TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	ID (x/y)	SD (x/y)	TD (x/y)	UD (x/y)	SMD (x/y)
			TOTAL NUMBER* OF INSTRUMENTS		
REACTOR BUILDING ELEV 422'-3"					
CRD PUMP ROOM	4/0				
AUX COND. PUMP ROOM	2/0				
REACTOR BUILDING ELEV 441'-0"					
RAILROAD AIRLOCK		4/0			
REACTOR BUILDING ELEV 444'-0"					
RHR-2A PUMP RM R2	3/0				
RHR-2B PUMP RM R1	3/0				
RHR-2C PUMP RM R4	3/0				
RCIC PUMP RM R3	3/0				
LPCS PUMP RM R5	2/0				
HPCS PUMP RM R6	3/0				
REACTOR BUILDING ELEV 471'-0"					
MCC ROOM	1/0				
GENERAL AREA	24/0				
REACTOR BUILDING ELEV 501'-0"					
GENERAL AREA	23/0				
REACTOR BUILDING ELEV 522'-0"					
MCC ROOM DIV. 2	1/0				
GENERAL AREA	28/0				
RHR VALVE ROOM	1/0				
REACTOR BUILDING ELEV 548'-0"					
FUEL POOL HT. EXCHGR ROOM A AND PUMP ROOM	1/0				
GENERAL AREA	29/0				
RHR HT. EXCHGR B ROOM	1/0				
REACTOR BUILDING ELEV 572'-0"					
HYDROGEN RECOMBINER CONT. RM DIV. 2	2/0				
RHR HT. EXCHGR RM 1A	1/0				
RHR HT. EXCHGR RM 1B	1/0				
GENERAL FLOOR AREA	25/0				
REACTOR BUILDING ELEV 606'-10.5"					
GENERAL FLOOR AREA				6/0	
RADWASTE CONTROL BUILDING ELEV 467'-0"				Misaligned	
ELECTRICAL EQUIPMENT ROOM NO. 1	2/0				
BATTERY ROOM NO. 1	4/0				
SWITCHGEAR ROOM NO. 1	3/0				
ELECTRICAL EQUIPMENT ROOM NO. 2	3/0				
BATTERY ROOM NO. 2	2/0				

TABLE 3.3.7.9-1 (Continued)

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	TOTAL NUMBER* OF INSTRUMENTS ID (x/y)	SD (x/y)	TD (x/y)	UD (x/y)	SMD (x/y)
RADWASTE CONTROL BUILDING ELEV 467'-0" (Continued)					
SWITCHGEAR ROOM NO. 2	3/0				
REMOTE SHUTDOWN ROOM	1/0				
CORRIDOR C-205	5/0				
RADWASTE AND CONTROL BUILDING ELEV 484'-0"					
CABLE SPREADING ROOM	0/36				
RADWASTE AND CONTROL BUILDING ELEV 501'-0"					
CABLE CHASE	0/5				
CONTROL ROOM (CEILING)	12/0	1/0			
CONTROL ROOM (PGCC)					
U679	8/0	0/9			
U680	11/0	0/14			
U681	7/0	~~0/8~~ 0/9			
U682	6/0	0/8			
U683	8/0	0/8			
U684	6/0	0/8			
U685	6/0	0/8			
U686	8/0	0/8			
U687	8/0	0/8			
U688	6/0	0/8			
U689	4/0	0/8			
U690	5/0	0/8			
U800	5/0	0/6			
U840	5/0	0/6			
U891	8/0	0/8			
U892	8/0	0/10			
U893	9/0	~~0/8~~ 0/9			
U894	8/0	0/9			
RADWASTE AND CONTROL BUILDING ELEV 525'-0"					
CABLE CHASE	0/6				
UNIT A - AIR CONDITIONING ROOM	5/0				2/1
UNIT B - AIR CONDITIONING ROOM	5/0				2/1
STANDBY SERVICE WATER PUMP HOUSE 1A					
PUMP HOUSE	1/0				
ELECTRICAL VAULT	1/0				
STANDBY SERVICE WATER PUMP HOUSE 1B					
PUMP HOUSE	1/0				
ELECTRICAL VAULT	1/0				

