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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 100
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District dated December 10, 1985, as supplemented by submittal dated January 13, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to 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 licensee is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

(2) Technical Specification

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 100, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 20, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 100

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

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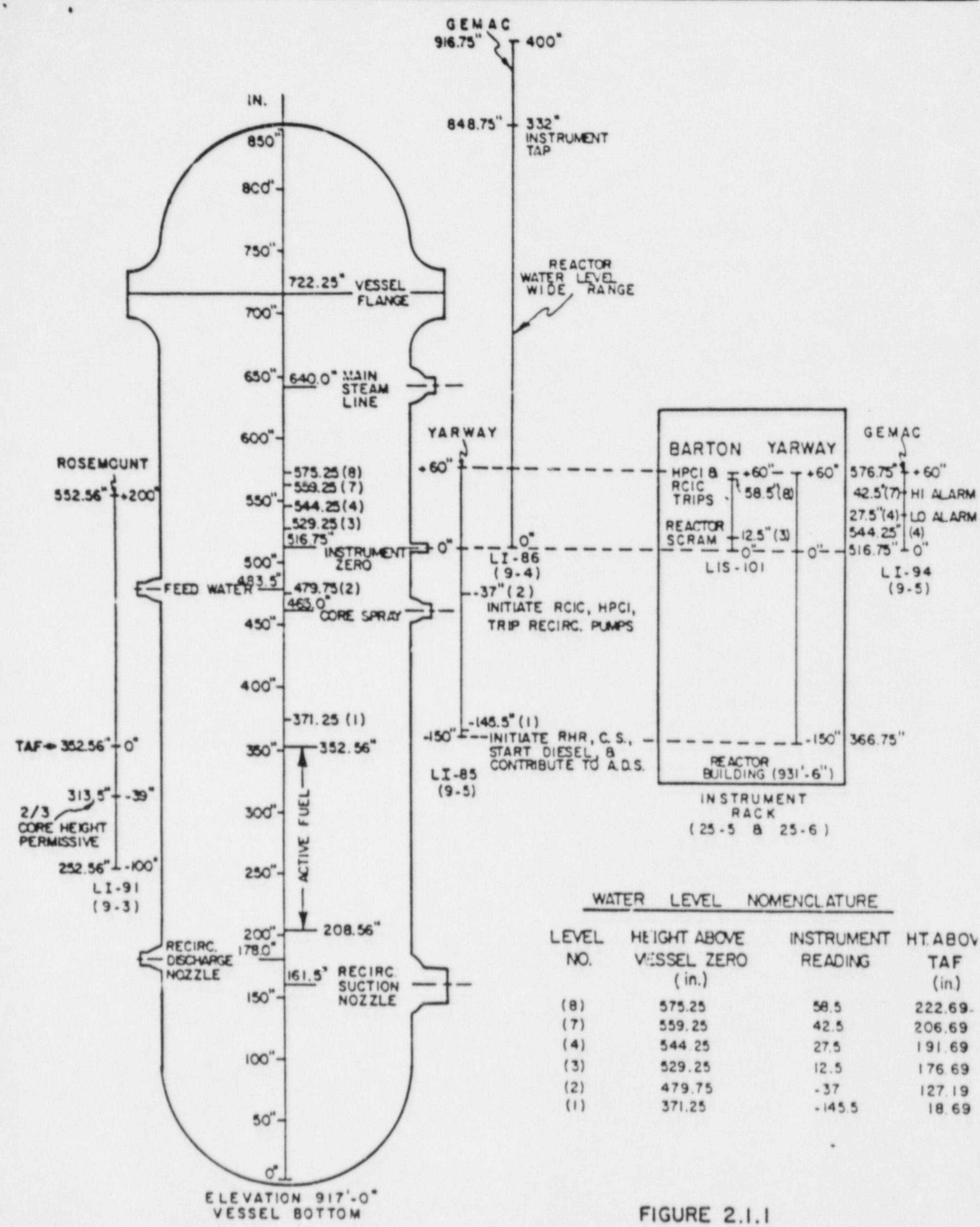

WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (in.)	INSTRUMENT READING	HT. ABOVE TAF (in.)
(8)	575.25	58.5	222.69
(7)	559.25	42.5	206.69
(4)	544.25	27.5	191.69
(3)	529.25	12.5	176.69
(2)	479.75	-37	127.19
(1)	371.25	-145.5	18.69

FIGURE 2.1.1

REACTOR WATER LEVEL INDICATION CORRELATION

3.1 BASES (Cont.d)

ance paragraph VII.5.7 FSAR). Thus the IRM System is not required in the "Run" mode. The APRM's cover only the power range. The IRM's and APRM's provide adequate coverage in the startup and intermediate range.

