, 3, and 4 or when irradiated fuel is in the fuel storage pool.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the actions of (a) or (b) or (c) are performed:

a. Initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter(s).

b. Place moveable air monitors in-line

c. Take grab samples at least once per 12 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of the GASEOUS RADWASTE SYSTEM may continue provided grab samples are taken and analyzed daily. With both channels inoperable operation may continue provided grab samples are taken and analyzed (1) every 4 hours during degassing operations, and (2) daily during other operations.

TABLE 3.3-~~13~~ 12 (Continued)

TABLE NOTATION

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 within one hour after the channel has been declared inoperable.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION b of Specification 3.9.12 or operate the fuel building essential ventilation system while moving irradiated fuel.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement restore the channel to OPERABLE status within 72 hours or:

a. Initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter(s) when it is needed.

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action(s) taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS RADWASTE SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release RU-12	P	P	R(3)	Q(1),(2),P###	#
b. Flow Rate Monitor	P	N.A.	R	Q,P###	#
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor (continuous)	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (sequential)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor (continuous)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (sequential)	D	N.A.	Q(5)	M	**

TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
3. CONDENSER EVACUATION SYSTEM (RU-141 and RU-142)					
a. Noble Gas Activity Monitor	D(6)	M	R(3)	Q(2)	1, 2, 3***, 4***
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3***, 4***
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3***, 4***
d. Flow Rate Monitor	D(7)	N.A.	R	Q	1, 2, 3***, 4***
e. Sampler Flow Rate Measuring Device	D(7)	N.A.	R	Q	1, 2, 3***, 4***
4. PLANT VENT SYSTEM (RU-143 and RU-144)					
a. Noble Gas Activitv Monitor	D(6)	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D(7)	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D(7)	N.A.	R	Q	*

TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
5. FUEL BUILDING VENTILATION SYSTEM (RU-145 and RU-146)					
a. Noble Gas Actvity Monitor	D(6)	M	R(3)	Q(2)	##
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	##
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	##
d. Flow Rate Monitor	D(7)	N.A.	R	Q	##
e. Sampler Flow Rate Measuring Device	D(7)	N.A.	R	Q	##

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE SYSTEM operation.

*** Whenever the condenser air removal system is in operation, or whenever turbine glands are being supplied with steam from sources other than the auxiliary boiler(s).

During waste gas release.

During MODES 1, 2, 3 or 4 or with irradiated fuel in the fuel storage pool.

Functional test should consist of, but not be limited to, a verification of system isolation capability by the insertion of a simulated alarm condition.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway occurs if the instrument indicates measured levels above the alarm/trip setpoint.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) The channel check for channels in standby status shall consist of verification that the channel is "on-line and reachable."

(7) Daily channel check not required for flow monitors in standby status.

3/4.7.~~13~~11 SHUTDOWN COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~13~~11 Two independent shutdown cooling subsystems shall be OPERABLE, with each subsystem comprised of:

a. One OPERABLE low pressure safety injection pump, and

b. An independent OPERABLE flow path capable of taking suction from the RCS hot leg and discharging coolant through the shutdown cooling heat exchanger and back to the RCS through the cold leg injection lines.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one shutdown cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within 1 hour, be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the next 30 hours and continue action to restore the required subsystem to OPERABLE status.

b. With both shutdown cooling subsystems inoperable, restore one subsystem to OPERABLE status within 1 hour or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 6 hours and continue action to restore the required subsystems to OPERABLE status.

c. With both sutdown cooling subsystems inoperable and both reactor coolant loops inoperable, initiate action to restore the required subsystems to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.~~13~~11 Each shutdown cooling subsystem shall be demonstrated OPERABLE:

a. At least once per 18 months, during shutdown, by establishing shutdown cooling flow from the RCS hot legs, through the shutdown cooling heat exchangers, and returning to the RCS cold legs.

b. At least once per 18 months, during shutdown, by testing the automatic and interlock action of the shutdown cooling system connections from the RCS. The shutdown cooling system suction valves shall not open when RCS pressure is greater than 410 psia. The shutdown cooling system suction valves located outside containment shall close automatically when RCS pressure is greater than 500 psia. The shutdown cooling system suction valve located inside containment shall close automatically when RCS pressure is greater than 700 psia.

PLANT SYSTEMS

3/4.7.~~14~~12 CONTROL ROOM AIR TEMPERATURE

LIMITING CONDITION OF OPERATION

3.7.~~14~~12 The control room air temperature shall be maintained less than or equal to 80°F.

APPLICABILITY: ALL MODES

ACTION:

With the control room air temperature greater than 80°F, reduce the air temperature to less than or equal to 80°F within 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.~~14~~12 At least once per 12 hours, verify that the control room air temperature is less than or equal to 80°F.

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M2805	E-NHN-M2827A	SG1 COLD LEG BLOWDOWN ISO VLV J-SGE-HV-41
E-NHN-M2806	E-NHN-M2827B	SG HOT LEG BLOWDOWN ISOLATION VALVE J-SGE-HV-43
E-NHN-M2827	E-NHN-M2827A	REACTOR COOL ANT PUMP OIL LIFT PUMP 1B M-RCN-P02BP
E-NHN-M2828	E-NHN-M2827A	REACTOR COOLANT PUMP OIL LIFT PUMP 2B M-RCN-P02DP
E-NHN-M2809	E-NHN-M2827C	CONTAINMENT EQUIP HATCH J-ZCN-E02
E-NHN-M2811	E-NHN-M2832A	30A RECEPTACLES FOR CTMT BLDG JIB CRANE M-ZCN-G04A, B
E-NHN-M2818	E-NHN-M2832A	30A RECEPTACLES FOR SEAL CRANE ASSY MOT
E-NHN-M2817	E-NHN-M2832B	CTMT BLDG MONORAIL HOIST 1 TON M-ZCN-G03
E-NHN-M2819	E-NHN-M2832B	30A RECEPTACLES FOR CTMT BLDG JIB CRANE M-ZCN-G04 A, B
E-NHN-M2820	E-NHN-M2832D	CTMT BLDG ELEV #2 CONTROLLER J-ZCN-E01
E-NHN-M2821	E-NHN-M2828C	MULTIPLE STUD TENSIONER M-ZCN-M15
E-NHN-M2822	E-NHN-M2828B	WELDING RECPTS E-NHN-I09 B, C, D
E-NHN-M2801A	E-NHN-M2827B	FUEL TRANSFER SYS CONTROL CONSOLE E-PCN-D02
E-NHN-M2833	E-NHN-M2827B	REFUELING MACHINE E-PCN-J02
E-NHN-M2833A	E-NHN-M2827B	CEA CHANGE PLATFORM E-PCN-J01

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-PGB-L34D2	E-NGN-B34D2 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A01D
E-PGB-L34D3	E-NGN-B34D3 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02D
E-PGB-L36D3	E-NGN-B36~~0~~3 (FUSE) D	CTMT NOR ACU FAN M-HCN-A01B
E-PHA-M3318	E-PHA-M3334	SAFETY INJECT TANK 4 ISOL VLV J-SIA-UV-644
E-PHA-M3316	E-PHA-M3316A	SAFETY INJECT TANK 3 ISOL VLV J-SIA-UV-634
E-PHB-M3404	E-PHB-M3405B	NCWS RET INT CTMT ISOL VLV J-NCB-UV-403
E-PHA-M3517	E-PHA-M3521	CTMT PRG RFL MODE ISO VLV J-CPA-UV-2B
E-PHA-M3503	E-PHA-M3507A	SHUT DN CLG ISOL LOOP 1 VLV J-SIA-UV-651
E-PHA-M3508	E-PHA-M3511A	CTMT/RAD SUMP CTMT INT ISO VLV J-RDA-UV-23
E-PHA-M3512	E-PHA-M3513A	CTMT SUMP ISOL TRAIN A VLV J-SIA-UV-673
E-PHB-M3622	E-PHB-M3629	CTMT PRG REFULING MODE ISO VLV J-CPB-UV-3A
E-PHB-M3604	E-PHB-M3604A	SHUT DN CLG ISOL LOOP 2 VLV J-SIB-UV-652
E-PHB-M3619	E-PHB-M3641A	SAFETY INJECTION TANK ISOL VLV J-SIB-UV-614

X

REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The analysis determined a Power Operating Limit (POL) power and assumed a CEA misalignment occurred from this power level. The power penalty factor ~~that~~ would accommodate changes in radial peaks and one hour xenon redistribution that would occur if there were a CEA misalignment with CEACs out of service. The quotient of the POL power and the CEA misalignment Power Penalty factor is the maximum power (50% power) at which DNBR SAFDL violation will occur even if there is a CEA misalignment from POL conditions. Below this power, extra thermal margin will be available to the plant. Thus, for CEA misalignment, power reduction below this limiting power is unnecessary.

The lowest core power for a POL was calculated to be 70% of rated power. This was based on the following worst COLSS fluid conditions.

High Temperature	:	580°F
Low Pressure	:	1785 psia
ASI	:	-.3
Underflow fraction	:	0.865
Low Flow	:	95% of full flow
High Radial Peak	:	1.70 (Bank 5+4+PLR; PDIL = 40% Power)

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The response times in Table 3.3-2 are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time). The response times are taken from the sequence-of-events Tables in Section 15 of CESSAR.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the

PROCEDURES AND PROGRAMS (Continued)

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

k. Pre-planned Alternate Sampling Program implementation.

l. Secondary water chemistry program implementation.

NOTE: The licensee shall perform a secondary water chemistry monitoring and control program that is in conformance with the program discussed in Section 10.3.4.1 of the CESSAR FSAR or another NRC approved program.

m. Post-Accident Sampling System implementation.*

n. Settlement Monitoring Program implementation.

NOTE: The licensee shall maintain a settlement monitoring program throughout the life of the plant in accordance with the program presented in Table 2.5-18 of the PVNGS FSAR or another NRC approved program.

o. CEA Symmetry Test Program implementation

NOTE: The licensee shall perform a CEA symmetry test program in conformance with the program discussed in Section 4.2.2 of the PVNGS SER dated November 11, 1981.

p. Fuel Assembly Surveillance Program Implementation

NOTE: The licensee shall perform a fuel assembly surveillance program in conformance with the program discussed in Section 4.2.4 of the PVNGS SER dated November 11, 1981.

6.8.2 Each program or procedure of Specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5 and approved prior to implementation. Programs, administrative control procedures and implementing procedures shall be approved by the PVNGS Plant Manager, or designated alternate who is at supervisory level or above. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom is a Shift Supervisor or Assistant Shift Supervisor with an SRO on the affected unit.

c. The change is documented, reviewed in accordance with Specification 6.5.2 and approved by the PVNGS plant manager or cognizant department head, as designated by the PVNGS plant manager, within 14 days of implementation. X X

P M

*Not required until prior to exceeding 5% of RATED THERMAL POWER.

PAGE

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SECTION

Table 3.3-6

CHANGE

Item 1.D; Measurement Range, 10^{-1} to 10^{-4} mR/hr is incorrect it should be changed to 10^{-1} to 10^{4} mR/hr.

JUSTIFICATION

This change is necessary due to a typographical error.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
A. Fuel Pool Area RU-31	1	**	$\leq$15mR/hr	10^{-1} to 10^{4}mR/hr	22 & 24
B. New Fuel Area RU-19	1	*	$\leq$15mR/hr	10^{-1} to 10^{4}mR/hr	22
C. Containment RU-148 & RU-149	2	1,2,3,4	$\leq$10R/hr	1R/hr to 10^{7}R/hr	27
D. Containment Power Access Purge Exhaust RU-37 & RU-38	1	#	$\leq$2.5mR/hr	10^{-1} to 10^{4}mR/hr	25
E. Main Steam					
1) RU-139 A&B	1	1,2,3,4	##	10^{-3} to 10^{4}R/hr	27
2) RU-140 A&B	1	1,2,3,4	##	10^{-3} to 10^{4}R/hr	27
2. Process Monitors					
A. Containment Building Atmosphere RU-1	2	1,2,3,4			23 & 27
1) Particulate			$\leq 2.3\times10^{-6}\mu$Ci/cc Cs-137	10^{-9} to $10^{-4}\mu$Ci/cc	
2) Gaseous			$\leq 6.6\times10^{-2}\mu$Ci/cc Xe-133	10^{-6} to $10^{-1}\mu$Ci/cc	
B. Noble Gas Monitors Control Room Ventilation Intake RU-29 & RU-30	1	ALL MODES	$\leq 2\times10^{-5}\mu$Ci/cc	10^{-6} to $10^{-1}\mu$Ci/cc	26
3. Post Accident Sampling System	1###	1,2,3	N.A.	N.A.	28

*With fuel in the storage pool or building.
**With irradiated fuel in the storage pool.
#When purge is being used.
##Three (3) times background in Rem/hour.
###The Minimum Channels Operable will be defined in the Preplanned Alternate Sampling Program.

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3/4 3-39

SECTION

Table 3.3-6, Action Statement 27

CHANGE

Add ...."when required" to item 1 in the action statement.

JUSTIFICATION

Item 2 is being upgraded to require a monitor, which performs the same functions as RU-1, is used as a backup. Since the level of protection provided by the backup unit is the same as RU-1, there is no increased risk to the public, therefore, there is no need to provide a special report when RU-1 is removed from service for greater than 72 hours.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12 or operate the fuel building essential ventilation system while handling irradiated fuel.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the essential filtration mode of operation.

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1. For area monitors RU-139 A and B, RU-140 A and B, RU-148 and RU-149, initiate a preplanned alternate program to monitor the appropriate parameters, when required. X

2. ~~For process monitors, place moveable air monitor in line.~~

   For RU-1, place a backup monitor in service which satisfies requirements a. and c. of Limiting Condition for Operation of Specification 3.4.5.1. No special report need be submitted if this action is completed.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 28 - With the number of OPERABLE Channels one less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 7 days, or:

1. Initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter(s).

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action(s) taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

[illegible]

PAGE

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SECTION

footnote

CHANGE

In ##footnote add the following after option. ....Not required in Mode 5.

JUSTIFICATION

Adding "Not required in Mode 5", would provide consistency between this table, and Sections 3.3.9 and 3.6.1.7. The Containment Purge Isolation System is only required to be operable during core alterations or movement of irradiated fuel within the containment. The containment purge supply and exhaust isolation valves are required to be operable in Modes 1, 2, 3 and 4. Adding this change would reduce unnecessary surveillance testing and decrease the time necessary to perform a containment purge in Mode 5.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
A. Fuel Pool Area RU-31	S	R	M	**
B. New Fuel Area RU-19	S	R	M	*
C. Containment Power Access Purge Exhaust RU-37 & RU-38	P#	R	P###,W##	##
D. Containment RU-148 & RU-149	S	R	M	1,2,3,4
E. Main Steam RU-139 A&B RU-140 A&B	S	R	M	1,2,3,4
2. Process Monitors				
A. Containment Building Atmosphere RU-1				
1) Particulate	S	R	M	1,2,3,4
2) Gaseous	S	R	M	1,2,3,4
B. Control Room Ventilation Intake RU-29 & RU-30	S	R	M	All MODES
3. Post Accident Sampling System	N.A.	R	M***	1,2,3

*With fuel in the storage pool or building.
**With irradiated fuel in the storage pool.
***The functional test should consist of, but not be limited to, a verification of system sampling capabilities.
#If purge is in service for greater than 12 hours, perform once per 12-hour period.
##When purge system is in operation. Not required in Mode 5.
###The functional test should consist of, but not be limited to, a verification of system isolation capability by the insertion of a simulated alarm condition.

PAGE

3/4 3-61 through 3-68: 3/4 7-29 through 7-44; B3/4 3-5; B3/4 7-7; B3/4 7-8 and 6-20

SECTIONS

3/4.3.3; 3/4.7.11; 3/4.7.12; B3/4.3.3.7; B3/4.7.11; B3/4.7.12, 6.9.3

CHANGE

Delete the fire protection program from the technical specifications.

Add Section 6.9.3 to the Administrative Controls special reports.

JUSTIFICATION

The basis for this change is that the fire protection program is covered by the operating license (NPF-41, condition 2.c.7 and NPF-51, condition 2.c.6). The technical specifications are redundant and somewhat conflicting. By deleting the fire protection requirements from the technical specifications, the entire fire protection program will fall under one governing document (license conditions as committed to in the FSAR). The requirements of the license condition are sufficient to ensure that the fire protection program is maintained.

The operation of the fire protection program will not be changed by deleting the requirements from the technical specifications. Periodic testing and inspections will be done and appropriate compensatory measures will be initiated. Significant changes to the program will be reviewed by a qualified fire protection engineer. The attached changes have been submitted to the staff by letter dated July 14, 1986, ANPP-37384.

Section 6.9.3 was added at the request of the NRC.

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each FPER detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

DELETE

ACTION:

a. With any, but not more than one-half the total in any fire zone Function X fire detection instrument shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

b. With more than one-half of the Function X fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function Y fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - CONTROL			
1	74'	Essential Chiller Rm. - Train A			24/0
2	74'	Essential Chiller Rm. - Train B			21/0
3A	74'	Cable Shaft - Train A			1/0
3B	74'	Cable Shaft - Train B			1/0
86A	74-156'4"	Deadspace Compartment - Train A	0/1		0/3
86B	74-156'4"	Deadspace Compartment - Train B	0/1		0/3
4A	100'	Cable Shaft - Train A			1/0
4B	100'	Cable Shaft - Train B			1/0
5A	100'	ESF Switchgear Room - Train A	0/5		0/5
5B	100'	ESF Switchgear Room - Train B	0/5		0/5
6A	100'	DC Equip. Rm. - Train A (Channel C)			2/0
6B	100'	DC Equip. Rm. - Train B (Channel D)			2/0
7A	100'	DC Equip. Rm. - Train A (Channel A)			2/0
7B	100'	DC Equip. Rm. - Train B (Channel B)			2/0
8A	100'	Battery Rm. - Train A (Channel C)	0/2		0/2
8B	100'	Battery Rm. - Train B (Channel D)	0/2		0/2

DELETE

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
9A	100'	Battery Rm. - Train A (Channel A)	0/2		0/2
9B	100'	Battery Rm. - Train B (Channel B)	0/2		0/2
10A	100'	Remote Shutdown Rm. - Train A	0/1		1/1
10B	100'	Remote Shutdown Rm. - Train B	0/1		1/1
11A	120'	Cable Shaft - Train A			1/0
11B	120'	Cable Shaft - Train B			1/0
12	120'	Communications Rm.			0/2
13	120'	Inverter Rm.			0/2
14	120'	Lower Cable Spreading Rm.			
		System 1	0/1		0/6
		System 2	0/1		0/6
		System 3	0/1		0/8
		System 4	0/1		0/8
		System 5	0/1		0/8
		System 6	0/1		0/8
15A	140'	Cable Shaft - Train A			1/0
16	140'	Computer, Office and Storage Rm.			4/4
17	140'	Control Rm. - MCB's & Relay Cabinets	2/0		110/0
18A	160'	Cable Shaft - Train A			1/0
18B	160'	Cable Shaft - Train B			1/0
19	160'	Normal Smoke Exhaust Rm.			1/0
20	160'	Upper Cable Spreading Rm.			
		System 1	0/1		0/12
		System 2	0/1		0/8
		System 3	0/1		0/8
		System 4	0/1		0/8
		System 5	0/1		0/8

DELETE

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - DIESEL GENERATOR			
21A	100'	Diesel Generator - Train A	0/3	0/4	
21B	100'	Diesel Generator - Train B	0/3	0/4	
22A	100'	Diesel Generator Control Rm. - Train A			1/0
22B	100'	Diesel Generator Control Rm. - Train B			1/0
24A	115'	Combustion Air Intake Rm. - Train A			1/0
24B	115'	Combustion Air Intake Rm. - Train B			1/0
23A	131'	Fuel Oil Day Tank - Train A	0/1		
23B	131'	Fuel Oil Day Tank - Train B	0/1		
25A	131'	Exhaust Silencer Rm. - Train A		3/0	
25B	131'	Exhaust Silencer Rm. - Train B		3/0	
		BUILDING - FUEL			
27	100'	Exhaust Essential Air Filtration Unit and Railroad Bay			4/0
28	100'	Spent Fuel Pool Cooling and Cleanup Pump Areas			3/0
29	120'	Electrical Equipment Area			3/0
29A	140'	New and Spent Fuel Storage Area			5/0

DELETE

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
		BUILDING - AUXILIARY			
88A	51'-6" & 40'	West Corridors			6/0
88B	51'-6" & 40'	East Corridors			6/0
90	51'-6" & 40'	Equipment Drain Tank			2/0
30A	51'-6" & 40'	Containment Spray Pump Rm. - Train A			0/2
30B	51'-6" & 40'	Containment Spray Pump Rm. - Train B			0/2
31A	51'-6" & 40'	HPSI Pump Rm. - Train A			0/2
31B	51'-6" & 40'	HPSI Pump Rm. - Train B			0/2
32A	51'-6" & 40'	LPSI Pump Rm. - Train A			0/2
32B	51'-6" & 40'	LPSI Pump Rm. - Train B			0/2
34A	70'	ECW Pump Rm. - Train A			2/0
34B	70'	ECW Pump Rm. - Train B			2/0
35A	70'	Shutdown Cooling Heat Exchanger Train A			4/0
35B	70'	Shutdown Cooling Heat Exchanger Train B			4/0
36	70'	Reactor Makeup and Boric Acid Makeup Pump Room			1/0
37C	70'& 88'	Piping Penetration Rm. - Train A			5/0
37D	70'& 88'	Piping Penetration Rm. - Train B			4/0
37B	70'	Corridors - East			11/0
37A	70'	Corridors - West			11/0

DELETE

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
39A	88'	Pipeways - Train A			8/0
39B	88'	Pipeways - Train B			8/0
42A	100'	Elect. Penetration Rm. - Tr. A (Chan. C)	0/1		0/25
42B	100'	Elect. Penetration Rm. - Tr. B (Chan. B)	0/1		0/24
42C	100'	Corridors - East & Southeast	0/2		3/35
42D	100'	Corridor - West	0/1		2/29
46A	100'	Charging Pump and Valve Gallery Rm. - Train A			0/3
46B	100'	Charging Pump and Valve Gallery Rm. - Train B			0/3
46E	100'	Charging Pump and Valve Gallery Rm. - Train E			0/3
47A	120'	Elect. Penetration Rm. - Tr. A (Chan. A)	0/1		0/28
47B	120'	Elect. Penetration Rm. - Tr. B (Chan. D)	0/1		0/24
48	120'	ECW Surge Tanks Corridor - Tr. A & B			3/0
50B	120'	Valve Gallery			1/0
51B	120'	Spray Chemical Storage Tk Rm.			1/0
52A	120'	Central Corridor - West	0/1		5/17
52D	120'	Central Corridor - East	0/1		7/18
54	120'	Reactor Trip Switchgear Rm.	1/0		6/0
55C	140'	Clean Issue Rm.			1/0
55E	140'	Hot Clothing Rm.			1/0

DELETE

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
56A	140'	Storage and Elect. Rm.			1/0
56B	140'	Storage and Elect. Equip. Rm. - East			6/0
57I	140'	Clothing Issue and Men's Locker Rm.			5/0
57J	140'	Women's Locker, Clean Storage and Lunch Rms.			7/0
57N	140'	Corridor Area			4/0
		BUILDING - CONTAINMENT**			
66A&66B	100' & 120	Southwest and Southeast Perimeter	1/0		
		DELETE			
67A&67B	100'	Northwest and Northeast Perimeter	1/0		
66A	120'	Southwest Perimeter	1/0		
66B	120'	Southeast Perimeter	1/0		
67A&67B	120'	Northwest and Northeast Perimeter	1/0		
63A	120'	No. 1 RCPs and SG Area			6/0
63B	120'	No. 2 RCPs and SG Area			6/0
66A&66B 67A&67B	140'	Southwest, Southeast, Northwest and Northeast Perimeters	1/0		
63A	140'	No. 1 RCPs and SG Area			5/0
63B	140'	No. 2 RCPs and SG Area			5/0
70	140'	Refueling Pool and Canal Area			4/0
71A	140'	North Preaccess Normal AFU Area			2/0

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

FIRE ZONE	ELEVATION	INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
71B	140'	South Preaccess Normal AFU Area			2/0
		MAIN STEAM SUPPORT STRUCTURE			
72	80'	Turbine Driven Aux. Feedpump Rm.			0/3
73	80'	Motor Driven Aux. Feedpump Rm.			1/1
74A	100',120' & 140'	Main Steam Isol. & Dump Valve Area North	0/6		
74B	100',120' & 140'	Main Steam Isol. & Dump Valve Area South	0/6		
		OUTSIDE AREAS			
83		Condensate Storage Tank Pump House			2/0
84A		Spray Pond Pump House - Train A			2/0
84B		Spray Pond Pump House - Train B			2/0

DELETE

~~** The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.~~

*(x/y): x is the number of instruments associated with early fire detection and notification only.

y is the number of instruments associated with actuation of fire suppression systems and early fire detection and notification.

PLANT SYSTEMS

3/4.7.11 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11.1 The fire suppression water system shall be OPERABLE with:

a. Three 50% capacity fire suppression pumps, each with a capacity of at least 1350 gpm, with their discharge aligned to the fire suppression header,

b. Two separate water supply tanks, each with a minimum contained volume of 300,000 gallons (23 feet 1.5 inches), and

c. An OPERABLE flow path capable of taking suction from the T01-A tank and the T01-B tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.7.11.2, 3.7.11.4, and 3.7.11.5.

DELETE

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days by starting the electric motor-driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position, when required to be operable.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 6 months by performance of a system flush.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each pump develops at least 1350 gpm at an indicated differential pressure of 125 psid by recording readings for at least 3 points on the test curve,

   2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   3. Verifying that each fire suppression pump starts sequentially to maintain the fire suppression water system pressure greater than or equal to 85 psig. DELETE

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.11.1.2 The fire pump diesel engines shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by verifying:

   1. The diesel fuel oil day storage tanks each contain at least 315 gallons of fuel, and

   2. The diesel engines start from ambient conditions and operate for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D4176-82, is within the acceptable limits specified in Table 1 of ASTM D975-81 when checked for viscosity, water, and sediment.

c. At least once per 18 months during shutdown, by subjecting the diesels to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.11.1.3 Each fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

DELETE

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The spray and/or sprinkler systems, listed in Table 3.7-3, shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS DELETE

4.7.11.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

c. At least once per 18 months:

   1. By performing a system functional test which includes simulated automatic actuation of the system, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct positions on a thermal/smoke test signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

DELETE

TABLE 3.7-3

SPRAY AND/OR SPRINKLER SYSTEMS

1. Lower Cable Spreading Room Zone 14 - Control Building 120 ft Elevation
   a. System 1
   b. System 2
   c. System 3
   d. System 4
   e. System 5
   f. System 6
2. Upper Cable Spreading Room Zone 20 - Control Building 160 ft Elevation
   a. System 1
   b. System 2
   c. System 3
   d. System 4
   e. System 5
3. Diesel Generator Room, Train A, Zone 21A - Diesel Generator Building 100 ft Elevation
4. Diesel Generator Room, Train B, Zone 21B - Diesel Generator Building 100 ft Elevation
5. Fuel Oil Day Tank Vault, Train A, Zone 23A - Diesel Generator Building 131 ft Elevation
6. Fuel Oil Day Tank Vault, Train B, Zone 23B - Diesel Generator Building 131 ft Elevation

DELETE

7. Low Pressure Safety Injection Pump Room, Train A, Zone 32A - Auxiliary Building 40 ft & 51 ft 6 inch Elevation
8. Low Pressure Safety Injection Pump Room, Train B, Zone 32B - Auxiliary Building 40 ft and 51 ft 6 in. Elevation
9. Electrical Penetration Room, Train A (Channel C) Zone 42A - Auxiliary Building 100 ft Elevation
10. Electrical Penetration Room, Train B (Channel B) Zone 42B - Auxiliary Building 100 ft Elevation
11. Charging Pumps A, B and E Zones 46A, 46B and 46E East Corridors, Zone 42C - Auxiliary Building 100 ft Elevation
12. West Corridors, Zone 42D - Auxiliary Building 100 ft Elevation
13. Electrical Penetration Room, Train A (Channel A) Zone 47A - Auxiliary Building 120 ft Elevation
14. Electrical Penetration Room, Train B (Channel D) Zone 47B - Auxiliary Building 120 ft Elevation
15. Central Corridors, Zone 52A - Auxiliary Building 120 ft Elevation
16. Central Corridors, Zone 52D - Auxiliary Building 120 ft Elevation
17. Turbine-Driven Auxiliary Feed Pump Room Zone 72 - Main Steam Support Structure 81 Ft Elevation
18. Train A Compartments between Auxiliary & Control Buildings, 74 ft & 156 ft 4 inch Elevation Zone 86A.
19. Train B Compartments between Auxiliary & Control Buildings, 74 ft & 156 ft 4 inch Elevation on Zone 86B.
20. Train A Main Steam Support Structure, Zone 74A 100 ft through 140 ft Elevation

PLANT SYSTEMS

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following low-pressure CO_2 systems shall be OPERABLE.

a. ESF Switchgear Room; one Train A, one Train B Zone 5A and 5B Control Building 100 ft Elevation

b. Battery Rooms; one Train A (Channel C) one Train B (Channel D) Zone 8A and 8B Control Building 100 ft Elevation

c. Battery Rooms; one Train A (Channel A) one Train B (Channel B) Zone 9A and 9B Control Building 100 ft Elevation

APPLICABILITY: Whenever equipment protected by the CO_2 system is required to be OPERABLE.

ACTION:

DELETE

a. With one or more of the above required CO_2 systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path is in its correct position.

4.7.11.3.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank weight to be greater than 10000 lb and pressure to be greater than 275 psig, and

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months by verifying:

1. The system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2. By visual inspection that there are no obstructions in the discharge path of the nozzles or during a "Puff Test."

DELETE

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The fire hose stations shown in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE, except that fire hose stations located in containment shall have their containment isolation valves closed in MODES 1, 2, 3, 4, and 5*.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-4 inoperable, provide a gated wye on the nearest OPERABLE hose station. One outlet of the wye shall be connected to the standard length of hose provided for the OPERABLE hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. The above action shall be accomplished within one hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose in 24 hours.

   The hose for the unprotected area shall be stored at the OPERABLE hose station. Signs identifying the purpose and location of the fire hose and related valves shall be mounted above the hose and at the inoperable hose station. DELETE

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the fire hose stations shown in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.
   2. Removing the hose for inspection and reracking, and
   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

*If hot work or other work relating to the use of combustable material or flammable liquids is to be performed in containment during MODE 5, the fire hose stations located in containment shall have their containment isolation valves open during the period the hot work or other work relating to the use of combustable material or flammable liquids is being performed.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

DELETE

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK IDENTIFICATION
Containment NE	80'	HS #01
Containment SE	80'	HS #02
Containment SW	80'	HS #03
Containment NW	80'	HS #04
Containment NE	100'	HS #05
Containment SE	100'	HS #06
Containment SW	100'	HS #07
Containment NW	100'	HS #08
Containment NE	120'	HS #09
Containment SE	120'	HS #10
Containment SW	120'	HS #11
Containment NW	120'	HS #12
Containment NE	140'	HS #13
Containment SW	140'	HS #14
Auxiliary Bldg. North Corridor - W	40'	HS #17
Auxiliary Bldg. North Corridor - E	40'	HS #18
Auxiliary Bldg. North Corridor - W	51'6"	HS #21
Auxiliary Bldg. North Corridor - E	51'6" DELETE	HS #22
Auxiliary Bldg. SE	70'	HS #23
Auxiliary Bldg. SW	70'	HS #24
Auxiliary Bldg. NW	70'	HS #25
Auxiliary Bldg. North Center Corridor	70'	HS #26
Auxiliary Bldg. NE	70'	HS #27
Auxiliary Bldg. NW	88'	HS #30
Auxiliary Bldg. NE	88'	HS #31
Auxiliary Bldg. SW	100'	HS #33
Auxiliary Bldg. East Corridor	120'	HS #37
Auxiliary Bldg. SW	120'	HS #38
Control Bldg. SW	74'	HS #86
Control Bldg. E	74'	HS #87
Control Bldg. SW	100'	HS #88
Control Bldg. East by Elevator	100'	HS #89
Control Bldg. SW	120'	HS #90
Control Bldg. SW	140'	HS #92
Control Bldg. SW	160'	HS #94
Control Bldg. SE	100'	HS #108
Fuel Bldg. South	100'	HS #97

PLANT SYSTEMS

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.11.5 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2-1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

DELETE

SURVEILLANCE REQUIREMENTS

4.7.11.5 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months by visually inspecting each yard fire hydrant for damage.

c. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

   2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings.

   3. Performing a flow check of each hydrant to verify its OPERABILITY.

TABLE 3.7-5

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION	HYDRANT NUMBER
150' Plant North of Fuel Bldg.	F. H. #15
100' Plant West of Rad Waste Bldg.	F. H. #17
150' Plant Northwest of Fuel Bldg.	F. H. #16*

DELETE

*No hose house, however, the hose station is used to service condensate transfer pump

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.6 The following Halon systems shall be OPERABLE.

a. Train A Remote Shutdown Panel Room, Zone 10A - Control Building 100 ft. Elevation

b. Train B Remote Shutdown Panel Room, Zone 10B - Control Building 100 ft. Elevation

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

DELETE

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.6 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight and pressure to be at least 90% of full charge pressure.

c. At least once per 18 months by:

1. Verifying the system actuates manually and automatically, upon receipt of a simulated test signal, and

2. Performance of an air flow test through headers and nozzles to assure no blockage.

PLANT SYSTEMS

3/4.7.12 FIRE-RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.12 All fire-rated assemblies (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire-rated assembly penetrations (fire doors, fire dampers, cable, piping and ventilation duct penetration seals) shall be OPERABLE.

APPLICABILITY: When the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire-rated assemblies (including sealing devices) inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

DELETE

SURVEILLANCE REQUIREMENTS

4.7.12.1 At least once per 18 months the above required fire-rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire damper and associated hardware.

c. Performing a visual inspection of at least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

d. Performing a functional test of at least 10% of the fire dampers that are installed in fire barriers separating redundant trains important to safe shutdown. If any dampers fail to operate correctly, an additional 10% of the dampers shall be sampled. This process shall continue until a 10% sample is verified OPERABLE. Samples shall be selected such that each damper will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.12.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. That each locked-closed fire door is closed at least once per 7 days.

b. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours.

c. Performing a functional test at least once per 18 months.

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

DELETE

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

DELETE

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

The fire zones listed in Table 3.3-11, Fire Detection Instruments, are discussed in Section 9B of the PVNGS FSAR.

7
3/4.3.3.8 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

8
3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

There are two separate radioactive gaseous effluent monitoring systems: the low range effluent monitors for normal plant radioactive gaseous effluents

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, CO_2, Halon, fire hose stations, and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected area(s) until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

DELETE

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. An allowance is made for ensuring a sufficient volume of CO_2/Halon in the CO_2/Halon storage tank by verifying either the weight or the level of the tank.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24-hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.12 FIRE-RATED ASSEMBLIES

DELETE

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire dampers, and fire doors are periodically inspected and functionally tested to verify their OPERABILITY.

3/4.7.~~13~~.11 SHUTDOWN COOLING SYSTEM

The OPERABILITY of two separate and independent shutdown cooling subsystems ensures that the capability of initiating shutdown cooling in the event of an accident exists even assuming the most limiting single failure occurs. The safety analysis assumes that shutdown cooling can be initiated when conditions permit.

The limits of operation with one shutdown cooling inoperable for any reason minimize the time exposure of the plant to an accident event occurring concurrent with the failure of a component on the other shutdown cooling subsystem.

3/4.7.~~14~~.12 CONTROL ROOM AIR TEMPERATURE

Maintaining the control room air temperature less than or equal to 80°F ensures that (1) the ambient air temperature does not exceed the allowable air temperature for continuous duty rating for the equipment and instrumentation in the control room and (2) the control room will remain habitable for operations personnel during plant operation. The 30 days to return the control room air temperature to less than or equal to 80°F in the Action Statement is consistent with the equipment qualification program for the control room.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with 10CFR50.73.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

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SECTION

Table 3.3-12

CHANGE

Add an "or" between instrument numbers JSVNYE-1, JSVNYE-2; JSVNYE-3, JSVNYE-4; JSVNYE-5, JSVNYE6; JSVNYE-7, JSVNYE-8.

JUSTIFICATION

Regulatory Guide 1.133, Rev. 1, Section c.1.a takes the position that a minimum of two sensors, suitably located to provide broad coverage, should be located at each natural collection region (e.g., reactor vessel upper and lower plenums and each pressurized water reactor steam generator coolant inlet plenum). Table 3.3-12 lists two upper vessel and two lower vessel locations as well as two locations in each steam generator. By placing an "or" between each instrument number, the Table will be in compliance with regulatory guidance.

TABLE 3.3-12

LOOSE PARTS SENSOR LOCATIONS

INSTRUMENT NO.	LOCATION
JSVNYE - 1	UPPER VESSEL A (STUD BOLTS)
OR	
JSVNYE - 2	UPPER VESSEL B (STUD BOLTS)
JSVNYE - 3	LOWER VESSEL A (INCORE NOZZLE)
OR	
JSVNYE - 4	LOWER VESSEL B (INCORE NOZZLE)
JSVNYE - 5	SG-1A (HOT LEG)
OR	
JSVNYE - 6	SG-1B (COLD LEG 1A)
JSVNYE - 7	SG-2A (HOT LEG)
OR	
JSVNYE - 8	SG-2B (COLD LEG 2A)

PAGE

3/4 3-75 and 76

SECTION

Table 3.3-13, Action 37 and 42.

CHANGE

Delete item "a", and change b & c to a & b. Editorial changes to action 42.a. Change "the" to "a", and delete the word Sampling.

JUSTIFICATION

By placing movable air monitors in-line or by taking grab samples at least once per 12 hours, is sufficient for monitoring purposes.

The change is to clarify what is actually done.

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.
** During GASEOUS RADWASTE SYSTEM operation.
*** Whenever the condenser air removal system is in operation, or whenever turbine glands are being supplied with steam from sources other than the auxiliary-boiler(s).
\# During waste gas release.
\#\# In MODES 1, 2, 3, and 4 or when irradiated fuel is in the fuel storage pool.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the actions of (a) or (b) ~~or (c)~~ are performed:

~~a. Initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter(s).~~

a. ~~b.~~ Place moveable air monitors in-line

b. ~~c.~~ Take grab samples at least once per 12 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of the GASEOUS RADWASTE SYSTEM may continue provided grab samples are taken and analyzed daily. With both channels inoperable operation may continue provided grab samples are taken and analyzed (1) every 4 hours during degassing operations, and (2) daily during other operations.

TABLE 3.3-13 (Continued)

TABLE NOTATION

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 within one hour after the channel has been declared inoperable.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION b of Specification 3.9.12 or operate the fuel building essential ventilation system while moving irradiated fuel.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement restore the channel to OPERABLE status within 72 hours or:

a. Initiate ~~the~~ a Preplanned Alternate ~~Sampling~~ Program to monitor the appropriate parameter(s) when it is needed.