PLANT SYSTEMS

SPRAY AND SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.6.2 The following pre-action and deluge spray and sprinkler systems shall be OPERABLE:

a. Radwaste Building:

1. Cable spreading room, elev. 484', system #65.
2. Cable chase and corridor, elev. 441' to 525', system #66.
3. Control Bldg. emergency charcoal filters, elev. 525', system #WMA-DV-54A and WMA-DV-54B.
4. Control Room, Elev. 501', a/s office areas only.

b. Diesel Generator Building:

1. DG room 1A and day tank room, elev. 441', system #79.
2. DG 1A day tank pump room, elev. 441', system #80.
3. DG room 1B and day tank room, elev. 441', system #81.
4. DG 1B day tank pump room, elev. 441', system #82.
5. HPCS DG room and day tank room, elev. 441', system #83.
6. HPCS DG day tank pump room, elev. 441', system #84.

c. Reactor Building:

1. Standby gas treatment system charcoal filters, elev. 572', system #SGT-DIV-1A-1, #SGT-DIV-1A-2, #SGT-DIV-1A-3, #SGT-DIV-1B-1, #SGT-DIV-1B-2, and #SGT-DIV-1B-3.

2. Sump vent filter system charcoal filters, elev. 572', system #REA-DV-2A and #REA-DV-2B.

APPLICABILITY: Whenever equipment protected by the spray and/or sprinkler systems 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 Specification 3.0.3 and 3.0.4 are not applicable.

TABLE 4.3.7.11-1 (Continued)

TABLE NOTATIONS

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

~~2. High voltage abnormally low.~~

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

(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. Instrument indicates a downscale failure.

3. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more 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. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours when continuous, periodic, or batch releases are made.

(5) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. High voltage abnormally low.

3. Instrument indicates a downscale failure.

SURVEILLANCE REQUIREMENTS (Continued)

e. For the ADS by:

1. At least once per 31 days by verifying that the accumulator backup compressed gas system pressure in each bottle is $\geq$ 2200 psig.

2. At least once per 31 days, performing a CHANNEL FUNCTIONAL TEST of the accumulator backup compressed gas system low pressure alarm system.

3. At least once per 18 months:

   a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

   b) Manually opening each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig* and observing that either:

      1) The control valve or bypass valve position responds accordingly, or

      2) There is a corresponding change in the measured steam flow.

   c) Performing a CHANNEL CALIBRATION of the accumulator backup compressed gas system low pressure alarm system and verifying an initiation setpoint of ≥ 140 ~~±3~~ psig on decreasing pressure and an alarm setpoint of ≥ 135 ~~±3~~ psig on decreasing pressure.

   d) Verifying the nitrogen capacity in at least two accumulator bottles per division within the backup compressed gas system.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

CONTAINMENT SYSTEMS

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system warmup test that the minimum recombiner heater outlet temperature increases to greater than or equal to 500°F within 90 minutes.

b. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

   2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

   3. Verifying during a recombiner system functional test that, upon introduction of 1% by volume hydrogen in a 140-180 scfm stream containing at least 1% by volume oxygen, that the catalyst bed temperature rises in excess of 120°F within 20 minutes.

   4. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e, loose wiring or structural connections, deposits of foreign materials, etc.

c. By measuring the system leakage rate:

   1. As a part of the overall integrated leakage rate test required by Specification 3.6.1.2, or

   2. By measuring the leakage rate of the system outside of the containment isolation valves at P_a, 34.7 psig, on the schedule required by Specification 4.6.1.2, and including the measured leakage as a part of the leakage determined in accordance with Specification 4.6.1.2.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 At least the following A.C. circuits inside primary containment shall be deenergized*:

a. Circuits supplied by breakers 2AR, and 8BR ~~2BR~~, MCC E-MC-8C.

b. Circuits supplied by panel E-LP-6BAG.

c. Circuits supplied by panel E-LP-3DAG.

d. Circuits supplied by breakers 2BL, 1D, MC-3DR, 2CR

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required A.C. circuits shall be determined to be deenergized at least once per 24 hours** by verifying that the associated circuit breakers are in the tripped condition.