The requirement to have the scram functions indicated in Table 3.1.1 operable in the Refuel mode assures that shifting to the Refuel mode during reactor power operation does not diminish the protection provided by the reactor protection system.

Turbine stop valve scram occurs at 10% of valve closure. Below 30% of the rated turbine first stage pressure, the scram signal due to the turbine stop valve closure may be bypassed because the flux and pressure scrams are adequate to protect the reactor. The actual scram bypass setpoint, however, is implemented at ≤25% of rated turbine first stage pressure (or 179 psig) to compensate for possible turbine trips during bypass valve testing. During bypass valve testing, the first stage pressure is reduced due to flow through the bypass valves while reactor power is unchanged.

Turbine control valves fast closure initiates a scram based on pressure switches sensing Electro-Hydraulic Control (EHC) system oil pressure. The switches are located on the Control Valve Emergency Trip oil header, and detects the loss of oil to hold the valves open.

This scram signal is also bypassed when turbine first stage pressure is less than 179 psig.

The requirements that the IRM's be inserted in the core when the APRM's read 2.5 indicated on the scale in the Startup and Refuel modes assures that

4.1 BASES (Cont.d)

zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the Flow Biasing Network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup; i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis.

2. Vacuum tube or semi-conductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc., drift specifications call for drift to be less that 0.4%/month; i.e., in the period of a month a maximum drift of 0.4% could occur, thus providing for adequate margin.

COOPER NUCLEAR STATION
TABLE 3.2.A (Page 1)
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION INSTRUMENTATION

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured (2)
Main Steam Line High Rad.	RMP-RM-251, A,B,C,&D	$\leq$ 3 Times Full Power	2	A or B
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D #1	$\geq$+12.5" Indicated Level	2(4)	A or B
Reactor Low Low Water Level	NBI-LIS-57 A & B #2 NBI-LIS-58 A & B #2	$\geq$-37" Indicated Level	2	A or B
Reactor Low Low Low Water Level	NBI-LIS-57 A & B #1 NBI-LIS-58 A & B #1	$\geq$-145.5" Indicated Level	2	A or B
Main Steam Line Leak Detection	MS-TS-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	$\leq$ 200°F	2(6)	B
Main Steam Line High Flow	MS-dPIS-116 A,B,C,&D 117, 118, 119	$\leq$ 140% of Rated Steam Flow	2(3)	B
Main Steam Line Low Pressure	MS-PS-134, A,B,C,&D	$\geq$ 825 psig	2(5)	B
High Drywell Pressure	PC-PS-12, A,B,C,&D	$\leq$ 2 psig	2(4)	A or B
High Reactor Pressure	RR-PS-128 A & B	$\leq$ 75 psig	1	D
Main Condenser Low Vacuum	MS-PS-103, A,B,C,&D	$\geq$ 7" Hg (7)	2	A or B
Reactor Water Cleanup System High Flow	RWCU-dPIS-170 A & B	$\leq$ 200% of System Flow	1	C

COOPER NUCLEAR STATION

TABLE 4.2.B (Page 2)

