UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

WPPSS NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

License No. NPF-21
Amendment No. 6

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Washington Public Power Supply System (WPPSS, also the licensee) dated March 9 and March 13, 1984, and supplemented on April 25 and May 4, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulation set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-21 is hereby amended as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 6, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
Changes to the Technical
Specifications

Date of Issuance: OCT 12 1984

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
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Specifications

Date of Issuance: OCT 12 1984

SER Conclusion is wrong. Revise & return to Patton for approval

LB#2/DL	LB#2/DL	LB#2/DL	OELD	ADYL/DL
RAuluck:dh	EButton	ASchwencer	WPatton	TMNovak
09/16/84	09/18/84	09/21/84	~~09/ /84~~ 10/1/84	09/21/84

10/9

10/4/84

ATTACHMENT TO LICENSE AMENDMENT NO. 6
FACILITY OPERATING LICENSE NO. NPF-21
DOCKET NO. 50-397

Replace the following pages of the Appendix "A" Technical Specifications with enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3/4 3-80	3/4 3-80
3/4 3-81	3/4 3-81
3/4 7-21	3/4 7-21
3/4 5-5	3/4 5-5
3/4 6-43	3/4 6-43
3/4 8-20	3/4 8- 20
3/4 8-23	3/4 8-23
3/4 8-24	3/4 8-24
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6-3	6-3
6-4	6-4
6-8	6-8
6-9	6-9
6-10	6-10
6-11	6-11
6-13	6-13
6-14	6-14
6-16	6-16
3/4 8-13	3/4 8-13
3/4 8-14	3/4 8-14

TABLE 3.3.7.9-1

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	TOTAL NUMBER* OF INSTRUMENTS				
	ID (x/y)	SD (x/y)	TD (x/y)	UD (x/y)	SMD (x/y)
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"					
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		0/9		
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		0/9		
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				

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', automatic sprinklers 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.

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 ≥ 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 psig on decreasing pressure and an alarm setpoint ≥ 135 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.

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.

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 8AR, 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 in cubicles 2BL, 1D, and 2CR of MC-3DA.

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

EQUIPMENT	PRIMARY PROTECTION	BACKUP PROTECTION	
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

EQUIPMENT	PRIMARY PROTECTION		BACKUP PROTECTION	
RRC-V-67A	MC-7C	25AF	MC-7C	50AF
RCC-V-71A	MC-7C	1.125AF	MC-7C	25AF
RCC-V-72A	MC-7C	1.125AF	MC-7C	25AF
RCC-V-72B	MC-8C	1.25AF	MC-8C	25AF
RCC-V-17A	MC-7C	1.125AF	MC-7C	25AF
CRA-FN-1C-2	MC-8B	110AF	MC-8B	200AF
RRC-V-67B	MC-8C	25AF	MC-8C	90AF
RRC-V-23B	MC-8C	15AF	MC-8C	25AF
RWCU-V-102	MC-8C	5AF	MC-8C	25AF
RWCU-V-106	MC-8C	3AF	MC-8C	25AF
RRC-V-23A	MC-8C	1AF	MC-8C	25AF
RWCU-V-101	MC-8C	3AF	MC-8C	25AF
RWCU-V-100	MC-8C	3AF	MC-8C	25AF
RCC-V-17B	MC-8C	1.25AF	MC-8C	25AF
RCC-V-71C	MC-8C	1.25AF	MC-8C	25AF
RCC-V-71B	MC-8C	1.25AF	MC-8C	25AF
CRA-FN-1A-2	MC-7B	100AF	MC-7B	200AF
CRA-FN-1A-1	MC-7B	100AF	MC-7B	200AF
CRA-FN-2A-2	MC-7B	60AF	MC-7B	90AF
CRA-FN-2A-1	MC-7B	150AF	MC-7B	300AF
CRA-FN-5A	MC-7B	25AF	MC-7B	50AF
CRA-FN-4A	MC-7B	15AF	MC-7B	50AF
CRA-FN-5C	MC-7B	25AF	MC-7B	50AF
CRA-FN-3A	MC-7B	25AF	MC-7B	50AF
CRA-FN-1B-2	MC-8B	100AF	MC-8B	200AF
CRA-FN-1B-1	MC-8B	100AF	MC-8B	200AF
CRA-FN-1C-1	MC-8B	100AF	MC-8B	200AF
CRA-FN-2B-1	MC-8B	150AF	MC-8B	300AF
CRA-FN-2B-2	MC-8B	60AF	MC-8B	90AF
MS-V-16	MC-8B-A	3.5AF	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
RCIC-V-76	MC-8B-A	1.125AF	MC-8B	125ACB
CRA-FN-5B	MC-8B	25AF	MC-8B	50AF