*Except during entry into the drywell.
**Except at least once per 31 days if locked, sealed, or otherwise secured in the tripped condition.

TABLE 3.8.4.2-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

EQUIPMENT	PRIMARY PROTECTION		BACKUP PROTECTION	
a. 6900V Circuit Breakers				
RRC-P-1A	E-CB-RRA (Relay)		E-CB-S5 (Relay)	E-CB-N2/5 (Relay)
RRC-P-1B	E-CB-RRB (Relay		E-CB-S6 (Relay)	E-CB-N2/6 (Relay)
b. 480VAC Fused Disconnects				
RRC-V-67A	MC-7C	25AF	MC-7C	50AF
RCC-V-71A	MC-7C	1.125 ~~1.25~~AF	MC-7C	25AF
RCC-V-72B → RCC-V-72A	MC-8C MC-7C	1.125 ~~1.25~~AF 1.25AF	MC-7C MC-8C	25AF 25AF
RCC-V-17A	MC-7C	1.125 ~~1.25~~AF	MC-7C	25AF
CRA-FN-1C-2	MC-8B	110AF	MC8B ~~SL-81~~	~~1000A Asst~~ 200AF
RRC-V-67B	MC-8C	25AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 90AF
RRC-V-23B	MC-8C	15 ~~12~~AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
RWCU-V-102	MC-8C	5AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
RWCU-V-106	MC-8C	3AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
RRC-V-23A	MC-8C	1AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
RWCU-V-101	MC-8C	3AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
RWCU-V-100	MC-8C	3AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
RCC-V-17B	MC-8C	1.25AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
RCC-V-71C	MC-8C	1.25AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
RCC-V-71B	MC-8C	1.25AF	MC-8C ~~SL-81~~	~~1000A Asst~~ 25AF
CRA-FN-1A-2	MC-7B	100 ~~110~~AF	MC-7B	~~150AF~~ 200AF
~~MT-HOI-18~~	~~MC-3D-A~~	~~50AF~~	~~MC-3D~~	~~200ACB~~
CRA-FN-1A-1	MC-7B	100 ~~70~~AF	MC-7B	~~90AF~~ 200AF
CRA-FN-2A-2	MC-7B	60 ~~50~~AF	MC-7B	90AF
CRA-FN-2A-1	MC-7B	150 ~~110~~AF	MC-7B	~~110AF~~ 300AF
CRA-FN-5A	MC-7B	25AF	MC-7B ~~SL-71~~	~~800A Asst~~ 50AF
CRA-FN-4A	MC-7B	15AF	MC-7B SL-71	~~800A Asst~~ 50AF
CRA-FN-5C	MC-7B	25AF	MC-7B ~~SL-71~~	~~800A Asst~~ 50AF
CRA-FN-3A	MC-7B	25AF	MC-7B ~~SL-71~~	~~800A Asst~~ 50AF
~~MT-HOI-19C~~	~~MC-3D-A~~	~~10AF~~	~~MC-3D~~	~~200ACB~~
CRA-FN-1B-2	MC-8B	100 ~~110~~AF	MC-8B	~~150AF~~ 200AF
CRA-FN-1B-1	MC-8B	100 ~~70~~AF	MC-8B	~~90AF~~ 200AF
CRA-FN-1C-1	MC-8B	100 ~~70AF~~AF	MC-8B	~~90AF~~ 200AF
CRA-FN-2B-1	MC-8B	150 ~~110~~AF	MC8B ~~SL-81~~	~~1000A Asst~~ 300AF
CRA-FN-2B-2	MC-8B	60 ~~30~~AF	MC8B ~~SL-81~~	~~1000A Asst~~ 90AF
MS-V-16	MC-8B-A	3.5 ~~1.25~~AF	MC-8B	125ACB
RWCU-V-1	MC-8B-A	5.6AF	MC-8B	125ACB
RHR-V-9	MC-8B-A	15AF	MC-8B	125ACB
RCIC-V-63	MC-8B-A	15AF	MC-8B	125ACB
RCC-V-40	MC-8B-A	3.2AF	MC-8B	125ACB
RHR-V-123B	MC-8B-A	1.125AF	MC-8B	125ACB