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action(s) taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

PAGE

3/4 4-9

SECTION

3.4.3.1

CHANGE

Change "nominal capacity" to "minimum capacity". Delete the words "at least" and change "150 kW" to 125 kW".

SECTION

4.4.3.1.2

CHANGE

Change "150 kW" to "125 kW"

JUSTIFICATION

Technical specification surveillance requirement 4.4.3.1.2 states that the capacity of the required groups of pressurizer heaters shall be verified to be at least 150 kW at least once per 92 days. The required pressurizer heaters are powered from a class 1E bus and have a nominal rating of 150 kW. During surveillance testing the pressurizer heaters may fail the surveillance criteria because of normal variations of the bus voltage. This technical specification change proposes reducing the required measured capacity of the pressurizer heaters to allow for variations in the bus voltage.

The basis for the pressurizer heater requirement is to enhance the capability to control reactor coolant system pressure and establish and maintain natural circulation. The original technical specification value of 150 kW was based on analytical pressurizer heat loss value. Since the calculation of this value, heat loss tests have been performed at PVNGS. The measured value of pressurizer heat loss is 118 kW (400,225 Btu/hr). Thus, the revised pressurizer heater capacity requirement of 125 kW is sufficient to offset pressurizer heat loss and allows for expected variation in bus voltage.

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3.1 The pressurizer shall be OPERABLE with a minimum steady-state water level of greater than or equal to 27% indicated level (425 cubic feet) and a maximum steady-state water level of less than or equal to 56% indicated level (948 cubic feet) and at least two groups of pressurizer heaters capable of being powered from Class 1E buses each having a ~~nominal~~ minimum capacity of ~~at least~~ ~~150~~ 125 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of the above required pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, restore the pressurizer to OPERABLE status within 1 hour, or be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.3.1.2 The capacity of the above required groups of pressurizer heaters shall be verified to be at least ~~150~~ 125 kW at least once per 92 days.

4.4.3.1.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power:

a. The pressurizer heaters are automatically shed from the emergency power sources, and

b. The pressurizer heaters can be reconnected to their respective buses manually from the control room.

PAGE

3/4 4-18

SECTION

3.4.5.1

CHANGE

Change "and/or" in the Action Statement to "or".

JUSTIFICATION

The use of the word "and" would mean that two of the required RCS leakage detection systems are inoperable, but the Action Statement only applies to one system inoperable. The action statement addresses operability of 2 of the 3 leakage detection systems with gaseous "and/or" particulate radioactivity systems inoperable. It's not possible to have both of these systems inoperable and still have 2 out of 3 leakage detection systems operable.

CHANGE

Break up the Action Statement into two Action Statements.

JUSTIFICATION

The use of the word "otherwise" in the existing technical specifications has lead to confusion in Units 1 and 2, since it is not clear what the word "otherwise" is referring to. This Action Statement is more conservative than 3.0.3. This proposed change does not change the required Action, but simply makes it more understandable.

REACTOR COOLANT SYSTEM

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.5.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level and flow monitoring system, and

c. The containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; ~~otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

b.

SURVEILLANCE REQUIREMENTS

4.4.5.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere gaseous and particulate monitoring system-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment sump level and flow monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

b. With less than two of the above required leakage detection systems inoperable, or if the requirements to obtain and analyze grab samples of the containment atmosphere at least once per 24 hours of ACTION a. above cannot be met, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

PAGE

3/4 4-20

SECTION

4.4.5.2.2d and e

CHANGE

add; .....of returning to 2250 ±20 psia.... to both of the above sections.

JUSTIFICATION

The Limiting Condition for Operation specifies the limit as 1 GPM at a reactor coolant system pressure of 2250 ±20 psia. This change allows reasonable time to conduct the surveillance upon reaching the necessary test conditions.

SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours**.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit**:

a. At least once per 18 months,

b.* Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d.* Within 24 hours or returning to 2250 ± 20 psia following valve actuation due to automatic or manual action or flow through the valve,

e.* Within 72 hours or returning to 2250 ± 20 psia following a system response to an Engineered Safety Feature actuation signal.

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*The provisions of Specifications 4.4.5.2.2.b, 4.4.5.2.2.d, and 4.4.5.2.2.e are not applicable for valves UV 651, UV 652, UV 653 and UV 654 due to position indication of valves in the control room.

**The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

PAGE

3/4 4-25

SECTION

3.4.7

CHANGE

Add "#" symbol following LIMITING CONDITION FOR OPERATION 3.4.7b, and add the following note at the bottom of the page:

> #Not applicable until a minimum of 2 EFPD and 20 days of power operation have elapsed since reactor was last subcritical for 48 hours or longer.

JUSTIFICATION

If the minimum requirement for sampling has not yet elapsed, any E-BAR value obtained would be invalid. This is consistent with the surveillance requirement (see Note on Table 4.4-4, pg. 4-26).

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/E microcuries/gram.#

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

b. With the specific activity of the primary coolant greater than 100/E microcuries/gram, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/E microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With T_{cold} greater than or equal to 500°F.

Not applicable until a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since Reactor was last subcritical for 48 hours or longer.

PAGE

B3/4 4-2

SECTION

B3/4.4.3

CHANGE

Delete the last sentence ...."Use of the auxiliary pressurizer spray is required during the recovery from a steam generator tube rupture and a small loss of coolant accident".

JUSTIFICATION

The reason for deleting the sentence is that it is not correct. By letter dated 6-26-86 (ANPP-37162), ANPP submitted a revised safety analysis for the Auxiliary Pressurizer Spray System. This new analysis does not take credit for the use of the auxiliary pressurizer spray during the recovery from a steam generator tube rupture and a small loss of coolant accident.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., there is no direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

The auxiliary pressurizer spray is required to depressurize the RCS by cooling the pressurizer steam space to permit the plant to enter shutdown cooling. The auxiliary pressurizer spray is required during those periods when normal pressurizer spray is not available, such as during natural circulation and during the later stages of a normal RCS cooldown. The auxiliary pressurizer spray also distributes boron to the pressurizer when normal pressurizer spray is not available. ~~Use of the auxiliary pressurizer spray is required during the recovery from a steam generator tube rupture and a small loss of coolant accident.~~

PAGE

B3/4 4-7, 4-8 and 4-9

SECTION

3/4.4.8
Table B3/4.4-1

CHANGE

Change F-773-1 to F-6411-2; and delete the phase ...."with the exception of the reactor pressure vessel",.... (See revised pages).

Change reactor vessel thoughness Tables.

JUSTIFICATION

Unit 3 has different reactor vessel thoughness forgings and plates than Units 1 and 2, i.e., Unit 3 specific.

PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimens are removed and examined to determine changes in material properties. The results of these examinations shall be used to update Figure 3.4-2 based on the greater of the following:

(1) the actual shift in reference temperature for plate F-~~773-1~~ 6411-2 and weld 101-142 as determined by impact testing, or

(2) the predicted shift in reference temperature for the limiting weld and plate as determined by RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, ~~with the exception of the reactor pressure vessel~~, has been determined to be 40°F. The Lowest Service Temperature limit is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. However, based upon the 10 CFR Part 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service Temperature line. Therefore, only the isothermal line is shown on Figure 3.4-2.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing these capsules are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

(FORGINGS)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a)(b) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb	MINIMUM UPPER SHELF C_V ENERGY ft-lb
128-201	F-774-01	SA 508-CL3	Inlet Nozzle	-20	-20	-15	+16	N.A.
128-201	F-774-02	SA 508-CL3	Inlet Nozzle	-30	-30	-8	+30	N.A.
128-201	F-774-03	SA 508-CL3	Inlet Nozzle	-40	-30	-6	+30	N.A.
128-201	F-774-04	SA 508-CL3	Inlet Nozzle	-40	-40	+15	+32	N.A.
131-102	F-767-01	SA 508-CL1	Outlet Nozzle Safe End	-30	-10	0	+45	N.A.
131-102	F-767-02	SA 508-CL1	Outlet Nozzle Safe End	-30	-10	0	+45	N.A.
128-301	F-764-01	SA 508-CL2	Outlet Nozzle	-10	-10	0	+30	N.A.
128-301	F-764-02	SA 508-CL2	Outlet Nozzle	-10	-10	0	+30	N.A.
131-101	F-766-01	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+34	N.A.
131-101	F-766-02	SA 508-CL1	Inlet Nozzle Safe End	0	+10	+27	+54	N.A.
131-101	F-766-03	SA 508-CL1	Inlet Nozzle Safe End	-30	+10	+27	+54	N.A.
131-101	F-766-04	SA 508-CL1	Inlet Nozzle Safe End	-30	-20	+20	+49	N.A.
126-101	F-762-01	SA 508-CL2	Vessel Flange	-40	-40	-36	+25	N.A.
106-101	F-761-01	SA 508-CL2	Closure Head Flange	-50	-50	-51	-16	N.A.

N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values of transverse specimens.
(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
(b) = 0° and 180° specimens had the same values.

Revised - see attached page

- NEW PAGE -

TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS
(FORGINGS)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (F)	RTNDT(a) (b) (F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft-lb (F)	TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft-lb (F)	MINIMUM UPPER SHELF CVN ENERGY (ft-lb)
128-3201	F-6409-1	SA 508-CL3	Inlet Nozzle	-50	-50	-22	5	NA
128-3201	F-6409-2	SA 508-CL3	Inlet Nozzle	-60	-60	4	29	NA
128-3201	F-6409-3	SA 508-CL3	Inlet Nozzle	-30	-30	-10	17	NA
128-3201	F-6409-4	SA 508-CL3	Inlet Nozzle	-40	-40	-16	12	NA
131-3302	F-6405-1	SA 508-CL1	Outlet Nozzle Safe End	-30	10	34	76	NA
131-3302	F-6405-2	SA 508-CL1	Outlet Nozzle Safe End	-30	10	34	76	NA
128-3301	F-6404-1	SA 508-CL3	Outlet Nozzle	-20	10	40	75	NA
128-3301	F-6404-2	SA 508-CL3	Outlet Nozzle	-20	10	40	75	NA
131-3301	F-6406-1	SA 508-CL1	Inlet Nozzle Safe End	-20	20	-4	41	NA
131-3301	F-6406-2	SA 508-CL1	Inlet Nozzle Safe End	-20	20	-4	41	NA
131-3301	F-6406-3	SA 508-CL1	Inlet Nozzle Safe End	-20	20	-4	41	NA
131-3301	F-6406-4	SA 508-CL1	Inlet Nozzle Safe End	-20	20	-4	41	NA
126-101	F-6402-1	SA 508-CL2	Vessel Flange	-40	-40	-43	-22	NA
106-101	F-6401-1	SA 508-CL2	Closure Head Flange	-10	-10	-42	0	NA

(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
(b) = 0 and 180 deg. specimens had the same values.
* = Lower bound curve values of transverse specimens.
NA = Not Applicable (no minimum upper shelf requirement).

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

(PLATES)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	@ 50 ft - lb	MINIMUM UPPER SHELF C_v ENERGY ft-lb
142-102	F-773-01	SA 533-GRB-CL1	Lower Shell Plate	-40	+10	+21	+65	105
142-102	F-773-02	SA 533-GRB-CL1	Lower Shell Plate	-50	0	-11	+21	127
142-102	F-773-03	SA 533-GRB-CL1	Lower Shell Plate	-60	-60	-32	-8	129
124-102	F-765-04	SA 533-GRB-CL1	Intermed. Shell Plate	-30	-20	+12	+48	114
124-102	F-765-05	SA 533-GRB-CL1	Intermed. Shell Plate	-20	+10	+15	+52	121
124-102	F-765-06	SA 533-GRB-CL1	Intermed. Shell Plate	-30	+10	+43	+69	126
122-102	F-765-01	SA 533-GRB-CL1	Upper Shell Plate	-30	0	+30	+62	N.A.
122-102	F-765-02	SA 533-GRB-CL1	Upper Shell Plate	-40	+10	+42	+70	N.A.
122-102	F-765-03	SA 533-GRB-CL1	Upper Shell Plate	-30	0	+16	+57	N.A.
102-102	F-770-01	SA 533-GRB-CL1	Closure Head Dome	-60	-20	-6	+36	N.A.
102-102	F-770-02	SA 533-GRB-CL1	Closure Head Dome	-50	-40	-10	+18	N.A.
150-102	F-771-01	SA 533-GRB-CL1	Bottom Head Dome	-90	-50	-37	-4	N.A.
150-102	F-771-02	SA 533-GRB-CL1	Bottom Head Dome	-70	-50	-23	-6	N.A.

(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
* = Lower bound curve values of transverse specimens.

REVISED - See attached page

-NEW PAGE-

TABLE B 3/4.4-1 (Continued)
REACTOR VESSEL TOUGHNESS
(PLATES)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (F)	RTNDT(a) (b) (F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft-lb (F)	@ 50 ft-lb (F)	MINIMUM UPPER SHELF CVN ENERGY (ft-lb)
142-102	F-6411-1	SA 533-GRB-CL1	Lower Shell Plate	-40	-40	-56	-37	156
142-102	F-6411-2	SA 533-GRB-CL1	Lower Shell Plate	-10	0	0	40	111
142-102	F-6411-3	SA 533-GRB-CL1	Lower Shell Plate	-60	-60	-2	30	107
124-102	F-6407-4	SA 533-GRB-CL1	Interm. Shell Plate	-30	-30	10	30	129
124-102	F-6407-5	SA 533-GRB-CL1	Interm. Shell Plate	-20	-20	30	60	114
124-102	F-6407-6	SA 533-GRB-CL1	Interm. Shell Plate	-20	-20	15	40	133
122-102	F-6407-1	SA 533-GRB-CL1	Upper Shell Plate	-20	-20	16	36	NA
122-102	F-6407-2	SA 533-GRB-CL1	Upper Shell Plate	-30	-30	-5	27	NA
122-102	F-6407-3	SA 533-GRB-CL1	Upper Shell Plate	-20	-20	7	32	NA
102-102	F-6414-1	SA 533-GRB-CL1	Closure Head Dome	-60	0	22	57	NA
102-102	F-6414-2	SA 533-GRB-CL1	Closure Head Dome	-40	-10	12	46	NA
150-102	F-6410-1	SA 533-GRB-CL1	Bottom Head Dome	-70	-60	-19	5	NA
150-102	F-6410-2	SA 533-GRB-CL1	Bottom Head Dome	-70	-70	-32	-17	NA

(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
(b) = 0 and 180 deg. specimens had the same values.
* = Lower bound curve values of transverse specimens.
NA = Not Applicable (no minimum upper shelf requirement).

PAGE

3/4 5-1

SECTION

3.5.1

CHANGE

Add the following words to the last sentence in the † footnote. ...."and the level and pressure need not be maintained".

JUSTIFICATION

Chapter 15, Accident Analysis, consider the Safety Injection Tanks to be depressurized and isolated by 650 psig, (well above the Technical Specification Limit of 430 psig for isolation). Per CESSAR Table 6.3.3.2-2, the initial conditions assumed for the large break and small break LOCA spectrums is 2250 psia. CESSAR paragraph 6.3.2.2.1 states that a SIAS below 400 psig would open the SIT isolation valves. In addition, Section 6.3.2.5.1 states: "In the unlikely event of a LOCA during shutdown cooling a SIAS will automatically open the safety injection tank isolation valves". However, a LOCA below 430 psig is not considered in the plant design bases.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve key-locked open and power to the valve removed,

b. A contained borated water level of between 1802 cubic feet (28% narrow range indication) and 1914 cubic feet (72 % narrow range indication),

c. A boron concentration between 2300 and 4400 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

e. Nitrogen vent valves closed and power removed**.

f. Nitrogen vent valves capable of being operated upon restoration of power.

APPLICABILITY: MODES 1*, 2*, 3,*†, and 4*†.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

†With pressurizer pressure greater than or equal to 1837 psia. When pressurizer pressure is less than 1837 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 1415 cubic feet (60% wide range indication) and 1914 cubic feet (83% wide range indication). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 962 cubic feet (39% wide range indication) and 1914 cubic feet (83% wide range indication). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated, and the level and pressure need not be maintained.

*See Special Test Exceptions 3.10.6 and 3.10.8.

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

PAGE

3/4 5-4

SECTION

4.5.2.C.2

CHANGE

Add the following words to the surveillance requirement ....""and tools or material were taken into containment. Containment entries made only for the purpose of inspection or data taking do not require post entry inspection to be performed.""

JUSTIFICATION

The basis for the post entry inspection requirement is to assure no material is left in containment which could restrict emergency pump suction in the event of a LOCA. Entries made only for the purpose of equipment inspection or data taking do not introduce loose debris, etc. to the containment.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with the valves key-locked shut:

Valve Number	Valve Function	Valve Position
1. SIA HV-604	1. HOT LEG INJECTION	1. SHUT
2. SIC HV-321	2. HOT LEG INJECTION	2. SHUT
3. SIB HV-609	3. HOT LEG INJECTION	3. SHUT
4. SID HV-331	4. HOT LEG INJECTION	4. SHUT

b. At least once per 31 days by:

1. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

2. Verifying that the ECCS piping is full of water by venting the accessible discharge piping high points.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. For all the affected areas within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established, and tools or material were taken into Containment. Containment entries made for the purpose of inspection or data taking do not require post entry inspection to be performed.

d. At least once per 18 months by:

PAGE

3/4 6-8

SECTION

4.6.1.6.1

CHANGE

Add an "s" to the word tendon.

JUSTIFICATION

Editorial.

PAGE

3/4 6-9

SECTION

4.6.1.6.2a(3)

CHANGE

(add) If no degradation of the tendon can be detected the tendon shall be retensioned to the prescribed value and considered acceptable.

(Delete) The term defective.

JUSTIFICATION

The present technical specification calls for the defective tendon to be detensioned and add additional lift-off testing shall be performed.... Lift-offs cannot be performed on a detensioned tendon. This change is necessary for clarification of the procedure to be used.

Editorial; the term defective is misleading.

PAGE

3/4 6-9

SECTION

4.6.1.6.2b

CHANGE

Delete the term "or strands".

JUSTIFICATION

Editorial.

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

SURVEILLANCE REQUIREMENTS (Continued)

1) If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability;

2) If the measured prestressing force of the selected tendon in a group lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon, shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered acceptable. If the measured prestressing force of any two tendons falls below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence;

3) If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be completely detensioned and additional lift-off testing shall be performed to determine the cause and extent of such occurrence. If no degradation of the tendon can be detected the tendon shall be retensioned to the prescribed value and considered acceptable.

4) If the average of all measured prestressing forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at anchorage location for that group, the condition shall be considered as below the acceptance criteria for containment vessel structural integrity; and

5) Unless there is degradation of the containment vessel below the acceptance criteria during the first three inspections, the sample population for subsequent inspections shall include at least 6 tendons (3 hoop and 3 inverted U).

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires. A previously stressed tendon wire or strands from one tendon of each group shall be removed for testing and examination over the entire length to determine (which should include the broken wire if so identified) that:

1) The tendon wires are free of corrosion, cracks, and damage;
2) There are no changes in the presence or physical appearance of the sheathing filler-grease; and

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.6 except for Specification 4.6.1.6.2a.4), restore the containment vessel to the required level of integrity within 15 days, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.6.2a.4), restore the containment vessel to the required level of integrity within 72 hours, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the containment vessel shall be demonstrated at the end of 1, 3 and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. All of the acceptance testing of tendons and visual examinations of end anchorages, adjacent concrete surfaces and containment vessel surfaces shall be performed sequentially and within the same time frame.

4.6.1.6.2 The structural integrity of the tendons shall be demonstrated by:

a. Determining from a random but representative sample of at least 10 tendons (6 hoop and 4 inverted U) that each group (hoop, and inverted U) has an observed lift-off force within the predicted limits for that group. For each subsequent inspection one tendon from each group shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows:

PAGE

3/4 6-10

SECTION

4.6.1.6.2b3

CHANGE

Add the word "tensile" after ultimate.

Delete the "d" in evidenced.

JUSTIFICATION

Editorial.

PAGE

3/4 6.10

SECTION

4.6.1.6.2C

CHANGE

Add;....The average force level (the average force between the tendon ends)....

JUSTIFICATION

The average force level can be achieved without modification to the original shim stack height and without influence to the adjacent tendons.

CHANGE

add; ....or the predicted lower limit value....

JUSTIFICATION

The present technical specification does not specify which predicted value.

CHANGE

add; ....of the predicted lower limit....

JUSTIFICATION

The present technical specification does not identify the predicted value.

CHANGE

add; ....for the predicted upper limit less the value for any broken and/or missing wires and any unseated button heads,....

JUSTIFICATION

The ineffective wires must be considered in order not to exceed 70% of GUTS for the tendon.

CHANGE

add; ....that calculated....

JUSTIFICATION

The technical specification refers ....three approximately equally spaced levels of force.... The original installation data sheets do not reflect elongations and force at three equally spaced levels consequently any comparison to be made during retensioning can only be done on calculated values.

CHANGE

Delete the words "or strands"

JUSTIFICATION

Editorial.

SURVEILLANCE REQUIREMENTS (Continued)

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate tensile strength of the tendon material) exists for at least three wire samples (one from each end and one at mid-length) cut from each removed wire. Failure of any one of the wire samples to meet the minimum tensile strength test is evidenced that structural integrity is below the acceptance criteria.

c. Performing tendon retensioning of those tendons detensioned for inspection to at least force level recorded prior to detensioning or the predicted value, ~~whichever is greater,~~ with the tolerance within minus zero to plus 6%, except that the final seating force shall be such that the stress in the wire ~~or strand~~ shall not exceed 70% of the guaranteed ultimate tensile strength of the tendons. During retensioning of these tendons, the stress in the tendon shall not exceed 80% of its ultimate strength, and the changes in load and elongation shall be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 10% from that ~~recorded during installation~~, an investigation shall be made to ensure that the difference is not related to wire failures or slips of wires in anchorages; and

of the predicted lower limit

or the predicted upper limit but less the value for any broken and/or missing wires and unseated button heads,

d. Verifying the OPERABILITY of the sheathing filler-grease by assuring:

1) No voids in excess of 5% of the net duct volume,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits specified as follows:

Water content	0 - 5% by wt.
Chlorides	0 - 10 ppm
Nitrates	0 - 10 ppm
Sulfides	0 - 5 ppm
Reserved Alkalinity (Base Numbers)	0 - 50% of the installed value (installed value 0-5 for older grease).

4.6.1.6.3 As an assurance of the structural integrity of the containment vessel, tendon anchorage assembly hardware (such as bearing plates, stressing washers, ~~wedges,~~ and buttonheads) of all tendons selected for inspection shall be visually examined. For those containments in multiple unit plants for which only visual inspection need be performed, tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load-bearing components of the anchorages. The surrounding concrete shall also be checked visually for indication of any abnormal condition.

PAGE

3/4 6-21 and 3/4 6-31

SECTION

Table 3.6-1

CHANGE

Delete valve number SIA-UV682 #28 on Page 3/4 6-31 and add it to page 3/4 6-21.

JUSTIFICATION

Valve SIA-UV682, Safety Injection Tank Drain, should be included in Section A-Containment Isolation (CIAS) and should be deleted from Section F-Normally Open/ESF Actuated Closed. SIA-UV682 is normally closed.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A. CONTAINMENT ISOLATION (CIAS)	
RDA-UV 023	9	Containment radwaste sump pump to LRS holdup tank	30
RDB-UV 024	9	Containment radwaste sump pump to LRS holdup tank	5
RDB-UV 407	9	Containment radwaste sump post-accident sampling system	5
SGB-HV 200#	11	Downcomer feedwater chemical injection	1
SGB-HV 201#	12	Downcomer feedwater chemical injection	1
SIA-UV 708#	23	Containment recirc sump to post-accident sampling system	5
HCB-UV 044	25A	Containment air radioactivity monitor (inlet)	12
HCA-UV 045	25A	Containment air radioactivity monitor (inlet)	12
HCA-UV 046	25B	Containment air radioactivity monitor (outlet)	12
HCB-UV 047	25B	Containment air radioactivity monitor (outlet)	12
SIA-UV 682#	28	Safety Injection Tank Drain Line	5
GAA-UV 002	29	N_2 to steam generator and reactor drain tank	10
GAA-UV 001	30	N_2 to SI tanks	10

#Not Type C tested.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
	F. NORMALLY OPEN - ESF ACTUATED CLOSED (Continued)		
SGA-UV 175#	12	Downcomer FIV	N.A.
~~SIA-UV-682#~~	~~28~~	~~SI-drain-from-drain-tank~~	~~N.A.~~
SGA-UV 211#	37A	Steam generator blowdown sample	N.A.
SGB-UV 228#	37A	Steam generator blowdown sample	N.A.
SGA-UV 204#	37B	Steam generator blowdown sample	N.A.
SGB-UV 219#	37B	Steam generator blowdown sample	N.A.
SGA-UV 500P#	46	Steam generator blowdown to SCCS	N.A.
SGB-UV 500Q#	46	Steam generator blowdown to SCCS	N.A.
SGB-UV 500R#	47	Steam generator blowdown to SCCS	N.A.
SGA-UV 500S#	47	Steam generator blowdown to SCCS	N.A.
SGB-UV 226#	48	Steam generator blowdown to downcomer blowdown sample	N.A.
SGA-UV 227#	48	Steam generator blowdown to downcomer blowdown sample	N.A.
SGA-UV 220#	49	Steam generator blowdown to downcomer blowdown sample	N.A.
SGB-UV 221#	49	Steam generator blowdown to downcomer blowdown sample	N.A.
SGB-UV 224#	63A	SG2 blowdown sample	N.A.
SGA-UV 225#	63A	SG2 blowdown sample	N.A.
SGB-UV 222#	63B	SG2 blowdown sample	N.A.
SGA-UV 223#	63B	SG2 blowdown sample	N.A.

#Not Type C tested.

PAGE

3/4 6-38; 6-39; 7-17; 7-20; 9-15; B3/4 6-4; 7-5; 9-3

SECTION

4.6.4.3b.2; 4.6.4.3c; 4.7.7b.2; 4.7.8b.2; 4.9.12b.2; 4.9.12c; B3/4.6.4; B3/4 7-8

CHANGE

The proposed change will allow testing of the charcoal filter units in accordance with ANSI N509-1980 in lieu of ANSI N509-1976. ANSI N509-1980 has been accepted by the NRC and the acceptance is documented in Section 6.5.1 of the Standard Review Plan, Revision 2, July 1981 (NUREG-0800).

JUSTIFICATION

Portions of the Unit 1 and 2 technical specifications reference ANSI N509-1980 (Amendment No. 1 to NPF-51, dated 8-11-86; Amendment No. 7 to NPF-41, 8-11-86).

DESCRIPTION OF THE PROPOSED CHANGE REQUEST

The requested change regarding the ESF air filtration unit charcoal involves only the initial qualification tests which are performed by the manufacturers to certify suitability of the impregnated activiated carbon for removal of radio-iodines from air streams and the verification tests which are performed by the user prior to installation of the charcoal into the filter unit. The differences betwen ANSI N509-1976 and ANSI N509-1980, Table 5-1, reflect a refinement in the test methods used for initial qualification. (See Table 1).

TABLE 1

Summary of Differences Between Table 5-1
of ANSI N509-1976 and ANSI N509-1980

A. Physical Characteristics:

	1976	1980
1. Particle Size Distribution:		
Retained on #6 Sieve	0.0%	0.1% max.
Retained on #8 Sieve	5.0%	5.0% max.
Thru #8, Retained on #12	40-60%	60% max.
Thru #12, Retained on #16	40-60%	40% min.
Thru #16, Retained on #16	5.0% max.	5.0% max.
Thru #18, Retained on #16	1.0% max.	1.0% max.

		1976	1980
2.	Hardness No.	95 min.	92 min.
3.	Ignition Temp (Min)	330°C	330°C
4.	CCL4 Activity	60 min.	60 min.
5.	Bulk Density (min.)	0.38 gm/ML	0.38g/cm^3

B. Performance Efficiency

		1976	1980
1.	Methyl Iodide @ 95% RM	99% at 25°C	97% @ 30°C
2.	Methyl Iodide @ 80°C & 95% RH	99%	99%
3.	Methyl Iodide @ 130°C, 95% RH	98%	98%
4.	Elemental Iodine Retention	99.9% Loading 99.9% Incl. Elution	99.9% min.
5.	Elemental Iodine @ 180°C	NA	99.5% min.

The essential filtration system is not directly used to help the plant achieve safe shutdown. This system ensures that the offsite radiation exposures and exposures to operations personnel in the control room are within the guideline values during and following all credible accident conditions. The PVNGS accident analysis assumes a filter efficiency of 95%. ANSI N509-1976 required new charcoal filter for Methyl Iodide 25°C, 95% RH to be 99% efficient. ANSI N509-1980 requires an efficiency of 97% at 30°C and 95% RH. As called out by Regulatory Guide 1.52 Revision 2, this ANSI Standard is used only for new charcoal filters. The PVNGS filters meet all the criteria for used filters when surveillance tests were performed. Due to the information presented, it is clear that updating the surveillance requirements in the Technical Specifications to include the 1980 version of the ANSI Standard will not increase the probability of occurrence of accidents or malfunctions of equipment important to safety as analyzed in the Final Safety Analysis Report (FSAR).

Because the change does not involve a change to the plant design or the manner in which the plant is operated and the current analyses in the FSAR remain valid, the possibility of any new accident or malfunctions is not created.

The NRC has reviewed ANSI N509-1980 and incorporated it as acceptance criteria in the Standard Review Plan, NUREG 0800, Section 6.5.1 Revision 2, 1981. No change is being made to the surveillance interval or the testing method. The essential filtration system will still serve the same purpose and function in the same manner as before this proposed change.

HYDROGEN PURGE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A containment hydrogen purge cleanup system, shared among the three units, shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With the containment hydrogen purge cleanup system inoperable and one hydrogen recombiner OPERABLE as determined by Specification 4.6.4.2, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE.

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 50 scfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2**, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978**.

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*With less than two hydrogen recombiners OPERABLE.

** ANSI N509-1980 is applicable for this specification.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying a system flow rate of 50 scfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 50 scfm ± 10%.

2. Verifying that the heaters dissipate at least 0.5 kW when tested in accordance with ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 50 scfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 50 scfm ± 10%.

---

* ANSI N509-1980 is applicable for this specification.

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 28,600 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a system flow rate of 28,600 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 28,600 cfm ± 10%.

2. Verifying that on a Control Room Essential Filtration Actuation Signal and on a SIAS, the system is automatically placed into a filtration mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8-inch Water Gauge relative to adjacent areas during system operation at a makeup flow rate to the control room of less than or equal to 1000 cfm.

4. Verifying that the emergency chilled water system will maintain the control room environment at a temperature less than or equal to 80°F for a period of 30 minutes.

---

* ANSI N509-1980 is applicable for this specification.

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,* meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.*

3. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10%.

2. Verifying that the system starts on an SIAS test signal.

e. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.0% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

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*ANSI N509-1980 is applicable for this specification

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 6000 cfm ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

3. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978*, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978*.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters, pre-filters, heaters, and charcoal adsorber banks is less than 8.4 inches Water Gauge while operating the system at a flow rate of 6000 cfm ± 10%.

2. Verifying that on a high radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the fuel building at a measurable negative pressure relative to the outside atmosphere during system operation.

---

* ANSI N509-1980 is applicable for this specification

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment automatic isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The only valves in the Table 6.2.4-1 of the PVNGS FSAR that are not required to be listed in Table 3.6-1 are the following: main steam safety valves and main steam atmospheric dump valves. The main steam safety valves and the atmospheric dump valves have very high pressure setpoints to actuate and are covered by Specifications 3/4.7.1.1 and 3/4.7.1.6, respectively.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971. The use of ANSI standard N509-1980 in lieu of ANSI standard N509-1976 to meet the guidance of Regulatory Guide 1.52, Revision 2, Positions C.6.a and C.6.b, has been found acceptable as documented in Revision 2 to Section 6.5.1 of the Standard Review Plan (NUREG-0800).

The guidance of Regulatory Guide 1.52, Revision 2, Regulatory positions C.6.a and C.6.b are followed by using the qualification criteria for new charcoal of Table 5-1 of ANSI N509-1980, as documented in Section 6.5.1 of the Standard Review Plan (NUREG-0800).

3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

The OPERABILITY of the control room essential filtration system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

3/4.7.8 ESF PUMP ROOM AIR EXHAUST CLEANUP SYSTEM

The OPERABILITY of the ESF pump room air exhaust cleanup system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Plant Review Board. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommedations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number

REFUELING OPERATIONS

BASES

A shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during surveillance testing of ECCS pumps. This is necessary to meet Surveillance 4.5.2, flow testing of the HPSI pumps without other pumps running, and 4.3.3.5, testing of the containment spray pumps and LPSI pumps during surveillance of the remote shutdown system.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth (at least 23 feet above the top of the spent fuel) is available to remove a nominal 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly for a maximum fuel rod pressurization of 1200 psig. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL BUILDING ESSENTIAL VENTILATION SYSTEM

The limitations on the fuel building essential ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. The use of ANSI standard N509-1980 in lieu of ANSI standard N509-1976 to meet the guidance of Regulatory Guide 1.52, Revision 2, Positions C.6.a and C.6.b, has been found acceptable as documented in Revision 2 to Section 6.5.1 of the Standard Review Plan (NUREG-0800).

PAGE

3/4 7-1 and 3/4 7-3

SECTION

3.7.1.1 and Table 3.7-2

CHANGE

1) Delete the word ..."Maximum" in Variable Overpower trip setpoint.... and add the word "ceiling" after setpoint.

2) Delete the word ...."MAXIMUM" in the title before VARIABLE OVERPOWER.... and also in the column heading. (See markup)

JUSTIFICATION

The title "Maximum Variable Overpower Trip Setpoint" is misleading since Variable Overpower Trip (VOPT) is equiped with ceiling, rate, and floor setpoints.

[illegible]

TABLE 3.7-2

MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL AND ~~MAXIMUM~~ VARIABLE OVERPOWER TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	~~MAXIMUM~~ VARIABLE OVERPOWER TRIP SETPOINT CEILING (% OF RATED THERMAL POWER)	MAXIMUM ALLOWABLE STEADY STATE POWER LEVEL (% OF RATED THERMAL POWER)
1	108.0	98.2
2	97.1	87.3
3	86.2	76.4
4	75.3	65.5

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2, 3, and 4*.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam safety valves inoperable per steam generator, operation in MODES 1 and 2 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or ~~the Maximum~~ Variable Overpower trip setpoint ceiling and the Maximum Allowable Steady State Power Level are reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with at least one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

---

*Until the steam generators are no longer required for heat removal.

**The maximum number of inoperable safety valves on any operating steam generator is four (4).

[illegible]

PAGE

3/4 8-4

SECTION

4.8.1.1.2.C

CHANGE

Add the words ....""from time of breaker closure""....

JUSTIFICATION

The time required for synchronization may vary widely between individual operators and/or generator response to speed raise/lower switch manipulations. Therefore, it would extremely difficult to verify the synchronization (which in itself may take us a significant portion of 60 seconds) occurs expeditiously enough to allow loading within 60 seconds.

4.8.1.1.2 (Continued)

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM-D4176-82, is within the acceptable limits specified in Table 1 of ASTM D975-81 when checked for viscosity, water and sediment.

c. At least once per 184 days the diesel generator shall be started** and accelerated to generator voltage and frequency at 4160 ± 420 volts and 60 ± 1.2 Hz in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal. The generator shall be manually synchronized to its appropriate emergency bus, loaded to an indicated 5200-5400*** kW in less than or equal to 60 seconds, and operate for at least 60 minutes. from time of breaker closure. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4, may also serve to concurrently meet those requirements as well.

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a single largest load of greater than or equal to 839 kW (Train B AFW pump) for emergency diesel generator B or 696 kW for emergency diesel generator A (Train A HPSI pump) while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz.

3. Verifying that the automatic load sequencers are OPERABLE with the interval between each load block within ± 1 second of its design interval.

4. Simulating a loss of offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesel starts** on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is

**This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

***This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring of the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

PAGE

3/4 8-10

SECTION

4.8.2.1.C.3

CHANGE

Change surveillance requirement 4.8.2.1.C.3 to read: "The connection resistance between the cable terminal lugs and the cell terminal post at each end of the cable jumpers, and the bolted bus bar connections between cells is less than or equal to 150×10^{-6} ohms", and

JUSTIFICATION

The intent of this requirement is to ensure the bolted connection resistance between cells do not increase due to corrosion, de-torqueing, etc. Since the cable resistance values were approved during initial installation and will remain essentially constant, it is satisfactory to measure the connection resistance between the cable terminal lugs and cell terminal post at each end of the cable jumpers to meet the acceptance criteria.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 105 volts, or battery overcharge with battery terminal voltage above 145 volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of six connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3. ~~The resistance of each cell-to-cell and terminal connection is less than or equal to 150 x 10-6 ohms, and~~

The connection resistance between the cable terminal lugs and the cell terminal post at each end of the cable jumpers, and the bolted bus bar connections between cells is less than or equal to 150×10^{-6} ohms, and

4. The battery charger will supply at least 400 amperes for batteries A and B and 300 amperes for batteries C and D at 125 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

PAGE

3/4 8-17

SECTION

3.8.4.1

CHANGE

Change ACTION a. to read as follows:

a. Restore the protection device(s) to OPERABLE Status or deenergize the circuit(s) by tripping or racking out the associated backup circuit circuit breaker or racking out or removing the inoperable device within 72 hours and declare the affected system or component inoperable and verify the backup circuit to be tripped or racked out or the inoperable circuit breaker racked out at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers racked out or tripped or the inoperable device racked out or removed, or

JUSTIFICATION

The Technical Specification bases for providing containment electrical penetrations and penetration conductors protection is by either demonstrating operability of the primary and backup overcurrent protection, or by deenergizing circuits. Protection by deenergizing circuits is provided by having the backup breaker or inoperable in-line device racked out (to either the test or full-out position) or by it being removed.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-2 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective devices shown in Table 3.8-2 inoperable:

or racking out

a. Restore the protection device(s) to OPERABLE status or deenergize the circuits(s) by tripping, the associated backup circuit breaker or racking out or removing the inoperable device within 72 hours and declare the affected system or component inoperable and verify the backup circuit breaker to be tripped, or racked out or the inoperable circuit breaker racked out at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers racked out or tripped, or the inoperable device racked out or removed, or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.1 All containment penetration conductor overcurrent protective devices (except fuses) shown in Table 3.8-2 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 kV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protection relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8-2.

PAGE

3/4 11-4

SECTION

Table 4.11-1, Note c

CHANGE

Delete "and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specificant 6.9.1.8".

JUSTIFICATION

Table 4.11-1 does not involve releases offsite. The Semiannual Radioactive Effluent Release Report is a report on radioactive materials released to areas at and beyond the site boundary, which does not include the onsite evaporation pond. In the standard Technical Specification (NUREG-0472), Section 3.11.1.1 limits liquid releases to unrestricted areas, which is not applicable to PVNGS. When Section 3.11.1.1 was written for PVNGS' onsite evaporation pond, the reference to the semiannual Radioactive Effluent Release Report was mistakenly left in.

