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.62
License No. DPR-46

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

   A. The applications for amendment by Nebraska Public Power District (the licensee) dated (1) February 8, 1980, (2) March 17, 1980 as revised April 18, 1980, (3) February 26, 1980, and (4) October 9, 1978, comply 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 applications, 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 license 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 Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 62 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8005300038

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

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Dated: May 20, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 62

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

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

Remove	Insert
13	13
17	17
19	19
22	22
25	25
26	26
50	50
61	61
68	68
104	104
136	136
167a	167a
184	184
194	194
212	212
214	214
214a	214a
214b	214b
214c	214c
214d	214d
214e	214e
217	217
219	219
226a	226a
237	273

Add page 211c

1.1 Bases: (Cont'd)

turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Cooper has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied on to determine if a Safety Limit has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 18 inches above the top of the fuel provides adequate margin.

References

1. General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, November 1973 (NEDO-10958).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June 1974 (NEDO-20340).

3. "Licensing Topical Report GE-BWR Generic Reload Fuel Application," NEDE-24011-P, (most current approved submittal).

2.1 Bases:

The abnormal operational transients applicable to operation of the CNS Unit have been analyzed throughout the spectrum of planned operating conditions up to 105% of rated steam flow. The analyses were based upon plant operation in accordance with Reference 3. In addition, 2381 MWt is the licensed maximum power level of CNS, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greater significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.a.1.a, when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR values specified in Reference 3.

b. <u>APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)</u>

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedure backed up by the rod worth minimizer, and the rod sequence control system. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This change can occur when reactor pressure is greater than Specification 2.1.A.6.

2.1 **Bases**: (Cont'd)

5. Main Steam Line Isolation Valve Closure on Low Pressure

The low pressure isolation of the main steam lines (Specification 2.1.A.6) was provided to protect against rapid reactor depressurization.

B. Reactor Water Level Trip Settings Which Initiate Core Standby Cooling Systems (CSCS)

The core standby cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature, to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each Core Standby Cooling System component was established based on the reactor low water level scram set point. To lower the set point of the low water level scram would increase the capacity requirement for each of the CSCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of CSCS capacity requirements.

The design for the CSCS components to meet the above guidelines was dependent upon three previously set parameters: The maximum break size, low water level scram set point and the CSCS initiation set point. To lower the set point for initiation of the CSCS may lead to a decrease in effective core cooling. To raise the CSCS initiation set point would be in a safe direction, but it would reduce the margin established to prevent actuation of the CSCS during normal operation or during normally expected transients.

Transient and accident analyses reported in Section 14 of the Final Safety Analyses Report demonstrate that these conditions result in adequate safety margins for the fuel.

C. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10801, Feb., 1973.

2. Station Safety Analysis Report (Section XIV).

3. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (most current approved submittal).

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

REFERENCES

1. Station Safety Analysis (Section XIV)

2. ASME Boiler and Pressure Vessel Code, Section III

3. USAS Piping Code, Section B31.1

4. Reactor Vessel and Appurtenances Mechanical Design (Subsection IV-2)

5. Station Nuclear Safety Operational Analysis (Appendix G)

6. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (most current approved submittal).

2.2 BASES

The 8 relief valves and 3 safety valves are sized and set pressures are established in accordance with the requirements of Section III of the ASME Code. A turbine trip without bypass is assumed. Relief valves are taken to operate normally, and credit is taken for a high pressure scram at 1045 psig. This analysis is discussed in Subsection IV-4 and Question 4.20 of Amendment 11 to the Safety Analysis Report.

The relief valve settings satisfy the Code requirements that the lowest valve set point be at or below the vessel design pressure of 1250 psig. These settings are also sufficiently above the normal operating pressure range to prevent unnecessary cycling caused by minor transients. The results of postulated transients where inherent relief valve actuation is required are given in Section XIV of the Safety Analysis Report.

