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OCT 17 1975

Docket No. 50-298

Nebraska Public Power District
ATTN: Mr. J. M. Pilant, Director
Licensing and Quality Assurance
Post Office Box 499
Columbus, Nebraska 68601

Gentlemen:

The Commission has issued the enclosed Amendment No. 14 to Facility License No. DPR-46 for the Cooper Nuclear Station. This amendment includes Change No. 17 to the Technical Specifications and is in response to your request dated July 3, 1975.

This amendment changes the fuel grapple hoist load switch setting from $\leq$ 1200 lbs. to $\leq$ 650 lbs. and corrects several clerical errors and oversights in various sections of the Technical Specifications. Two of the clerical errors (Reactor Highwater Level switch designation on page 4 of Table 4.2.B and the specification number change on page 165 of the Technical Specifications) were previously corrected by Changes 4 and 10 dated April 18, 1974 and February 26, 1975, respectively.

An additional change requested in your July 3 letter, concerning section 3.10.A.2. of the Technical Specifications, was the subject of an NRC letter dated September 19, 1975.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 14 to License No. DPR-46 w/Change No. 17
2. Safety Evaluation
3. Federal Register Notice

OFFICE ➤	RL:ORB #2	RL:ORB #2	OELD	RL:ORB #2		
SURNAME ➤	RMDiggs	~~MFletcher~~ RDSilver	B. Smith	DLZiemann		
DATE ➤	10/8/75	10/9/75	10/11/75	10/17/75		

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Lincoln, Nebraska 68509

OFFICE➤						
SURNAME➤						
DATE➤						

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. 14
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated July 3, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility 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, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 17."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Attachment:
Change No. 17 to the
Technical Specifications

Date of Issuance: OCT 17 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 14

CHANGE NO. 17 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

The Technical Specifications contained in Appendix A of Facility License No. DPR-46 are hereby changed by replacing pages 55, 56, 58, 65, 75, 80, 194, 204 and 208 of Appendix A to the Technical Specifications with the attached revised pages bearing the same numbers. Changed areas on the revised pages are reflected by marginal lines.

COOPER NUCLEAR STATION

TABLE 3.2.B (Page 3)

RESIDUAL HEAT REMOVAL SYSTEM (LPCI MODE) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured
RHR Pump Low Flow	RHR-dPIS-125 A & B	≥2500 gpm	1	A
Break Detection Time Delays	RHR-TDR-K28, A & B	0.25≤T≤0.75 sec.	1	A
	RHR-TDR-K40, A & B	0.25≤T≤0.75 sec.	1	A
	RHR-TDR-K34, A & B	1.5≤T≤2.5 sec.	1	A
	(RHR-TDR-K51, A & B)	9≤T≤11 min.	1	A
	RHR-TDR-K86, A & B	9≤T≤11 min.	1	A
	RHR-TDR-K45, 1A & 1B	4.25≤T≤5.75 min.	1	A
RHR Pump Start Time Delay	RHR-TDR-K75, A & B	4.5≤T≤5.5 sec.	1	A
	RHR-TDR-K70, A & B	≤.5 sec.	1	A
RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	1.8≤T≤2.2 min.	1	B
RHR Crosstie Valve Position	RHR-LMS-2	N.A.	(3)	B
Bus 1A Low Volt. Aux. Relay	27 X 3/1A	Loss of Voltage	1	B
Bus 1B Low Volt. Aux. Relay	27 X 3/1B	Loss of Voltage	1	B
Bus 1F Low Volt. Aux. Relays	27 X 1/1F	Loss of Voltage	1	B
	27 X 2/1F	Loss of Voltage	1	B
Bus 1G Low Volt. Aux. Relays	27 X 1/1G	Loss of Voltage	1	B
	27 X 2/1G	Loss of Voltage		
Pump Discharge Line Low Pressure	CM-PS-266	≥5 psig	(3)	D

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COOPER NUCLEAR STATION
TABLE 3.2.B (Page 4)
HPCI SYSTEM CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured
Reactor Low Water Level	NBI-LIS-72, A,B,C, & D #3	≥-37" Indicated Level	2	A
Reactor High Water Level	NBI-LIS-101, B & D #2	≤+58.5" Indicated Level	2(2)	A
High Drywell Press.	14A-K5, A & B (6) 14A-K6, A & B (6)	≤2 psig	2(2)	A
HPCI Turbine High Exhaust Pressure	HPCI-PS-97, A & B	≤150 psig	1(2)	A
HPCI Pump Low Suction Press.	HPCI-PS-84-1	≤15" Hg Vacuum	1(2)	A
HPCI Pump Low Discharge Flow	HPCI-FS-78	≥400 gpm	1(2)	A
HPCI Low Steam Supply Pressure	HPCI-PS-68, A,B,C & D	≥100 psig	2(2)	A
HPCI Steam Line High ΔP	HPCI-dPIS-76 HPCI-dPIS-77	130 ≤S≤210" H_2O -130 ≥S≥-210" H_2O	1	A
HPCI Steam Line Space Hi 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	≤200° F	2(4)	A
Emerg. Cond. Storage Tank Low Level	HPCI-LS-74 A & B HPCI-LS-75 A & B	≥0" H_2O (10,000 gal. usable remaining)	1(2)	A