RHR SYSTEM TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrumentation				
1. Drywell High Pressure	PC-PS-101, A, B, C & D	Once/Month (1)	Once/3 Months	None
2. Reactor Vessel Shroud Level	NBI-LITS-73, A & B #1	Once/Month (1)	Once/3 Months	Once/Day
3. Reactor Low Pressure	RR-PS-128 A & B	Once/Month (1)	Once/3 Months	None
4. Reactor Low Pressure	NBI-PS-52 A & C	Once/Month (1)	Once/3 Months	None
	NBI-PIS-52 B & D			
5. Drywell Press.-Containment Spray	PC-PS-119, A,B,C & D	Once/Month (1)	Once/3 Months	None
6. RHR Pump Discharge Press.	RHR-PS-120, A,B,C & D	Once/Month (1)	Once/3 Months	None
7. RHR Pump Discharge Press.	RHR-PS-105, A,B,C & D	Once/Month (1)	Once/3 Months	None
8. RHR Pump Low Flow Switch	RHR-dPIS-125 A & B	Once/Month (1)	Once 3 Months	None
9. RHR Pump Start Time Delay	RHR-TDR-K70, A & B	Once/Month (1)	Once/Oper. Cycle	None
10. RHR Injection Valve Close T.D.	RHR-TDR-K45 1A & 1B	Once/Month (1)	Once/Oper. Cycle	None
11. RHR Pump Start Time Delay	RHR-TDR-K75, A & B	Once/Month (1)	Once/Oper. Cycle	None
12. RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	Once/Month (1)	Once/Oper. Cycle	None
13. RHR Cross Tie Valve Position	RHR-LMS-2	Once/Month (1)	N.A.	
14. Low Voltage Relays	27 X 3/1A	(7)		None
15. Low Voltage Relays	27 X 3/1B	(7)		None
16. Low Voltage Relays	27 x 2/1F, 27 X 2/1G	(7)		None
17. Low Voltage Relays	27 X 1/1F, 27 X 1/1G	(7)		None
18. Pump Disch. Line Press. Low	CM-PS-266, CM-PS-270	Once/3 Months	Once/3 Months	None
19. Emergency buses Undervoltage Relays (Degraded Voltage)	27/1F-2, 27/1FA-2, 27/1G-2, 27/1GB-2	Once/Month	Once/18 Months	Once/12 hrs.
20. Emergency Buses Loss of Voltage Relays	27/1F-1, 27/1FA-1, 27/1G-1, 27/1GB-1, 27/ET-1, 27/ET-2	Once/Month	Once/18 Months	Once/12 hrs.
21. Emergency Buses Undervoltage Relays Timers	27X7/1F, 27X7/1G	Once/Month	Once/18 Months	None

COOPER NUCLEAR STATION
TABLE 4.2.B (Page 4)
HPCI TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
1. Reactor Low Water Level	NBI-LIS-72, A,B,C, & D, #3	Once/Month (1)	Once/3 Months	Once/Day
2. Reactor High Water Level	NBI-LIS-101, (B & D #2)	Once/Month (1)	Once/3 Months	Once/Day
3. High Drywell Pressure	14A - K5 A & B	(7)	(7)	None
	14A - K6 A & B	(7)	(7)	None
4. HPCI Turbine High Exhaust Press.	HPCI-PS-97 A & B	Once/Month (1)	Once/3 Months	None
5. HPCI Pump Low Suction Press.	HPCI-PS-84-1	Once/Month (1)	Once/3 Months	None
6. HPCI Pump Low Discharge Flow	HPCI-FS-78	Once/Month (1)	Once/3 Months	None
7. HPCI Low Steam Supply Press.	HPCI-PS-68, A,B,C, & D	Once/Month (1)	Once/3 Months	None
8. HPCI Steam Line High ΔP	HPCI-dPIS-76	Once/Month (1)	Once/3 Months	None
	HPCI-dPIS-77	Once/Month (1)	Once/3 Months	None
9. HPCI Steam Line Space High Temp.	HPCI-TS-101, A,B,C, & D 102, 103, 104, HPCI-TS-125, 126, 127, 128 RHR-TS-150,151,152,153,154, 155,156,157,158,159,160,161	Once/Month (1)	Once/Oper. Cycle	None
10. Emergency Cond. Stg. Tk. Low Level	HPCI-LS-74 A & B	Once/Month (1)	Once/3 Months	None
	HPCI-LS-75 A & B	Once/Month (1)	Once/3 Months	None
11. Suppression Chamber High Water Level	HPCI-LS-91 A & B	Once/Month (1)	Once/3 Months	None
12. HPCI Gland Seal Cond. Hotwell Level	HPCI-LS-356 B	Once/Month (1)	Once/3 Months	None
	HPCI-LS-356 A	Once/Month (1)	Once/3 Months	None
13. HPCI Control Oil Pressure Low	HPCI-PS-2787-H	Once/Month (1)	Once/3 Months	None
	HPCI-PS-2787-L	Once/Month (1)	Once/3 Months	None
14. Turbine Condition Supr. Alarm Actuation Timer	HPCI-TDR-K14	Once/Month (1)	Once/Oper. Cycle	None
15. Pump Disch. Line Low Press.	CM-PS-268	Once/3 Months	Once/3 Months	None
16. HPCI Turbine Stop Valve Mon.	HPCI-LMS-4	Once/Month	N.A.	None
17. Sup. Chamber HPCI Suction Vlv.	HPCI-LMS-2	Once/Month	N.A.	None
18. HPCI Steam Line High ΔP Actuation Timer	HPCI-TDR-K33,	Once/Month	Once/Oper. Cycle	None
	HPCI-TDR-K43	Once/Month	Once/Oper. Cycle	None

Logic (4)(6)

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
1. Logic Bus Power Monitor		Once/6 Months	N.A.	
2. HPCI Initiation		Once/6 Months	N.A.	
3. HPCI Turbine Trip		Once/6 Months	N.A.	

5. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator who withdraws control rods according to written sequences. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod with drawal errors when this condition exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR equals the operating limit as defined on Figure 3.11, and LHGR = as defined in 1.0.A.4). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the Division Manager of Nuclear Operations.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the safety limit. The limiting power transient is defined in Reference 3. Analysis of this transient shows that the negative reactivity rates resulting from the scram provide the required protection, and MCPR remains greater than the safety limit.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Cooper Nuclear Station.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays; at this point, the pilot scram solenoid deenergizes. Approximately 120 milliseconds later, 3.6.E **Jet Pumps**

1. Whenever the reactor is in the start-up or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, or if two or more jet pump flow instruments failures occur and cannot be corrected within 24 hours, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

F. Recirculation Pump Flow Mismatch

1. Deleted.

2. Following one recirculation pump operation, the discharge valve of the low speed recirculation pump may not be opened unless the speed of the faster pump is equal to or less than 50% of its rated speed.

G. Inservice Inspection

To be considered operable, components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for continued service of ASME Code Class 1, 2, and 3 components except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

4.6.E. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

a. The recirculation pump flow differs by more than 15% from the established speed flow characteristics.

b. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%.

c. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from the mean of all jet pump differential pressures by more than 10%.

F. Recirculation Pump Flow Mismatch

1. Deleted.

G. Inservice Inspection

Inservice inspection shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle riser system failure.

F. Recirculation Pump Flow Mismatch

Requiring the discharge valve of the lower speed loop to remain closed until the speed of faster pump is equal to or less than 50% of its rated speed provides assurance when going from one to two recirculation pump operation that excessive vibration of the jet pump risers will not occur.

G. Inservice Inspection

The inservice inspection program conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, the inspection of components conforms to the requirements of ASME Code Class 1, 2, and 3 components contained in Section XI of the ASME Boiler and Pressure Vessel Code. If a Code required inspection is impractical, a request for a deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

Deviations which are needed from the procedures prescribed in Section XI of the ASME Code and applicable Addenda will be reported to the Commission prior to the beginning of each 10-year inspection period if they are known to be required at that time. Deviations which are identified during the course of inspection will be reported quarterly throughout the inspection period.

TABLE 3.7.2

TESTABLE PENETRATIONS WITH DOUBLE O-RING SEALS

PEN. NO	DESCRIPTION
X-1A	Drywell equipment hatch
X-1B	Drywell equipment hatch
X-2	Drywell airlock door
X-4	Drywell head access hatch
X-6	CRD removal hatch
X-35A	TIP "D" Penetration
X-35B	TIP "A" Penetration
X-35C	TIP "C" Penetration
X-35D	TIP "B" Penetration
X-35E	TIP N_2 Purge Connection
X-200A	Suppression chamber access hatch
X-200B	Suppression chamber access hatch
X-213B	Suppression chamber drain flange
----	Drywell head
----	Stabilizer Assembly Inspection Ports (8)

TABLE 3.7.4
PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-7A	MS-AO-80A and MS-AO-86A, Main Steam Isolation Valves	Air
X-7B	MS-AO-80B and MS-AO-86B, Main Steam Isolation Valves	Air
X-7C	MS-AO-80C and MS-AO-86C, Main Steam Isolation Valves	Air
X-7D	MS-AO-80D and MS-AO-86D, Main Steam Isolation valves	Air
X-8	MS-MO-74 and MS-MO-77, Main Steam Line Drain	Air
X-9A	RF-15CV and RF-16CV, Feedwater Check Valves	Air
X-9A	RCIC-AO-22, RCIC-MO-17, and RWCU-15CV, RCIC/RWCU Connection to Feedwater	Air
X-9B	RF-13CV and RF-14CV, Feedwater Check Valves	Air
X-9B	HPCI-AO-18 and HPCI-MO-57, HPCI Connection to Feedwater	Air
X-10	RCIC-MO-15 and RCIC-MO-16, RCIC Steam Line	Air
X-11	HPCI-MO-15 and HPCI-MO-16, HPCI Steam Line	Air
X-12	RHR-MO-17 and RHR-MO-18, RHR Suction Cooling	Air
X-13A	RHR-MO-25A and RHR-MO-27A, RHR Supply to RPV	Air
X-13B	RHR-MO-25B and RHR-MO-27B, RHR Supply to RPV	Air
X-14	RWCU-MO-15 and RWCU-MO-18, Inlet to RWCU System	Air
X-16A	CS-MO-11A and CS-MO-12A, Core Spray to RPV	Air
X-16B	CS-MO-11B and CS-MO-12B, Core Spray to RPV	Air
X-17	RHR-MO-32 and RHR-MO-33, RPV Head Spray	Air
X-18	RW-732AV and RW-733AV, Drywell Equipment Sump Discharge	Air
X-19	RW-765AV and RW-766AV, Drywell Floor Drain Sump Discharge	Air
X-25	PC-232MV and PC-238AV, Purge and Vent Supply to Drywell	Air
X-25	ACAD-1305MV and ACAD-1306MV, Supply to Drywell	Air
X-26	PC-231MV, PC-246AV, and PC-306MV Purge and Vent Exhaust from Drywell	Air
X-26	ACAD-1310MV, Bleed from Drywell	Air