Reanalysis in Reference 6 for the case of MSIV-Closure with flux scram transient results in a peak pressure at the vessel bottom which is below the maximum of 110 percent of design pressure allowed by the Code. This is adequate margin to ensure that the 1375 psig pressure safety limit is not exceeded. A sensitivity study on peak vessel pressure to the failure to open of one of the lowest set-point safety valves was performed for a typical high power density BWR (Reference 7). The study is applicable to the Cooper reactor and shows that the sensitivity of a high power density plant to the failure of a safety valve is approximately 20 psi. A plant specific analysis for the Cooper overpressure transient would show results equal to or less than this value.

The design pressure of the shutdown cooling piping of the Residual Heat Removal System is not exceeded with the reactor vessel steam dome less than 75 psig.

REFERENCES

1. Topical Report, "Summary of Results Obtained from a Typical Startup and Power Test Program for a General Electric Boiling Water Reactor", General Electric Company, Atomic Power Equipment Department (APED-5698)

2. Station Nuclear Safety Operational Analysis (Appendix G)

3. Station Safety Analysis (Section XIV)

4. Control and Instrumentation (Section VII)

5. Summary Technical Report of Reactor Vessel Overpressure Protection (Question 4.20, Amendment 11 to SAR)

6. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (most current approved submittal).

7. Letter from I. F. Stewart (GE) to v. Stello (NRC) dated December 23, 1975.

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	$\geq$+12.5" Indicated Level	2(4)	A or B
Reactor Low Low Water Level	NBI-LIS-57 A & B NBI-LIS-58 A & B	$\geq$-37" 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	$\geq$ 850 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

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TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System (5)
APRM Upscale (Flow Bias)	$\leq (0.66W + 42\%) \left[\frac{FRP}{MFLPD}\right]$ (2)	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq$ (0.66W + 41%) (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1)(6)
SRM Detector Not Full In (4)(8)	($\geq$ 100 cps)	1(1)(6)
SRM Inoperative (8)	(10a)	1(1)(6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. flow	1
SRM Downscale (8)(7)	$\geq$ 3 Counts/Second (11)	1(1)(6)

COOPER NUCLEAR STATION
TABLE 4.2.A (Page 1)
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION SYSTEM
TEST AND CALIBRATION FREQUENCIES

Item	Item I.D. No.	Function Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D	Once/Month (1)	Once/3 Months	Once/Day
Reactor Low Low Water Level	NBI-LIS-57, A & B NBI-LIS-58, A & B	Once/Month (1)	Once/3 Months	Once/Day
Main Steam Line Leak Detection	MS-TE-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	Once/Month (1)	Once/Operating Cycle	None
Main Steam Line High Flow	MS-dPIS-116, A,B,C,&D	Once/Month (1)	Once/3 Months	None
	117	Once/Month (1)	Once/3 Months	None
	118	Once/Month (1)	Once/3 Months	None
	119	Once/Month (1)	Once/3 Months	None
Main Steam Line Low Press.	MS-PS-134, A,B,C,&D	Once/Month (1)	Once/3 Months	None
High Reactor Pressure	RR-PS-128, A & B	Once/Month (1)	Once/3 Months	None
Condenser Low Vacuum	MS-PS-103, A,B,C,&D	Once/Month (1)	Once/3 Months	None
Reactor Water C.U. High Flow	RWCU-dPIS-170, A & B	Once/Month (1)	Once/3 Months	None
Reactor Water C.U. High Space Temp.	RWCU-TS-150 A-D, 151, 152, 153, 154, 155, 156, 157, 158, 159, RWCU-TS-81 A,B,E,F, RWCU-TS-81 C,D,G,H	Once/Month (1)	Once/Operating Cycle	None

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, 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, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in the allowable scram insertion times of Specification 3.3.C. The time to deenergize the pilot valve scram solenoid is measured during the calibration tests required by Spec 4.1.