TABLE 4.11-1 (Continued)

TABLE NOTATION

[c]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137 and Ce-141. Ce-144 shall also be measured, but with an LLD of 5 x 10^{-6}. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed. ~~and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.~~

[d]A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

PAGE

3/4 11-8

SECTION

Table 4.11-2, item D.

CHANGE

Change the words ...."All Radwaste Types as listed in A., B., and C. above." To "All Release Types as listed in C. above".

JUSTIFICATION

At other facilities each release type as listed in Table 4.11-2 may vent directly to atmosphere, however, at Palo Verde the release types listed in A and B exhaust through the plant vent. The waste gas decay tanks and containment purges are not separate effluent release points and therefore do not require the LLD specified.

PAGE

3/4 11-10

SECTION

Table 4.11-2, Note d.

CHANGE

Replace "....samples collected for 24 hours..." with "....samples collected for 24 hours or less...".

JUSTIFICATION

Provides clarification.

PAGE

3/4 11-10

SECTION

Table 4.11-2, new note h

CHANGE

Add note "h", which reads "Continuous sampling is only required in the modes specified for each effluent monitor in 3.3.3.9".

Add note "h", to all Continuous SAMPLING FREQUENCIES in Table 4.11-2.

JUSTIFICATION

This would provide clarification between Table 4.11-2 and Specification 3.3.3.9, which specifies in which Modes the effluent monitors are required.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) ($\mu Ci/ml$)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
B. Containment Purge	P Each Purge[b,c] Grab Sample	P Each Purge[b,c]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. 1. Condenser Vacuum Pump Exhaust 2. Plant Vent 3. Fuel Bldg. Exhaust	M[b,e] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[f,h]	4/M[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f,h]	4/M[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f,h]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f,h]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
D. All Release ~~Radwaste~~ Types as listed in A., B., and C. above.	Continuous[f,h]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

[b]Analyses shall also be performed following SHUTDOWN, STARTUP, or a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period if 1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and 2) the noble gas activity monitor on the plant vent shows that effluent activity has increased by more than a factor of 3. If the associated noble gas vent monitor is inoperable, samples must be obtained as soon as possible. Analyses shall be performed within a four-hour period. This requirement does not apply to the Fuel Building Exhaust.

[c]Sampling and analyses shall also be performed at least once per 31 days when purging time exceeds 30 days continuous.

[d]Samples shall be changed at least 4 times a month and analyses shall be completed within 48 hours after changing (or after removal from sampler). When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. or less

[e]Tritium grab samples shall be taken at least monthly from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

[f]The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

[g]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported in the Semiannual Radioactive Effluent Release Report.

[h] Continuous sampling is only required in the MODES specified for each effluent monitor in 3.3.3.9.

PAGE

5-9

SECTION

Table 5.7-2

CHANGE

Add to the Auxiliary Spray Δ Ta....601-650.... and to N_A....75.

JUSTIFICATION

A Δ T occurred in the 601-650 range in Unit 1 on July 12, 1986. The occurrence was addressed by EER #86-RC-159. The Technical Specification Table 5.7-2 needs to be expanded to reflect this area of plant operation. This additional transient is within the bounds of the PVNGS safety analysis.

TABLE 5.7-2

PRESSURIZER SPRAY NOZZLE USAGE FACTOR

Main Spray				Auxiliary Spray			
ΔT_m	N_A	N	N/N_A	ΔT_a	N_A	N	N/N_A
201-250	7900			201-250	50000		
251-300	4500			251-300	2200		
301-350	2900			301-350	1300		
351-400	1900			351-400	850		
401-450	1200			401-450	550		
451-500	850			451-500	375		
501-550	555			501-550	225		
				551-600	150		
				601-650	75		
		$\Sigma N/N_A$ =	______			$\Sigma N/N_A$ =	______

Cumulative Usage Factor

$\Sigma N/N_A$ (Main Spray) ____________

$\Sigma N/N_A$ (Aux. Spray) ____________

Total ____________ = Cumulative Usage Factor

PAGE

6-3 and 6-4

SECTION

Figure 6.2-1 and 6.2-2

CHANGE

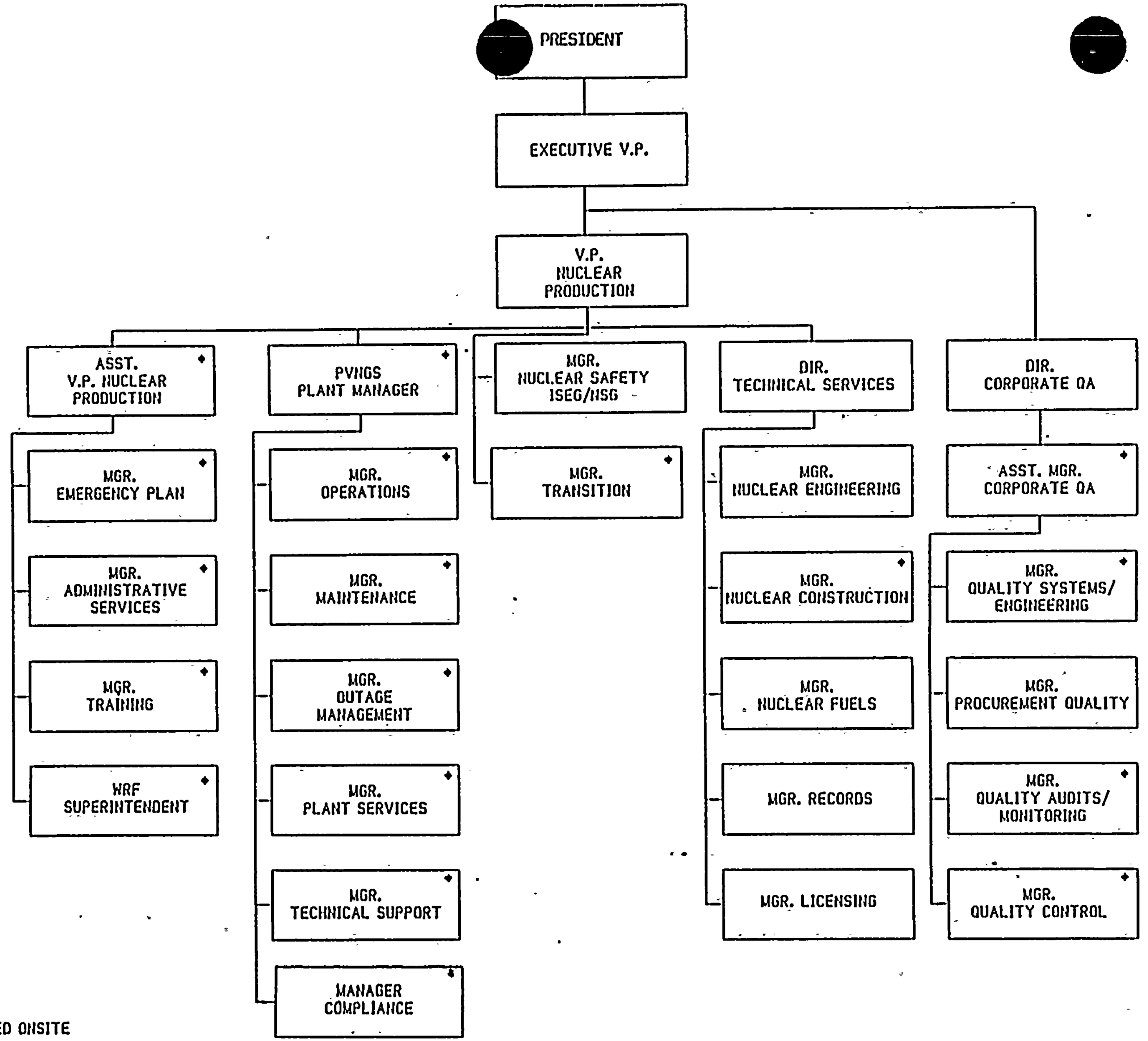
1) The Executive V.P. will report directly to the President instead of the CEO.

2) The Manager of Compliance will report to the Plant Manager instead of the Technical Support Manager.

JUSTIFICATION

1) The changes to Figures 6.2-1 and 6.2-2 represent the current organization. The Executive Vice President of Arizona Nuclear Power Project will be reporting to the President and Chief Operating Officer of Arizona Public Service Co. Previously, the Executive Vice President reported directly to the Chief Executive Officer. This change was made because Palo Verde is shifting from its construction phase to its commercial operation phase.

2) Additional changes can also be seen on these figures in that; a) the Compliance Department will report to the Plant Manager. This change is intended to increase the effectiveness of Compliance in responding to regulatory and plant issues. And b), the title of Figures 6.2-2, has been changed from "ONSITE UNIT ORGANIZATION" to "ONSITE ORGANIZATION".



♦ LOCATED ONSITE

FIGURE 6.2-1
OFFSITE ORGANIZATION

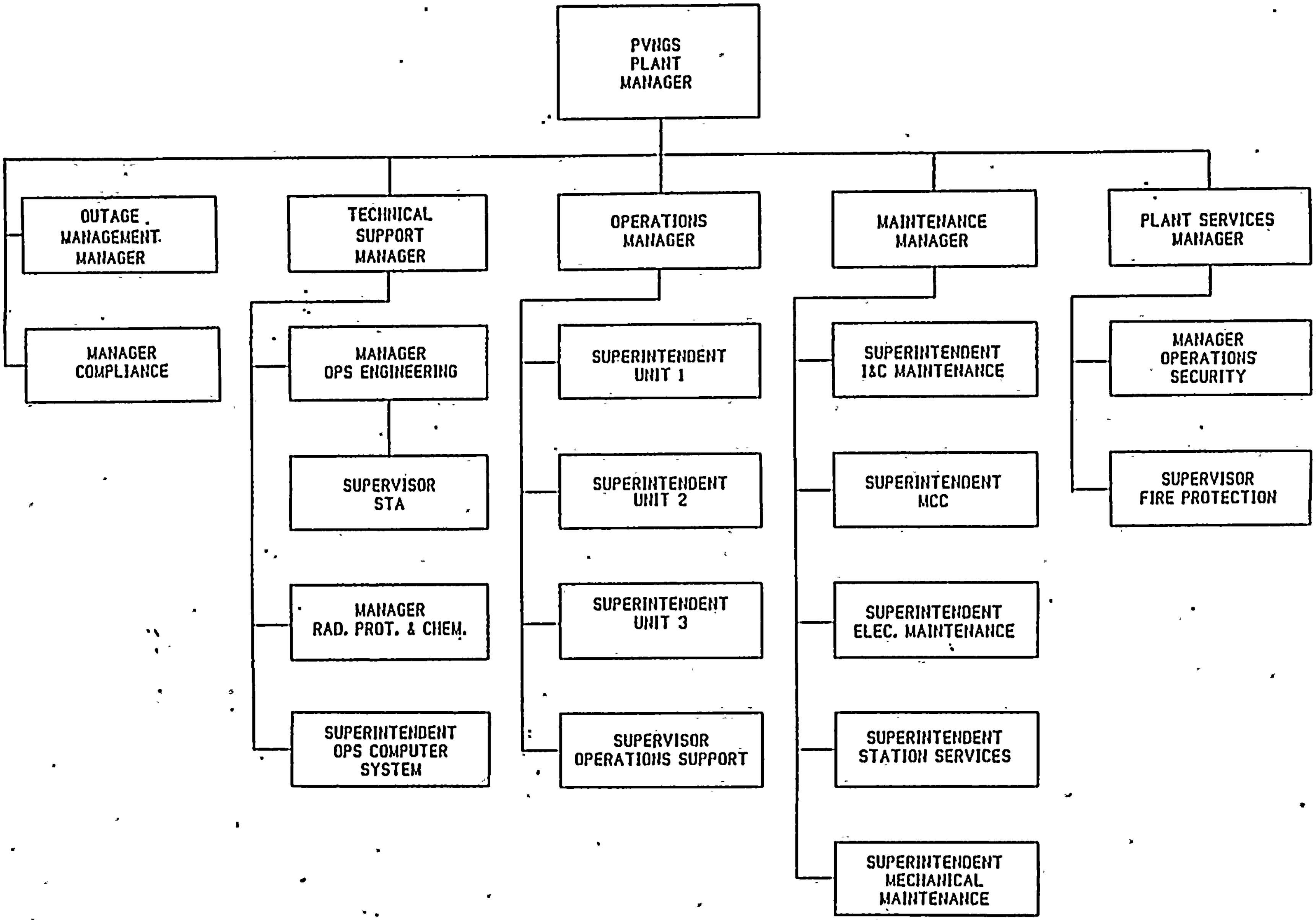


FIGURE 6.2-2
ONSITE ORGANIZATION

OLD ORGANIZATION

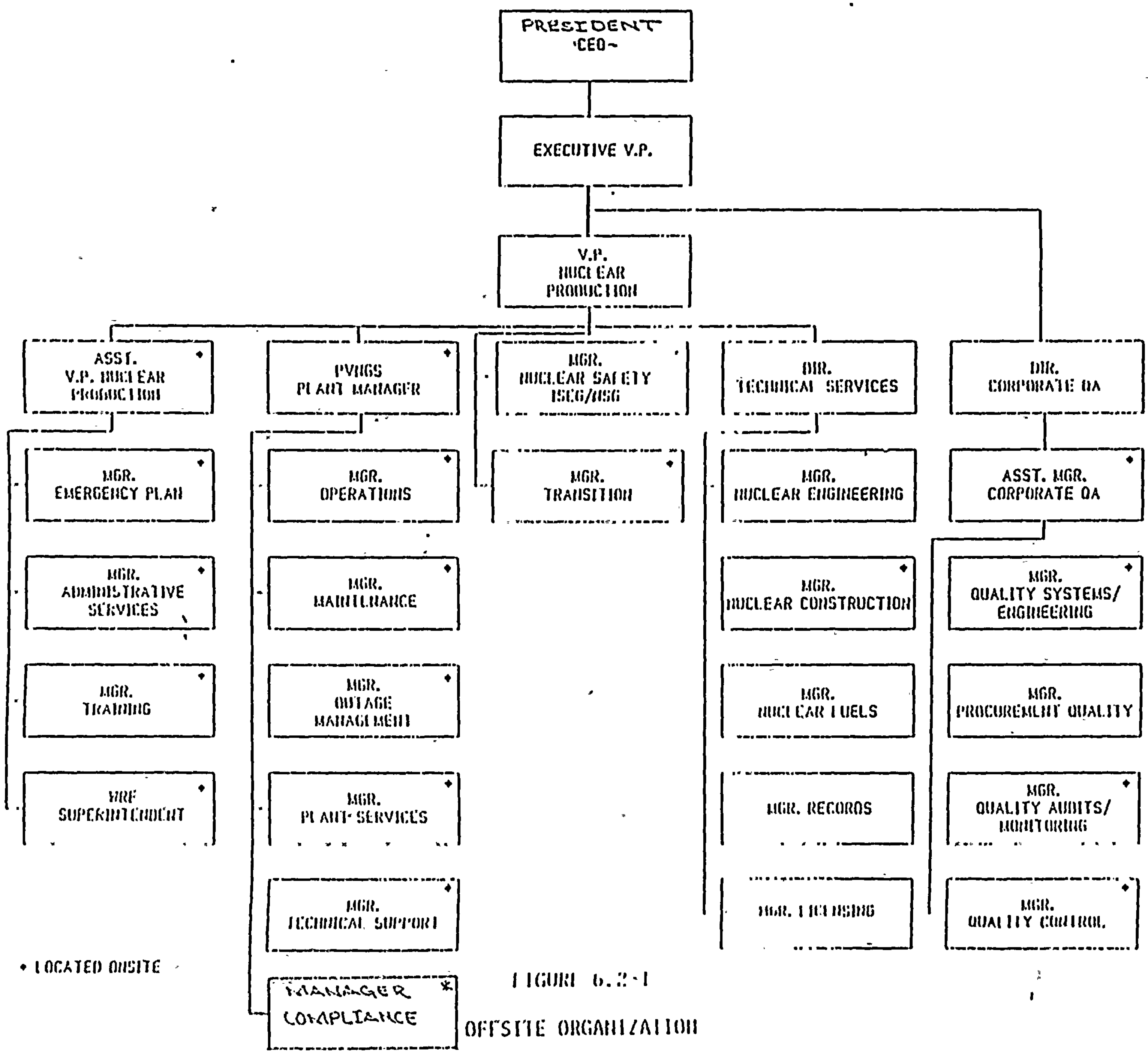


FIGURE 6.2-1

OFFSITE ORGANIZATION

OLD ORGANIZATION

PALO VERDE - UNIT

CONTROLLED BY USER

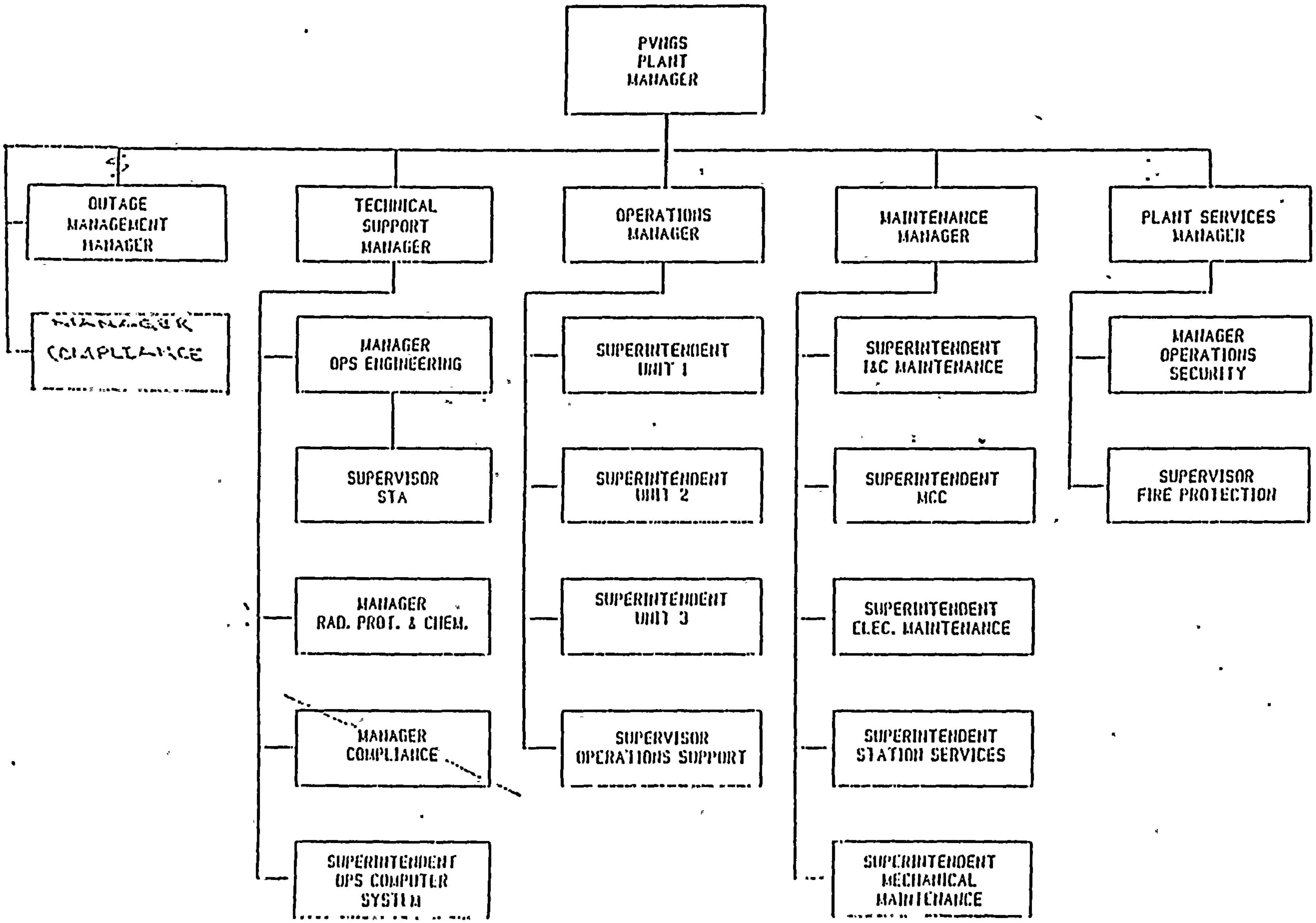


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FIGURE 6.2-2

ONSITE UNIT ORGANIZATION

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LCO's and Surveillance Requirements: 3/4 1-1, 1-2, 1-2a, 1-3, 1-5, 1-8, 1-10, 1-13, 3-5, 10-1, 10-9.

Bases: B3/4 1-1, 1-1a, 1-2, 10-2.

SECTIONS

Index,

Definitions

Section 2; Table 2.2-1, Table 2.2-1; Note (5); 3/4.1; Table 3.3-1, Note (c), 3/4.10.1; 3/4.10.9; B3/4.1, B3/4.10.9

CHANGE

(See marked-up technical specification pages).

JUSTIFICATION

(See attachment pages 1-30; Figures 1-1 through 1-32 and 4-1 through 4-5).

INDEX

DEFINITIONS

DEFINITIONS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

X

X

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

KN-1

1.16 KN-1 is the k-effective calculated assuming the fully or partially inserted full-length control element assembly of highest inserted worth is fully withdrawn.

MEMBER(S) OF THE PUBLIC

1.1~~6~~7 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.1~~7~~8 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.1~~8~~9 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.~~19~~20 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and cold leg reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.2~~0~~1 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.2~~1~~2 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.2~~2~~3 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

DEFINITIONS

PROCESS CONTROL PROGRAM (PCP)

1.234 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full-scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full-scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites. X

PURGE - PURGING

1.245 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. X

RATED THERMAL POWER

1.256 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3800 MWt. X

REACTOR TRIP SYSTEM RESPONSE TIME

1.267 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism. X

REPORTABLE EVENT

1.278 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.72 and 50.73 to 10 CFR Part 50. X

SHUTDOWN MARGIN

1.289 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: X

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.~~29~~30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.3~~0~~1 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION

1.3~~1~~2 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.3~~2~~3 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.3~~3~~4 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.3~~4~~5 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.3~~5~~6 UNIDENTIFIED LEAKAGE shall be all leakage which does not constitute either IDENTIFIED LEAKAGE or reactor coolant pump controlled bleed-off flow.

UNRESTRICTED AREA

1.3~~6~~7 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.3~~7~~8 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.3~~8~~9 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Logarithmic Power Level - High (1)		
a. Startup and Operating	< 0.~~798%~~ .010% of RATED THERMAL POWER	< 0.~~895%~~ .011% of RATED THERMAL POWER
b. Shutdown	< 0.~~798%~~ .010% of RATED THERMAL POWER	< 0.~~895%~~ .011% of RATED THERMAL POWER
C. Core Protection Calculator System		
1. CEA Calculators	Not Applicable	Not Applicable
2. Core Protection Calculators	Not Applicable	Not Applicable
D. Supplementary Protection System		
Pressurizer Pressure - High	$\leq$ 2409 psia	$\leq$ 2414 psia
II. RPS LOGIC		
A. Matrix Logic	Not Applicable	Not Applicable
B. Initiation Logic	Not Applicable	Not Applicable
III. RPS ACTUATION DEVICES		
A. Reactor Trip Breakers	Not Applicable	Not Applicable
B. Manual Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATIONS

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) In MODES 3-4, value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) In MODES 3-4, value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level wide range instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below ~~1%~~ 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to ~~1%~~ 10^{-4}% of RATED THERMAL POWER.

The approved DNBR limit is 1.231 which includes a partial rod bow penalty compensation. If the fuel burnup exceeds that for which an increased rod bow penalty is required, the DNBR limit shall be adjusted. In this case a DNBR trip setpoint of 1.231 is allowed provided that the difference is compensated by an increase in the CPC addressable constant BERR1 as follows:

$$BERR1_{new} = BERR1_{old} \left[1 + \frac{RB - RB_o}{100} \times \frac{d\ (\%\ POL)}{d\ (\%\ DNBR)}\right]$$

where $BERR1_{old}$ is the uncompensated value of BERR1; RB is the fuel rod bow penalty in % DNBR; RB_o is the fuel rod bow penalty in % DNBR already accounted for in the DNBR limit; POL is the power operating limit; and d (% POL)/d (% DNBR) is the absolute value of the most adverse derivative of POL with respect to DNBR.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - ~~T_{cold} GREATER THAN 210°F~~ ALL CEAS FULLY INSERTED

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~6.0%~~ 1.0% delta k/k.

APPLICABILITY: MODES ~~1, 2*~~, 3*, ~~and~~ 4*, and 5 with all Full-length CEAS fully inserted.

ACTION:

With the SHUTDOWN MARGIN less than ~~6.0%~~ 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 26 gpm to reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~6.0%~~ 1.0% delta k/k:

~~a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).~~

~~b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.~~

~~c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.~~

* See Special Test Exception 3.10.~~1~~ 9.

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d. ~~Prior to initial~~ operation above 5% RATED THERMAL POWER ~~after each~~ fuel loading, by consideration ~~of the factors of~~ e. below, with the ~~CEA groups at the~~ Transient Insertion Limits of Specification ~~3.1.3.6~~.

a. ~~e.~~ When in MODE 3 or 4, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1~~e~~, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

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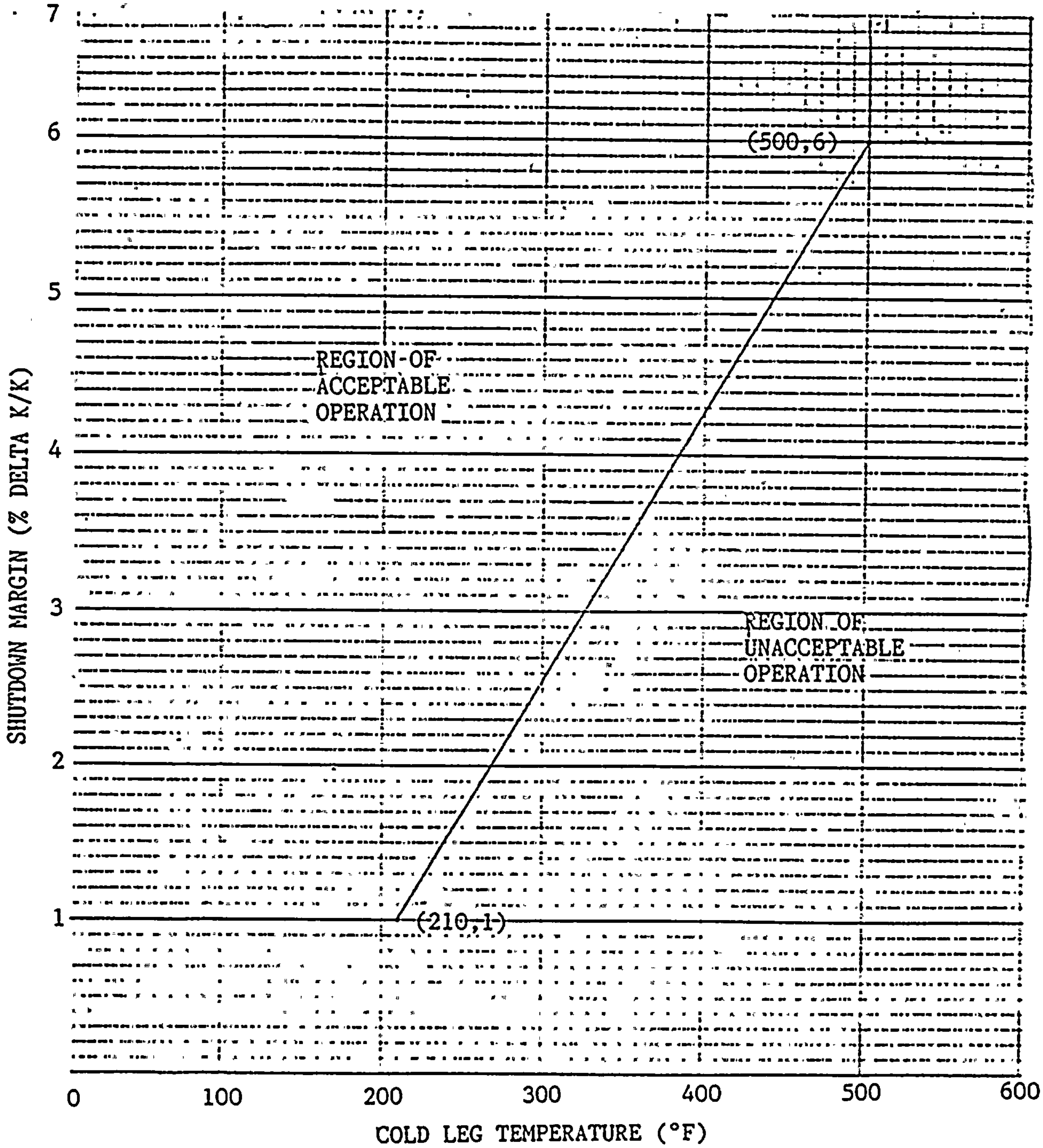


FIGURE 3.1 - 1A

SHUTDOWN MARGIN VERSUS COLD LEG TEMPERATURE

SHUTDOWN MARGIN - K_{N-1} - ANY CEA WITHDRAWN ~~T_{cold} LESS THAN OR EQUAL TO 210°F~~

LIMITING CONDITION FOR OPERATION

3.1.1.2

~~3.1.1.2~~ a. The SHUTDOWN MARGIN shall be greater than or equal to ~~4.0% delta k/k.~~ that shown in Figure 3.1-1A, and

b. For T_{cold} less than or equal to 500°F, K_{N-1} shall be less than 0.99.

APPLICABILITY: MODES ~~5~~, 1, 2*, 3*, 4* and 5* with any full-length CEA fully or partially withdrawn.

ACTION:

a. With the SHUTDOWN MARGIN less than that in Figure 3.1-1A, ~~4.0% delta k/k~~, immediately initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored, and

b. With T_{cold} less than or equal to 500°F and K_{N-1} greater than or equal to 0.99, immediately vary CEA positions and/or initiate and continue boration at greater than or equal to 26 gpm to the reactor coolant system of a solution containing greater than or equal to 4000 ppm of boron or equivalent until the required K_{N-1} is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 With any full-length CEA fully or partially withdrawn, the SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~4.0% delta k/k:~~ that in Figure 3.1-1A:

a. Within 1 hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable as a result of excessive friction or mechanical interference or known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

~~b. At least once per 24 hours by consideration of the following factors:~~

INSERT

~~1. Reactor Coolant System boron concentration,~~
~~2. CEA position,~~
~~3. Reactor Coolant System average temperature,~~
~~4. Fuel burnup based on gross thermal energy generation,~~
~~5. Xenon concentration, and~~
~~6. Samarium concentration.~~

*See Special Test Exception 3.10.1 and 3.10.9

INSERT

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3, 4, or 5, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration
2. CEA position
3. Reactor Coolant System average temperature
4. Fuel burnup based on gross thermal energy generatio
5. Xenon concentration, and
6. Samarium concentration

4.1.1.2.2 When in MODE 3, 4, or 5, with any full-length CEA fully or partially withdrawn, and T_{cold} less than or equal to 500°F, K_{N-1} shall be determined to be less than 0.99 at least once per 24 hours by consideration of at least.

the following factors:

1. Reactor Coolant System boron concentration
2. CEA position
3. Reactor Coolant System average temperature
4. Fuel burnup based on gross thermal energy generation
5. Xenon concentration, and
6. Samarium concentration

4.1.1.2.3 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.2.1e or 4.1.1.2.2. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY ~~and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F~~ within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. X

SURVEILLANCE REQUIREMENTS

4.1.2.2.1 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm for 1 charging pump and 68 gpm for two charging pumps to the Reactor Coolant System.

4.1.2.2.2 The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or Mode 4 to perform the surveillance testing of Specification 4.1.2.2.b provided the testing is performed within 24 hours after achieving normal operating pressure in the reactor coolant system.

X

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY ~~and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F~~ within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. X

SURVEILLANCE REQUIREMENTS

4.1.2.4 No additional Surveillance Requirements other than those required by Specification 4.0.5.

X

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

APPLICABILITY: MODES 1, 2,* 3,* and 4*.

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and ~~borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F,~~ and restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and
2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

---

*See Special Test Exception 3.10.7.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above $10^{-4}\%$ of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to $10^{-4}\%$ of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below ~~1%~~ $10^{-4}\%$ of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to ~~1%~~ $10^{-4}\%$ of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN AND K_{N-1} -CEA WORTH TESTS

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN and K_{N-1} requirement of Specification 3.1.1.~~1~~.2 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

APPLICABILITY: MODES 2, 3* and 4*#.

ACTION:

a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN and K_{N-1} required by Specification 3.1.1.~~1~~2 ~~is~~ are restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3 or MODE 4, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

a. Reactor Coolant System boron concentration,

b. CEA position,

c. Reactor Coolant System average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration, and

f. Samarium concentration.

---

*Operation in MODE 3 and MODE 4 shall be limited to 6 consecutive hours.

#Limited to low power PHYSICS TESTING at the 320°F plateau.

3/4.10.9 SHUTDOWN MARGIN AND K_{N-1} - CEDMS TESTING

LIMITING CONDITION FOR OPERATION

3.10.9 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 and the SHUTDOWN MARGIN and K_N-1 requirements of Specification 3.1.1.2 may be suspended for pre-startup tests to demonstrate the OPERABILITY of the control element drive mechanism system provided:

a. No more than one CEA is withdrawn at any time.

b. No CEA is withdrawn more than 7 inches.

c. The K_N-1 requirement of Specification 3.1.1.2 is met prior to the start of testing.

d. All other operations involving positive reactivity changes are suspended during the testing.

APPLICABILITY: MODES 4 and 5

ACTION: With any of the above requirements not met, suspend testing and comply with the requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable.

SURVEILLANCE REQUIREMENTS

4.10.9 Surveillance Requirements 4.1.1.2.1 e and 4.1.1.2.2 shall be conducted within one hour prior to the start of testing, and at least once per 12 hours during testing.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN AND KN-1

~~A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made~~ subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits assuming the insertion of the regulating CEAs are within the limits of Specification 3.1.3.6, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SEE INSERT A

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{cold}. The most restrictive condition occurs at EOL, with T_{cold} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 6.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with the criteria used to establish the power dependent CEA insertion limits and with the assumptions used in the FSAR Safety Analysis.

With T_{cold} less than or equal to 210°F, the reactivity transients resulting from uncontrolled RCS cooldown are minimal and a 4% Δk/k SHUTDOWN MARGIN requirement is set to ensure that reactivity transients resulting from an ~~inadvertent single CEA withdrawal event are minimal.~~

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). During operation in MODES 1 and 2, with K_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately followin a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and,

...therefore, the required SHUTDOWN MARGIN also decrease. Below T_{cold} of about 210°F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 210°F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with at least one CEA partially or fully withdrawn, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. In the analysis of the CEA ejection event, the K_{N-1} requirement ensures that the radially averaged enthalpy acceptance criterion is satisfied, considering power redistribution effects. Above T_{cold} of 500°F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

Other technical specifications that reference the Specifications on SHUTDOWN MARGIN or K_{N-1}

are: 3/4.1.2, BORATION SYSTEMS, 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, 3/4.9.1, REFUELING OPERATIONS - BORON CONCENTRATION, 3/4.10.1, SHUTDOWN MARGIN AN[D] K_N-1 - CEA WORTH TESTS, and 3/4.10.9, SHUTDOWN MARGIN AND K_N-1 - CEDMS TESTING.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) an emergency power supply from OPERABLE diesel generators, and (5) the volume control tank (VCT) outlet valve CH-UV-501, capable of isolating the VCT from the charging pump suction line. The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

INSERT. B

~~The boration capability of either system is sufficient to provide a~~ SHUTDOWN MARGIN ~~from expected~~ operating conditions of ~~4% delta~~ k/k after xenon decay and cooldown to 210°F. The ~~maximum expected~~ boration capability requirement occurs at EOL from ~~full power~~ equilibrium xenon ~~conditions~~ and requires 23,800 ~~gallons of 4000~~ ppm borated water from either the refueling ~~water tank or the spent fuel pool~~.

With the RCS temperature below 210°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

INSERT C

~~The boron capability required below 210°F is based upon providing a 4%~~ delta k/k SHUTDOWN MARGIN ~~after xenon decay and cooldown~~ from 210°F to 120°F. This condition ~~requires 9,700 gallons~~ of 4000 ppm borated ~~water from either the refueling water tank or the spent fuel pool.~~

- INSERT B -

Each system is capable of providing boration equivalent to a SHUTDOWN MARGIN of 4% delta k/k after xenon decay and cooldown to 210° F. Therefore, the boration capacity of either system is more than sufficient to satisfy the SHUTDOWN MARGIN and/or K_N-1 requirements of the Specifications. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

- INSERT C -

Each system is capable of providing boration equivalent to a SHUTDOWN MARGIN of 4% delta k/k after xenon decay and cooldown from 210°F to 120° F. Therefore, the boration capacity of the system required below 210° F is more than sufficient to satisfy the SHUTDOWN MARGIN and/or K_N-1 requirements of the Specifications. This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

BASES

3/4.10.6 SAFETY INJECTION TANKS

This special test exception permits testing the low pressure safety injection system check valves. The pressure in the injection header must be reduced below the head of the low pressure injection pump in order to get flow through the check valves. The safety injection tank (SIT) isolation valve must be closed in order to accomplish this. The SIT isolation valve is still capable of automatic operation in the event of an SIAS; therefore, system capability should not be affected.

3/4.10.7 SPENT FUEL POOL LEVEL

This special test exception permits loading of the initial core with the spent fuel pool dry.

3/4.10.8 SAFETY INJECTION TANK PRESSURE

This special test exception allows the performance of PHYSICS TESTS at low pressure/low temperature (600 psig, 320°F) conditions which are required to verify the low temperature physics predictions and to ensure the adequacy of design codes for reduced temperature conditions.

3/4.10.9 SHUTDOWN MARGIN AND KN-1 - CEDMS TESTING

This special test exception allows the performance of control element drive mechanism test prior to startup, without the operator having to be concerned as to whether Specification 3.1.1.1 or 3.1.1.2 is applicable as CEAs are moved. The logarithmic power level-high trip provides additional protection against inadvertant criticality during this test.

ATTACHMENT

A. DESCRIPTION OF THE TECHNICAL SPECIFICATION AMENDMENT REQUEST

The proposed changes to Technical Specification sections 1.0, 2.0, 3/4.1.1, 3/4.1.2 and 3/4.10.1 reduce the boration requirements when shut down, by modification of the shutdown margin requirements as follows:

1. A new parameter, K_{N-1}, is introduced and defined as the K_{eff} calculated assuming the partially or fully inserted rod of highest inserted worth is fully withdrawn.