TABLE 3.7.4 (page ...

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-39A	RHR-MO-26A and RHR-MO-31A, Drywell Spray Header Supply	Air
X-39B	RHR-MO-26B and RHR-MO-31B, Drywell Spray Header Supply	Air
X-39B	ACAD-1311MV and ACAD-1312MV, Supply to Drywell	Air
X-41	RRV-740AV and RRV-741AV, Reactor Water Sample Line	Air
X-42	SLC-12CV and SLC-13CV, Standby Liquid Control	Air
X-205	PC-233MV and PC-237AV, Purge and Vent Supply to Torus	Air
X-205	PC-13CV and PC-243AV, Torus Vacuum Relief	Air
X-205	PC-14CV and PC-244AV, Torus Vacuum Relief	Air
X-205	ACAD-1303MV and ACAD-1304MV, Supply to Torus	Air
X-210A	RCIC-MO-27 and RCIC-13CV, RCIC Minimum Flow Line	Air
X-210A	RHR-MO-21A, RHR to Torus	Air
X-210A	RHR-MO-16A, RHR-10CV, and RHR-12CV, RHR Minimum Flow Line	Air
X-210B	RHR-MO-21B, RHR to Torus	Air
X-210B	HPCI-17CV and HPCI-MO-25, HPCI Minimum Flow Line	Air
X-210B	RHR-MO-16B, RHR-11CV, and RHR-13CV, RHR Minimum Flow Line	Air
X-210A and 211A	RHR-MO-34A, RHR-MO-38A, and RHR-MO-39A, RHR to Torus	Air
X-210B and 211B	RHR-MO-34B, RHR-MO-38B, and RHR-MO-39B, RHR to Torus	Air
X-211B	ACAD-1301MV and ACAD-1302MV, Supply to Torus	Air
X-212	RCIC-15CV and RCIC-37, RCIC Turbine Exhaust	Air
X-214	HPCI-15CV and HPCI-44, HPCI Turbine Exhaust	Air
X-214	HPCI-AO-70 and HPCI-AO-71, HPCI Turbine Exhaust Drain	Air
X-214	RHR-MO-166A and RHR-MO-167A RHR Heat Exch. Vent	Air
X-214	RHR-MO-166B and RHR-MO-167B RHR Heat Exch. Vent	Air
X-220	PC-230MV, PC-245AV, and PC-305MV Purge and Vent Exhaust from Torus	Air
X-220	ACAD-1308MV, Bleed from Torus	Air
X-221	RCIC-12CV and RCIC-42, RCIC Vacuum Line	Air
X-222	HPCI-50 and HPCI-16CV, HPCI Turbine Drain	Air

6.2 REVIEW AND AUDIT

6.2.1 The organization and duties of committees for the review and audit of station operation shall be as outlined below:

A. Station Operations Review Committee (SORC)

1. Membership:

a. Chairman: Division Manager of Nuclear Operations

b. Technical Staff Manager

c. Operations Manager

d. Technical Manager

e. Operations Supervisor

f. Maintenance Supervisor

g. Instrument and Control Supervisor

h. Chemistry and Health Physics Supervisor

i. Plant Engineering Supervisor

j. Operations Engineering Supervisor

k. Computer Applications Supervisor

l. Maintenance Manager

m. Quality Assurance Manager - non-voting member.

Alternate members shall be appointed in writing by the Division Manager of Nuclear Operations to serve on a temporary basis; however, no more than two alternates shall serve on the Committee at any one time.