TABLE 3.8.4.2-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

EQUIPMENT	PRIMARY PROTECTION		BACKUP PROTECTION	
RCIC-V-76	MC-8B-A	1.125 .4AF	MC-8B	125ACB
CRA-FN-5B	MC-8B	25AF	MC-8B ~~SL-81~~	~~1000A Asst~~ 50AF
CRA-FN-5D	MC-8B	25AF	MC-8B ~~SL-81~~	~~1000A Asst~~ 50AF
CRA-FN-3B	MC-8B	25AF	MC-8B ~~SL-81~~	~~1000A Asst~~ 50AF
CRA-FN-3C	MC-8B	25AF	MC-8B ~~SL-81~~	~~1000A Asst~~ 50AF
CRA-FN-4B	MC-8B	15AF	MC-8B ~~SL-81~~	~~1000A Asst~~ 25AF
RCC-V-72B	MC-8C	1.25AF	SL-81	1000A Asst
MS-V-1	MC-8C-B	14AF	MC-8C-B	~~200ACB~~ 25AF
MS-V-2	MC-8C-B	14AF	MC-8C-B	~~200ACB~~ 25AF
MS-V-5	MC-8C-B	14AF	MC-8C-B	~~200ACB~~ 25AF
RHR-V-123A	MC-8B-A	1.125 ~~2.25~~AF	MC 8B ~~MC-81~~	125ACB

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TABLE 3.8.4.3-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED
a.	CAC-V-2	Containment Atmospheric Control System
	CAC-V-4	
	CAC-V-6	
	CAC-V-8	
	CAC-V-11	
	CAC-V-13	
	CAC-V-15	
	CAC-V-17	
b.	CIA-V-20	Containment Instrument Air System
	CIA-V-30A	
	CIA-V-30B	
c.	FPC-V-153	Fuel Pool Cooling System
	FPC-V-154	
	FPC-V-156	
d.	HPCS-V-1	High Pressure Core Spray System
	HPCS-V-4	
	HPCS-V-10	
	HPCS-V-11	
	HPCS-V-12	
	HPCS-V-15	
	HPCS-V-23	
e.	LPCS-V-1	Low Pressure Core Spray System
	LPCS-V-5	
	LPCS-FCV-11	
	LPCS-V-12	
f.	MS-V-1	Main Steam System
	MS-V-2	
	MS-V-5	
	MS-V-16	
	MS-V-19	
	MS-V-20	
	MS-V-67A	
	MS-V-67B	
	MS-V-67C	
	MS-V-67D	
	MS-V-146	
g.	MSLC-V-1A	Main Steam Isolation Valve Leakage Control System
	MSLC-V-1B	
	MSLC-V-1C	
	MSLC-V-1D	
	MSLC-V-2A	
	MSLC-V-2B	
	MSLC-V-2C	
	MSLC-V-2D	
	MSLC-V-3A	
	MSLC-V-3B	
	MSLC-V-3C	
	MSLC-V-3D	
	MSLC-V-4	
	MSLC-V-5	
	MSLC-V-9	
	MSLC-V-10	
h.	RCC-V-5	Reactor Closed Cooling Water System
	RCC-V-6	
	RCC-V-17A	
	RCC-V-17B	
	RCC-V-21	
	RCC-V-40	
	RCC-V-71A	
	RCC-V-71B	
	RCC-V-71C	
	RCC-V-72A	
	RCC-V-72B	
	RCC-V-104	
	RCC-V-129	
	RCC-V-130	
	RCC-V-131	
i.	RCIC-V-1	Reactor Core Isolation Cooling System
	RCIC-V-8	
	RCIC-V-10	
	~~RCIC-V-12~~	
	RCIC-V-13	
	RCIC-V-19	
	RCIC-V-22	
	RCIC-V-31	