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

TABLE 3.8.4.3-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED
a.	CAC-V-2 CAC-V-4 CAC-V-6 CAC-V-8 CAC-V-11 CAC-V-13 CAC-V-15 CAC-V-17	Containment Atmospheric Control System
b.	CIA-V-20 CIA-V-30A CIA-V-30B	Containment Instrument Air System
c.	FPC-V-153 FPC-V-154 FPC-V-156	Fuel Pool Cooling System
d.	HPCS-V-1 HPCS-V-4 HPCS-V-10 HPCS-V-11 HPCS-V-12 HPCS-V-15 HPCS-V-23	High Pressure Core Spray System
e.	LPCS-V-1 LPCS-V-5 LPCS-FCV-11 LPCS-V-12	Low Pressure Core Spray System
f.	MS-V-1 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	Main Steam System

	VALVE NUMBER	SYSTEM(S) AFFECTED
g.	MSLC-V-1A 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	Main Steam Isolation Valve Leakage Control System
h.	RCC-V-5 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	Reactor Closed Cooling Water System
i.	RCIC-V-1 RCIC-V-8 RCIC-V-10 RCIC-V-13 RCIC-V-19 RCIC-V-22 RCIC-V-31	Reactor Core Isolation Cooling System

TABLE 3.8.4.3-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	SYSTEM(S) AFFECTED
i.	RCIC-V-45 RCIC-V-46 RCIC-V-59 RCIC-V-63 RCIC-V-68 RCIC-V-69 RCIC-V-76 RCIC-V-110 RCIC-V-113	Reactor Core Isolation Cooling System
j.	RFW-V-65A RFW-V-65B	Reactor Feedwater System
k.	RHR-V-3A 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-16A RHR-V-16B RHR-V-17A RHR-V-17B RHR-V-21 RHR-V-23 RHR-V-24A RHR-V-24B 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-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-115 RHR-V-116 RHR-V-123A RHR-V-123B RHR-V-134A RHR-V-134B	Residual Heat Removal System
l.	RRC-V-16A RRC-V-16B RRC-V-23A RRC-V-23B	Reactor Recirculation System
m.	RWCU-V-1 RWCU-V-4 RWCU-V-31 RWCU-V-34	Reactor Water Cleanup System