2. Limiting Conditions for Operation (LCOs) 3.1.1.1 and 3.1.1.2 require that for Modes 1-4, the shutdown margin be greater than or equal to 6% delta K/K, and for Mode 5, greater than or equal to 4% delta K/K. The proposed changes revise the shutdown margin requirements for Modes 1-5 according to full length Control Element Assembly (CEA) position. The revised Tech Spec 3.1.1.1 is applicable when all full length CEAs are fully inserted and requires that for Modes 3-5, the shutdown margin be greater than or equal to 1% delta K/K. The revised Tech Spec 3.1.1.2 is applicable when any full length CEA is withdrawn and requires that for Modes 1-5, the shutdown margin be greater than or equal to that given in a new Figure 3.1-1A. For reactor coolant cold leg temperature less than or equal to 500°F, K_{N-1} shall be less than 0.99. The LCO action statements are also revised to require boration when the above shutdown margin requirements are not met.

   Surveillance Requirements 4.1.1.1.1 and 4.1.1.2 require that the shutdown margin be verified at given time intervals to satisfy the LCO requirements. The proposed changes revise Tech Spec 4.1.1.1.1 and 4.1.1.2 to require the shutdown margin to be verified to the proposed new LCO requirements, as described above. In addition, the proposed change requires K_{N-1} to be determined to be less than 0.99 at least once every 24 hours.

   The associated Bases 3/4.1.1 and 3/4.1.2 are also revised to reflect the proposed changes.

3. The action statements for LCOs 3.1.2.2, 3.1.2.4, and 3.1.2.6, for Modes 1-4, require in part that when the requirements of the LCOs are not met, boration to a shutdown margin equivalent to at least 6% delta K/K at 210°F be carried out. The proposed changes delete any reference to the shutdown margin requirements.

4. LCO 3.10.1 currently requires that a reactivity equivalent to at least the highest estimated CEA worth be available for trip insertion when the shutdown margin requirement of Tech Spec 3.1.1.1 is suspended for measurement of CEA worth and shutdown margin

during physics tests, and boration is required when that requirement is not met. The proposed change revises LCO 3.10.1 to state that the shutdown margin and K_{N-1} requirements of Tech Spec 3.1.1.2 may be suspended for measurement of CEA worth and shutdown margin, provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from operable CEAs or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

5. The proposed change adds Special Test Exception 3.10.9 to allow the facility to suspend the requirements of Tech Spec 3.1.1.1 and 3.1.1.2 for the demonstration of the operability of the control element drive mechanism system during pre-startup tests. Basis 3/4.10.9 is also added to reflect this.

6. As a result of proposed changes 1-5 above, it is necessary to propose changes to revise setpoints to provide reactor trips to prevent the core from exceeding its safety limits in terms of departure from nucleate boiling ratio (DNBR) and local power density. These proposed changes consist of two parts:

   a. Item B.2 of Table 2.2-1 specifies a trip setpoint for "Excore Neutron Flux - Logarithmic Power Level - High" of less than or equal to 0.798% of rated thermal power, and an allowable value of less than or equal to 0.815% of rated thermal power. The proposed change revises the trip setpoint and allowable value to 0.010% and 0.011% of rated thermal power, respectively.

   b. Table notation (c) of Table 3.3-1 and Table notation (5) of Table 2.2-1 state that CPC trips may be manually bypassed below 1% of rated thermal power and the bypass shall automatically be removed when thermal power is greater than or equal to 1% of rated thermal power. The proposed changes revise the value at which the CPC trip may be manually bypassed and at which the manual bypass is automatically removed, from 1% of rated thermal power to 10^{-4}% of rated thermal power.

7. The proposed change renumbers Special Test Exception 3.10.9 "Natural Circulation Testing Program" (Unit 1 only) to 3.10.10. This is necessary so the Special Test Exception added in item 5 above will be the same section number in both Unit 1 and 2 Technical Specifications.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

The purpose of the Technical Specifications affected by these proposed changes is to ensure that an adequate shutdown margin is maintained in the reactor at all times.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

(Will add later)

D. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

1. The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

   A discussion of these standards as they relate to the amendment request follows:

   Standard 1 -- Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

   The anticipated operational occurrences (AOOs) and accidents that have the potential for being impacted by the proposed changes are steam line break, CEA withdrawal, CEA ejection, inadvertent deboration and startup of an inactive reactor coolant pump (RCP). All the impacted AOOs and accidents have been reevaluated to determine the resulting consequences due to the proposed changes. The results of these evaluations show that the consequences are still within the appropriate acceptance criteria discussed below.

   The PVNGS safety analysis requires that steam line break events be evaluated considering potential for fuel damage. If the minimum DNBR during a steam line break event falls below specified limits based on acceptable correlations, fuel damage must be assumed. The results of the limiting steam line break analysis indicate that the minimum post-trip DNBR remains well above the specified safety limit.

   The PVNGS safety analysis requires that the consequences of an uncontrolled control element assembly (CEA) withdrawal from a subcritical or low power startup condition be evaluated on the basis that they are acceptable if the minimum DNBR remains above specified limits based on acceptable correlations. The reevaluation of the limiting CEA withdrawal analysis indicates that the minimum DNBR will remain above the plant specific safety limit of 1.231.

C. The PVNGS safety analysis requires that for a startup of an inactive RCP, fuel clad integrity should be maintained by ensuring that specified acceptable fuel design limits are not exceeded. The results of a limiting startup of an inactive RCP indicate that the reactor remains subcritical and the specified acceptable fuel design limits are not exceeded, thus maintaining fuel clad integrity.

The PVNGS safety analysis requires that for a CEA ejection, the reactivity excursion should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel rod. Reevaluation of the limiting CEA ejection accident concurrent with the introduction of the K_{N-1} requirement ensures that the safety analysis acceptance criterion will be met.

The proposed change to lower the setpoint of the high logarithmic power trip will provide the trip function earlier than the previous setpoint. This trip provides protection in the event of an inadvertent control element assembly (CEA) bank withdrawal from MODES 2 and 3 initial conditions with four reactor coolant pumps (RCPs) operating. The proposed change to lower the value of power below which the CPC trip can be bypassed and above which the manual bypass is automatically removed also provides added protection for an inadvertent CEA bank withdrawal postulated to occur in MODES 3, 4, or 5 with less than four reactor coolant pumps operating. If the reactor coolant pressure or temperature is outside the CPC wide range trip limits, a continuous reactor trip signal will be generated by all four CPC channels and an immediate reactor trip will terminate an inadvertent CEA bank withdrawal event before significant power is generated. The MODE 2 and 3, with four RCPs operating, and the MODE 3, 4, and 5, with less than four RCPs operating events have been determined to be less limiting than the CEA bank withdrawal event presented in the Final Safety Analysis Report. Also, the proposed changes do not alter how the CPCs respond to design basis events.

Therefore, operation of the facility in accordance with the proposed changes will not involve a significant increase in probability or consequences of an accident previously evaluated.

Standard 2 -- Create the Possibility of a New or Different Kind of Accident from Any Accident Previously Evaluated.

There is no change in the plant hardware or analysis method as a result of the proposed changes. Although some of the proposed changes will result in modification to the operating procedures and plant operation in the shutdown modes, operation of the facility in accordance with the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 — Involve a Significant Reduction in a Margin of Safety.

Operation of the facility in accordance with the proposed changes may reduce in some way a safety margin, but where the results of the change are clearly within the acceptance criteria.

The PVNGS safety analysis requires that for an inadvertent boron dilution, a minimum time interval of 15 minutes for MODES 1 through 5 and 30 minutes for MODE 6 be available from the time an alarm makes the operator aware of unplanned boron dilution before a loss of shutdown margin occurs. The time to a complete loss of shutdown margin for the limiting inadvertent boron dilution is now 50 minutes compared to 95 minutes for the previously analyzed incident. However, an alarm will alert the operator of an unplanned boron dilution at least 15 minutes prior to a complete loss of shutdown margin.

Although a numerically smaller value of SHUTDOWN MARGIN may appear to result in a reduction in the safety margin, operation of the facility in accordance with the proposed changes does not involve a significant reduction in a margin of safety as an alarm is available in time to satisfy the safety analysis criterion.

The lower logarithmic power level trip setpoint and automatic removal of the CPC manual bypass at a lower power level result in an earlier reactor protective system actuation for the postulated transients, which involves no significant reduction in the margin of safety.

2. The proposed changes in parts 1, 3, 4, and 5 match the guidance concerning the application of the standards for determining whether or not a significant hazards consideration exists (48 FR 14870) by the example:

   (i) A purely administrative change to technical specifications for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature.

   The proposed changes in part 2 match the guidance of 48 FR 14870 by the example:

   (iv) A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculation model or design method.

The proposed changes in part 6 match the guidance of 48 FR 14870 by the example:

(ii) A change that constitutes an additional limitation restriction or control not presently included in the technical specifications: for example, a more stringent surveillance requirement.

E. SAFETY EVALUATION FOR THE AMENDMENT REQUEST

The proposed Technical Specification amendment will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR. The anticipated operational occurrences (AOOs) and accidents that have the potential for being impacted by the proposed changes are steam line break, CEA withdrawal, CEA ejection, inadvertent deboration and startup of an inactive reactor coolant pump (RCP). All the impacted AOOs and accidents have been reevaluated to determine the resulting consequences due to the proposed changes. The results of these evaluations show that the consequences are still within the appropriate acceptance criteria discussed below.

1. The PVNGS safety analysis requires that steam line break events be evaluated considering potential for fuel damage. If the minimum DNBR during a steam line break event falls below specified limits based on acceptable correlations, fuel damage must be assumed. The results of the limiting steam line break analysis indicate that the minimum post-trip DNBR remains well above the specified safety limit.

2. The PVNGS safety analysis requires that the consequences of an uncontrolled control element assembly (CEA) withdrawal from a subcritical or low power startup condition be evaluated on the basis that they are acceptable if the minimum DNBR remains above specified limits based on acceptable correlations. The reevaluation of the limiting CEA withdrawal analysis indicates that the minimum DNBR will remain above the plant specific safety limit of 1.231.

3. The PVNGS safety analysis requires that for a startup of an inactive RCP, fuel clad integrity should be maintained by ensuring that specified acceptable fuel design limits are not exceeded. The results of a limiting startup of an inactive RCP indicate that the reactor remains subcritical and the specified acceptable fuel design limits are not exceeded, thus maintaining fuel clad integrity.

4. The PVNGS safety analysis requires that for a CEA ejection, the reactivity excursion should not result in a radially averaged enthalpy greater than 280 cal/gm at any axial location in any fuel

rod. Reevaluation of the limiting CEA ejection accident concurrent with the introduction of the K_{N-1} requirement ensures that the safety analysis acceptance criterion will be met.

The proposed change to lower the setpoint of the high logarithmic power trip will provide the trip function earlier than the previous setpoint. This trip provides protection in the event of an inadvertent control element assembly (CEA) bank withdrawal from MODES 2 and 3 initial conditions with four reactor coolant pumps (RCPs) operating. The proposed change to lower the value of power below which the CPC trip can be bypassed and above which the manual bypass is automatically removed also provides added protection for an inadvertent CEA bank withdrawal postulated to occur in MODES 3, 4, or 5 with less than four reactor coolant pumps operating. If the reactor coolant pressure or temperature is outside the CPC wide range trip limits, a continuous reactor trip signal will be generated by all four CPC channels and an immediate reactor trip will terminate an inadvertent CEA bank withdrawal event before significant power is generated. The MODE 2 and 3, with four RCPs operating, and the MODE 3, 4, and 5, with less than four RCPs operating events have been determined to be less limiting than the CEA bank withdrawal event presented in the Final Safety Analysis Report. Also, the proposed changes do not alter how the CPCs respond to design basis events.

Therefore, operation of the facility in accordance with the proposed changes will not involve a significant increase in probability or consequences of an accident previously evaluated.

The proposed Technical Specification amendment will not create the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR. There is no change in the plant hardware or analysis method as a result of the proposed changes. Although some of the proposed changes will result in modification to the operating procedures and plant operation in the shutdown modes, operation of the facility in accordance with the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Operation of the facility in accordance with the proposed changes may reduce in some way a safety margin, but where the results of the change are clearly within the acceptance criteria.

The PVNGS safety analysis requires that for an inadvertent boron dilution, a minimum time interval of 15 minutes for MODES 1 through 5 and 30 minutes for MODE 6 be available from the time an alarm makes the operator aware of unplanned boron dilution before a loss of shutdown margin occurs. The time to a complete loss of shutdown margin for the limiting inadvertent boron dilution is now 50 minutes compared to 95 minutes for the previously analyzed incident. However, an alarm will alert the operator of an unplanned boron dilution at least 15 minutes prior to a complete loss of shutdown margin.

Although a numerically smaller value of SHUTDOWN MARGIN may appear to result in a reduction in the safety margin, operation of the facility in accordance with the proposed changes does not involve a significant reduction in a margin of safety as an alarm is available in time to satisfy the safety analysis criterion.

The lower logarithmic power level trip setpoint and automatic removal of the CPC manual bypass at a lower power level result in an earlier reactor protective system actuation for the postulated transients, which involves no significant reduction in the margin of safety.

F. ENVIRONMENTAL IMPACT CONSIDERATION DETERMINATION

The proposed change request does not involve an unreviewed environmental question because operation of PVNGS Units 1 and 2, in accordance with this change, would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board, Supplements to the FES, Environmental Impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or

2. Result in a significant change in effluents or power levels; or

3. Result in matters not previously reviewed in the licensing basis for PVNGS which may have a significant environmental impact.

Supporting Analyses for the Technical Specification Amendment Request:

1. STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT - MODE 3 OPERATION: $T_C < 500°F$

1.1 Identification of Event and Causes

Refer to CESSAR Section 15.1.4, except that steam line break events during Mode 3 operation for reactor cold leg temperature (T_{cold}) less than 500°F are analyzed to demonstrate the adequacy of the shutdown margin as specified by Technical Specifications 3.1.1.1 and 3.1.1.2, to prevent degradation in fuel performance as a result of post trip return to power. The results show that the shutdown margin is sufficiently large to prevent a post trip return to power. The steam line breaks presented are:

A. A large steam line break inside containment during Mode 3 operation with concurrent loss of offsite power in combination with a single failure and technical specification shutdown margin.

B. A large steam line break inside containment during Mode 3 operation with offsite power available in combination with a single failure and technical specification shutdown margin.

For $T_{cold} > 500°F$, the shutdown margin specified by Technical Specification 3.1.1.2 is 6% $\Delta\rho$. Below 500°F, the shutdown margin decreases linearly with temperature. Here, cold leg temperatures below 500°F are considered. Hot zero power steam line breaks above 500°F using technical specification shutdown margin are found in PVNGS FSAR Section 15.1.5. The requirements of Technical Specification 3.1.1.1 are less limiting for steam line breaks than are those of Technical Specification 3.1.1.2.

The largest possible steam line break size is the double ended rupture of a steam line upstream of the main steam isolation valve (MSIV). In the PVNGS design, an integral flow restrictor exists in each steam generator outlet nozzle. The largest effective steam blowdown area for each steam line, which is limited by the flow restrictor throat area, is 1.28 square feet.

These two cases are analyzed for end of equilibrium core, self-generated plutonium recycle (SGR) conditions. PVNGS specific minimum safety injection flow rates, feedwater isolation valve closure time, and steam generator differential pressure (Δ P) isolation (lockout) setpoint are employed.

1.2 Sequence of Events and System Operation

Steam line breaks are characterized as cooldown events due to the increased steam flow rate, which causes excessive energy removal from the steam generators and the reactor coolant system (RCS). This results in a decrease in reactor coolant temperatures and in RCS and steam generator pressures. The cooldown causes an increase in core reactivity due to the negative moderator and Doppler reactivity coefficients.

Mode 3 steam line breaks are initiated from a subcritical reactivity condition. Detection of the cooldown is accomplished by the pressurizer and steam generator low pressure alarms, by the high reactor power alarm and by the low steam generator water level alarm. Reactor trip is provided by one of two available reactor trip signals. These are the low steam generator pressure and the high logarithmic power level trips.

For a steam line break that occurs with a concurrent loss of offsite power, termination of feedwater to both steam generators and coastdown of the reactor coolant pumps are assumed to be initiated simultaneously. In general, the depressurization of the affected steam generator results in actuation of a main steam isolation signal (MSIS). This closes the MSIVs, isolating the unaffected steam generator from blowdown, and closes the main feedwater isolation valves (MFIV), terminating main feedwater flow to both steam generators. After the reduction of steam flow that occurs following MSIV closure, the level in the intact steam generator falls below the auxiliary feedwater actuation signal (AFAS) setpoint. The resulting AFAS causes auxiliary feedwater (AFW) flow to be initiated to both steam generators. If the differential pressure between the two steam generators exceeds the setpoint, the AFW logic isolates flow to the affected steam generator and

diverts the flow from both AFW pumps to the intact steam generator. The pressurizer pressure may decrease to the point where a safety injection actuation signal (SIAS) is initiated. The isolation of the unaffected steam generator and subsequent emptying of the affected steam generator terminate the cooldown. The introduction of safety injection boron upon SIAS causes core reactivity to decrease. The operator, via the appropriate emergency procedures, may initiate plant cooldown by manual control of the atmospheric steam dump valves, or, in the event that offsite power is available, by using the unaffected steam generator and the turbine bypass valves, any time after the affected steam generator empties. The analyses presented herein conservatively assume operator action is delayed until 30 minutes after event initiation. The plant is then cooled to 350°F and 400 psia, at which point shutdown cooling is initiated.

A parametric study of single failures (see Appendix 15C of CESSAR) that would have an adverse impact on the SLB event has determined that the failure of one of the high pressure safety injection (HPSI) pumps to start following SIAS has the most adverse effect for those cases that result in generation of SIAS. For the two cases presented here, there is no SIAS actuation for the duration of the transient (500 seconds). For these events the most adverse effect is caused by the failure of a MSIV on one of the steam lines from the intact generator to close following MSIS. Consequently, for these cases steam is assumed to continue to be released from the intact steam generator at 1.5% of the design steam rate. This open flow path is represented by an effective flow area for steam blowdown from the intact steam generator of 0.034 square feet.

1.3 Analysis of Effects and Consequences

A. Mathematical Models

The mathematical models and data transfer between codes used in the SLB analysis are presented in PVNGS FSAR Appendix 15.C.

B. Input Parameters and Initial Conditions

The initial conditions assumed in the analysis of the NSSS response to Cases 1 and 2 are presented in Table 1-1. The initial K_{eff} of 0.99 is the highest value allowed by technical specifications for Mode 3 and leaves the least pretrip margin to criticality. There is no effect on the post trip margin to criticality. Above core inlet temperatures of 500°F, the shutdown margin is 6% $\Delta\rho$. Below 500°F the required shutdown margin decreases linearly with temperature. The initial core inlet temperature of 450°F was selected to demonstrate the adequacy of the shutdown margin in the temperature range where its magnitude is decreasing. This is a representative cold leg temperature. Analysis at other initial cold leg temperatures below 500°F will produce results and parameter trends similar to those presented here. Initially two reactor coolant pumps are assumed to be operating, as allowed in Mode 3. The initial pressurizer pressure of 830 psia falls within the range of normal Mode 3 operating procedures. The SIAS setpoint is set at 430 psia, 400 psi below the initial pressurizer pressure, the maximum offset allowed by technical specifications. This and the high initial pressurizer water volume have the effect of delaying SIAS actuation since SIAS generally occurs after the pressurizer empties. The technical specification shutdown margin at 450°F is 5.1% $\Delta\rho$. Since the reactor is 1% subcritical initially, a CEA worth at trip of 4.1% $\Delta\rho$ is assumed. The moderator and Doppler reactivity coefficients corresponding to the end of equilibrium cycle, self-generated plutonium recycle (SGR) are employed. For the purpose of conservatism the moderator reactivity coefficients correspond to the condition of no initial boron in the core.

C. Results

Case 1: Large Steam Line Break During Mode 3 Operation with Concurrent Loss of Offsite Power (SLBM3LOP)

The dynamic behavior of the salient NSSS parameters following the SLBM3LOP is presented in Figures 1-1 through 1-16. Table 1-2 summarizes the major events, times, and results for this transient.

Concurrent with the steam line break, a loss of offsite power occurs. At this time an actuation signal for the emergency diesel generators is initiated. Also at this time, the CEDM coils are assumed to lose power and, after a 0.34 second coil decay delay, the CEAs begin to drop into the core. At 21.3 seconds the steam generator pressure falls below the main steam isolation signal (MSIS) setpoint of 223 psia. This results in the generation of MSIS at 22.3 seconds, which initiates closure of the MSIVs and MFIVs. The MSIVs close by 26.9 seconds. The MFIVs close by 31.9 seconds.

During the first 500 seconds of the transient the pressurizer has not yet emptied and pressurizer pressure remains above the SIAS setpoint of 430 psia. Hence no safety injection flow and no boron reaches the RCS during this time.

AFAS is assumed to be actuated soon after the MSIVs close. Auxiliary feedwater is assumed to enter the steam generators after the level falls below the 80% high level setpoint; i.e., at 118 seconds. The pressure difference between the two steam generators remains below the analysis setpoint of 325 psid during the transient. Hence there is no automatic isolation of auxiliary feedwater to the affected steam generator.

At 500 seconds the transient reactivity is -2.1%, which indicates there is still a significant margin to recriticality. This margin will continue to decrease as the affected steam generator continues to blow down and the RCS continues to cool. After the pressurizer empties the RCS pressure is expected to fall more rapidly resulting in SIAS and subsequent inflow of boron into the RCS. Alternately SIAS may be manually actuated by the operator. In either case, after the inflow of boron into the RCS, the margin to recriticality is expected to increase.

Eventually, the affected steam generator is expected to blow down to atmospheric pressure. This would terminate further RCS cooldown. Even assuming the limiting case, where the affected steam generator has depressurized to atmospheric pressure and no safety injection boron has reached the RCS, the core will remain subcritical with a margin to criticality of no less than -0.4% $\Delta\rho$.

The discontinuity seen in some of the parameter plots at about 470 seconds (e.g., Figures 1-6 and 1-8) is due to safety injection tank (SIT) flow into the RCS for a short period of time. A SIT injection gas cover pressure of 608 psia was used in the analysis. The effect of this is small since no credit was taken for the SIT boron in the analysis.

The minimum DNBR remains above 10 during this transient. At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown by the manual control of the atmospheric dump valves. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

Case 2: Large Steam Line Break During Mode 3 Operation with Offsite Power Available (SLBM3)

The dynamic behavior of the salient NSSS parameters following the SLBM3 is presented in Figures 1-17 through 1-32. Table 1-3 summarizes the major event, times, and results for this transient.

At 24.3 seconds after the initiation of the steam line break, the steam generator pressure drops below the low steam generator pressure trip and MSIS setpoint of 223 psia. The reactor trip breakers open at 25.45 seconds. After a 0.34 second coil delay, the CEAs begin to drop into the core at 25.8 seconds. The MSIS initiates closure of the MSIVs and MFIVs. The MSIVs close by 29.9 seconds. The MFIVs close by 34.9 seconds.

During the first 500 seconds of the transient, as in Case 1, the pressurizer has not yet emptied and pressurizer pressure remains above the SIAS setpoint of 430 psia. Hence no safety injection flow and no safety injection boron reaches the RCS during this time.

AFAS is assumed to be actuated soon after the MSIVs close. Auxiliary feedwater is assumed to enter the steam generators after the level falls below the 80% high level setpoint, i.e., at 115 seconds. The pressure difference between the two steam generators stays below the analysis setpoint of 325 psid during the transient. Hence there is no automatic isolation of auxiliary feedwater to the affected steam generator.

At 500 seconds the transient reactivity is -1.9%, which indicates there is still a significant margin to criticality. This margin will continue to decrease as the affected steam generator continues to blow down and the RCS continues to cool. After the pressurizer empties the RCS pressure is expected to fall more rapidly resulting in SIAS and subsequent inflow of boron into the RCS. Alternately SIAS may be manually actuated by the operator. In either case, after the inflow of boron into the RCS, the margin to recriticality is expected to increase. Eventually, the affected steam generator is expected to blow down to atmospheric pressure (T=212°F). This would terminate further RCS cooldown. Even assuming the limiting case, where the affected steam generator has depressurized to atmospheric pressure and no safety injection boron has reached the RCS, the core will remain subcritical with a margin to criticality of no less than -0.4% $\Delta\rho$.

The discontinuity seen in some of the parameter plots at about 390 and 450 seconds (e.g., Figures 1-22 and 1-24) is due to safety injection tank (SIT) flow into the RCS for a short period of time. A SIT injection gas cover pressure of 608 psia was used in the analysis. The effect of this is small since no credit was taken for the SIT boron in the analysis.

The minimum DNBR remains above 10 during the transient. At a maximum of 30 minutes, the operator, via the appropriate emergency procedure, initiates plant cooldown. Shutdown cooling is initiated when the RCS reaches shutdown cooling entry conditions.

1.4 Conclusion

For the large steam line break during Mode 3 operation for reactor cold leg temperatures less than 500°F with or without a loss of offsite power, and in combination with a single failure the shutdown margin is sufficient to prevent a post trip return to power.

TABLE 1-1

ASSUMPTIONS AND INITIAL CONDITIONS FOR LARGE STEAM LINE BREAKS DURING MODE 3 OPERATION WITH AND WITHOUT CONCURRENT LOSS OF OFFSITE POWER (SLBM3LOP & SLBM3)

Parameters	Assumed Value
Initial Reactivity	0.99
Initial Core Inlet Coolant Temperature, F	450
Initial Core Mass Flow Rate, 10^6 lbm/hr (2 RCPs)	91.1
Initial Pressurizer Pressure, psia	830
Initial Pressurizer Water Volume, ft^3	1100
Doppler Coefficient Multiplier	1.15
Moderator Coefficient Multiplier	1.10
Axial Shape Index	+.3
CEA Worth at Trip, 10^{-2} $\Delta\rho$	-4.1
Initial Steam Generator Inventory, lbm	311,000
Core Burnup	End of Cycle
Blowdown Fluid	Saturated Steam
Blowdown Area for Each Steam Line, ft^2	1.283

TABLE 1-2

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING MODE 3 OPERATION WITH CONCURRENT LOSS OF OFFSITE POWER (SLBM3LOP)

Time (Sec)	Event	Setpoint or Value
0.0	Steam line break and loss of offsite power occur. Holding coils lose power.	--
21.3	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint, psia	230
22.3	MSIS generated	--
26.9	MSIVs completely closed	--
31.9	MFIVs completely closed	--
500	Transient reactivity, $10^{-2}\ \Delta\rho$	-2.1
>500	Pressurizer empties	--
>500	Pressurizer pressure reaches safety injection actuation signal (SIAS) analysis setpoint, psia	430
>500	SIAS generated	--
>500	Voids begin to form in reactor vessel upper head	--
>500	Safety injection flow begins	--
>500	Safety injection boron begins to reach reactor core	--
1800	Operator initiates cooldown	--

TABLE 1-3

SEQUENCE OF EVENTS FOR A LARGE STEAM LINE BREAK DURING MODE 3 OPERATION WITH OFFSITE POWER AVAILABLE (SLBM3)

Time (Sec)	Event	Setpoint or Value
0.0	Steam line break occurs	--
24.3	Steam generator pressure reaches main steam isolation signal (MSIS) analysis setpoint and low steam generator pressure trip setpoint, psia	230
25.3	Low steam generator pressure trip signal and MSIS generated	--
25.45	Reactor trip breakers open	--
29.9	MSIVs completely closed	--
34.9	MFIVs completely closed	--
500	Transient reactivity, $10^{-2}\Delta\rho$	-1.9
> 500	Pressurizer empties	--
> 500	Pressurizer pressure reaches safety injection actuation signal (SIAS) analysis setpoint, psia	430
>500	SIAS generated	--
>500	Voids begin to form in reactor vessel upper head	--
>500	Safety injection flow begins	--
>500	Safety injection boron begins to reach reactor core	--
1800	Operator initiates cooldown	--

2. STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT - MODE 4 OPERATION

2.1 Identification of Event and Causes

The steam system piping failure event during Mode 4 operation is evaluated to demonstrate the adequacy of the shutdown margin, as specified by technical specifications, to prevent degradation in fuel performance as a result of a post trip return to power.

The double ended rupture of a steam line upstream of the main steam isolation valve (MSIV) is considered. Reactor physics parameters (e.g., moderator and Doppler reactivity coefficients) for the end of equilibrium core, self-generated plutonium recycle (SGR) are assumed. The moderator reactivity coefficient corresponding to the condition of zero initial boron in the core is conservatively assumed. The more likely condition of boron existing in the RCS would cause the moderator reactivity to be less negative or even positive, thus reducing the reactivity increase during a steam line break.

Extrapolation of the results from Section 1 show that the shutdown margin in Mode 4 is sufficiently large to prevent a post trip return to power.

2.2 Sequence of Events and Systems Operation

Steam line breaks result in excessive cooldown of the reactor coolant which causes an increase in core reactivity due to the negative moderator and Doppler reactivity coefficients.

Mode 4 steam line breaks are initiated from a subcritical reactivity condition. The initial cold leg temperature may range from 210°F to 350°F. The sequence of events during a steam line break event will differ depending on the initial cold leg temperature and pressurizer pressure.

Detection of the cooldown is accomplished by the pressurizer and steam generator low pressure alarms and by the low steam generator water level alarm.

Reactor trip is provided by one of two available reactor trip signals. These are the low steam generator pressure and the high logarithmic power level trips. The depressurization of the affected steam generator may result in actuation of a main steam isolation signal (MSIS). This closes the MSIVs, isolating the unaffected steam generator from blowdown. If the level in either steam generator falls sufficiently low, an auxiliary feedwater actuation signal (AFAS) will occur. The pressurizer pressure may decrease to the point where a safety injection actuation signal (SIAS) is initiated. The introduction of safety injection boron upon SIAS causes core reactivity to decrease. Eventually the affected steam generator will depressurize to atmospheric pressure which will terminate the RCS cooldown.

2.3 Analysis of Effects and Consequences

The magnitude of the core reactivity increase during a steam line break during Mode 4 depends on the initial cold leg temperature. A steam line break initiated from 210°F or lower will result in negligible increases in core reactivity. A steam line break initiated from 350°F has the potential for moderate amounts of reactivity increase. Eventually, the affected steam generator will blow down to atmospheric pressure terminating further reactivity increase. If a SIAS is initiated, either automatically or by manual operator action, the core reactivity will decrease subsequent to the inflow of boron into the RCS. Even assuming the limiting case where the affected steam generator has depressurized to atmospheric pressure and no safety injection boron has reached the RCS, the technical specification shutdown margin will prevent a return to core criticality.

2.4 Conclusion

For the large steam line break during Mode 4 operation, the shutdown margin as specified by the technical specifications is sufficient to prevent a return to core criticality.

3. UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER CONDITION

Refer to CESSAR Section 15.4.1.

4. UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL FROM MODES 2 AND 3 SUBCRITICAL WITH 4 REACTOR COOLANT PUMPS OPERATING

4.1 Identification of Event and Causes

An uncontrolled sequential withdrawal of CEAs is assumed to occur as a result of a single failure in the Control Element Drive Mechanism (CEDM), Control Element Drive Mechanism Control System (CEDMCS), reactor regulating system, or as a result of operator error. This event is analyzed to justify reduced Technical Specification Shutdown Margin requirements in subcritical modes to below that required for Modes 1 and 2.

4.2 Sequence of Events and Systems Operation

The withdrawal of CEAs from subcritical conditions adds reactivity to the reactor core, causing both the core power level and the core heat flux to increase followed by corresponding increases in reactor coolant temperatures and reactor coolant system (RCS) pressure. The withdrawal of CEAs also produces a time dependent redistribution of core power. These transient variations in core thermal parameters may result in an approach to the specified acceptable fuel design limits (SAFDL), thereby requiring the protective action of the reactor protection system (RPS).

Reactivity Control

The reactivity insertion rate accompanying the uncontrolled CEA withdrawal is dependent primarily upon the CEA withdrawal rate and the CEA worth since, at subcritical conditions, the normal reactor feedback mechanisms do not occur until power generation in the core is large enough to cause changes in the fuel and moderator temperatures. The reactivity insertion rate determines the rate of approach to the fuel design limits. The uncontrolled CEA withdrawal

transient from subcritical conditions is terminated by a high logarithmic power level trip or a CPC range trip.

Reactor Heat Removal

Following the cooldown phase in which the steam bypass control system is used, the shutdown cooling system (SCS) is manually actuated when the RCS temperature and pressure have been reduced to 350°F and 400 psia, respectively. This system provides sufficient cooling flow to cool the RCS to cold shutdown.

Primary System Integrity

The RCS pressure remains below the pressurizer pressure safety valve setpoint and remains less than 110% of design pressure. The pressurizer pressure control system and the pressurizer level control system are manually operated to regulate RCS pressure and coolant inventory during the cooldown phase.

Secondary System Integrity

The secondary system pressure increases following reactor trip and is limited by the steam generator safety valves. The atmospheric dump valves are used to cool the plant down to shutdown cooling entry conditions. The feedwater flowrate is in manual mode and is very low because it matches steam flow rates.

Table 4-1 gives the sequence of events for the limiting CEA withdrawal transient from subcritical conditions identified in paragraph 4.3.

4.3 Analysis of Effects and Consequences

A. Mathematical Model

The Nuclear Steam Supply System (NSSS) response to a CEA sequential withdrawal from subcritical or low power conditions was simulated using

the CESEC computer program described in PVNGS FSAR Section 15.0. The thermal margin on DNBR in the reactor core was simulated using the TORC computer program described in PVNGS FSAR Section 15.0 with the CE-1 CHF correlation described in PVNGS FSAR Chapter 4.

B. Input Parameters and Initial Conditions

The initial conditions and NSSS characteristics assumed in this analysis have been determined to be the limiting conditions from which a CEA withdrawal could be initiated from subcritical modes. The range of initial conditions which were considered is limited both by Technical Specification LCO's and by the auxiliary trip functions in the CPC's. At 10^{-4} percent power, the CPC zero power bypass is automatically removed. If initial NSSS parameters or core coolant flow, temperature, pressure etc., are beyond specified values the CPC's will cause an immediate reactor trip upon bypass removal. The 10^{-4} percent power level setpoint at which the CPC bypass is automatically removed is well below the setpoint of the high logarithmic power level trip. A trip generated at this power level would cause a decrease in fission power before the point of adding sensible heat flux is reached, thereby precluding a challenge to the SAFDLs.

Parametric studies of initial conditions which would not generate an immediate CPC trip were performed. The initial conditions which resulted in the most adverse transient are presented in Table 4-2.

All control element assemblies are initially assumed to be fully inserted and the initial K_{eff} is 0.91. The reactivity insertion rate is conservatively taken to be 3.23×10^{-4} $\Delta\rho$/second which is greater than the maximum differential worth of the highest worth CEA bank.

C. Results

The dynamic behavior of important NSSS parameters following a CEA withdrawal from subcritical conditions is presented in Figures 4-1 through 4-5.

The withdrawal of CEA's from subcritical conditions gradually reduces the amount by which the core is shut down. During this time, subcritical multiplication causes core power to increase. The reactor reaches critical at 278 seconds. Following this, core power rises at a rate which increases as the CEA's continue to withdraw.

A reactor trip on high logarithmic power is generated before core power reaches the point of adding sensible heat. Due to the rapid rate of power increase at the time of trip generation and the effect of continued CEA withdrawal until the trip breakers open, a brief power excursion occurs past the point of adding sensible heat.

The CEA's begin dropping into the core at 293.7 seconds terminating the power escalation with a hot channel minimum DNBR greater than 2.0.

4.4 Conclusions

The uncontrolled CEA withdrawal from a subcritical condition event meets general design criteria 25 and 20 as specified in SRP 15.4.1. These criteria require that the specified acceptable fuel design limits are not exceeded and that protection system action is initiated automatically. The transient terminates with a hot channel minimum DNBR greater than 1.19 and the peak fuel centerline temperature during the transient is less than 1230°F.

TABLE 4-1

SEQUENCE OF EVENTS FOR THE
SUBCRITICAL CEA WITHDRAWAL EVENT

Time (Sec)	Event	Setpoint or Value
0.0	Withdrawal of CEA's - Initiating Event	--
292.8	Core Power reaches High Logarithmic power level reactor trip analysis setpoint, percent of design power	2.7×10^{-2}
293.2	High Logarithmic Power Level Trip Signal Generated	--
293.4	Trip Breakers Open	--
293.7	Maximum Core Power, % of Design Power	63%
293.9	Maximum Core Average Heat Flux, % of Full Power Heat Flux	7.8%
294.0	Minimum DNBR	2.0

TABLE 4-2

ASSUMPTIONS AND INITIAL CONDITIONS FOR THE SUBCRITICAL CEA WITHDRAWAL ANALYSIS

Parameters	Assumed Value
Initial Fission power level, MWt	2.9×10^{-8}
Core inlet coolant temperature, °F	565.5
Core mass flowrate, 10^{-6} lb_m/h	142.1
Reactor coolant system pressure, psia	1785
One pin 3-D peaking factor, with uncertainty	9.0
Steam generator pressure, psia	1178
Moderator temperature coefficient, 10^{-4} $\Delta\rho$/°F	+0.5
Doppler coefficient multiplier	.85
CEA reactivity addition rate, 10^{-4} $\Delta\rho$/sec	3.23
CEA Worth on trip, 10^{-2} $\Delta\rho$	3.79
Steam bypass control system	Automatic

5. UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL FROM MODES 3, 4 AND 5 WITH LESS THAN 4 REACTOR COOLANT PUMPS OPERATING

5.1 Identification of Event and Causes

Refer to Section 3.

5.2 Sequence of Events and Systems Operation

This event proceeds the same as the event in Section 4, the CEAW for 4 Reactor Coolant Pumps Operating. A trip is generated by the CPCs when the zero power bypass is automatically removed at $10^{-4}\%$ power since less than four pumps are in operation. This causes the shutdown of the reactor prior to the point of adding sensible heat flux.

5.3 Analysis of Effects and Consequences

Due to the prompt CPC trip at $10^{-4}\%$ power the consequences of this event are less adverse than for the CEAW presented in Section 3.

6. STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

6.1 Identification of Event and Causes

The Startup of an Inactive Reactor Coolant Pump (SIRCP) is presented here with respect to potential loss of minimum required shutdown margin. This event is also evaluated with respect to RCS pressure and fuel performance criteria.

Administrative procedures govern the starting of RCPs and reduce the effects of RCP starts.

6.2 Sequence of Events and Systems Operation

SIRCP can either raise or lower core average coolant temperature. The average temperature can be lowered by increased heat transfer to the steam generators caused by increased core coolant flow and by colder primary system water in the steam generators being forced into the core. The core average temperature can be raised by increased heat transfer from the steam generators to the RCS as a result of increased core coolant flow and by hotter primary system water in the steam generators being forced into the core.

The SIRCP event which lowers the core average temperature (the cooldown event) combined with a negative isothermal temperature coefficient (ITC) produces a positive reactivity insertion. The SIRCP event which increases core average temperature (the heatup event), combined with a positive ITC produces an increase in RCS pressure and a positive reactivity insertion.

6.3 Analysis of Effects and Consequences

SIRCP can cause either a heatup or cooldown of the primary system depending on the primary to secondary ΔT.