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TABLE 3.8.4.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED
i.	RCIC-V-45	Reactor Core Isolation Cooling System
	RCIC-V-46	
	RCIC-V-59	
	RCIC-V-63	
	~~RCIC-V-64~~	
	RCIC-V-68	
	RCIC-V-69	
	RCIC-V-76	
	RCIC-V-110	
	RCIC-V-113	
j.	RFW-V-65A	Reactor Feedwater System
	RFW-V-65B	
k.	RHR-V-3A	Residual Heat Removal System
	RHR-V-3B	
	RHR-V-4A	
	RHR-V-4B	
	RHR-V-4C	
	RHR-V-6A	
	RHR-V-6B	
	RHR-V-8	
	RHR-V-9	
	~~RHR-V-11A~~	
	~~RHR-V-11B~~	
	~~RHR-V-12A~~	
	~~RHR-V-12B~~	
	RHR-V-16A	
	RHR-V-16B	
	RHR-V-17A	
	RHR-V-17B	
	RHR-V-21	
	RHR-V-23	
	RHR-V-24A	
	RHR-V-24B	
	~~RHR-V-26A~~	
	~~RHR-V-26B~~	
	RHR-V-27A	
	RHR-V-27B	
	RHR-V-40	
	RHR-V-42A	
	RHR-V-42B	
	RHR-V-42C	
	RHR-V-47A	
	RHR-V-47B	
	RHR-V-48A	
	RHR-V-48B	
	RHR-V-49	
	~~RHR-V-52A~~	
	~~RHR-V-52B~~	
	RHR-V-53A	
	RHR-V-53B	
	RHR-V-64A	
	RHR-V-64B	
	RHR-V-64C	
	RHR-V-68A	
	RHR-V-68B	
	RHR-V-73A	
	RHR-V-74A	
	RHR-V-74B	
	~~RHR-V-87A~~	
	~~RHR-V-87B~~	
	RHR-V-115	
	RHR-V-116	
	RHR-V-123A	
	RHR-V-123B	
	~~RHR-V-124A~~	
	~~RHR-V-124B~~	
	~~RHR-V-125A~~	
	~~RHR-V-125B~~	
	RHR-V-134A	
	RHR-V-134B	
l.	RRC-V-16A	Reactor Recirculation System
	RRC-V-16B	
	RRC-V-23A	
	RRC-V-23B	
	~~RRC-V-67A~~	
	~~RRC-V-67B~~	
m.	RWCU-V-1	Reactor Water Cleanup System
	RWCU-V-4	
	RWCU-V-31	
	RWCU-V-34	