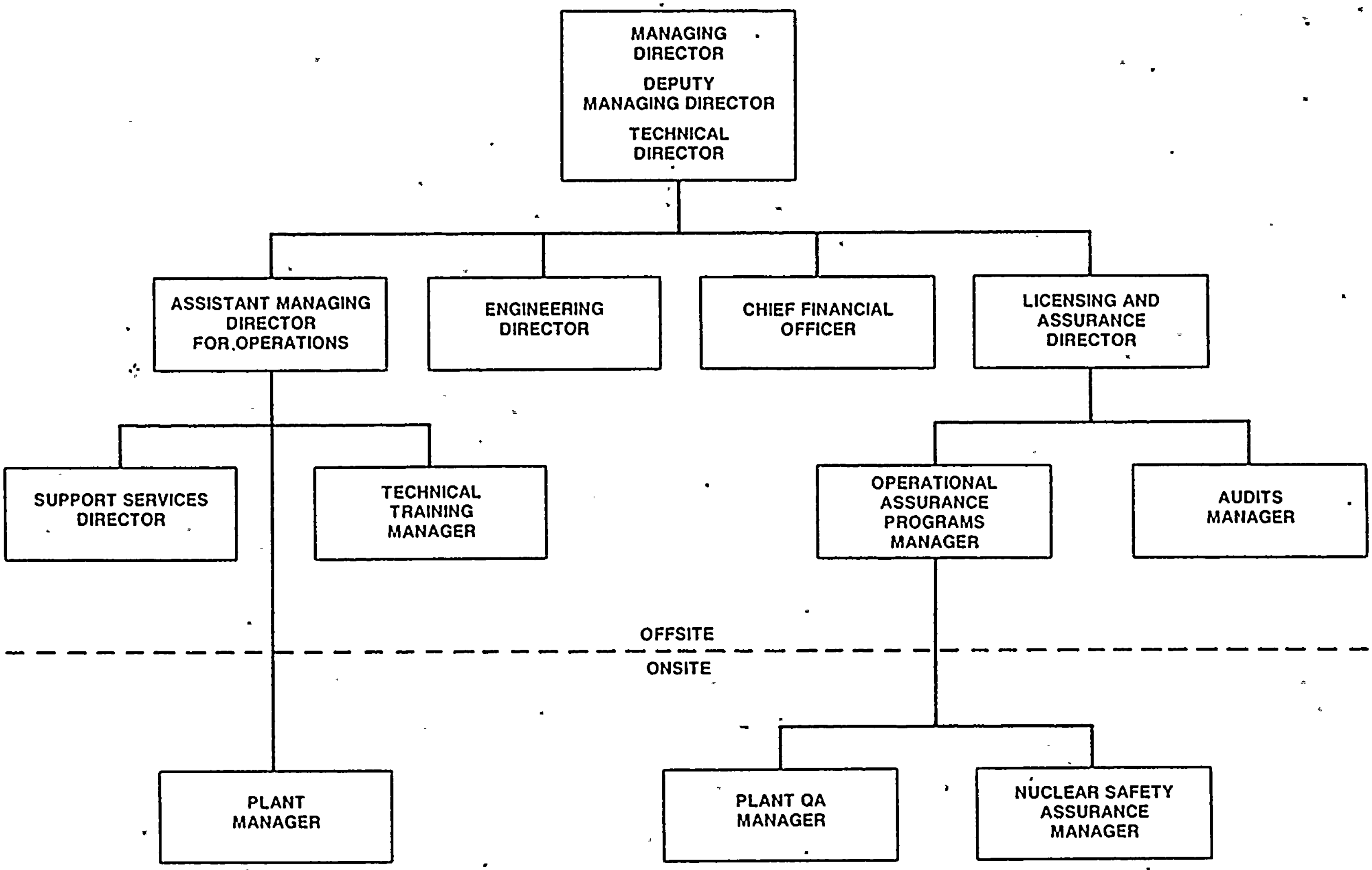

FIGURE 6.2.1-1
OFFSITE ORGANIZATION

840230-2A (2/84)