SIRCP was examined in Modes 3 through 6, since plant operation with less than 4 RCPs running is only permitted in these modes.

A. Mathematical Models

The reactivity added to the core during a heatup or cooldown SIRCP event was determined using conservative isothermal temperature coefficients (ITCs) with the maximum uncertainty applied. These ITCs were used with the maximum core temperature increase or decrease to determine the maximum reactivity inserted during SIRCP. This reactivity insertion is compared to the minimum shutdown margin required by the technical specifications.

B. Input Parameters and Initial Conditions

The initial conditions considered for this event ranged from a positive to a negative temperature difference between the secondary and primary system. Primary system temperature higher than the secondary (a positive temperature difference) would result in cooling down the RCS. Secondary system temperature initially higher than the primary temperature (a negative temperature difference) would result in heating up the RCS. Cooling the RCS would increase reactivity if there is a negative ITC. Heating the RCS would increase reactivity and RCS pressure if there is a positive ITC.

To conservatively calculate the reactivity added to the core during SIRCP, the most negative or positive ITCs are used with uncertainties applied in the most conservative direction. The initial core average moderator temperature during SIRCP is assumed to be at the temperature corresponding to the most positive ITC for the heatup event, or the most negative ITC for the cooldown event.

The following assumptions are made:

1) Prior to SIRCP all reactor coolant pumps are off. Normally at least one RCP must be running (or one shutdown cooling train during shutdown cooling operation). The technical specifications allow operation without any pumps running for up to one hour. This assumption maximizes the change in temperature during SIRCP.

2) Following SIRCP the core average temperature either (1) drops to the temperature of the coldest steam generator, for the cooldown event, or (2) increases to the temperature of the hottest steam generator, for the heatup event. This conservatively bounds the maximum change in core temperature that can occur during this event, by ignoring coolant mixing that would occur in the reactor coolant system.

C. Results

The results show that the maximum temperature change during SIRCP when used with the most conservative ITCs does not result in a loss of the minimum required shutdown margin.

When the RCS is above the conditions requiring low temperature over pressure (LTOP) protection, the SIRCP event that results in a heatup of the RCS will not result in a peak pressure greater than 110% of design pressure. While the RCS is in the LTOP mode, the shutdown cooling system (SCS) relief valves will prevent violation of RCS integrity limits. (See PVNGS FSAR Section 5.2 for a general discussion of RCS integrity.)

Since shutdown margin is not lost during the event, there is no increase in heat flux and therefore no decrease in minimum DNBR.

6.4 Conclusions

The SIRCP does not result in a loss of shutdown margin. The increase in pressure during this event will not result in peak pressures above the applicable limits. There is no increase in core heat flux and therefore no decrease in minimum DNBR.

7. INADVERTENT DEBORATION

7.1 Identification of Event and Causes

The Inadvertent Deboration (ID) event is presented here with respect to time available for operator corrective action prior to the loss of minimum required shutdown margin. Fuel integrity is not challenged by this event.

The ID event may be caused by improper operator action or by a failure in the boric acid makeup flow path which reduces the flow of borated water to the charging pump suction. Either cause can produce a boron concentration of the charging flow which is below the concentration of the reactor coolant.

The ID event initiated during each of the six operational modes defined in the technical specifications was evaluated. This evaluation shows that MODE 4 (hot shutdown) results in the least time available for detection and termination of the event. This is because the shutdown margin requirement which will be specified by the technical specifications is at its minimum value in the lower temperature range of MODE 4 and the boron dilution time constant which drives the dilution rate is also small in MODE 4. This combination of a minimum shutdown margin and small time constant results in the fastest dilution rate and, therefore, yields the shortest time to a complete loss of shutdown margin.

Since boron dilution is conducted under strict procedural controls which specify limits on the rate and the magnitude of any required change in boron concentration, the probability of a sustained and erroneous dilution due to operator error is very low.

The indications and/or alarms available to alert the operators that a boron dilution event is occurring in each of the operational modes are outlined below.

1. The following control indications and corresponding pre-trip alarms are available for MODES 1 and 2: a high power or, for some set of conditions, a high pressurizer pressure trip in MODE 1 or a high logarithmic power level trip in MODE 2. Furthermore, a high T_{AVG} alarm may also occur prior to trip.

2. In MODES 3 and 4 with CEAs withdrawn, the high logarithmic power level trip and pre-trip alarm, and a high neutron flux alarm will provide an indication to alert the operator of an inadvertent boron dilution.

3. In MODES 3, 4, and 5 with CEAs fully inserted except the worst rod stuck out and in MODE 6, a high neutron flux alarm on the startup flux channels will provide indication of any boron dilution event.

4. In MODE 5 with the RCS partially drained for system maintenance, the startup flux channel alarm will provide indication of any boron dilution event. In this plant condition, administrative controls would allow operation of only one charging pump at a maximum rate of 44 gpm. Plant operating procedure will require that the power to the other two charging pumps be removed and their breakers locked out. This drained-down case is less limiting than the MODE 4 event presented below.

The operational procedure guidelines, in addition to these indications and/or alarms, will assure detection and termination of the boron dilution event before the shutdown margin is lost.

7.3 Sequence of Events and Systems Operation

The core is initially subcritical with shutdown margin at the minimum value consistent with the technical specification limit. An inadvertent deboration occurs which causes unborated water to be pumped into the RCS. The resulting decrease in RCS boron concentration adds positive reactivity to the core. Assuming dilution continues at the maximum possible rate, 50 minutes would elapse before the core becomes critical.

The success path is as follows:

Reactivity Control:

The operator is alerted to a decrease in the reactor coolant system (RCS) boron concentration either through a high neutron flux alarm on the startup flux channel, sampling, boronometer indications, or boric acid flow rate. The operator turns off the charging pump(s) and closes the letdown control valves in order to halt further dilution. Next, the operator increases the RCS boron concentration by implementing the emergency boration procedure for achieving cold shutdown boron concentration.

7.3 <u>Analysis of Effects and Consequences</u>

A. Mathematical Model

Assuming complete mixing of boron in the RCS, the rate of change of boron concentration during dilution is described by the following equation.

$$M\frac{dC}{dt} = -WC$$

Where: M = RCS mass

C = RCS boron concentration

W = Charging mass flow rate of unborated water

dC/dt is maximized by maximizing W and minimizing M. Assuming:

W = Constant, equal to the maximum possible value,

and choosing:

M = Constant, equal to the minimum value occurring during the boron dilution incident,

the solution of Equation (1) can be written

$$C(t) = C(o)e^{-t/\tau} \tag{2}$$

Where: τ = M/W = Boron dilution time constant

$C(o)$ = Initial boron concentration

The time T required to dilute to criticality is given by

$$T = \tau \ln\frac{C(o)}{C_{crit}} \tag{3}$$

Where: C_{crit} = Critical boron concentration

B. Input Parameters and Initial Conditions

It is assumed that the inadvertent deboration proceeds at the maximum possible rate. For this to occur, all charging pumps must be on, the reactor makeup water tank must be aligned with the charging pump suction, a reactor makeup water pump must be on, letdown flow must be diverted from the volume control tank, and a failure in the boric acid makeup water flow path (e.g., flow control valve FV-210Y failing in the closed position) must terminate borated water flow to the charging pump suction.

Evaluation of ID events initiated during each of the six plant operational modes (defined in the technical specifications) shows that MODE 4 (hot shutdown) results in the shortest available time for detection and termination of the event. Therefore, the initial conditions and analysis parameters are chosen for the hot shutdown operational mode to minimize the interval from initiation of dilution to the time at which criticality is reached. The following are the analysis assumptions for the ID event:

1. Complete mixing of boron within the RCS is assumed.

2. The technical specification lower limit on shutdown margin for hot shutdown is assumed. The shutdown margin as specified in the technical specifications can vary as a function of reactor coolant cold leg temperature. The minimum value of shutdown margin at the technical specification lower limit of temperature range of MODE 4, i.e., 210°F, is 1% $\Delta\rho$.

3. The cold reactor coolant system volume, excluding pressurizer and surge line, is 12,016 ft^3. A conservatively low reactor coolant mass was assumed by using the cold RCS internal volume. Assuming the coolant temperature of 350°F, the technical specification upper limit for hot shutdown, the resulting mass is 667,927 lbm.

4. All three charging pumps are assumed to be on at their maximum rate; 44 gpm per pump, for a total of 132 gpm. The corresponding mass flow rate, assuming cold liquid flow, is 18.36 lbm/sec.

5. The critical boron concentration, with all rods in except the highest reactivity worth rod stuck out, and the inverse boron worth are 752 ppm and 65 ppm/% $\Delta\rho$, respectively, including uncertainties for the hot shutdown conditions. The initial subcritical boron concentration for the hot shutdown mode is found by adding the product of the inverse boron worth and the minimum shutdown margin (i.e., one percent) to the critical boron concentration. The resulting minimum initial boron concentration in MODE 4 is 817 ppm. Thus, the change in boron concentration from 1% $\Delta\rho$ subcritical to critical is 65 ppm.

The parameters discussed above are summarized in Table 7-1.

C. Results

Using conservative parameters as described above in Equation (3), the minimum possible time interval to dilute from 1.0%$\Delta\rho$ subcritical to criticality is 50 minutes. Given the numerous indications of improper operation and the high neutron flux alarm on the startup flux channel, sufficient time is available to assure detection of a boron dilution event at least 15 minutes prior to criticality. Boron dilution will then be terminated before loss of shutdown margin by the operator actions discussed in Section 7.2.

7.4 Conclusions

The inadvertent deboration event will result in acceptable consequences. Sufficient time is available for the operator to detect and to terminate an inadvertent deboration event if it occurs. Fuel integrity is not challenged during this event.

TABLE 7-1

ASSUMPTIONS FOR THE INADVERTENT DEBORATION ANALYSIS

Parameters	Assumed Value
Cold RCS Volume (excluding pressurizer surge line), ft^3	12,016
RCS Mass (excluding pressurizer and surge line), lbm	667,927
Volumetric Charging Rate, gpm	132
Mass Charging Rate, lbm/sec	18.36
Dilution Time Constant, τ , sec	36,380
Initial Boron Concentration - C(o), ppm	817
Critical Boron Concentration - C_{crit}, ppm	752

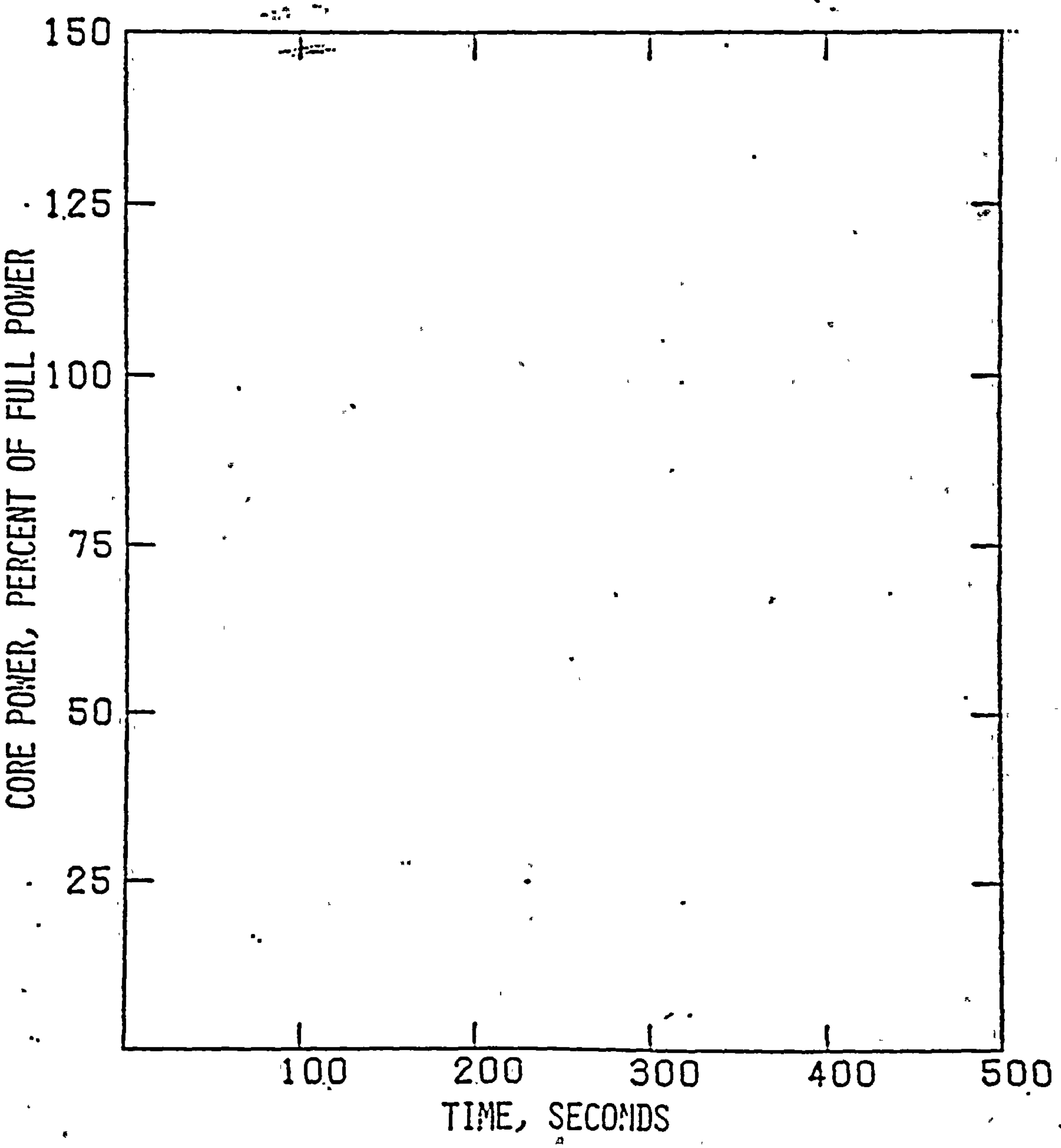
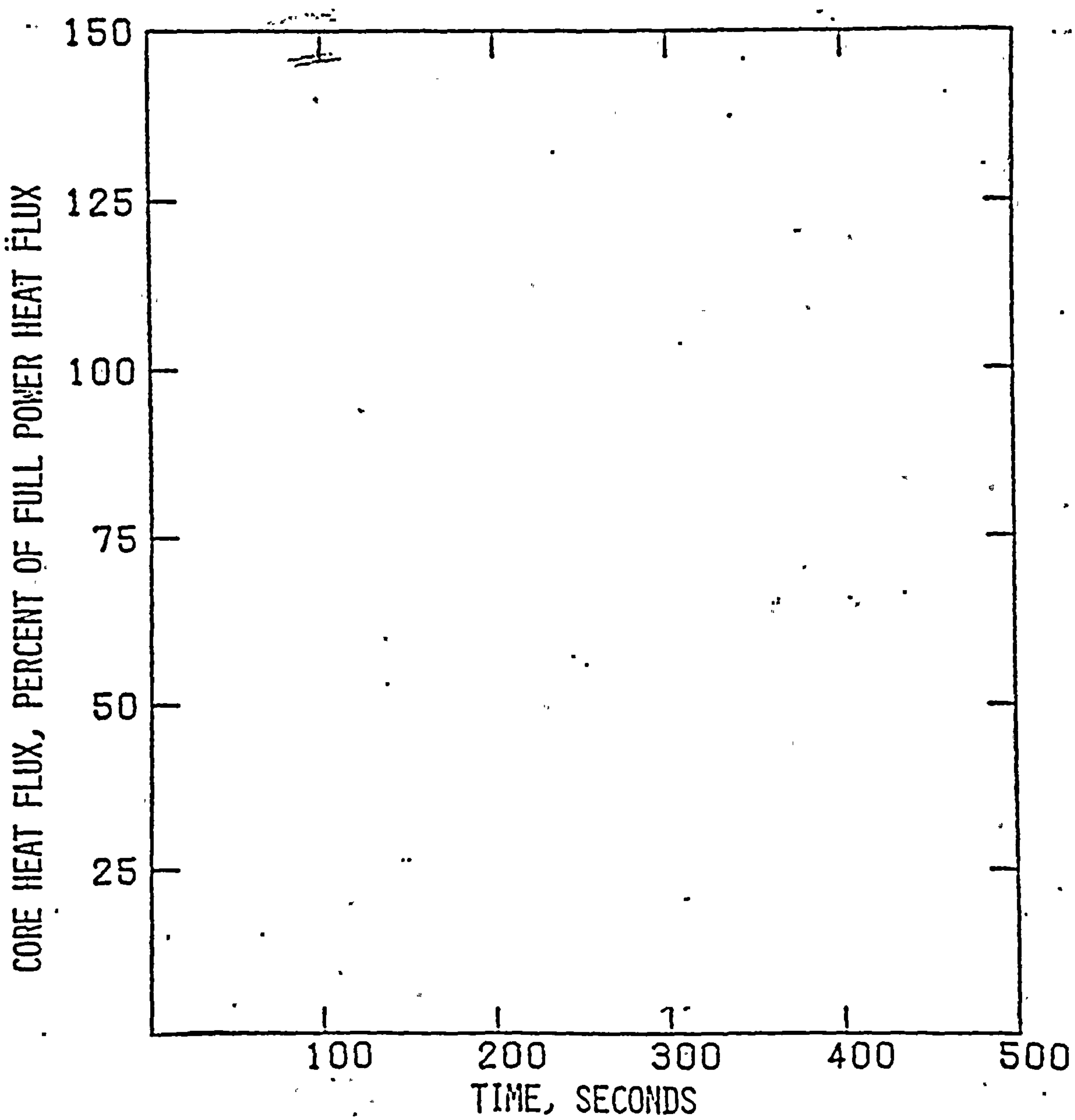
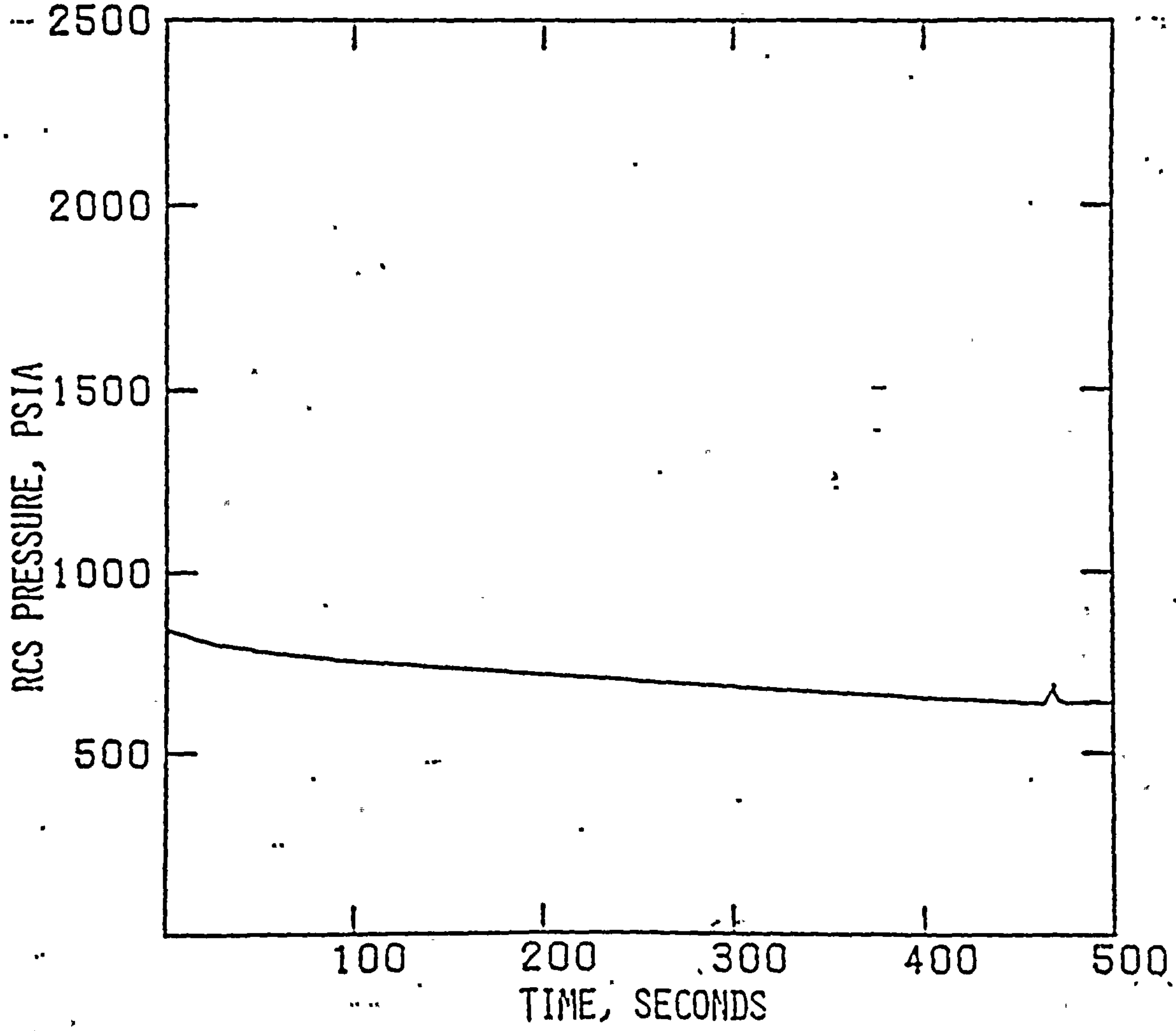

150
125
100
75
50
25
CORE POWER, PERCENT OF FULL POWER
100
200
300
400
500
TIME, SECONDS

Figure 1-1



150
125
100
75
50
25
CORE HEAT FLUX, PERCENT OF FULL POWER HEAT FLUX
100
200
300
400
500
TIME, SECONDS

Figure 1-2



2500
2000
1500
1000
500
RCS PRESSURE, PSIA
100
200
300
400
500
TIME, SECONDS

Figure 1-3

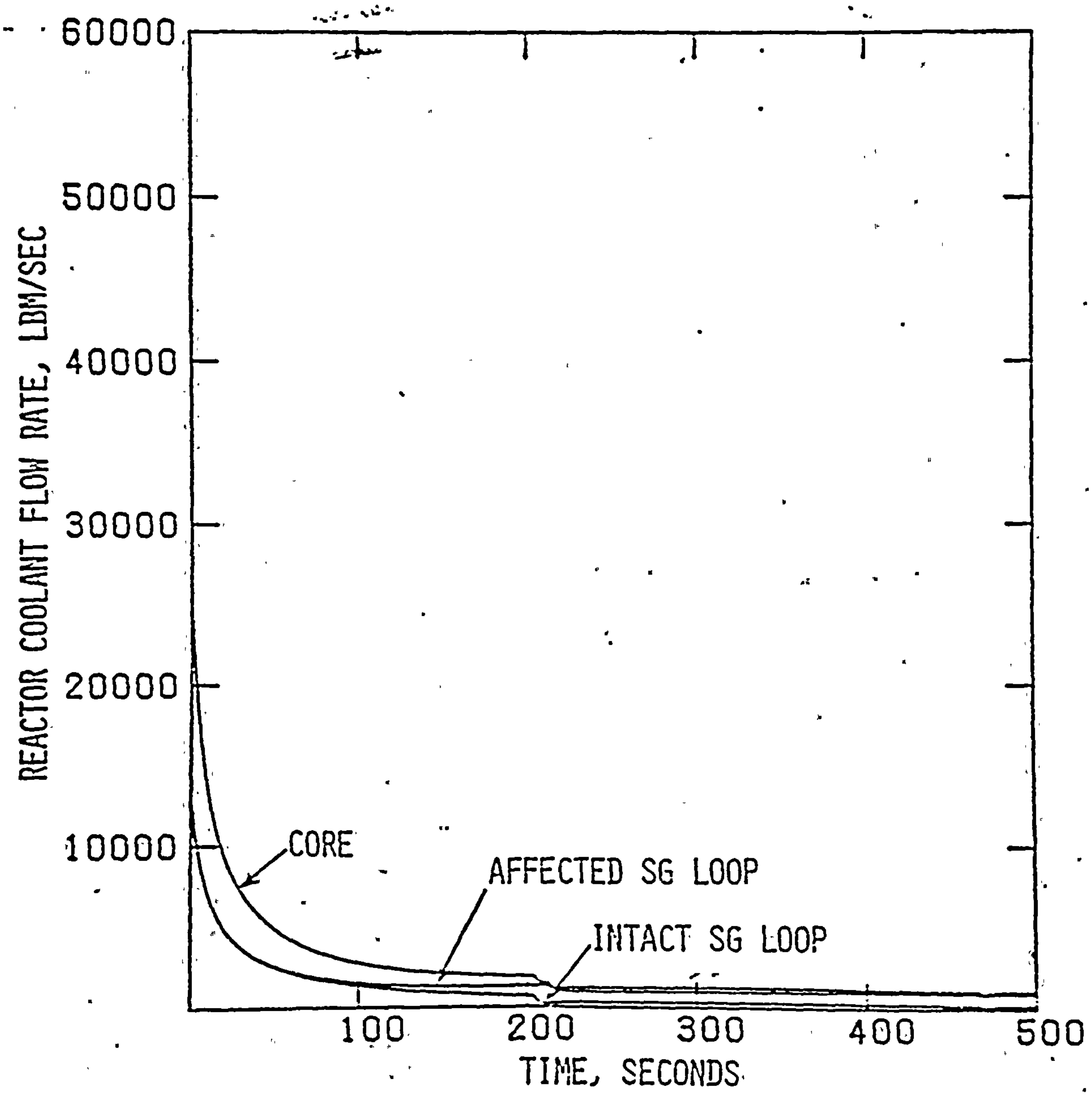

60000
50000
40000
30000
20000
10000
REACTOR COOLANT FLOW RATE, LBM/SEC
CORE
AFFECTED SG LOOP
INTACT SG LOOP
100
200
300
400
500
TIME, SECONDS

Figure 1-4

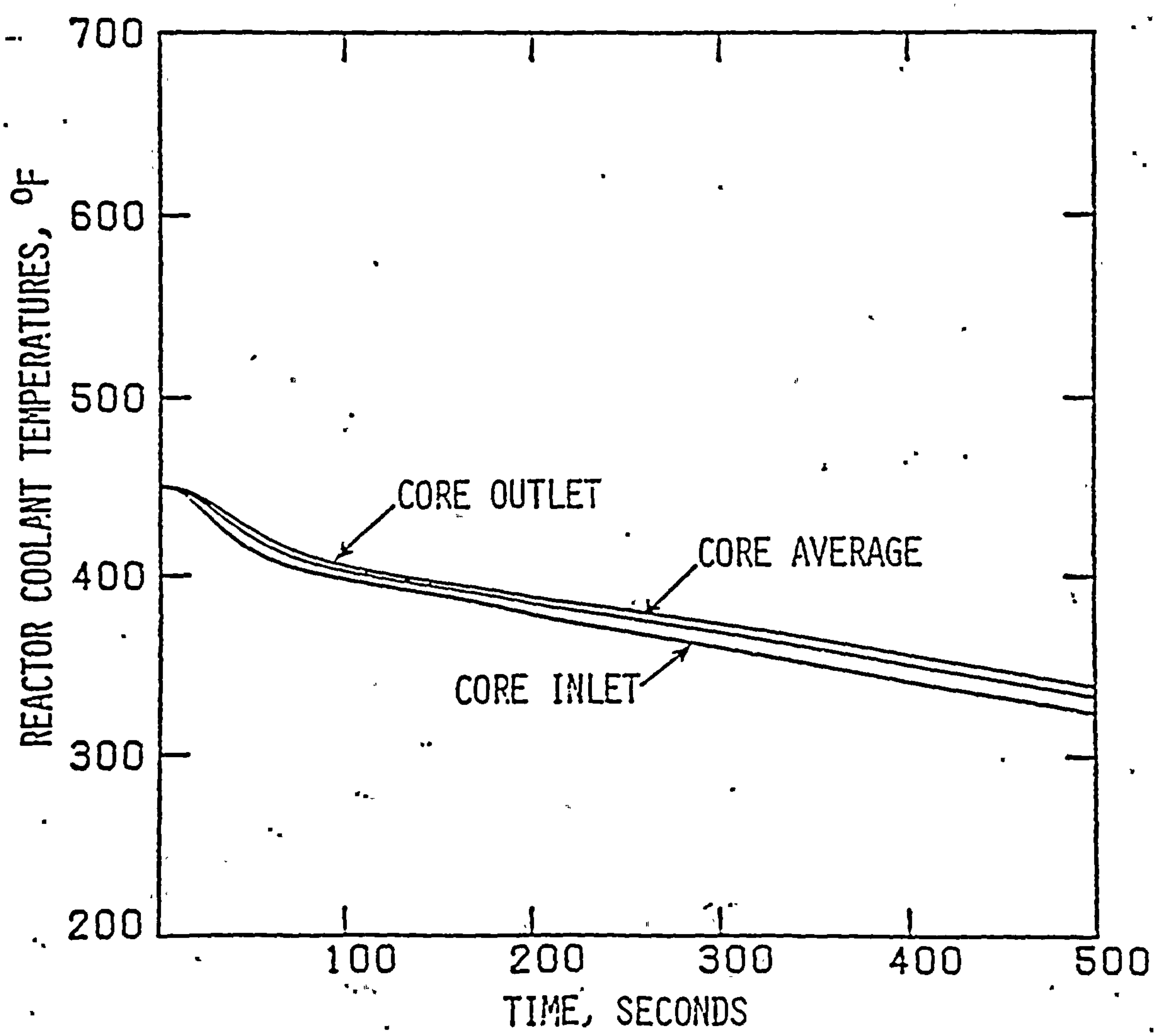
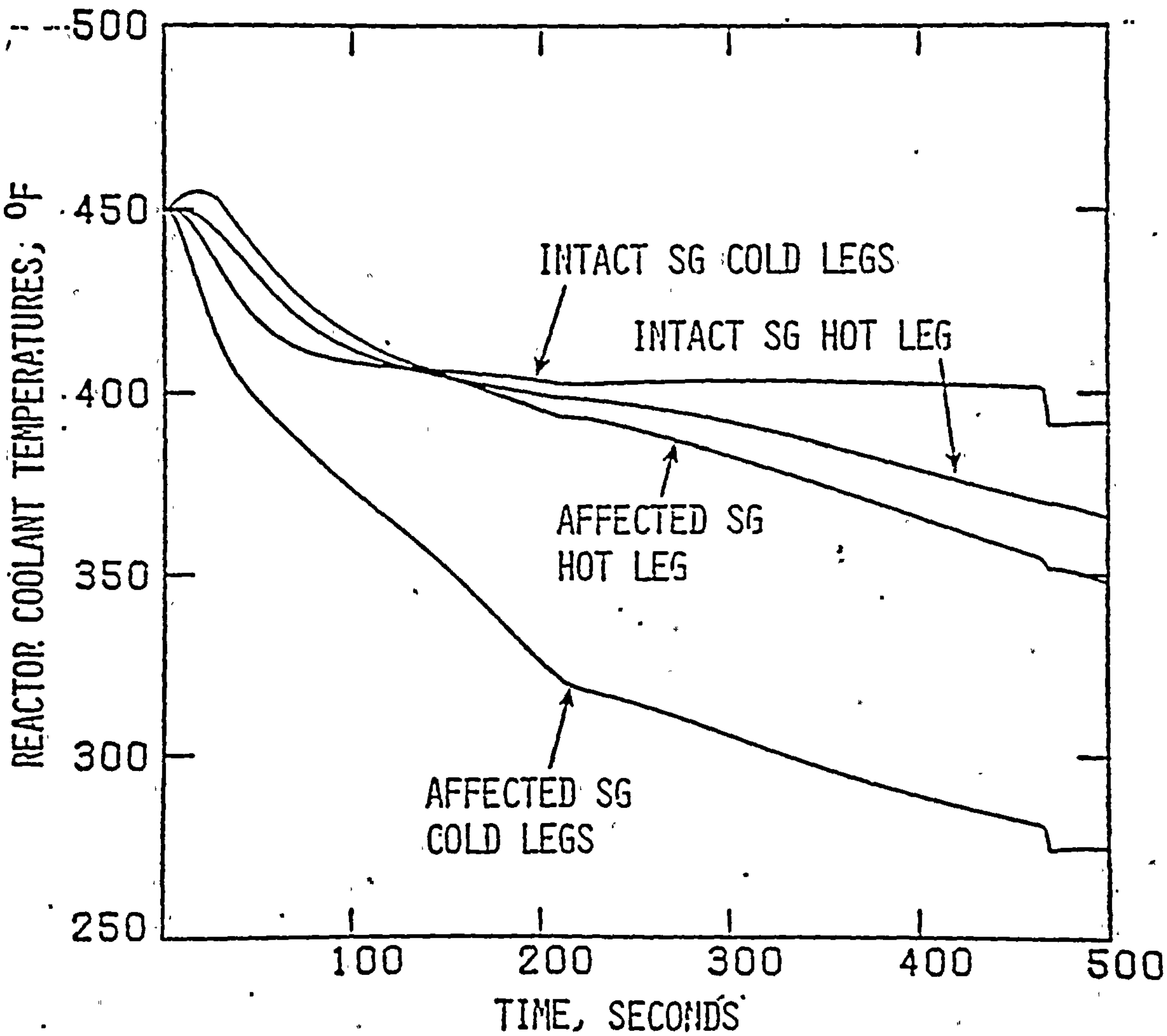
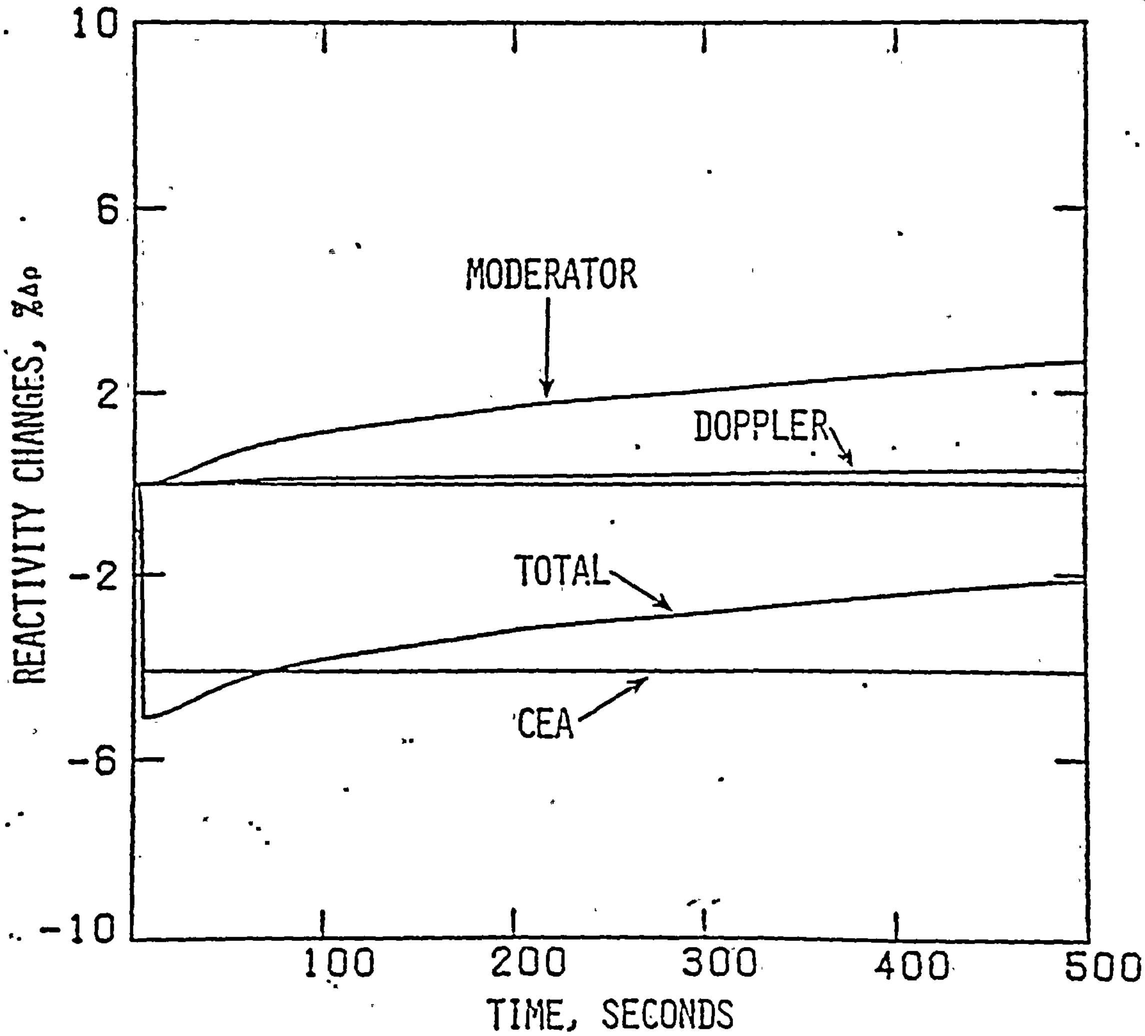

700
600
500
400
300
200
REACTOR COOLANT TEMPERATURES, ^oF
CORE OUTLET
CORE AVERAGE
CORE INLET
100
200
300
400
500
TIME, SECONDS

Figure 1-5



500
450
400
350
300
250
100
200
300
400
500
REACTOR COOLANT TEMPERATURES, °F
TIME, SECONDS
INTACT SG COLD LEGS
INTACT SG HOT LEG
AFFECTED SG
HOT LEG
AFFECTED SG
COLD LEGS