D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% Δk. Deviations in core reactivity greater than 1% Δk are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

E. Recirculation Pumps

Until analyses are submitted for review and approval by the NRC which prove that recirculation pump startup from natural circulation does not cause a reactivity insertion transient in excess of the most severe coolant flow increase currently analyzed, Specification 3.3.E prevents starting recirculation pumps while the reactor is in natural circulation above 1% of rated thermal power.

REFERENCES

1. NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," Paone, Stirn & Woolley, 3-72, Class I.

2. NEDO-10427, Supplement 1, "Rod Drop Accident Analysis for Large Boiling Water Reactors," Stirn, Paone & Yound, 7-72, Class I.

3. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (most current approved submittal).

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.D Safety and Relief Valves	4.6.D Safety and Relief Valves
1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, all three safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.	1. Approximately half of the safety valves and relief valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles. The set point of the safety valves shall be as specified in Specification 2.2.
2.	2. At least one of the relief valves shall be disassembled and inspected each refueling outage.
a. From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable.	
b. From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.	
3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to a cool shutdown condition within 24 hours.	3. The integrity of the relief safety valve bellows on any three stage valve shall be continuously monitored. 4. The operability of the bellows monitoring system shall be demonstrated once every three months when three stage valves are installed. 5. Once per operating cycle, with the reactor pressure ≥ 100 psig, each relief valve shall be manually opened until the main turbine bypass valves have closed to compensate for relief valve opening.

3.7 (cont'd)

E. Drywell-Suppression Chamber Differential Pressure

1. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.0 psid except as specified in a, b, and c below.

   a. This differential shall be established within 26 hours after placing the mode switch in run.

   b. This differential may be decreased to less than 1.0 psid 24 hours prior to placing mode switch in refuel or shutdown.

   c. This differential may be decreased to less than 1.0 psid for a maximum of four (4) hours during required operability testing of the HPCI system pump, the RCIC system pump, and the drywell-pressure suppression chamber vacuum breakers.

2. If the differential pressure of specification 3.7.E.1 cannot be maintained, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in Hot Standby in six (6) hours and in a Cold Shutdown condition within the following 18 hours.

4.7 (cont'd)

E. Drywell-Suppression Chamber Differential Pressure

1. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

3.7.D & 4.7.D (cont.d)

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment. A program for periodic testing and examination of the excess flow check valves is performed as follows:

1. Vessel at pressure sufficient to actuate valves. This could be at time of vessel hydro following a refueling outage.

2. Isolate sensing line from its instrument at the instrument manifold.

3. Provide means for observing and collecting the instrument drain or vent valve flow.

4. Open vent or drain valve.

    a. Observe flow cessation and any leakage rate.

    b. Reset valve after test completion.

5. The head seal leak detection line cannot be tested in this manner. This valve will not be exposed to primary system pressure except under unlikely conditions of seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source and therefore this valve need not be tested. This valve is in a sensing line that is not safety related.

6. Valves will be accepted if a marked decrease in flow rate is observed and the leakage rate is acceptable.

3.7.E <u>Bases</u>

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed as described in the licensee's letter of October 4, 1976, which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of drywell-suppression chamber differential pressure of 1.0 psid and a suppression chamber water level corresponding to a downcomer submergence range of three to four feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.A	4.9.A.2 (cont'd)

During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps and fuel oil day tank level switches shall be demonstrated, and the diesel starting time to reach rated voltage and frequency shall be logged.

b. Once every 18 months the condition under which the diesel generator is required will be simulated and a test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The results shall be logged.

c. Specification 4.9.A.2.c deleted.

d. Once a month the quantity of diesel fuel available shall be logged.

e. Every three months and upon delivery a sample of diesel fuel shall be checked for quality. The quality shall be within the acceptable limits specified in Table 1 of ASTM D975-68 for Nos. 1D or 2D and logged.