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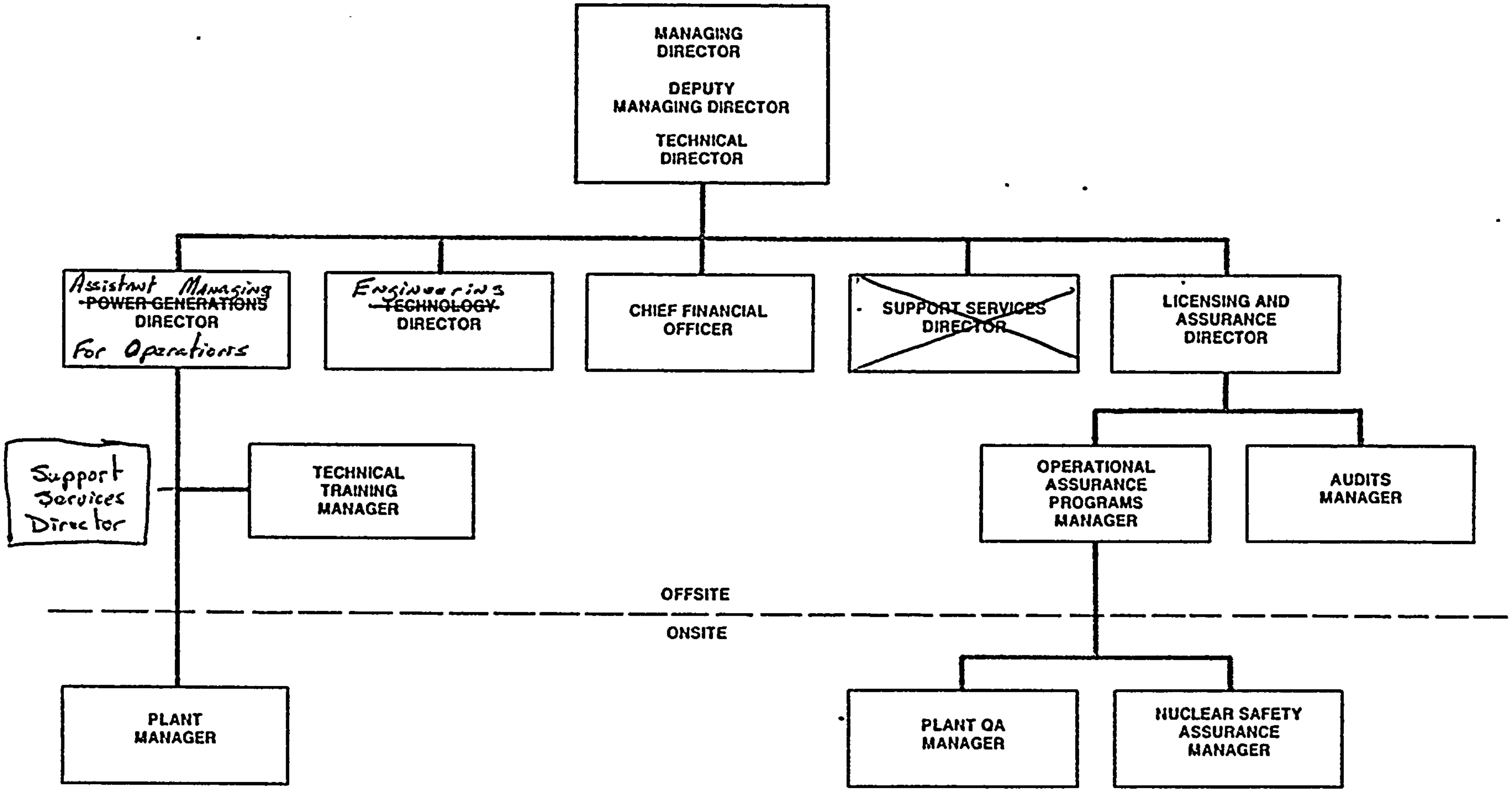