2. Meeting Frequency: Monthly, and as required on call of the Chairman.

3. Quorum: Division Manager of Nuclear Operations or his designated alternate plus four other members including alternates.

4. Responsibilities:

a. Review all proposed normal, abnormal, maintenance and emergency operating procedures specified in 6.3.1, 6.3.2, 6.3.3, and 6.3.4 and proposed changes thereto: and any other proposed procedures or changes thereto determined by any member to effect nuclear safety.

b. Review all proposed tests and experiments and their results, which involve nuclear hazards not previously reviewed for conformance with technical specifications. Submit tests which may constitute an unreviewed safety question to the NPPD Safety Review and Audit Board for review.

c. Review proposed changes to Technical Specifications.

d. Review proposed changes or modifications to station systems or equipment as discussed in the SAR or which involves an unreviewed safety question as defined in 10CFR50.59(c). Submit changes to equipment or systems having safety significance to the NPPD Safety Review and Audit Board for review.

e. Review station operation to detect potential nuclear safety hazards.

6.2 (cont'd)

f. Investigate all violations of Technical Specifications, including reporting evaluation and recommendations to prevent recurrence, to the Vice President - Nuclear and to the Chairman of the NPPD Safety Review and Audit Board.

g. Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Safety Review and Audit Board.

h. Review all reportable events specified in Section 50.73 to 10CFR Part 50.

i. Review drills on emergency procedures (including plant evacuation) and adequacy of communication with off site groups.

j. Periodically review procedures required by Specifications 6.3.1, 6.3.2, 6.3.3, and 6.3.4 as set forth in administrative procedures.

5. Authority

a. The Station Operations Review Committee shall be advisory.

b. The Station Operations Review Committee shall recommend to the Division Manager of Nuclear Operations approval or disapproval of proposals under items 4, a through e and j above. In case of disagreement between the recommendations of the Station Operations Review Committee and the Division Manager of Nuclear Operations, the course determined by the Division Manager of Nuclear Operations to be the more conservative will be followed. A written summary of the disagreement will be sent to the Vice President - Nuclear and to the NPPD Safety Review and Audit Board.

c. The Station Operations Review Committee shall report to the Chairman of the NPPD Safety Review and Audit Board on all reviews and investigations conducted under items 4.f, 4.g, 4.h, and 4.i.

d. The Station Operations Review Committee shall make determinations regarding whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review by the NPPD Safety Review and Audit Board.

6. Records:

Minutes shall be kept for all meetings of the Station Operations Review Committee and shall include identification of all documentary material reviewed; copies of the minutes shall be forwarded to the Chairman of the NPPD Safety Review and Audit Board and the Vice President - Nuclear within one month.

7. Procedures:

Written administrative procedures for Committee operation shall be prepared and maintained describing the method for submission and content of presentations to the committee, provisions for use of subcommittees, review and approval by members of written Committee evaluations and recommendations, dissemination of minutes, and such other matters as may be appropriate.

B. NPPD Safety Review and Audit Board (SRAB)

Function: The Board shall function to provide independent review and audit of designated activities.

6.2 (cont'd)

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Appendix A Technical Specifications or the CNS Operating License.

e. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All reportable events specified in Section 50.73 to 10CFR Part 50.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Minutes of meetings of the Station Operations Review Committee.

j. Disagreement between the recommendations of the Station Operations Review Committee and the Division Manager of Nuclear Operations.

k. Review of events covered under e,f,g, and h above include reporting to appropriate members of management on the results of investigations and recommendations to prevent or reduce the probability of recurrence.

5. Authority: The NPPD Safety Review and Audit Board shall report to and be advisory to the Vice President - Nuclear on those areas of responsibility specified in Specifications 6.2.1.B.4 and 6.2.1.B.7.

6. Records:

Minutes shall be recorded for all meetings of the NPPD Safety Review and Audit Board and shall identify all documentary material reviewed. Copies of the minutes shall be forwarded to the Vice President - Nuclear and the Division Manager of Nuclear Operations, and such others as the Chairman may designate within one month of the meeting.