Figure 1-6



MODERATOR
DOPPLER
TOTAL
CEA
REACTIVITY CHANGES, %Δρ
TIME, SECONDS
10
6
2
-2
-6
-10
100
200
300
400
500

Figure 1-7

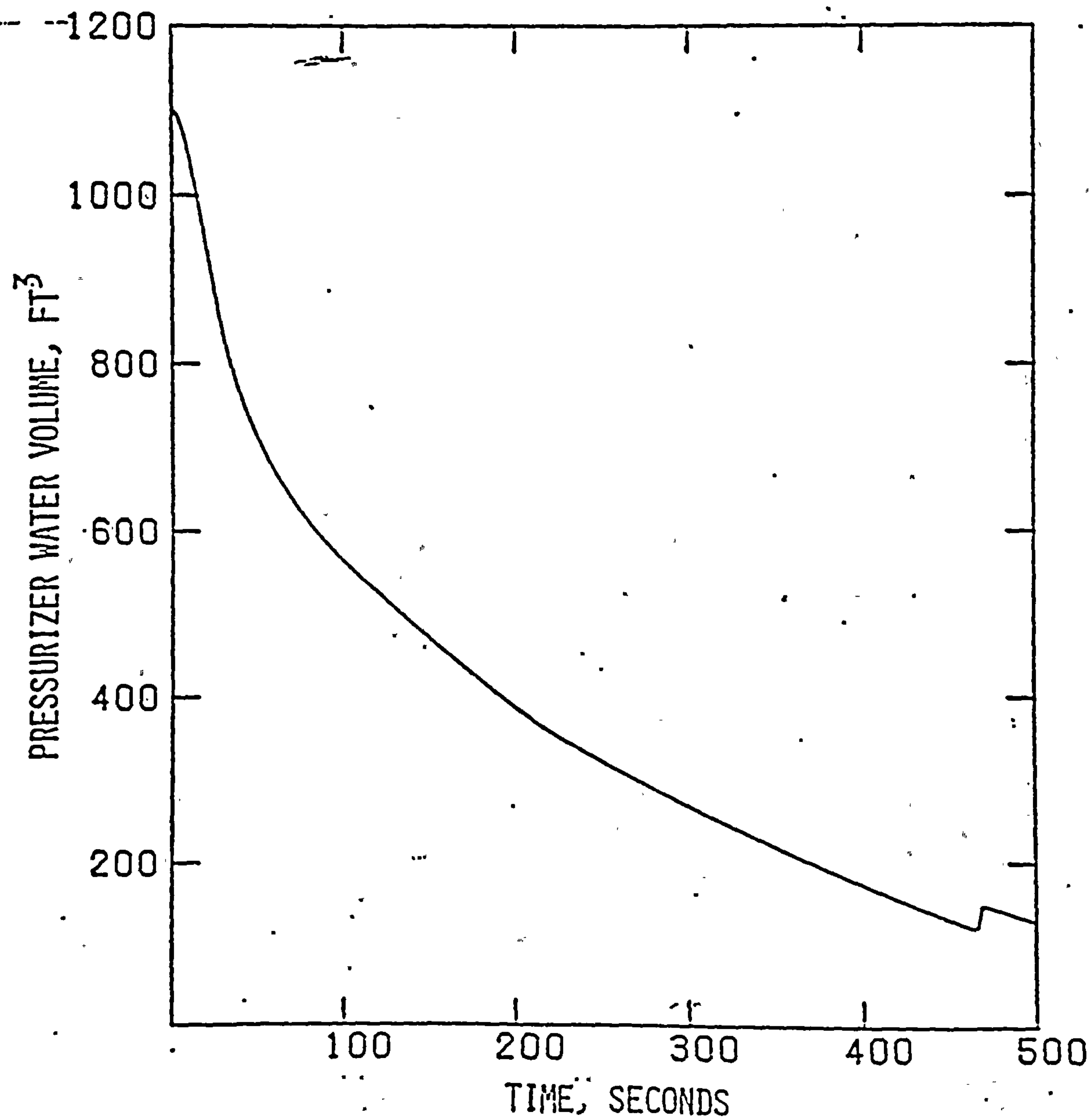

1200
1000
800
600
400
200
PRESSURIZER WATER VOLUME, FT^3
100
200
300
400
500
TIME, SECONDS

Figure 1-8

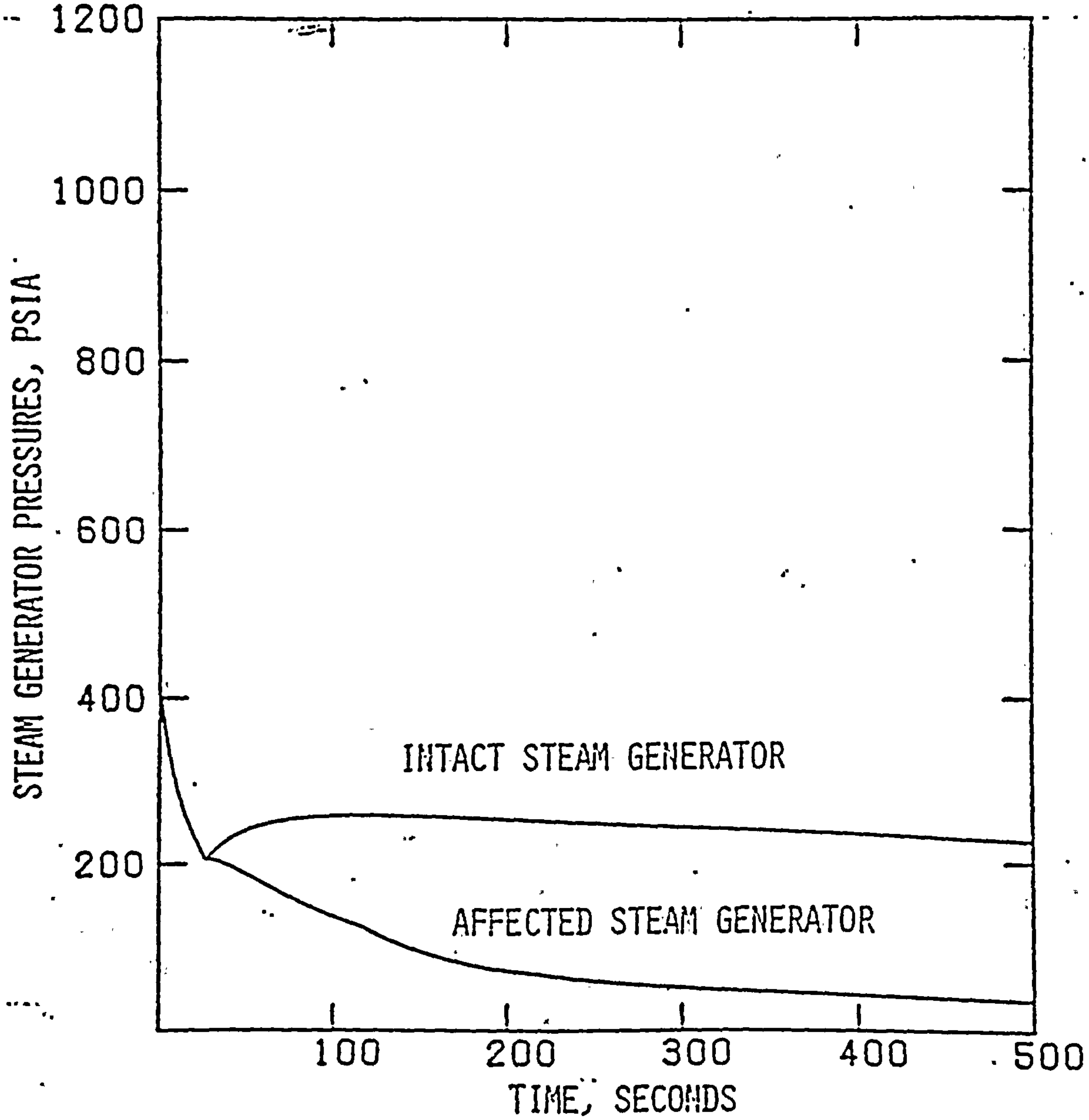
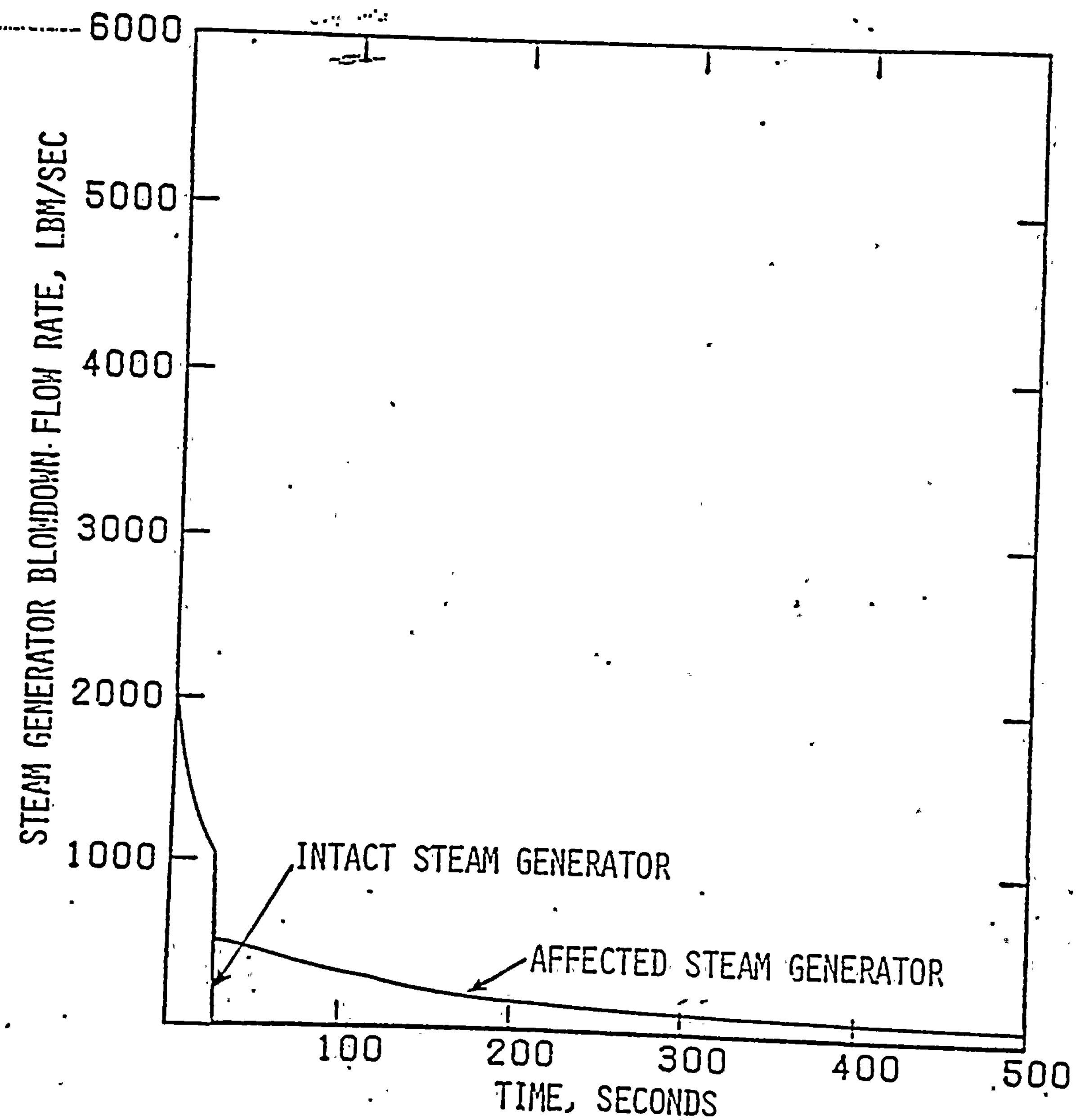
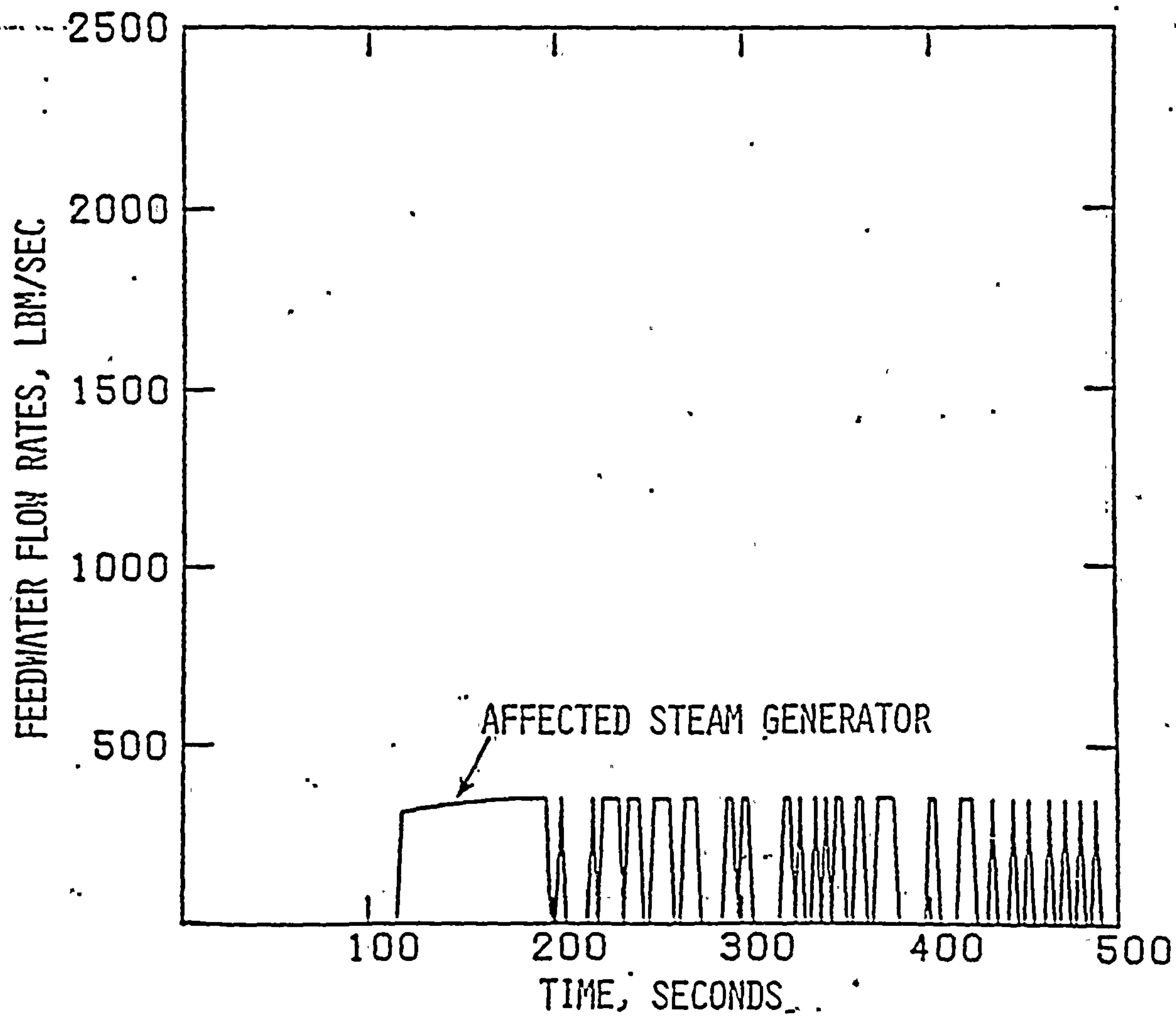

1200
1000
800
600
400
200
STEAM GENERATOR PRESSURES, PSIA
INTACT STEAM GENERATOR
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-9



6000
5000
4000
3000
2000
1000
STEAM GENERATOR BLOWDOWN FLOW RATE, LBM/SEC
INTACT STEAM GENERATOR
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-10



2500
2000
1500
1000
500
FEEDWATER FLOW RATES, LBM/SEC
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-11

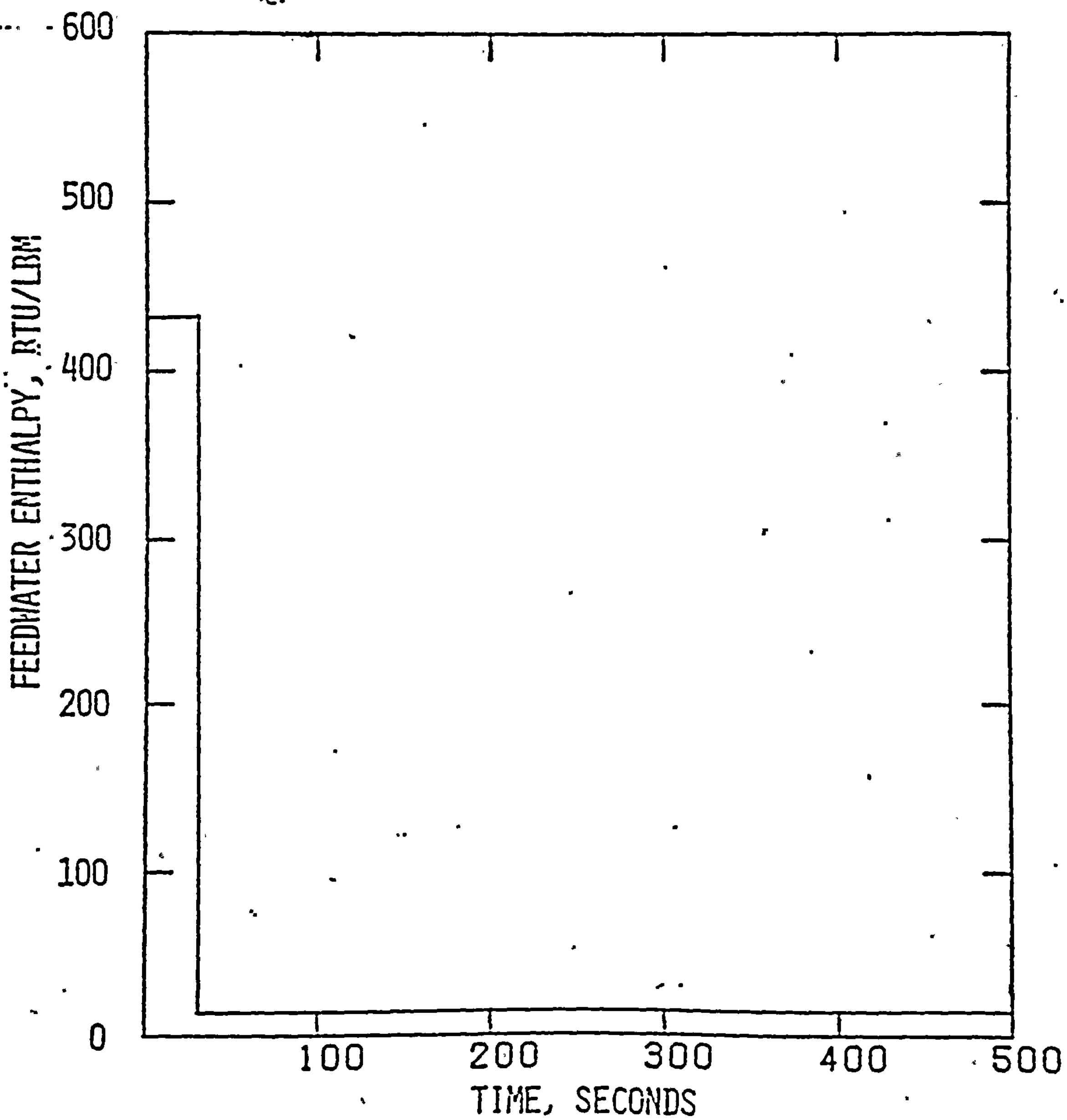
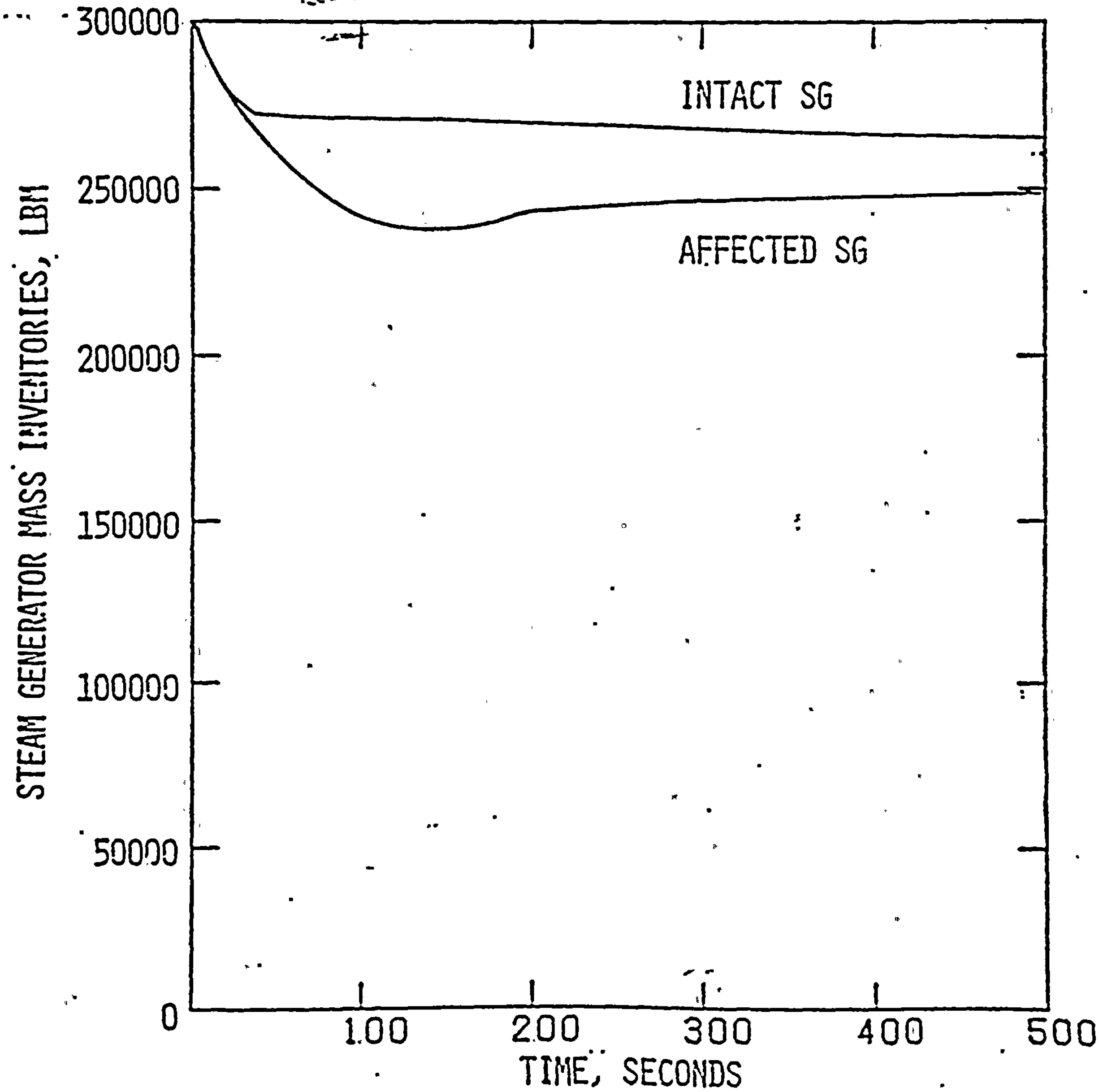
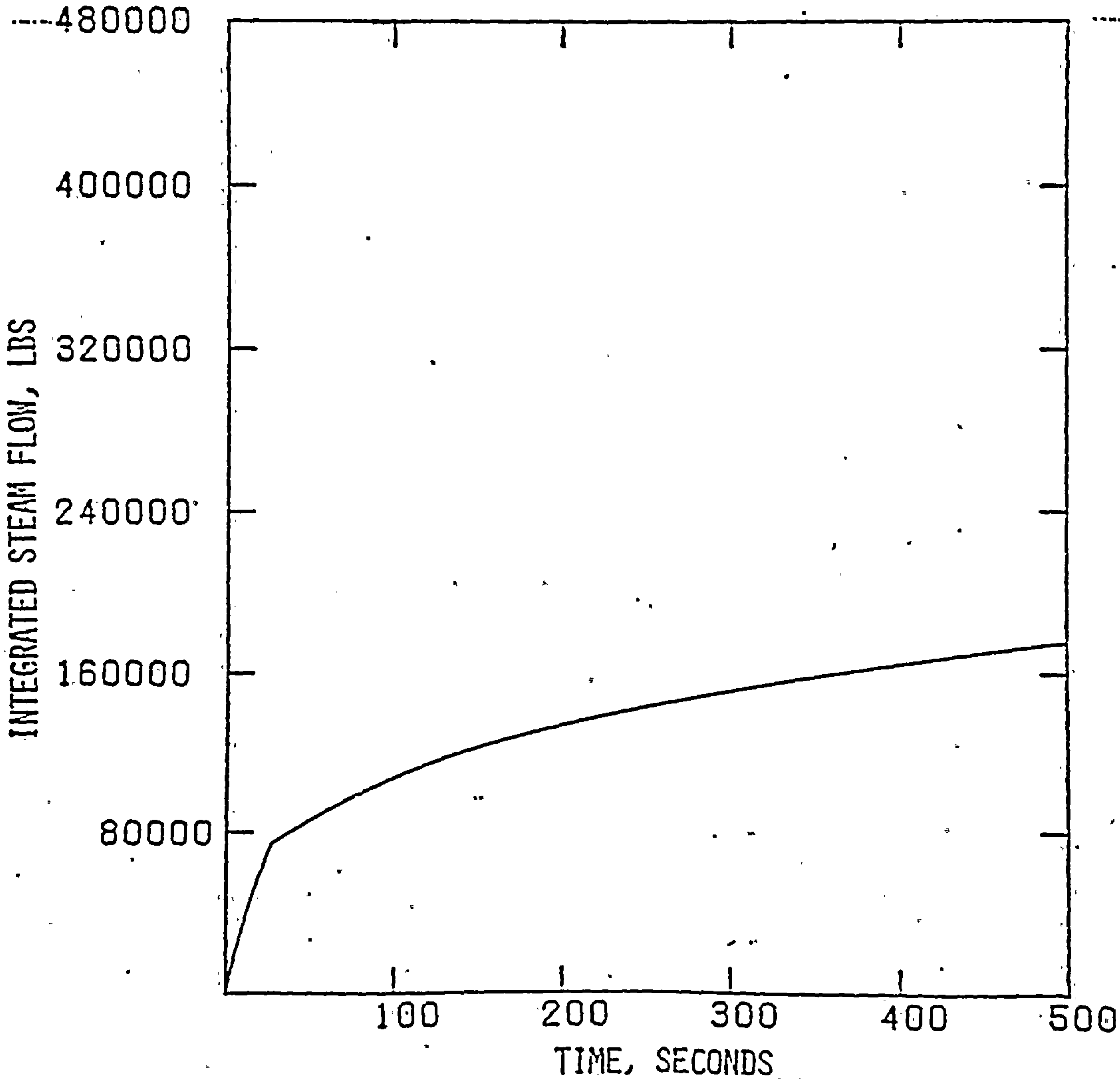

600
500
400
300
200
100
0
FEEDWATER ENTHALPY, BTU/LBM
100
200
300
400
500
TIME, SECONDS

Figure 1-12



300000
250000
200000
150000
100000
50000
0
100
200
300
400
500
TIME, SECONDS
STEAM GENERATOR MASS INVENTORIES, LBM
INTACT SG
AFFECTED SG

Figure 1-13



480000
400000
320000
240000
160000
80000
INTEGRATED STEAM FLOW, LBS
100
200
300
400
500
TIME, SECONDS

Figure 1-14

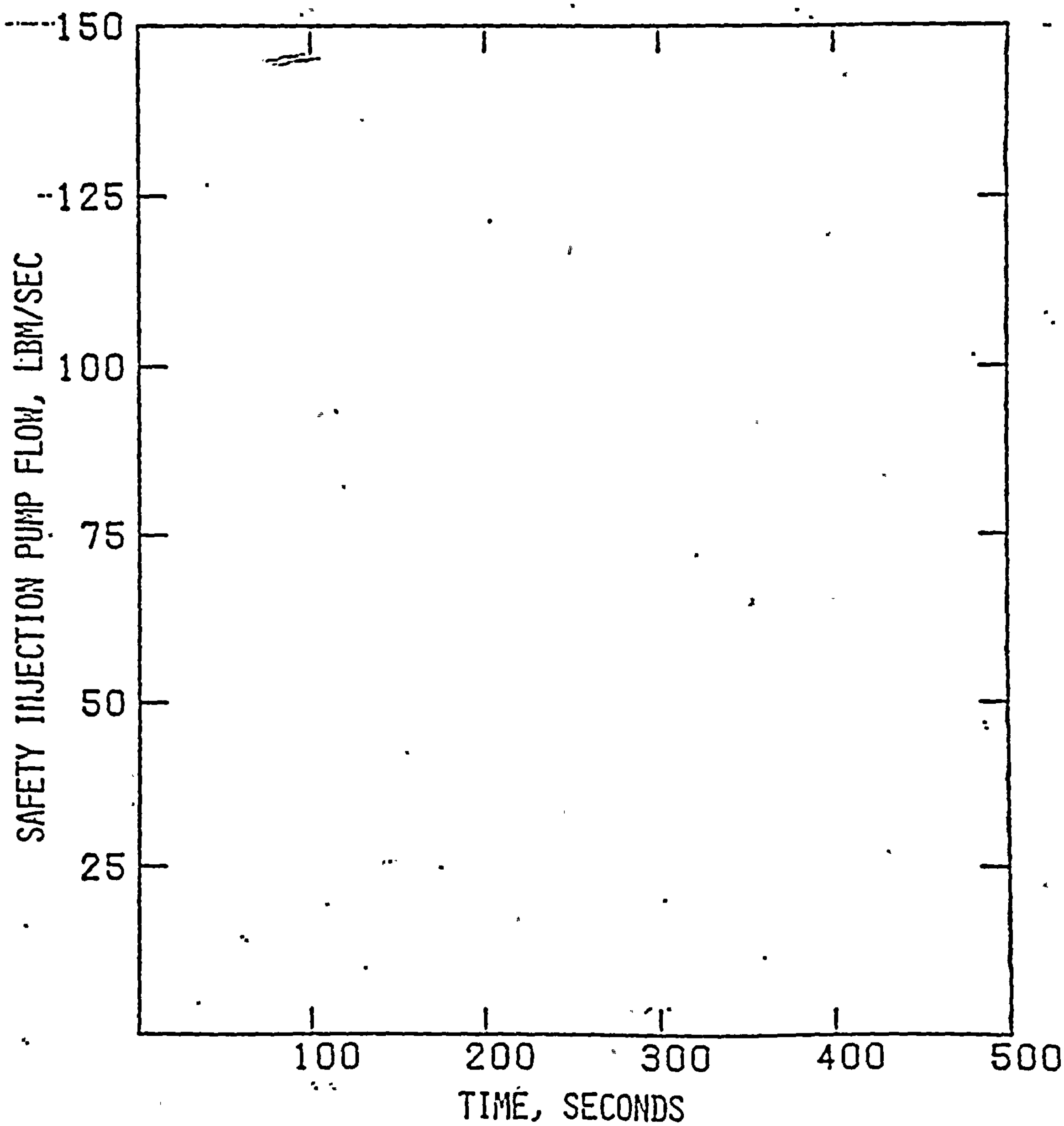
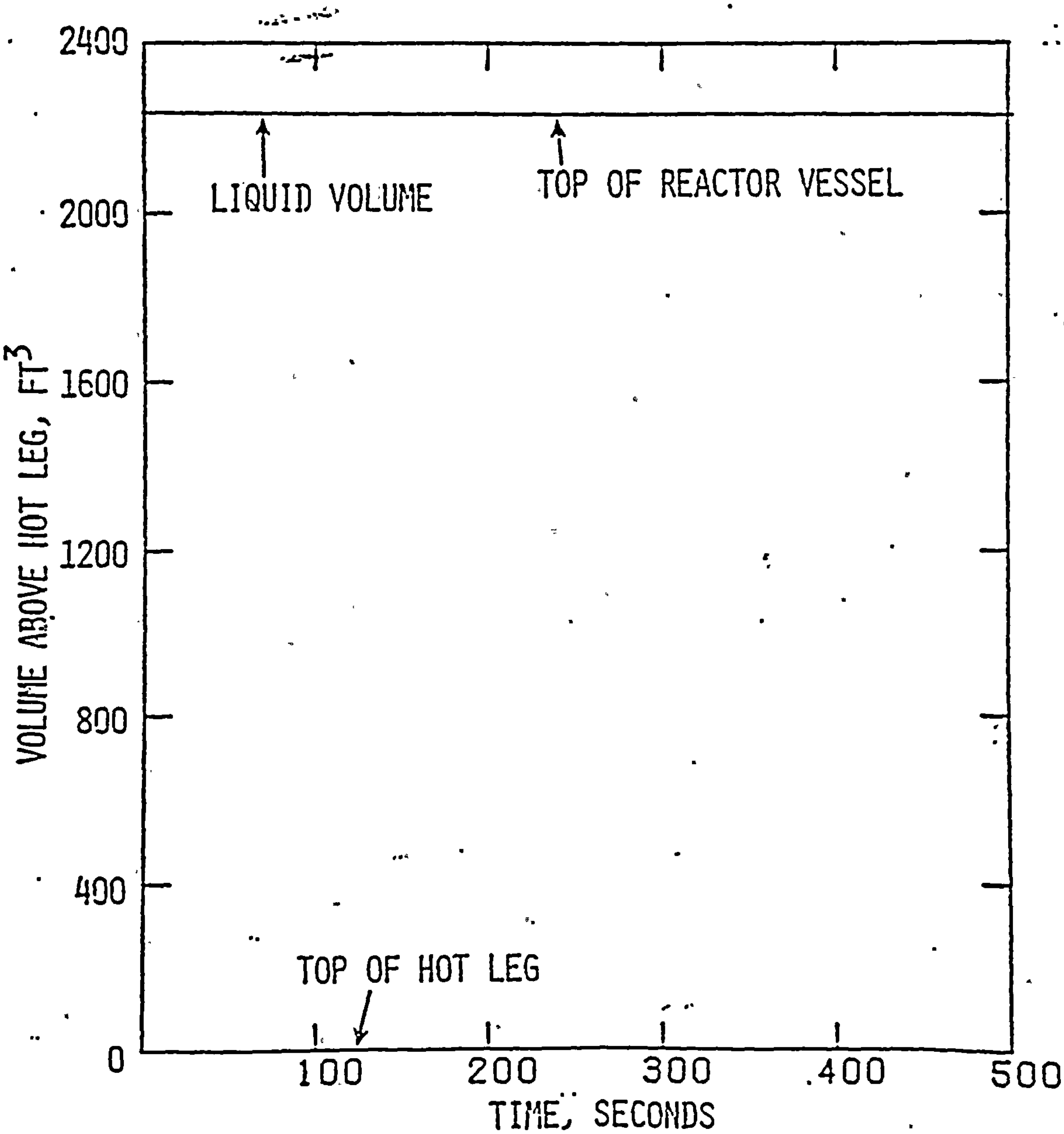
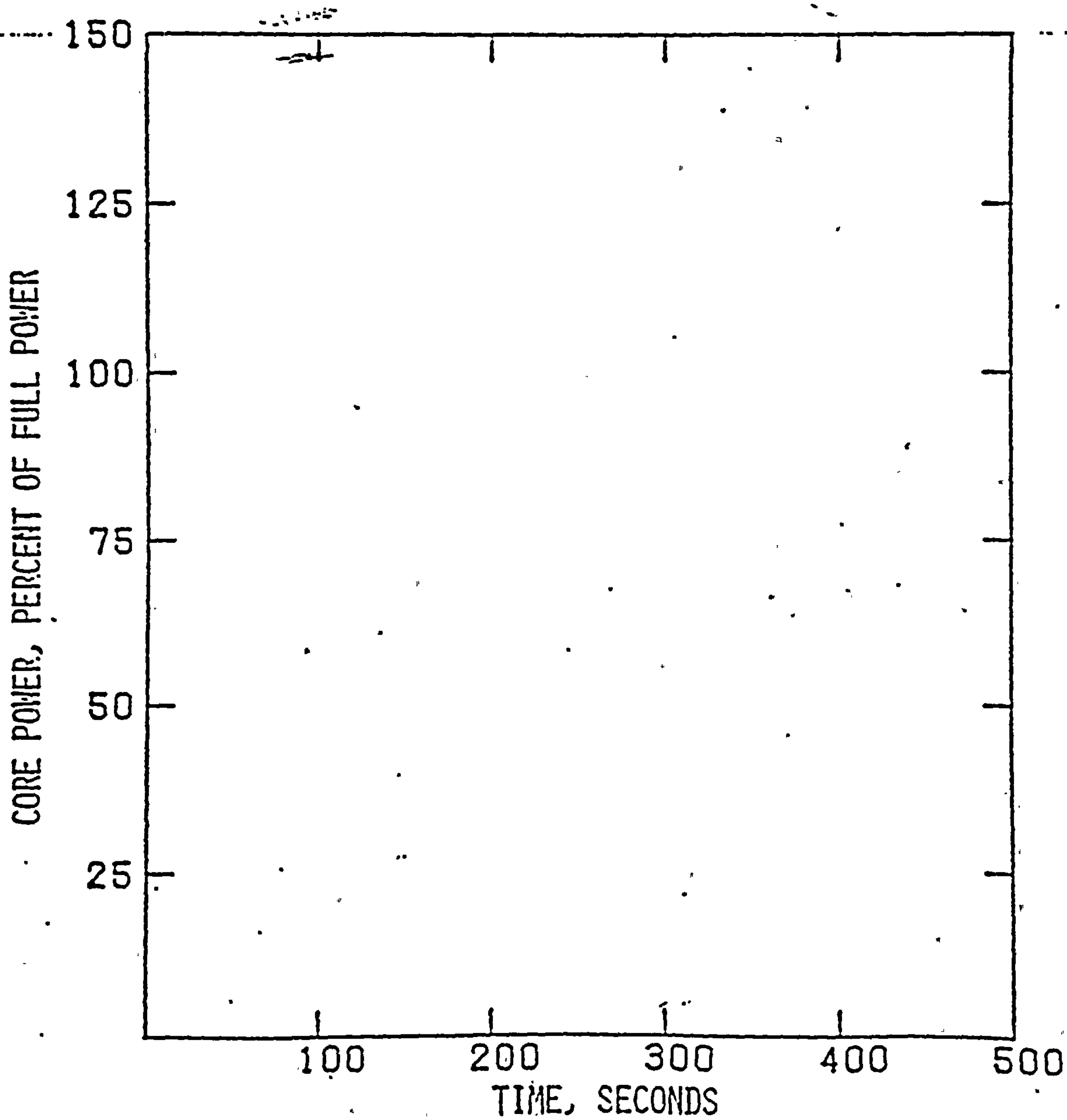

150
125
100
75
50
25
SAFETY INJECTION PUMP FLOW, LBM/SEC
100
200
300
400
500
TIME, SECONDS

Figure 1-15



2400
2000
1600
1200
800
400
0
LIQUID VOLUME
TOP OF REACTOR VESSEL
TOP OF HOT LEG
VOLUME ABOVE HOT LEG, FT3
100
200
300
400
500
TIME, SECONDS