f. Each diesel generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

3. Unit Batteries

a. Every week the specific gravity, the voltage and temperature of the pilot

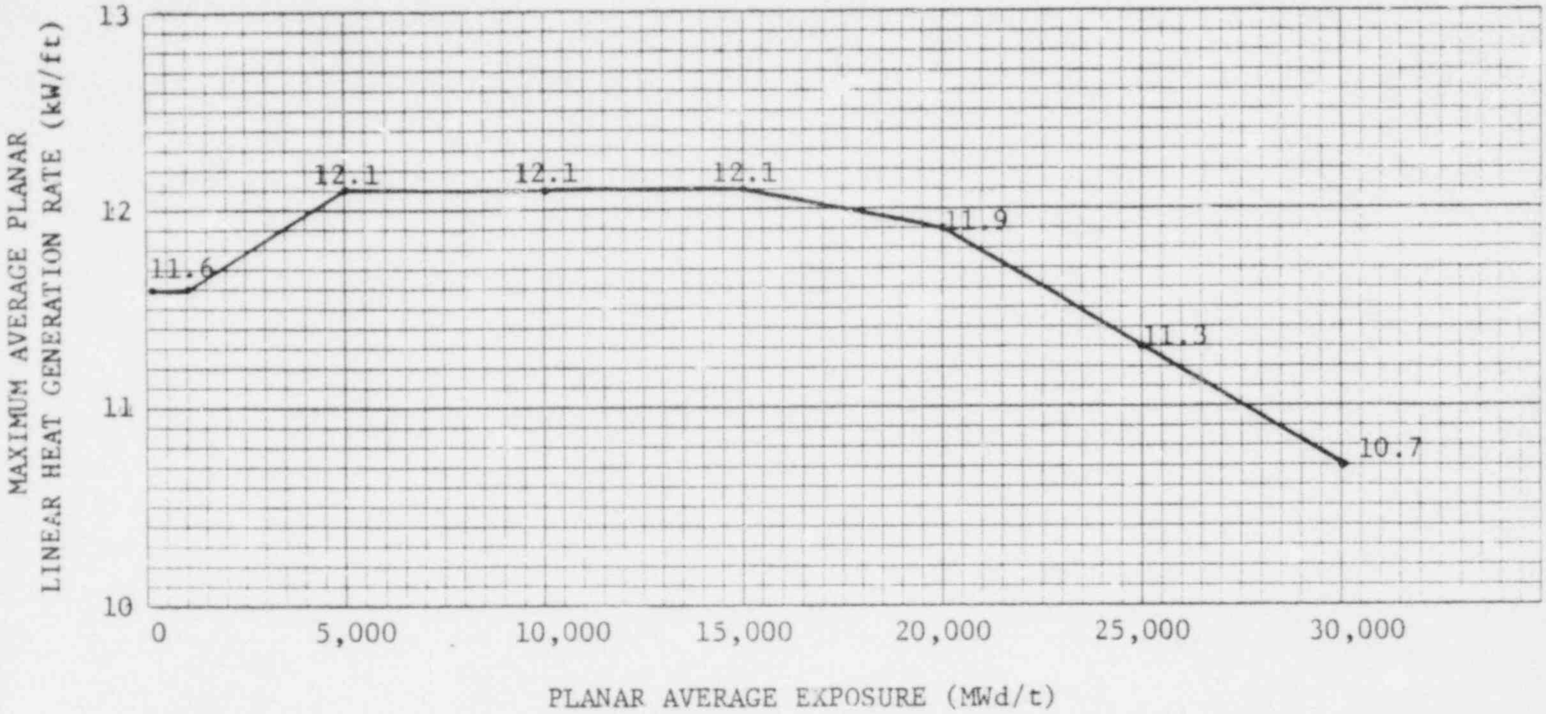

Figure 3.11-1.6 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, P8DRB265L Fuel.