7. Audits:

Audits of selected aspects of plant operation shall be performed under the cognizance of SRAB with a frequency commensurate with their safety significance. Audits performed by the Quality Assurance Department which meet this specification shall be considered to meet the SRAB audit requirements if the audit results are reviewed by SRAB. A representative portion of procedures and records of the activities performed during the audit period shall be audited and, in addition, observations of performance of operating and maintenance activities shall be included. These audits shall encompass:

1. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

2. A summary description of facility changes, tests or experiments in accordance with the requirements of 10CFR50.59(b).

3. Pursuant to 3.8.A, a report of radioactive source leak testing. This report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

4. Documentation of all challenges to relief valves or safety valves.

D. Monthly Operating Report

Routine reports of operating statistics, shutdown experience, and a narrative summary of operating experience relating to safe operation of the facility, shall be submitted on a monthly basis to the individual designated in the current revision of Reg. Guide 10.1 no later than the tenth of each month following the calendar month covered by the report.

6.5.2 Reportable Events

A Reportable Event shall be any of those conditions specified in Section 50.73 to 10CFR Part 50. The NRC shall be notified and a report submitted pursuant to the requirements of Section 50.73. Each Reportable Event shall be reviewed by SORC and the results of this review shall be submitted to SRAB and the Vice President - Nuclear.

1/ This tabulation supplements the requirements of §20.407 of 10CFR Part 20.