FIGURE 6.2.1-1
OFFSITE ORGANIZATION

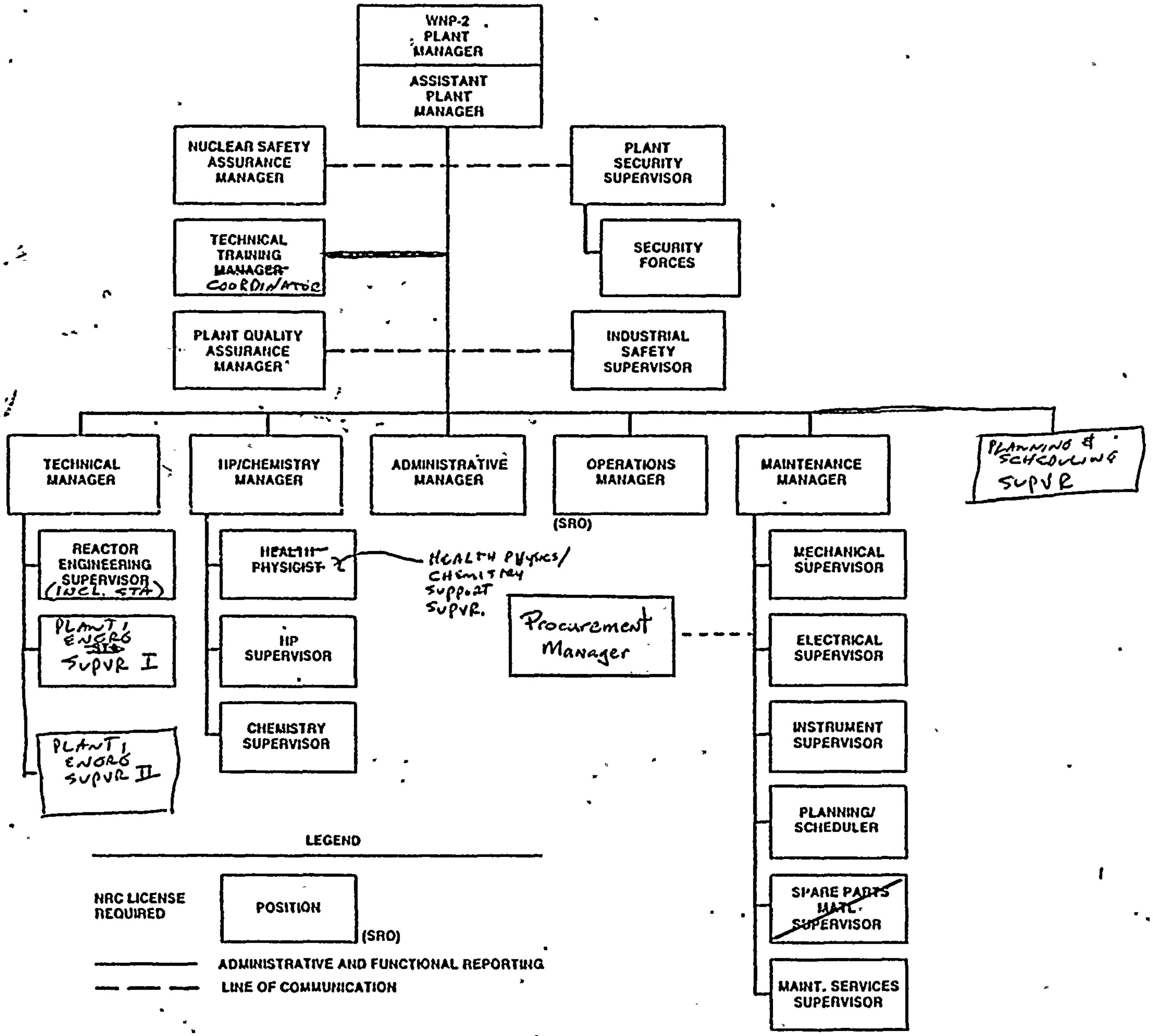

FIGURE 6.2.2-1a
UNIT ORGANIZATION

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Technical Training ~~Manager~~ Coordinator, shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS COMMITTEE (POC)

FUNCTION

6.5.1.1 The POC shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The POC shall be composed of the:

Chairman:	Plant Manager
Vice Chairman:	Assistant Plant Manager
Member:	Operations Manager
Member:	Technical Manager
Member:	Maintenance Manager
Member:	Administrative Manager
Member:	Plant QA/QC Manager
Member:	Health Physics/Chemistry Manager

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the POC Chairman or Vice Chairman to serve on a temporary basis.

MEETING FREQUENCY

6.5.1.4 The Plant Operations Committee shall meet at least once per calendar month and as convened by the POC Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the POC necessary for the performance of the POC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or Vice Chairman and four members including alternates. No more than two alternates shall make up the quorum.

RESPONSIBILITIES

6.5.1.6 The POC shall be responsible for:

a. Review of (1) all proposed procedures required by Specification 6.8 and changes thereto, (2) all proposed programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety;

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to the Appendix A Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the ~~Director of~~ Assistant Managing Director for Operations ~~Power Generation~~ and to the Corporate Nuclear Safety Review Board;

~~f. Review of events requiring 24-hour written notification to the Commission;~~ Review of all reportable events; (this change requested 1/20/84 G02-84-033)

g. Review of unit operations to detect potential hazards to nuclear safety;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager or the Corporate Nuclear Safety Review Board;

i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Corporate Nuclear Safety Review Board; ~~and~~

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Corporate Nuclear Safety Review Board;

k. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the ~~Director of~~ Assistant Managing Director for Operations ~~Power Generation~~ and to the Corporate Nuclear Safety Review Board; and

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

AUTHORITY

6.5.1.7 The POC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specification 6.5.1.6a. through d. prior to their implementation.

AUTHORITY (Continued)

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through e. constitutes an unreviewed safety question, as defined in 10 CFR 50.59.

c. Provide written notification within 24 hours to the Assistant Managing Director for Operations ~~Director of Power Generation~~ and the Corporate Nuclear Safety Review Board of disagreement between the POC and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The POC shall maintain written minutes of each POC meeting that, at a minimum, document the results of all POC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Assistant Managing Director for Operations ~~Director of Power Generation~~ and the Corporate Nuclear Safety Review Board.