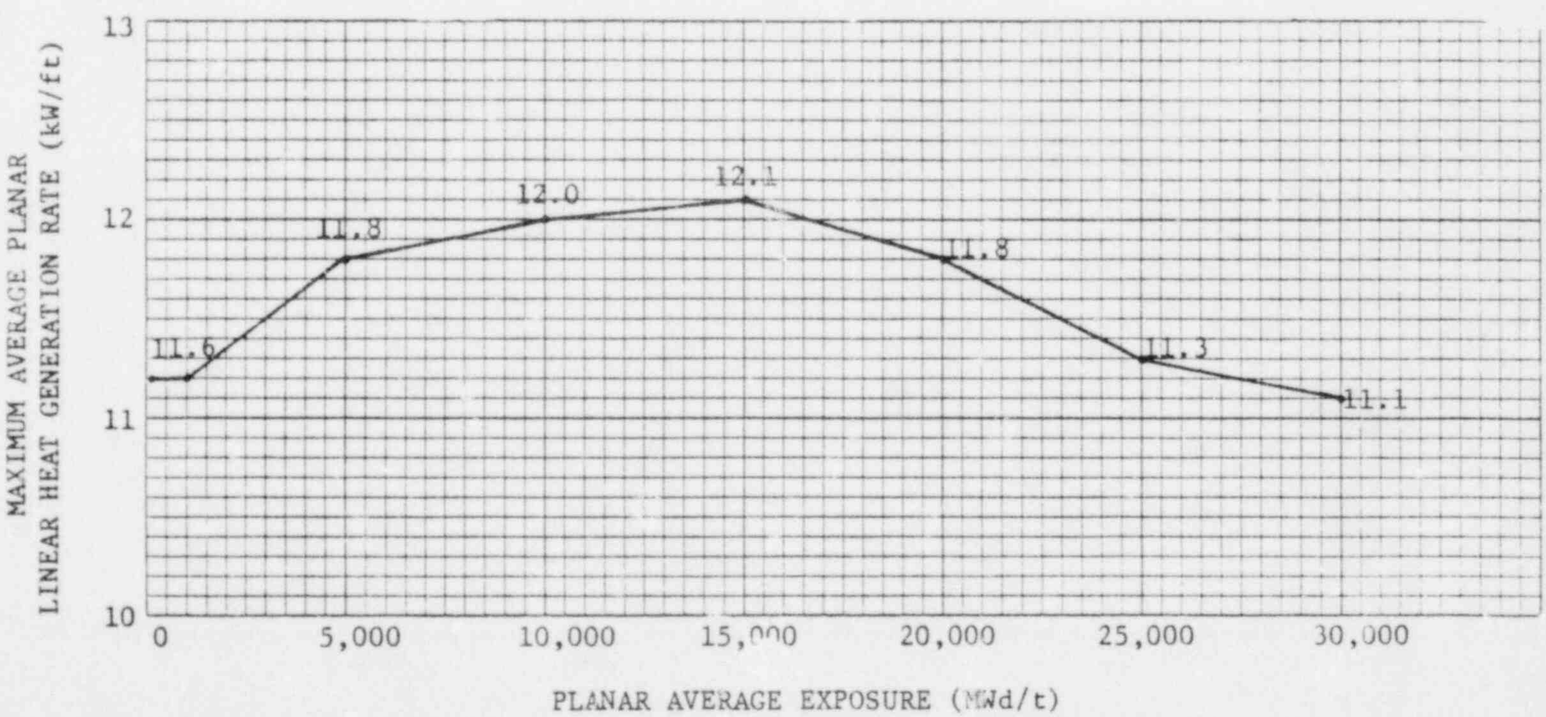

Figure 3.11-1.7 Maximum Average Planar Linear Heat Generation Rate versus Exposure with LPCI Modification and Bypass Flow Holes Plugged, P8DRB283 Fuel.