Figure 1-16



150
125
100
75
50
25
CORE POWER, PERCENT OF FULL POWER
100
200
300
400
500
TIME, SECONDS

Figure 1-17

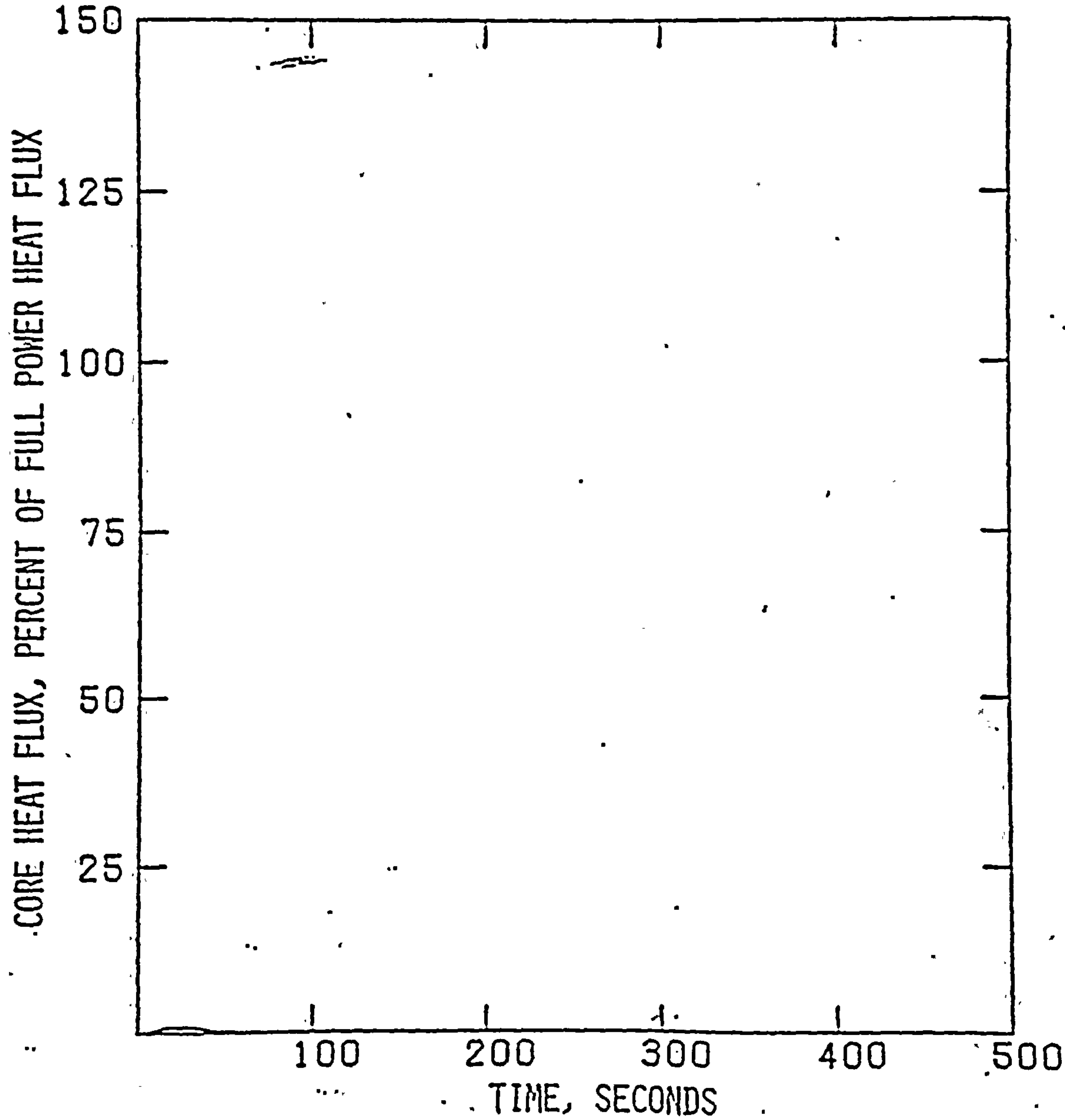

150
125
100
75
50
25
CORE HEAT FLUX, PERCENT OF FULL POWER HEAT FLUX
100
200
300
400
500
TIME, SECONDS

Figure 1-18

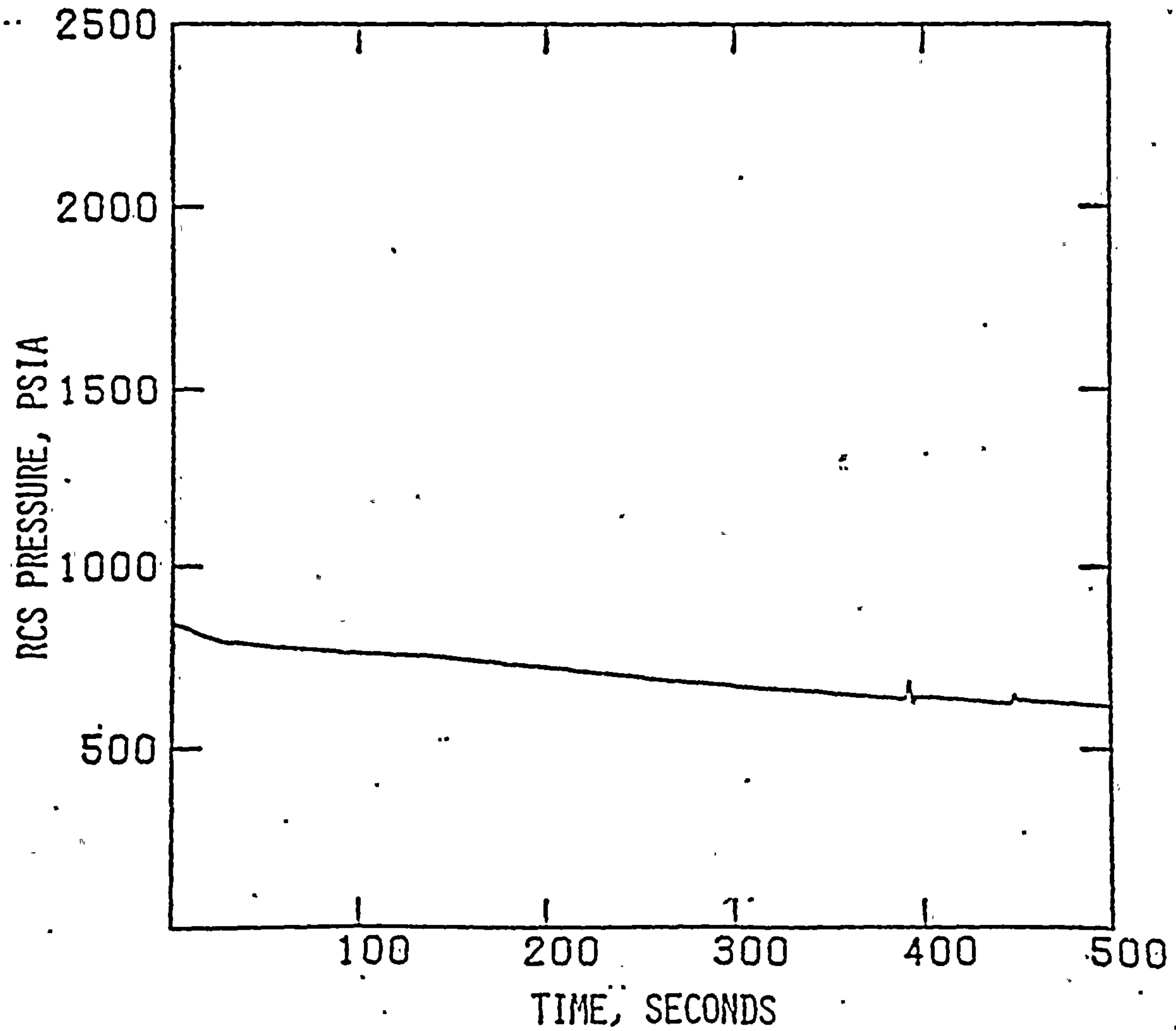

2500
2000
1500
1000
500
RCS PRESSURE, PSIA
100
200
300
400
500
TIME, SECONDS

Figure 1-19

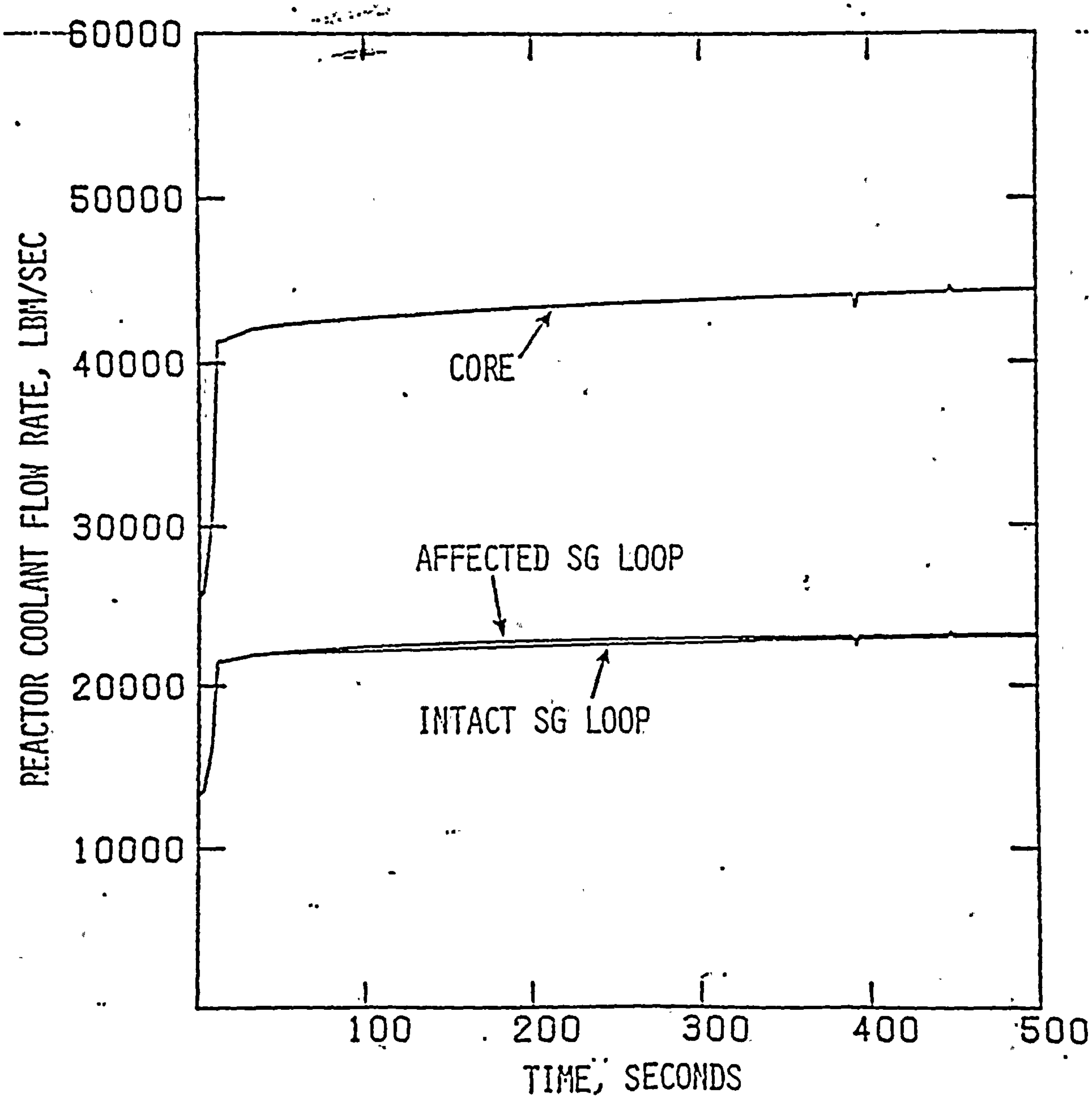
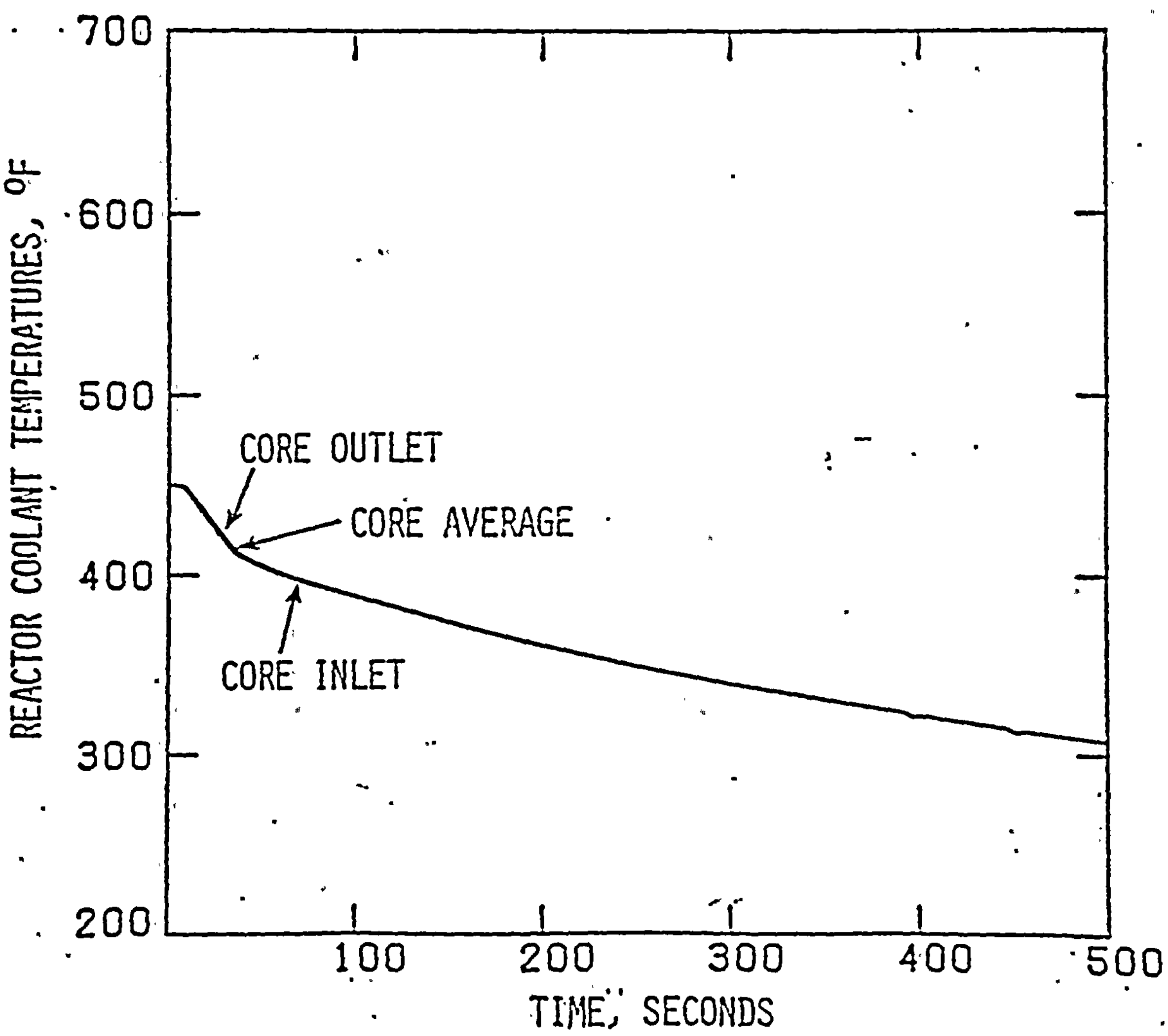
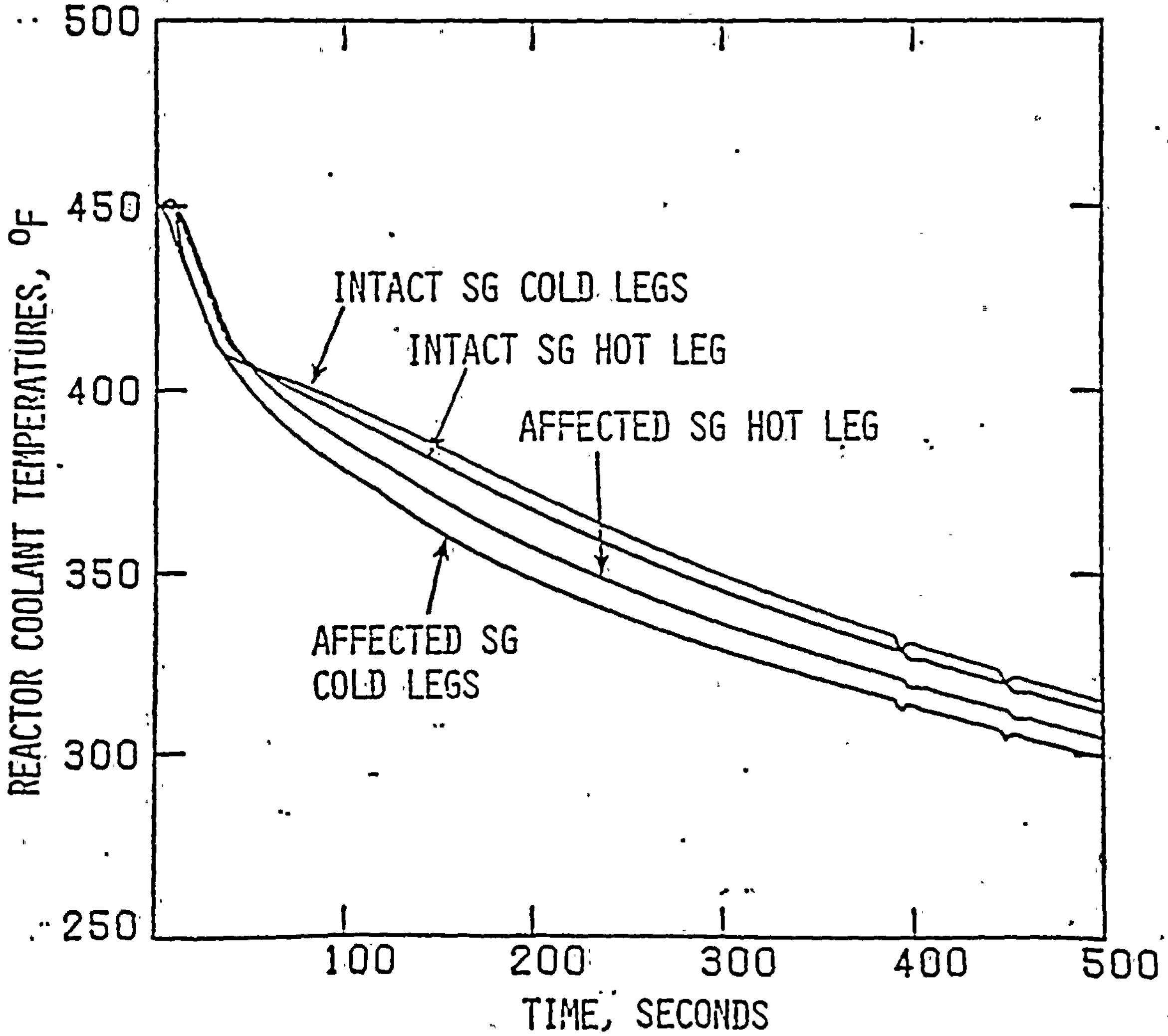

60000
50000
40000
30000
20000
10000
REACTOR COOLANT FLOW RATE, LBM/SEC
CORE
AFFECTED SG LOOP
INTACT SG LOOP
100
200
300
400
500
TIME, SECONDS

Figure 1-20



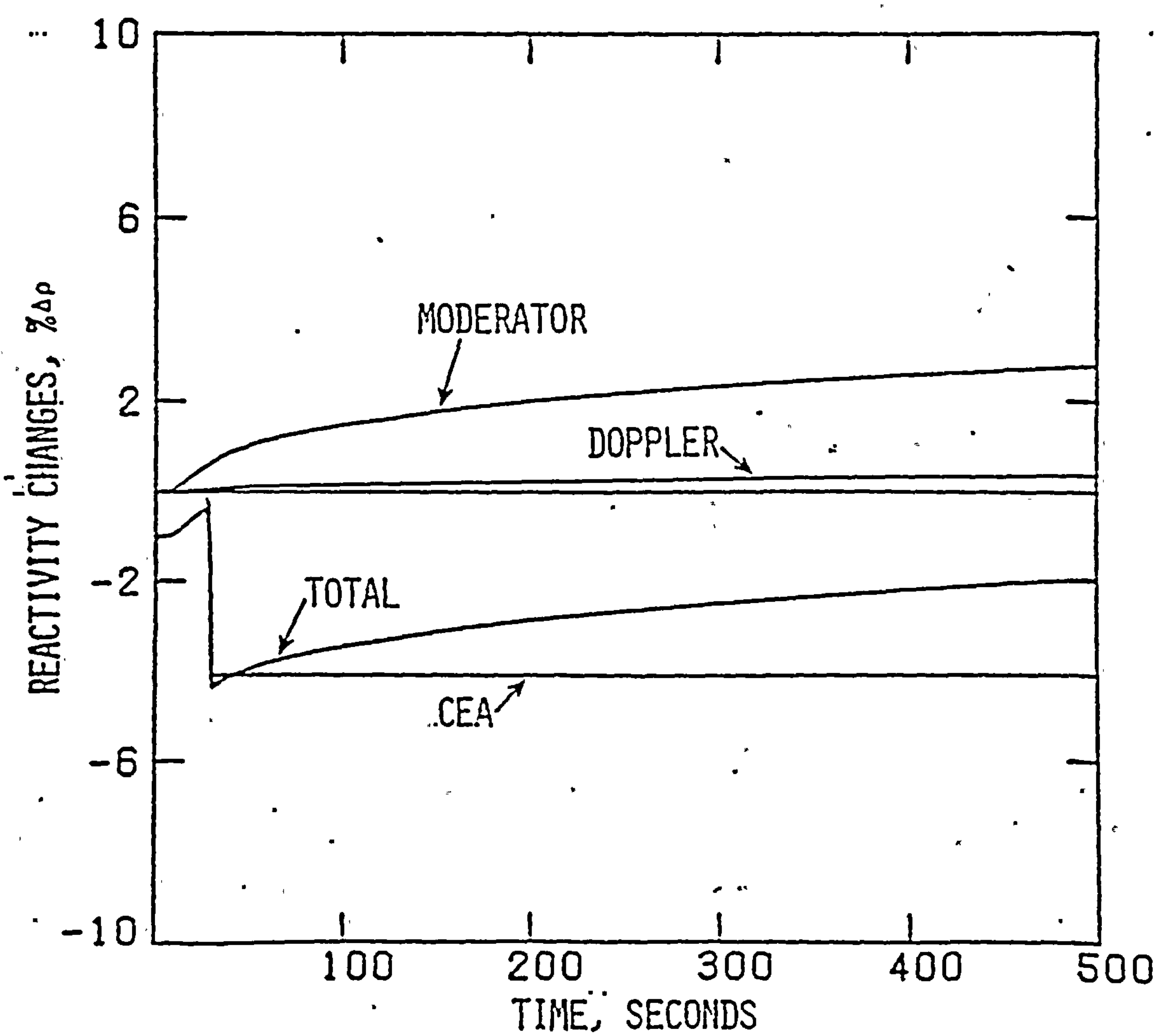
700
600
500
400
300
200
100
200
300
400
500
REACTOR COOLANT TEMPERATURES, °F
TIME, SECONDS
CORE OUTLET
CORE AVERAGE
CORE INLET

Figure 1-21



500
450
400
350
300
250
REACTOR COOLANT TEMPERATURES, °F
INTACT SG COLD LEGS
INTACT SG HOT LEG
AFFECTED SG HOT LEG
AFFECTED SG
COLD LEGS
100
200
300
400
500
TIME, SECONDS

Figure 1-22



10
6
2
-2
-6
-10
REACTIVITY CHANGES, %Δρ
MODERATOR
DOPPLER
TOTAL
CEA
100
200
300
400
500
TIME, SECONDS

Figure 1-23

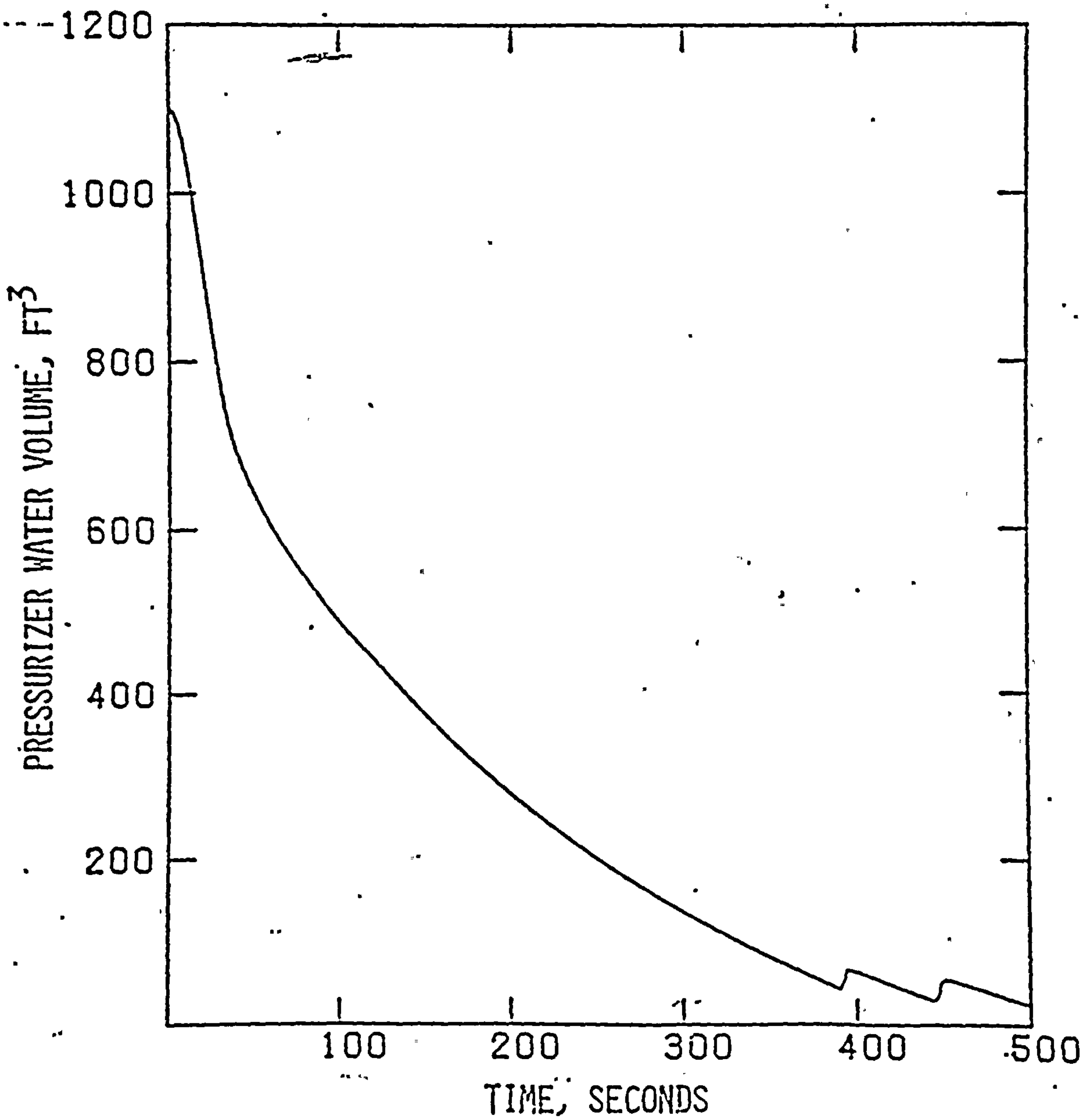

1200
1000
800
600
400
200
PRESSURIZER WATER VOLUME, FT^3
100
200
300
400
500
TIME, SECONDS

Figure 1-24

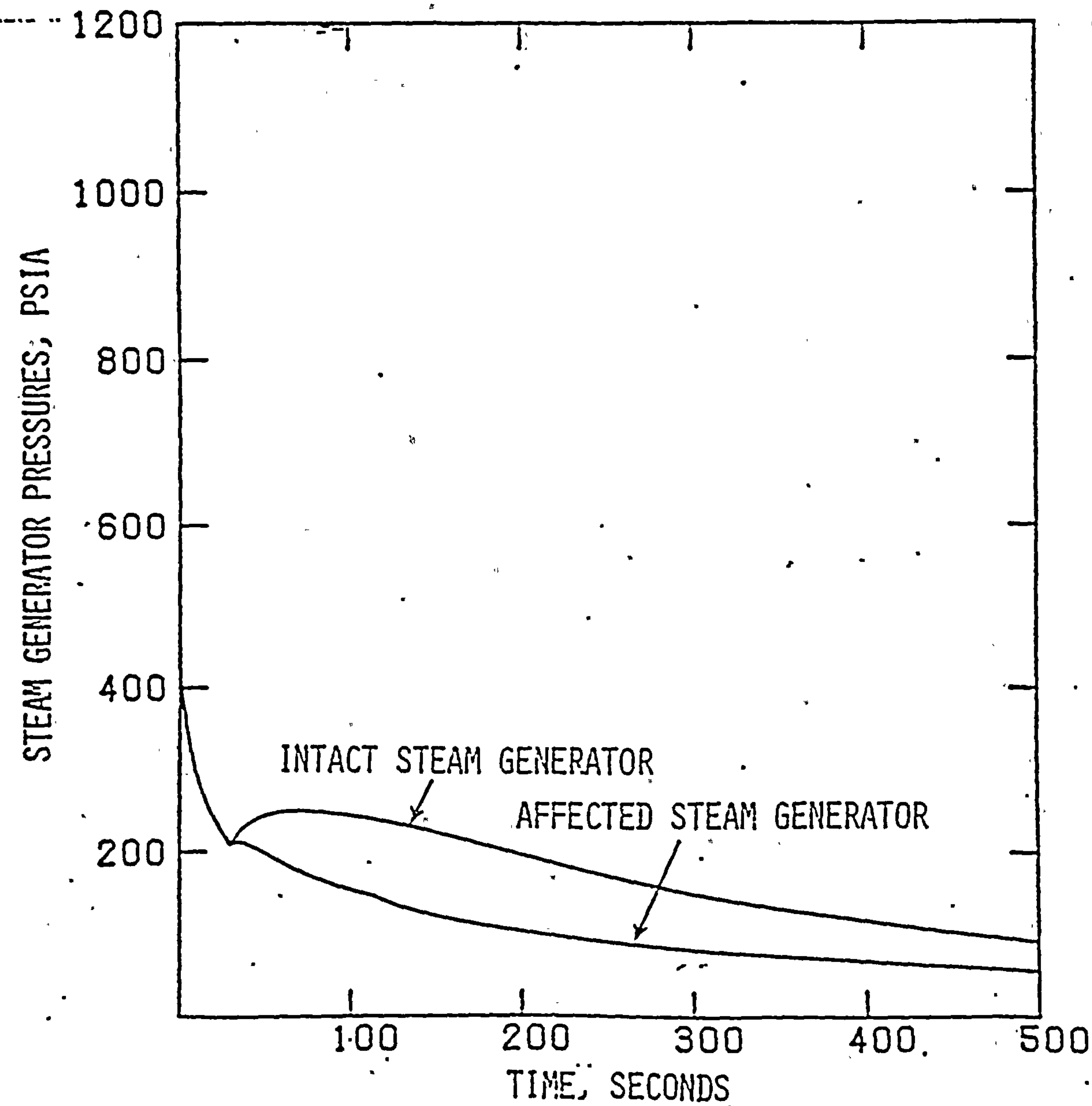

1200
1000
800
600
400
200
STEAM GENERATOR PRESSURES, PSIA
INTACT STEAM GENERATOR
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-25

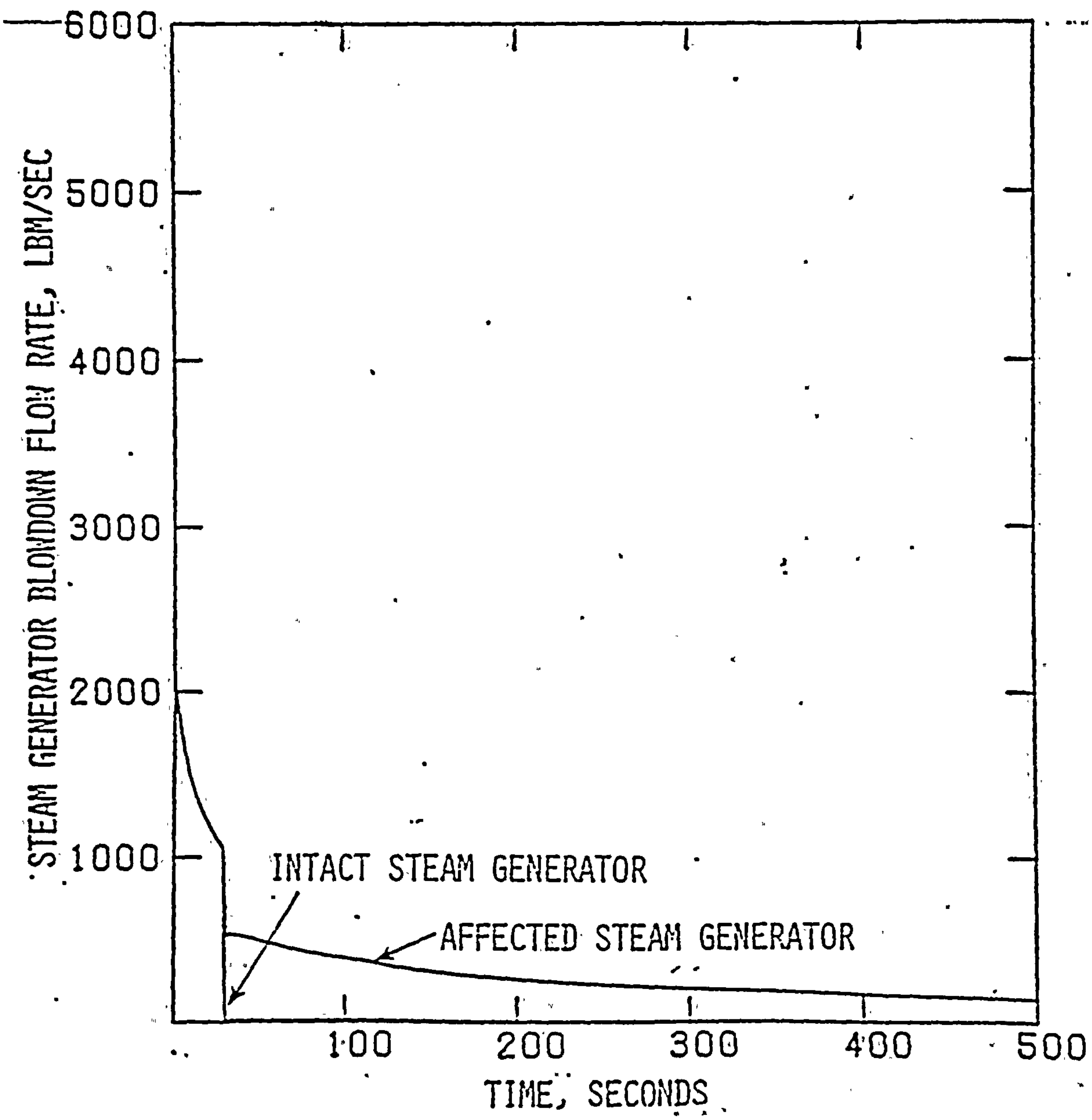

6000
5000
4000
3000
2000
1000
STEAM GENERATOR BLOWDOWN FLOW RATE, LBM/SEC
INTACT STEAM GENERATOR
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-26