6.5.2 CORPORATE NUCLEAR SAFETY REVIEW BOARD (CNSRB)

FUNCTION

6.5.2.1 The CNSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The CNSRB shall report to and advise the Managing Director on those areas of responsibility in Specifications 6.5.2.7 and 6.5.2.8.

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[illegible]

[illegible]

[illegible]

[illegible]

ADMINISTRATIVE CONTROLS

COMPOSITION

6.5.2.2 The CNSRB shall be composed of nine members appointed in writing by the Managing Director from his senior technical staff and/or from outside the Supply System. He shall designate from the members, a Chairman, an Alternate Chairman, and an Executive Secretary. The Directorates of ~~Power Generation,~~ Engineering ~~Technology;~~ Support Services, and Licensing and Assurance shall be represented. The qualifications of all members ~~, appointed and permanent,~~ shall meet the minimum requirements of Section 4.7 of ANSI/ANS 3.1-1981 and have, cumulatively, expertise in the areas listed in Specification 6.5.2.1, as a minimum.

plant organization and the

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNSRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNSRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNSRB Committee to provide expert advice to the CNSRB.

MEETING FREQUENCY

6.5.2.5 The CNSRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the CNSRB necessary for the performance of the CNSRB review and audit functions of these Technical Specifications shall consist of the Chairman or the alternate Chairman and at least four CNSRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The CNSRB shall review:

a. The safety evaluations for (1) changes to procedures, equipment or systems and (2) tests or experiments completed under the provision of 10 CFR 50.59 to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

AUDITS (Continued)

h. The fire protection equipment and program implementation, at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer(s) or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least once every third year; and

i. Any other area of unit operation considered appropriate by the CNSRB or the Managing Director.

j. The radiological environmental monitoring program and the results thereof at least once per 12 months.

k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

l. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

m. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

RECORDS

6.5.2.9 Records of CNSRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each CNSRB meeting shall be prepared, approved, and forwarded to the Managing Director 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 above, shall be prepared, approved, and forwarded to the Managing Director within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Managing Director and to the management positions responsible for the areas audited within 30 days after completion of the audit. ~~by the auditing organization.~~

6.6 ~~REPORTABLE OCCURRENCE ACTION~~ REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE ~~OCCURRENCES.~~ Events:

This change requested 1/20/84, G02-84-033

a. The Commission shall be notified and a report submitted pursuant to the requirements of ~~Specification 6.9, and~~ Section 50.73 to 10 CFR Part 50, and

~~b. Each REPORTABLE OCCURRENCE requiring 24-hour notification to the Commission shall be reviewed by the POC, and the results of this review shall be submitted to the CNSRB and the Director of Power Generation.~~

b. Each REPORTABLE EVENT shall be reviewed by the POC, and the results of this review shall be submitted to the CNSRB and the ~~Director of Power Generation.~~ Assistant Managing Director for Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The ~~Director of Power Generation~~ Assistant Managing Director for Operations and the CNSRB shall be notified ~~within 24 hours~~.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the POC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the CNSRB, and the ~~Director of Power Generation within 14 days of the violation.~~ Assistant Managing Director for Operations.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

b. The applicable procedures required to implement the requirements of NUREG-0737.

c. Refueling operations.

d. Surveillance and test activities of safety-related equipment.

e. Security Plan implementation.

f. Emergency Plan implementation.

g. Fire Protection Program implementation.

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environmental monitoring.

k. Health Physics/Chemistry Support Program.

6.8.2 Each procedure of Specification 6.8.1a. through j., and changes thereto, shall be reviewed by the POC and shall be approved by the Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

In addition the review and approval of the ~~The~~ implementing procedures supporting item k. in Specification 6.8.1 will be coordinated by the Director of Support Services ~~under the cognizance of the Manager of Radiological Programs~~ who will provide review and approval control. The WNP-2 Health Physics/Chemistry Support Program procedure will be reviewed by POC and approved by the Plant Manager.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS ~~AND REPORTABLE OCCURRENCES~~ (change requested 1/20/84, GO2-84-033)

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

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