LT = Total core length - 12 feet

L = Axial position above bottom of core

G = 18.5 kW/ft for 7x7 fuel bundles
= 13.4 kW/ft for 8x8 fuel bundles

N = 0.038 for 7x7 fuel bundles
= 0.0 for 8x8 fuel bundles

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded action shall then be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation MCPR shall be $\geq$ 1.20 for 7x7 bundles and $\geq$ 1.25 for 8x8 bundles, at rated power and flow. If, at any time during steady state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

For core flows other than rated the MCPR shall be the operating limit at rated flow times K_f, where K_f is as shown in Figure 3.11-2.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

3.11 BASES

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for APLHGR is shown in Figure 3.11-1.

The calculational procedure used to establish the APLHGR shown on Figure 3.11.1 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10CRF50. A complete discussion of each code employed in the analysis is presented in Reference 1.

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REFERENCES FOR BASES 3.11.A

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566, dated January 1976.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 5 of Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern. Pellet densification power spiking in 8x8 fuel has been accounted for in the safety analysis presented in Reference 5; thus no adjustment to the LHGR limit for densification effects is required for 8x8 fuels.

C. Minimum Critical Power Ratio (MCPR)
Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.11C are derived from the established fuel cladding integrity Safety Limit MCPR of 1.07, and an analysis of abnormal operational transients (Reference 5). For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the more limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

3.11 **Bases**: (Cont'd)

The limiting transient which determines the required steady state MCPR limit and thus yields the largest ΔCPR is discussed in Reference 5. When added to the safety limit MCPR of 1.07 the required minimum operating limit MCPR's of specification 3.11C are obtained.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4 of NEDO-24011[2] and on core parameters shown in Table 5-2 of Reference 2.

The evaluation of a given transient begins with the system initial parameters shown in Table 5-2 of Reference 2 that are input to a GE core dynamic behavior transient computer program described in NEDO-10802[3]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

D. <u>MCPR Limits for Core Flows Other than Rated</u>

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow, the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

3.11 Bases: (Cont'd)

The K_f factor curves shown in Figure 3.11-2 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11-2, are conservative for Cooper operation because the operating limit MCPR's are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

References for Bases 3.11.B, 3.11.C, 3.11.D

1. "Cooper Nuclear Station Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21072, October 1975.

2. Licensing Topical Report, General Electric Boiling Water Reactor, Generic Reload Fuel Application, (NEDE-24011-P), (most current approved submittal).

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566, dated January 1976.

5. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," (most current approved submittal).

6. April 18, 1978 letter from J. M. Pilant (NPPD) to G. E. Lear (NRC).

4.11 **Bases:**

A & B. Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

D. Core Stability

The calculations, regarding reactor core stability, presented in the most current approved submittal of "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Unit 1," show that the reactor is in compliance with the ultimate performance criteria, including the most responsive condition at natural circulation and rod block power. However, to preclude the possibility of operation under conditions which could result in reactor core instability, the NRC requested the incorporation of a specification limit.

The power level specified results in a decay ratio (X_2/X_0) which is significantly less than the ultimate stability limit of 1.0.

5.0 MAJOR DESIGN FEATURES

5.1 Site Features

The Cooper Nuclear Station site is located in Nemaha County, Nebraska, on the west bank of the Missouri River, at river mile 532.5. This part of the river is referred to by the Corps of Engineers as the Lower Brownville Bend. Site coordinates are approximately 40° 21' north latitude and 95° 38' west longitude. The site consists of 1351 acres of land owned by Nebraska Public Power District. About 205 acres of this property is located in Atchison County, Missouri, opposite the Nebraska portion of the station site. The land area upon which the station is being constructed is crossed by the Missouri River on the east and is bounded by privately owned property on the north, south, and west. At the west site boundary, a county road and Burlington Northern Railroad spur pass the site.

The reactor (center line) is located approximately 3600 feet from the nearest property boundary. No part of the present property shall be sold or leased by the applicant which would reduce the minimum distance from the reactor to the nearest site boundary to less than 3600 feet without prior NRC approval.

The protected area is formed by a seven foot chain link fence which surrounds the site buildings.

5.2 Reactor

A. The core shall consist of not more than 548 fuel assemblies of 7x7 (49 fuel rods) and 8x8 (63 fuel rods) and 8x8R/P8x8R (62 fuel rods).