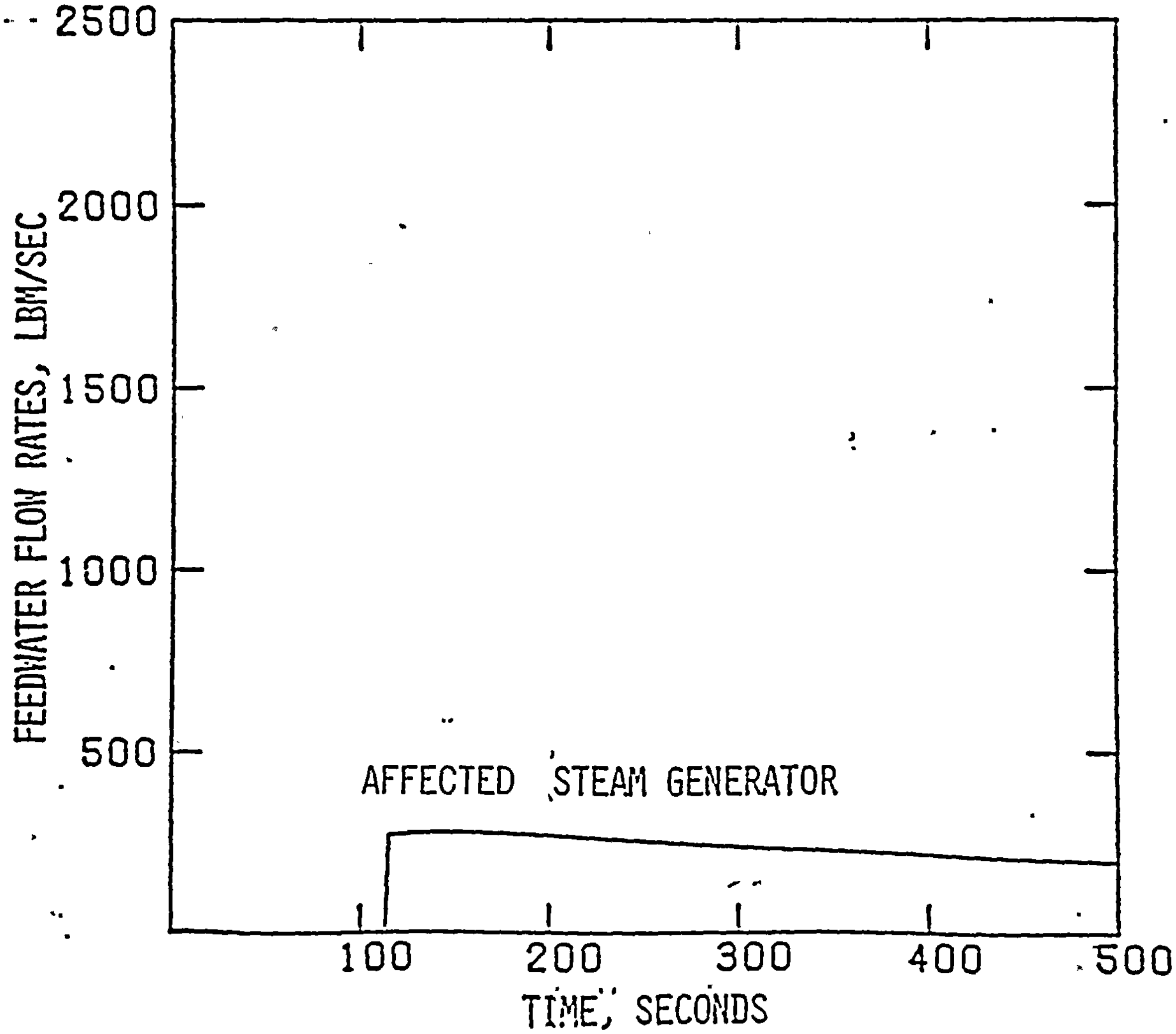

FEEDWATER FLOW RATES, LBM/SEC
2500
2000
1500
1000
500
AFFECTED STEAM GENERATOR
100
200
300
400
500
TIME, SECONDS

Figure 1-27

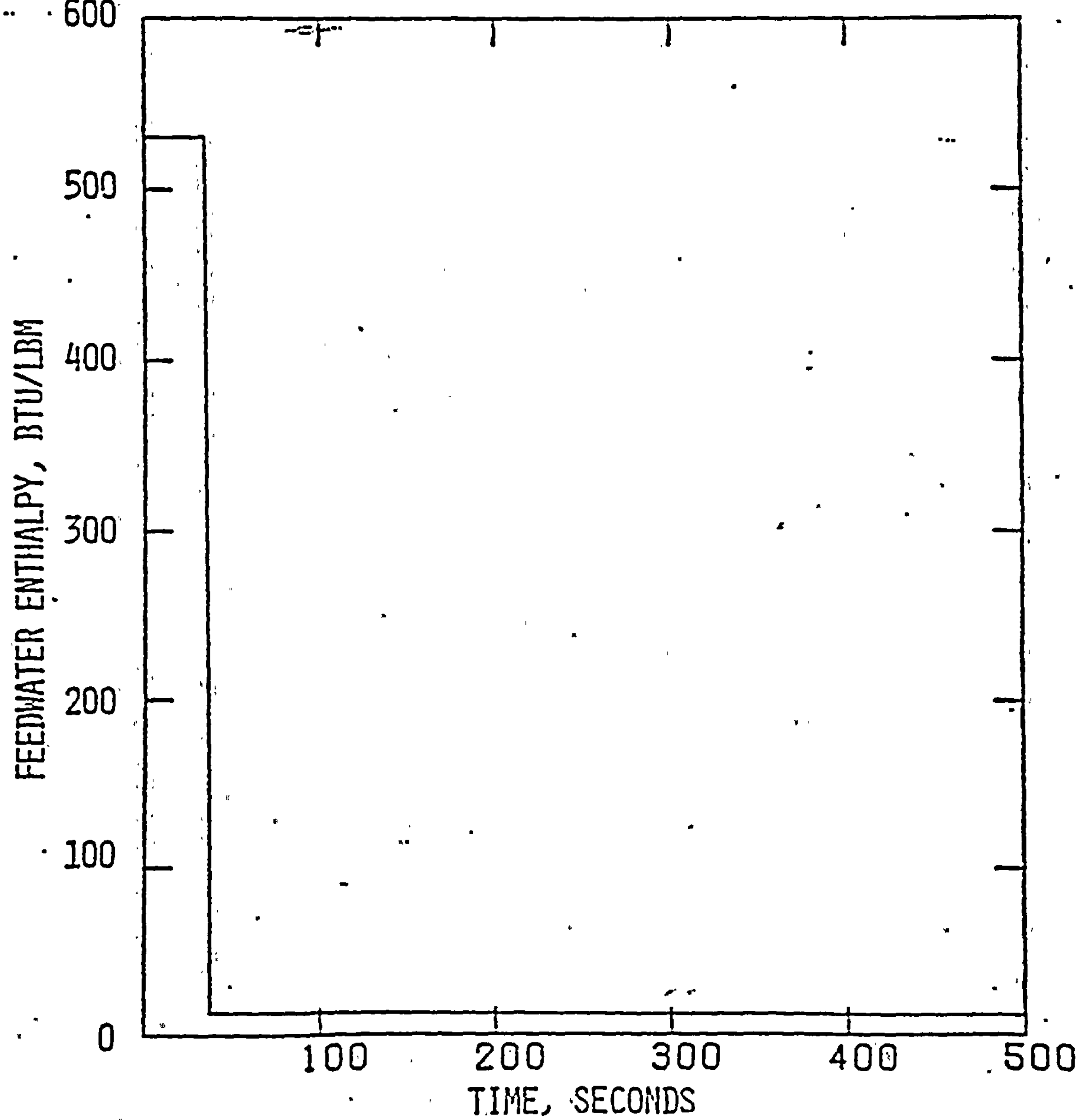
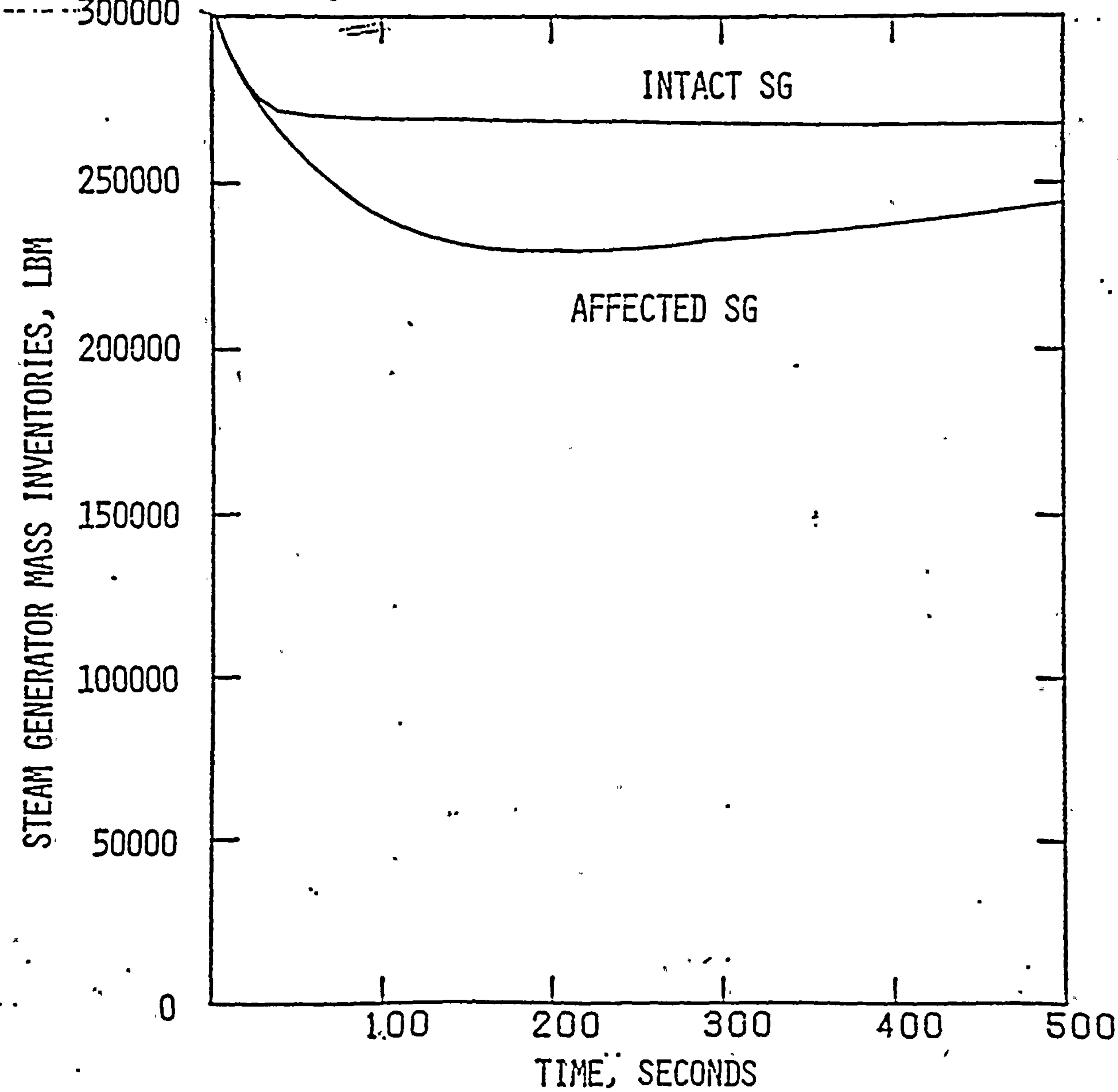

600
500
400
300
200
100
0
FEEDWATER ENTHALPY, BTU/LBM
100
200
300
400
500
TIME, SECONDS

Figure 1-28



300000
250000
200000
150000
100000
50000
0
STEAM GENERATOR MASS INVENTORIES, LBM
INTACT SG
AFFECTED SG
100
200
300
400
500
TIME, SECONDS

Figure 1-29

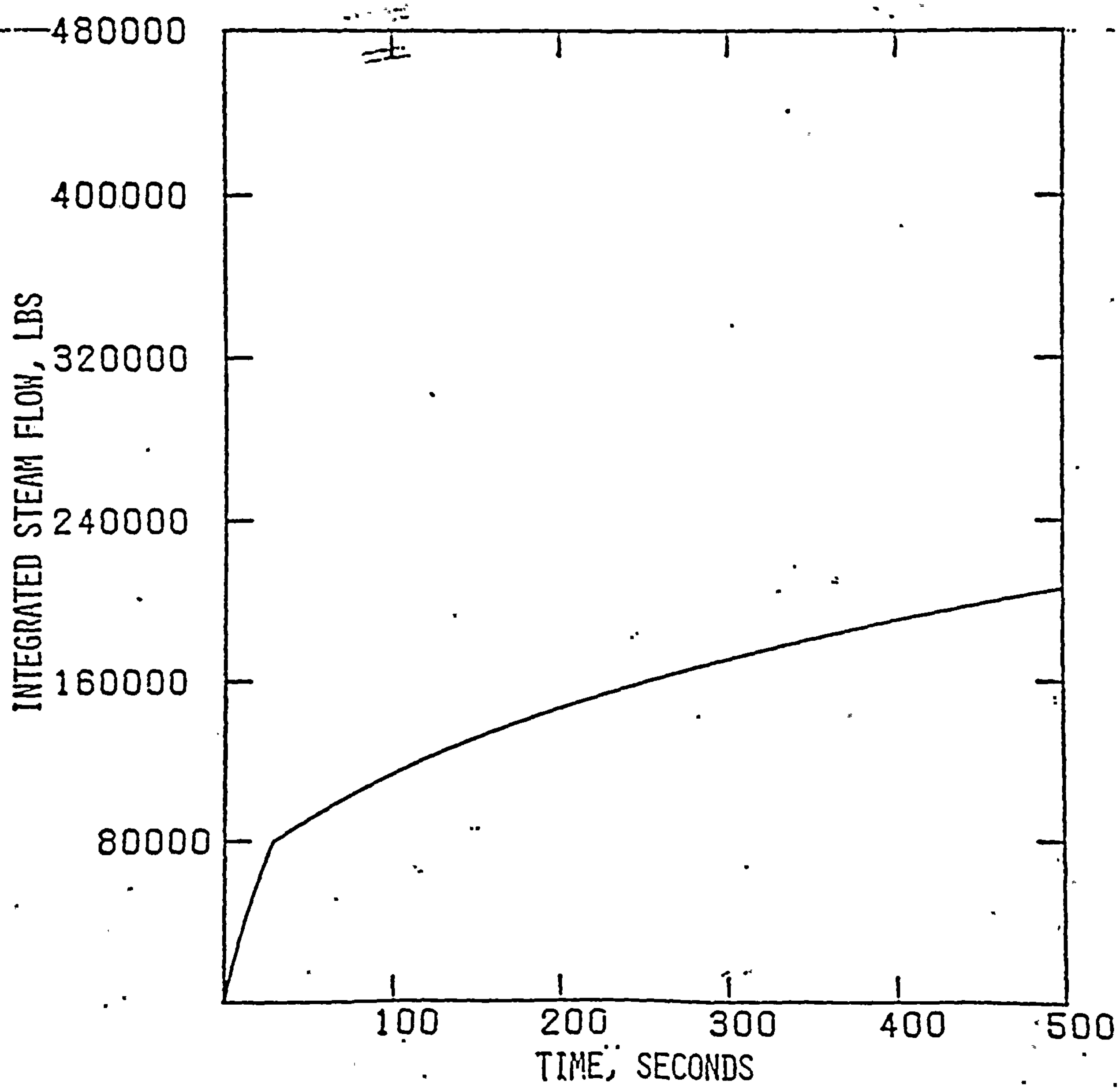

480000
400000
320000
240000
160000
80000
INTEGRATED STEAM FLOW, LBS
100
200
300
400
500
TIME, SECONDS

Figure 1-30

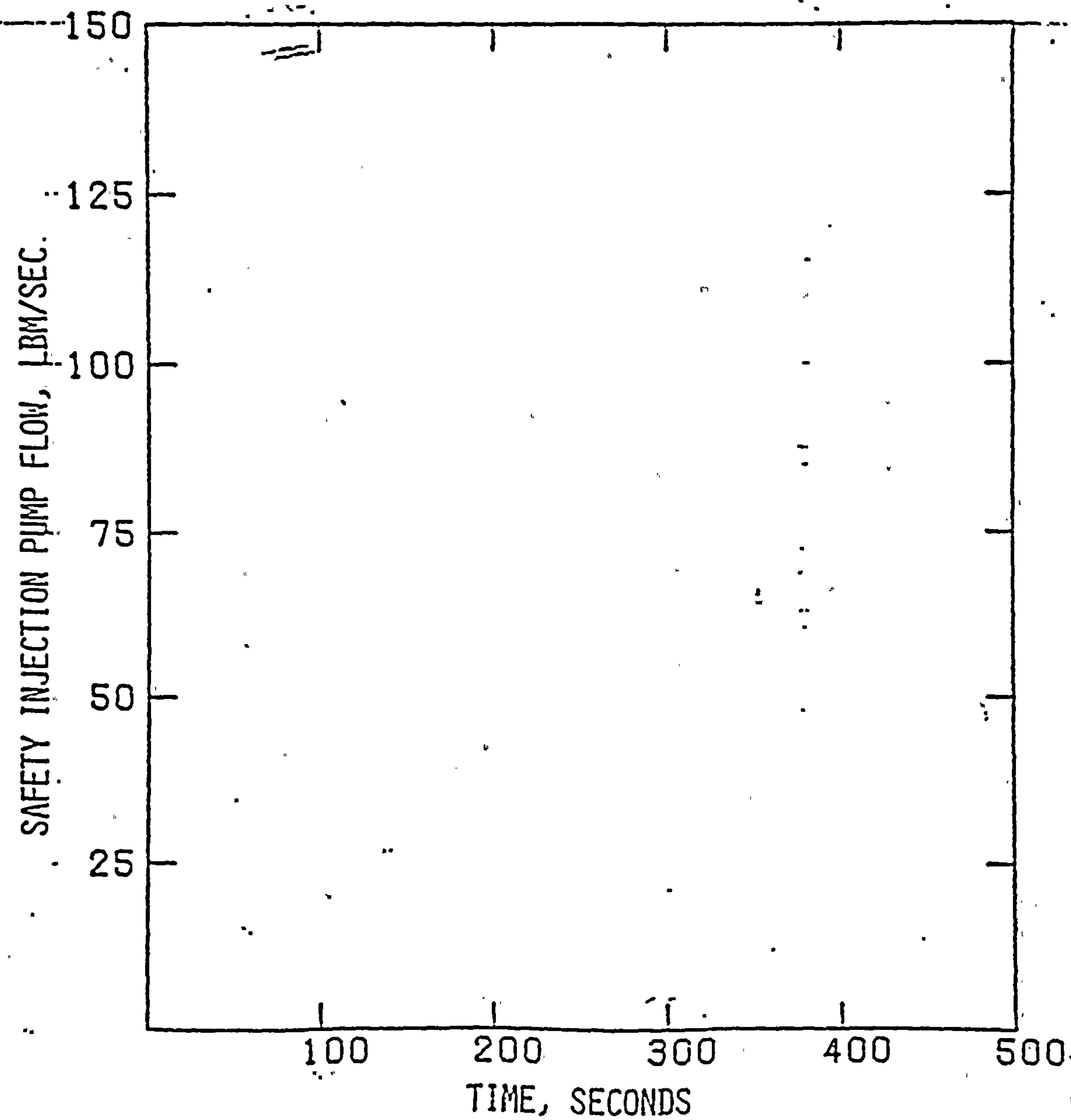
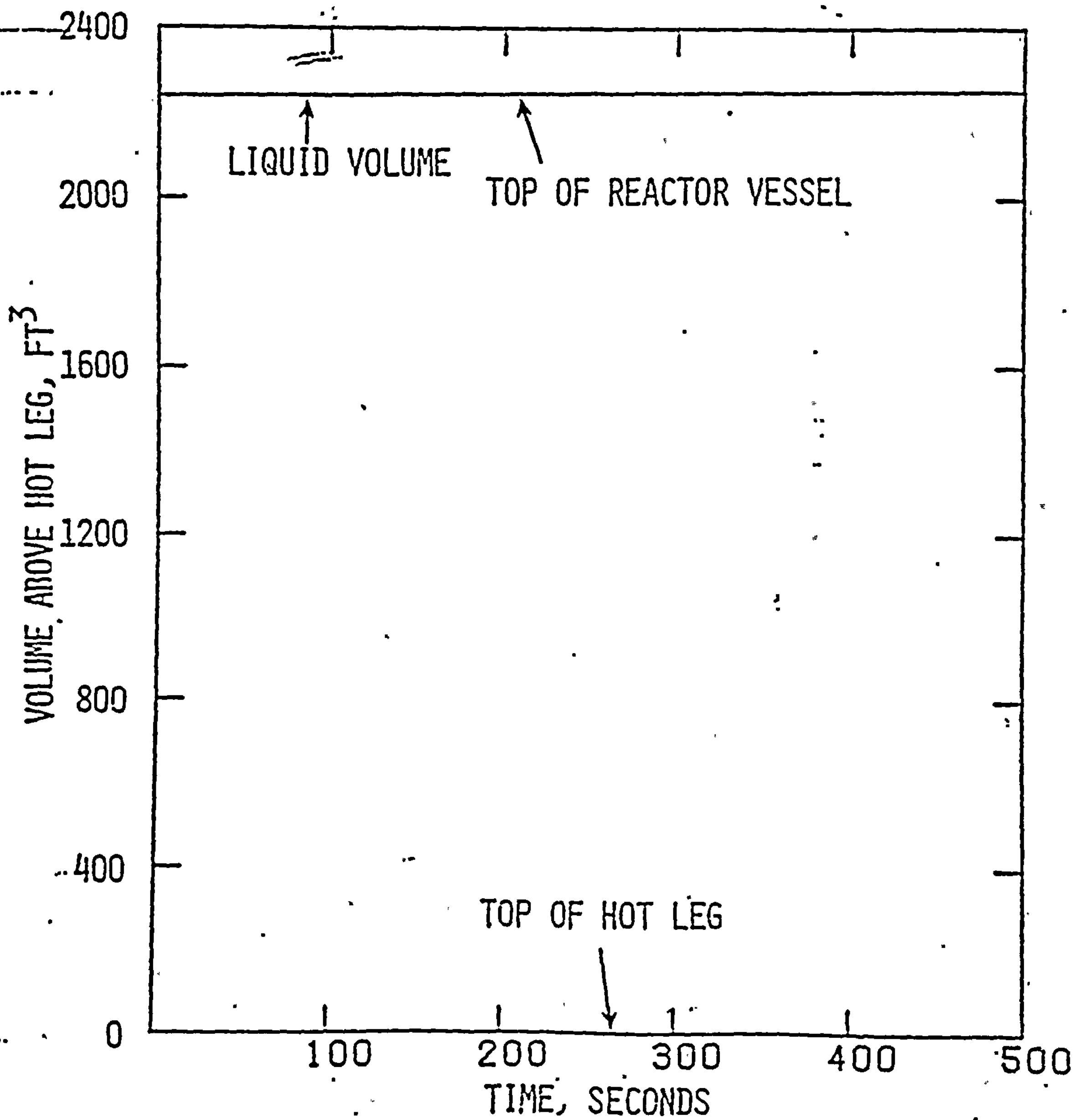
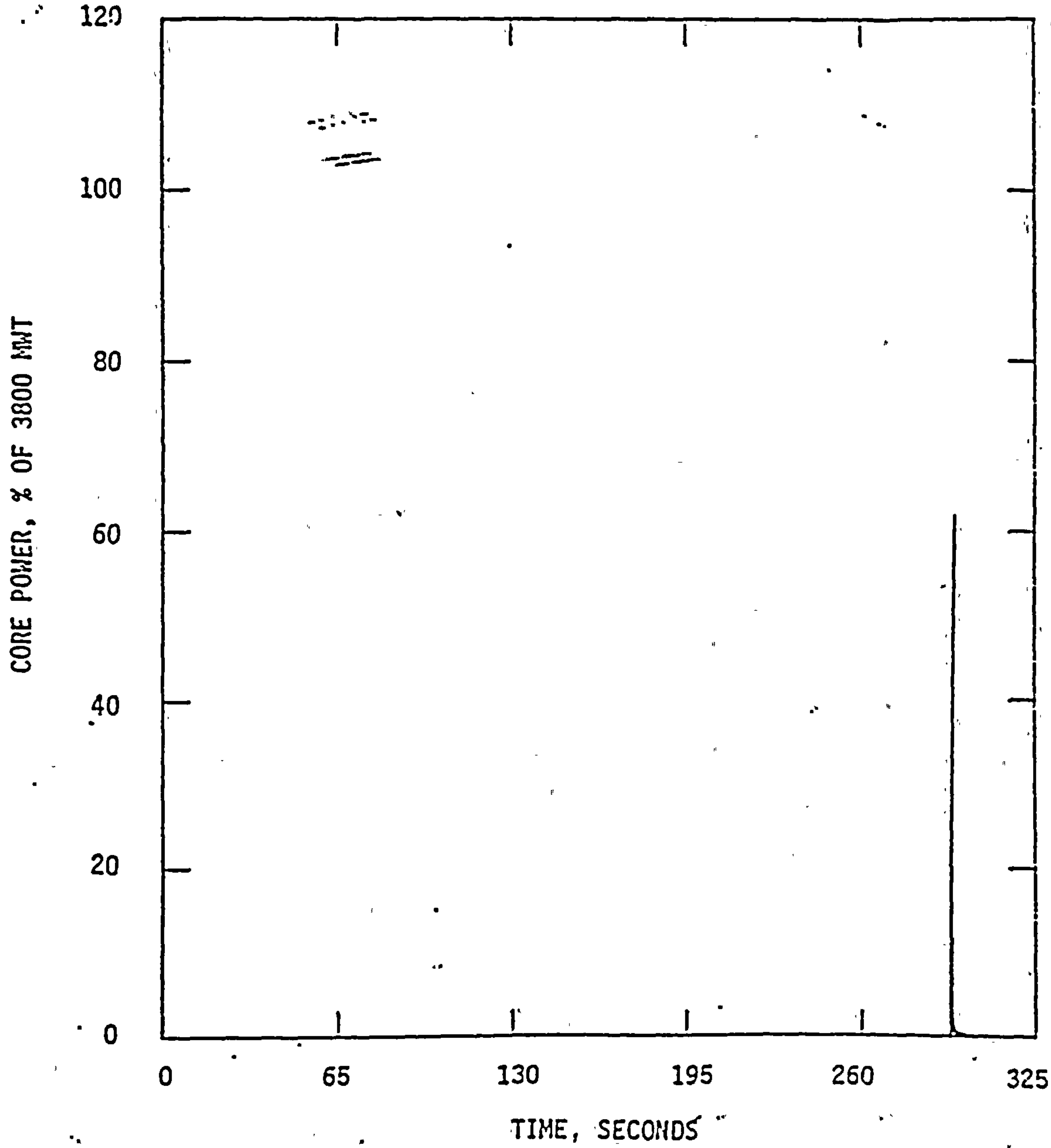

150
125
100
75
50
25
SAFETY INJECTION PUMP FLOW, LBM/SEC.
100
200
300
400
500
TIME, SECONDS

Figure 1-31



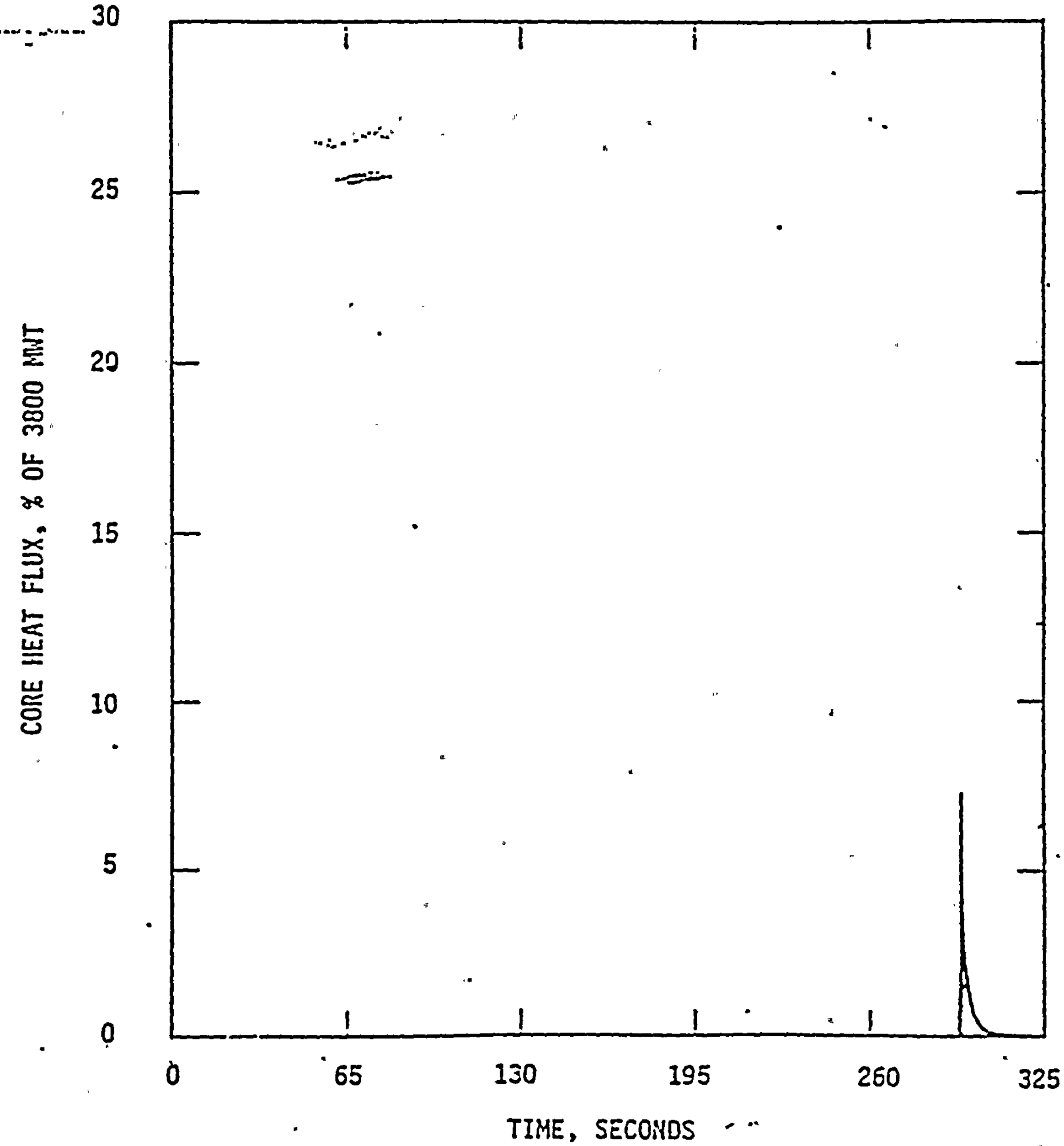
LIQUID VOLUME
TOP OF REACTOR VESSEL
TOP OF HOT LEG
VOLUME ABOVE HOT LEG, FT3
2400
2000
1600
1200
800
400
0
100
200
300
400
500
TIME, SECONDS

Figure 1-32



CORE POWER, % OF 3800 MWT
120
100
80
60
40
20
0
0
65
130
195
260
325
TIME, SECONDS

Figure 4-1



30
25
20
15
10
5
0
CORE HEAT FLUX, % OF 3800 MWT
0
65
130
195
260
325
TIME, SECONDS

Figure 4-2

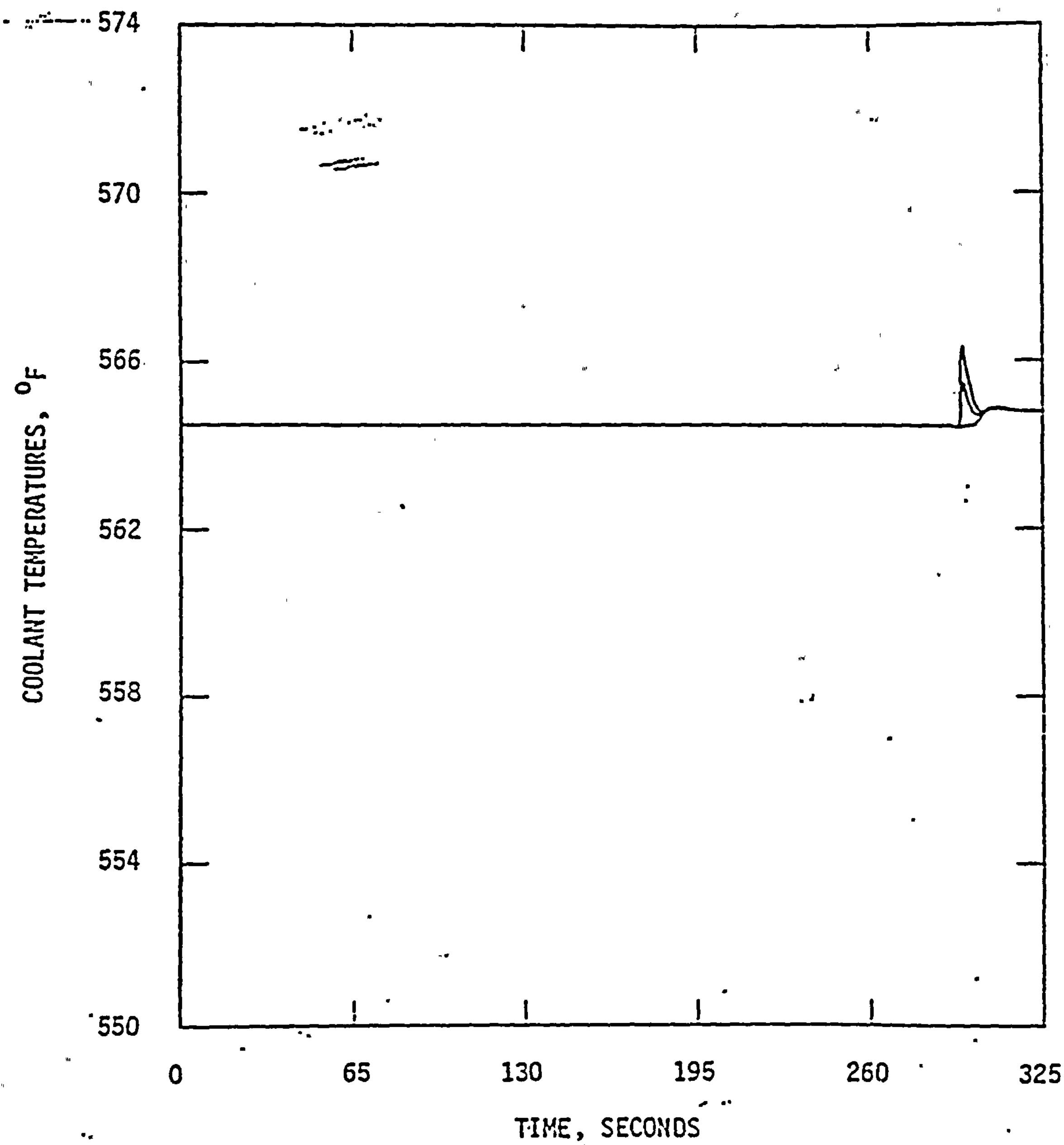

574
570
566
562
558
554
550
COOLANT TEMPERATURES, °F
0
65
130
195
260
325
TIME, SECONDS

Figure 4-3

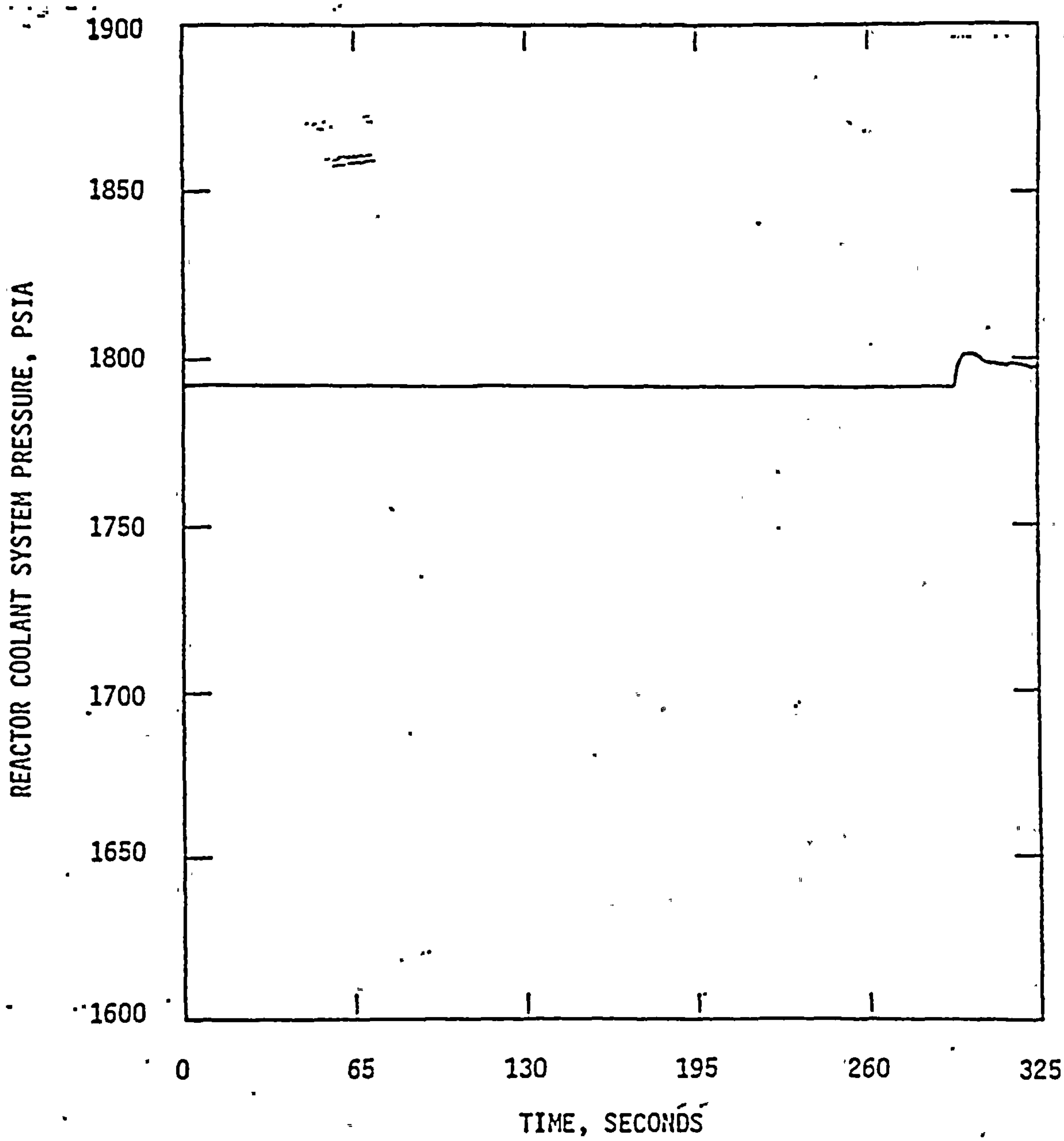

REACTOR COOLANT SYSTEM PRESSURE, PSIA
1900
1850
1800
1750
1700
1650
1600
0
65
130
195
260
325
TIME, SECONDS

Figure 4-4

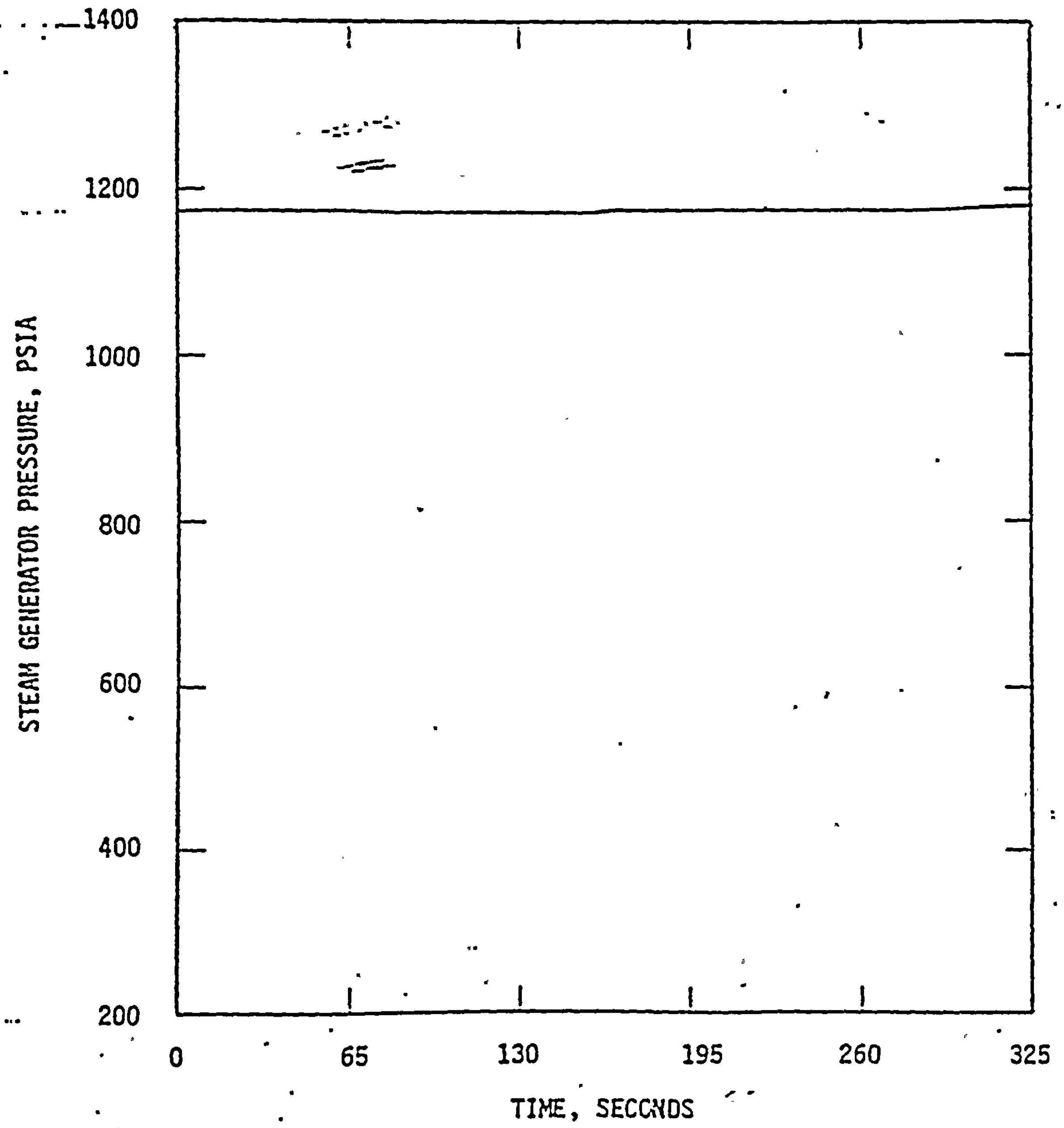

1400
1200
1000
800
600
400
200
STEAM GENERATOR PRESSURE, PSIA
0
65
130
195
260
325
TIME, SECCNDS

Figure 4-5