Amendment No. ~~75~~, ~~85~~, 100

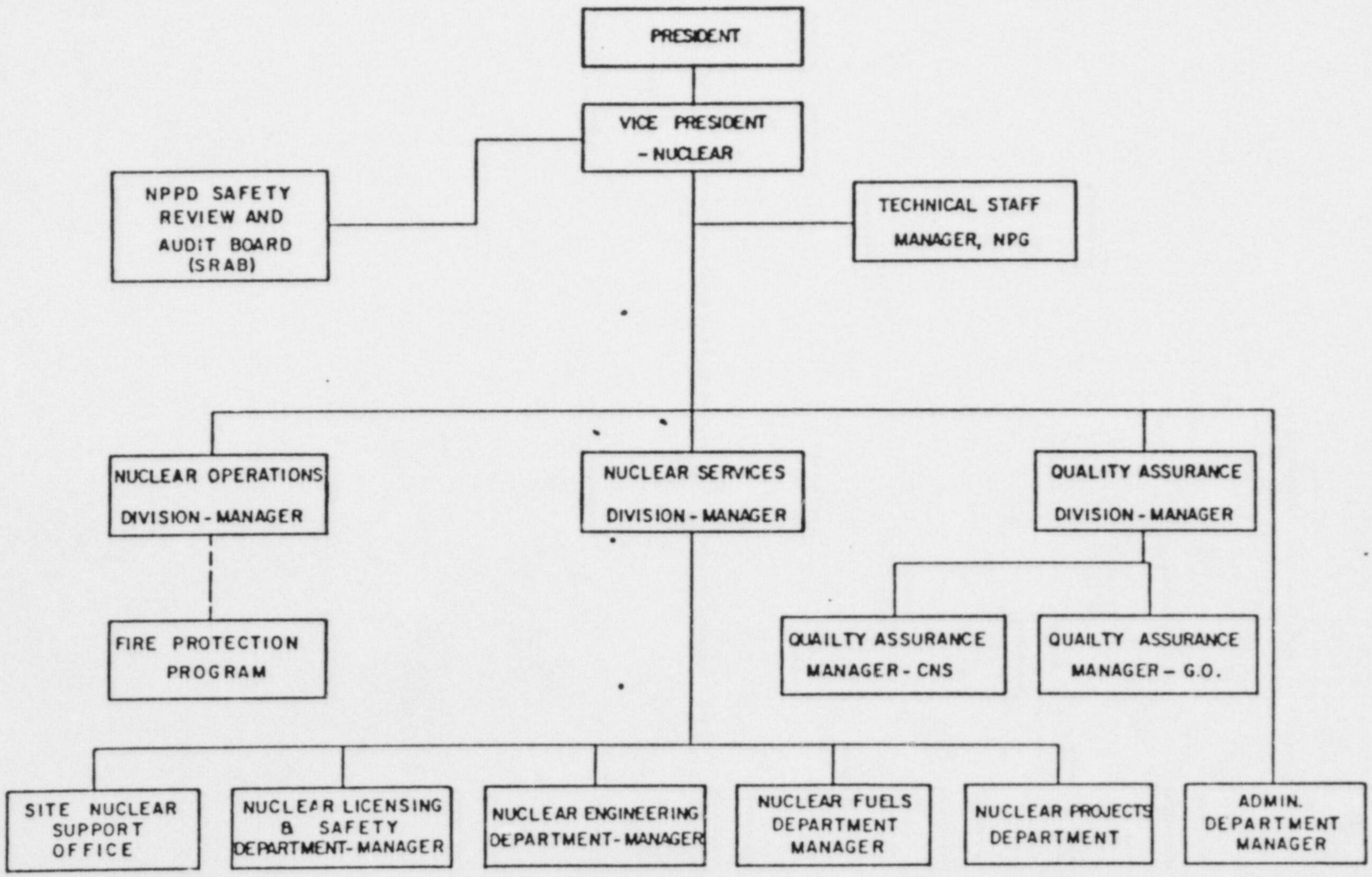

Figure 6.1.1
NPPD Nuclear Power Group
Organization Chart

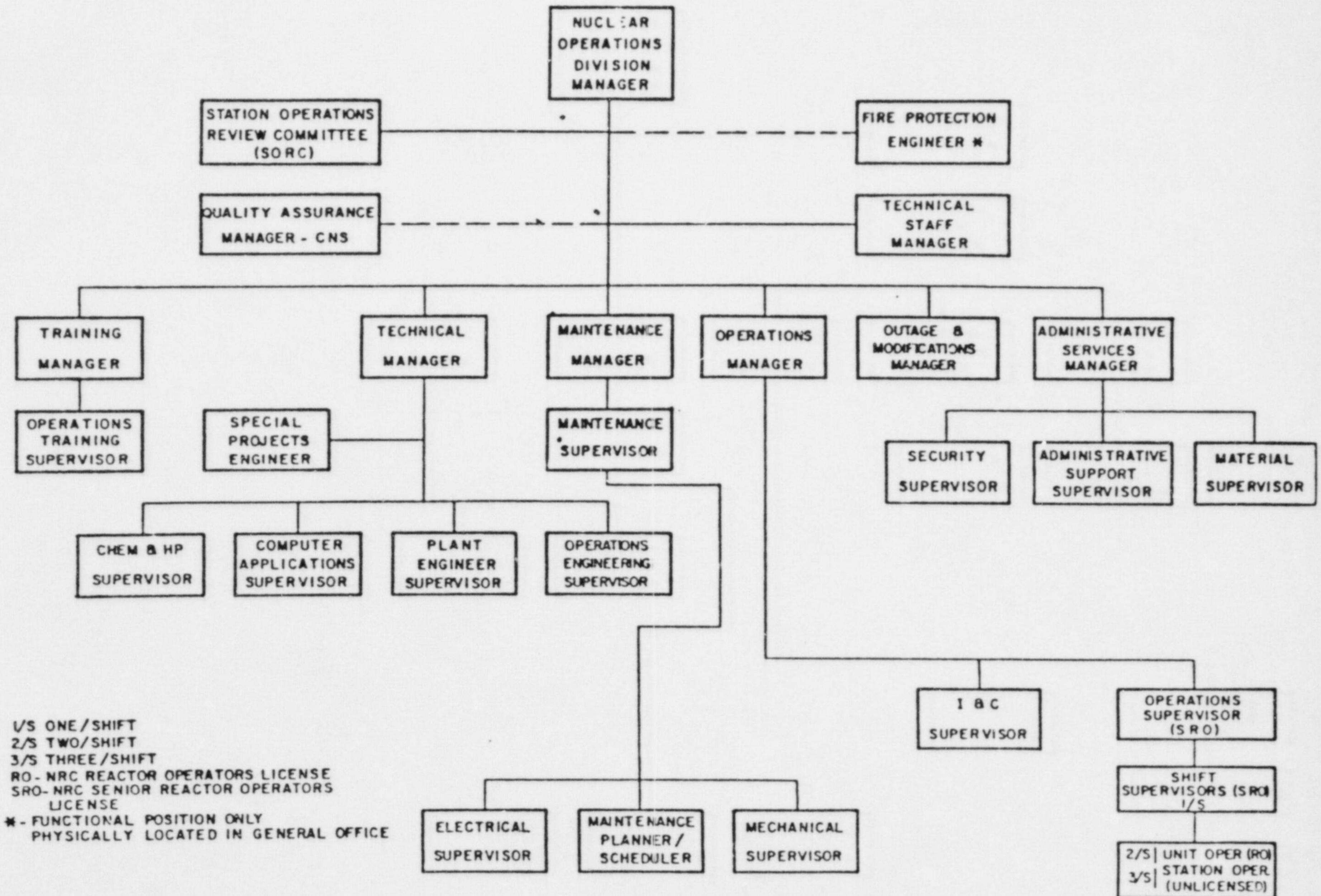

Figure 6.1.2
NPPD Cooper Nuclear Station
Organization Chart