B. The core shall contain 137 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C) compacted to approximately 70% theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Section IV-20 of the SAR. The applicable design shall be as described in this section of the SAR.

5.4 Containment

A. The principal design parameters for the primary containment shall be as given in Table V-2-1 of the SAR. The applicable design shall be as described in Section XII-2.3 of the SAR.

B. The secondary containment shall be as described in Section V-3.0 of the SAR.

C. Penetrations to the primary containment and piping passing through such

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 The Station Superintendent shall have the over-all fulltime onsite responsibility for the safe operation of the Cooper Nuclear Station. During periods when the Station Superintendent is unavailable, he may delegate his responsibility to the Assistant to Station Superintendent or, in his absence, to one of the Department Supervisors.

6.1.2 The portion of the Nebraska Public Power District management which relates to the operation of this station is shown in Figure 6.1.1.

6.1.3 The organization for conduct of operation of the station is shown in Fig. 6.1.2. The shift complement at the station shall at all times meet the following requirements. Note: Higher grade licensed operators may take the place of lower grade licensed or unlicensed operators.

A. A licensed senior reactor operator (SRO) shall be present at the station at all times when there is any fuel in the reactor.

B. A licensed reactor operator shall be in the control room at all times when there is any fuel in the reactor.

C. Two licensed reactor operators shall be in the control room during all startup, shutdown and other periods involving significant planned control rod manipulations. A licensed SRO shall either be in the Control Room or immediately available to the Control Room during such periods.

D. A licensed senior reactor operator (SRO) with no other concurrent duties shall be directly in charge of any refueling operation, or alteration of the reactor core.

A licensed reactor operator (RO) with no other concurrent duties shall be directly in charge of operations involving the handling of irradiated fuel other than refueling or reactor core alteration operations.

E. An individual who has been trained and qualified in health physics techniques shall be on site at all times that fuel is on site.

F. Minimum crew size during reactor operation shall consist of three licensed reactor operators (one of whom shall be licensed SRO) and three unlicensed operators. Minimum crew size during reactor cold shutdown conditions shall consist of two licensed reactor operators (one of whom shall be licensed SRO) and one unlicensed operator.

In the event that any member of a minimum shift crew is absent or incapacitated due to illness or injury a qualified replacement shall be designated to report on-site within two hours.

G. A fire brigade of at least 3 members shall be maintained at all times. This excludes the 3 members of the minimum shift crew necessary for safe shutdowns, and other personnel required for other essential functions during a fire emergency.

6.3 (cont'd.)

6.3.5 In lieu of the "control device" or "alarm signal" required by Paragraph 20.203 (c) (2) of 10 CFR 20 each High Radiation Area (100 mrem/hr or greater) shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by requiring notification and permission of the shift supervisor. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

6.3.6 All procedures described in 6.3.2, 6.3.3, & 6.3.4 above, and changes thereto, shall be reviewed by the Station Operations Review Committee and approved by the Station Superintendent prior to implementation, except as provided for in 6.3.7 below.

6.3.7 Temporary changes to procedures which do not change the intent of the original procedure may be made, provided such changes are approved by two members of the operating staff holding SRO licenses. Such changes shall be documented and subsequently reviewed by the Station Superintendent within one month.