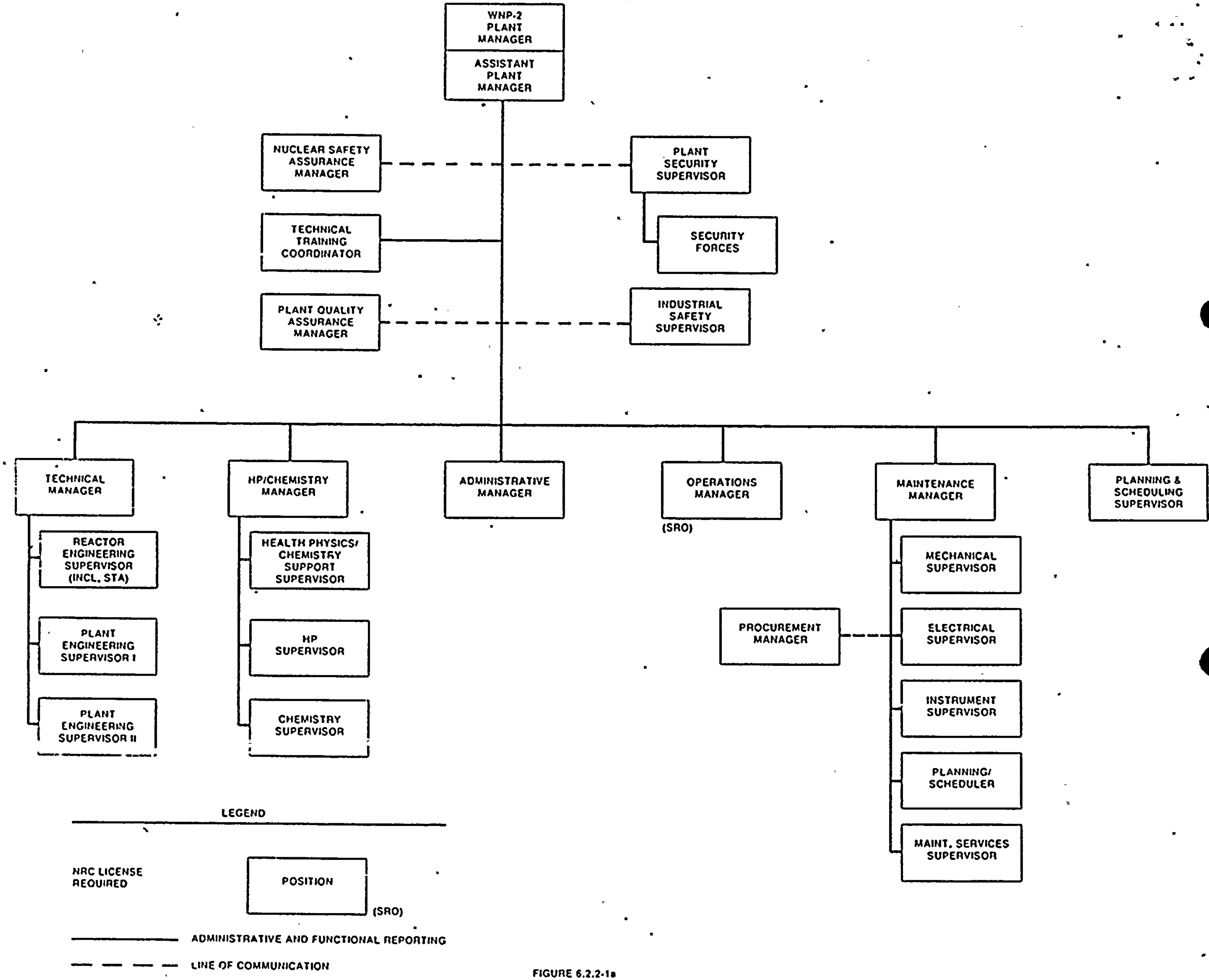

FIGURE 6.2.2-1a
UNIT ORGANIZATION

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 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 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 Assistant Managing Director for Operations and to the Corporate Nuclear Safety Review Board;

f. Review of all REPORTABLE EVENTS;

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;

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 Assistant Managing Director for Operations 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 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 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.

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 plant organization and the Directorates of Engineering, Support Services, and Licensing and Assurance shall be represented. The qualifications of all members 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.

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.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

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

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

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 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 implementing procedures supporting item k. in Specification 6.8.1 will be coordinated by the Director of Support Services, 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

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.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 2 hours for Divisions 1, 2 and 3 when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile while maintaining the battery terminal voltage greater than or equal to 21 volts for the ±24-volt battery, 105 volts for the 125-volt battery, and 210 volts for the 250-volt battery, and 105 volts for the 125-volt Div. 3 battery.

BATT SYS. TIME	0-3 sec	3-13 sec	13-30 sec	30-60 sec	1-60 min	1-2 hr
±24 v	17	17	17	17	17	17
125 v Div 1	671	252	237	153	86	86
125 v Div 2	426	224	209	125	99	99
125 v Div 3	73.4	73.4	73.4	73.4	18.4	19.4
250 v	1462	567	567	567	432	396

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. At this once per 60-month interval, this performance discharge test may be performed in lieu of the battery service test.

f. At least once per 18 months during shutdown 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.

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)		≥ 1.195	Not more than 0.020 below the average of all connected cells
	≥ 1.200(b)	Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(b)

(a)Corrected for electrolyte temperature. Level correction will be used when electrolyte level is outside the normal range.

(b)Or battery charging current is less than (2) amperes when on float charge.

(c)May be corrected for average electrolyte temperature.

(1)For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2)For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3)With any Category B parameter not within its allowable value declare the battery inoperable.