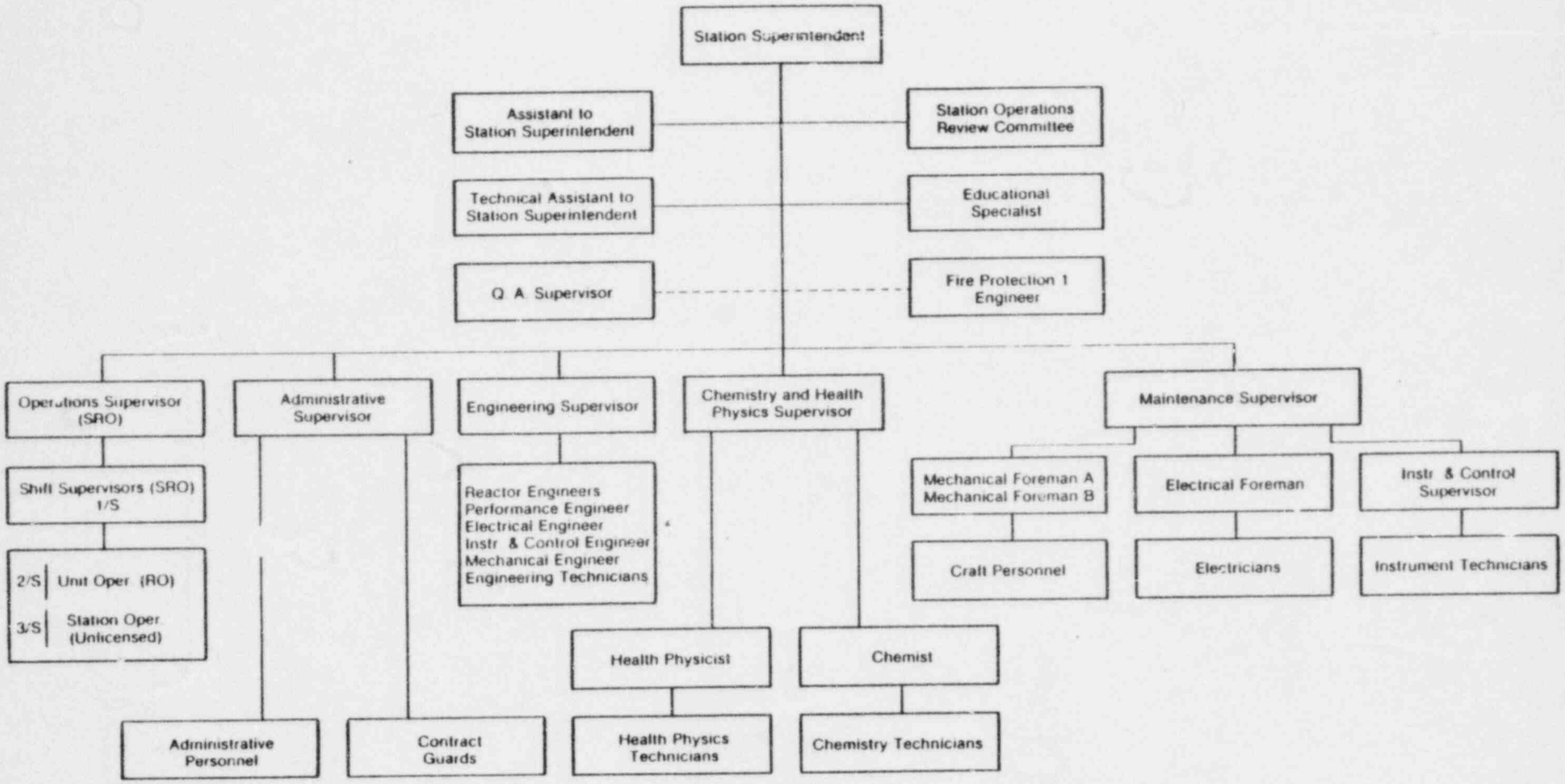
6.3.8 Drills of the Emergency Plan procedures shall be conducted annually, including a check of communications with offsite support groups. Drills on the procedures specified in 6.3.2.A, B, and C above shall be conducted as part of the retraining program.

REFERENCE

1. SAR Subsection XIII-6.

Amendment No. 62

CNS ORGANIZATION CHART



1/S one/shift
2/S two/shift
3/S three/shift
RO-NRC Reactor Operators License
SRO-NRC Senior Reactor Operators License
1-Functional Position Only
physically located in General Office

Figure 6.1.2
Cooper Nuclear Station
Organization Chart