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COOPER NUCLEAR STATION
TABLE 3.2.B (Page 6)
REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
RCIC High Turbine Exhaust Press.	RCIC-PS-72, A & B	≤25 psig	1(2)	A
RCIC Low Pump Suction Press.	RCIC-PS-67-1	≤-15" Hg	1(2)	
RCIC Steam Line Space Excess Temp.	RCIC-TS-79, A,B,C, & D RCIC-TS-80, A,B,C, & D RCIC-TS-81, A,B,C, & D RCIC-TS-82, A,B,C, & D	≤200°F	2(4)	A
RCIC Steam Line High ΔP	RCIC-dPIS-83 & 84	370" ≤S≤620" H_2O	1	A
RCIC Steam Supply Press. Low	RCIC-PS-87, A,B,C & D	≥50 psig	2(2)	A
RCIC Low Pump Disch. Flow	RCIC-FIS-57	≥40 gpm	1(2)	A
Pump Discharge Line Low Pressure	CM-PS-269	≥10 psig	(3)	D
RCIC Turbine Conditional Supervisory Alarm Timer	RCIC-TDR-K9	13.5 ≤ T ≤ 16.5	(3)	E
Reactor Low Water Level	10A-K80, A & B 10A-K79, A & B (NBI-LIS-72, A,B,C, & D)	≥-37" Indicated Level	2(2)	A
Reactor High Water Level	NBI-LIS-101, A & C#2	≤+58.5 Indicated Level	2(2)	A

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COOPER NUCLEAR STATION
TABLE 3.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION

Instrument	Instrument I.D. No.	Range	Minimum Number of Operable Instrument Channels	Action Required When Minimum Condition Not Satisfied (1)
Reactor Water Level	NBI-LI-85A NBI-LI-85B	-150" to +60" -150" to +60"	2	A,B,C
Reactor Pressure	RFC-PI-90A RFC-PI-90B	0 - 1200 psig 0 - 1200 psig	2	A,B,C
Drywell Pressure	PC-PI-512A PC-PR-512B	0 - 80 psia 0 - 80 psia	2	A,B,C
Drywell Temperature	PC-TR-503 PC-TI-505	50 - 170°F 50 - 350°F	2	A,B,C
Suppression Chamber Air Temperature	PC-TR-21A PC-TA-20 A,C	0 - 300°F 0 - 400°F	2	A,B,C
Suppression Chamber Water Temperature	PC-TR-21B PC-TA-20 B,D	0 - 300°F 0 - 400°F	2	A,B,C
Suppression Chamber Water Level	PC-LI-10 PC-LR-11 PC-LI-12	(-4' to +6') (-4' to +6') -10" to +10"	2	A,B,C
Suppression Chamber Pressure	PC-PR-20	0 - 2 psig	1	B,C
Control Rod Position	N.A.	Indicating Lights	1	A,B,C,D
Neutron Monitoring	N.A.	S.R.M., I.R.M., LPRM 0 - 100% power	1	A,B,C,D

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COOPER NUCLEAR STATION
TABLE 4.2.B (Page 6)
RCIC TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
1. Reactor High Water Level	NBI-LIS-101 A & C, #2	Once/Month (1)	Once/3 Months	Once/Day
2. Reactor Low Water Level	10A - K79 A & B 10A-K80 A & B	Once/Month (1)	Once/3 Months	Once/Day
3. RCIC High Turbine Exhaust Press.	RCIC-PS-72, A & B	Once/Month (1)	Once/3 Months	None
4. RCIC Low Pump Suction Press.	RCIC-PS-67-1	Once/Month (1)	Once/3 Months	None
5. RCIC Steam Line Space Excess Temp.	RCIC-TS-79 A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-80, A,B,C,& D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-81, A,B,C,& D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-82, A,B,C,&D	Once/Month (1)	Once/Oper. Cycle	None
6. RCIC Steam Line High ΔP	RCIC-dPIS-83	Once/Month (1)	Once/3 Months	None
	RCIC-dPIS-84	Once/Month (1)	Once/3 Months	None
7. RCIC Steam Supply Press. Low	RCIC-PS-87 A,B,C, & D	Once/Month (1)	Once/3 Months	None
8. RCIC Low Pump Disch. Flow	RCIC-FIS-57	Once/Month (1)	Once/3 Months	None
9. Pump Disch. Line Low Pressure	CM-PS-269	Once/3 Months	Once/3 Months	None
10. RCIC Turbine Conditional Supv. Alarm Timer	RCIC-TDR - K9	Once/Month (1)	Once/Oper. Cycle	None
Logic Systems (4)(6)				
1. Logic Buss Power Monitor		Once/6 Months	N.A.	
2. RCIC Initiation		Once/6 Months	N.A.	
3. Turbine Trip		Once/6 Months	N.A.	
4. RCIC Automatic Isolation		Once/6 Months	N.A.	

17

COOPER NUCLEAR STATION
TABLE 4.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION
TEST AND CALIBRATION FREQUENCIES

Instrument	Instrument I.D. No.	Calibration Frequency	Instrument Check
Reactor Water Level	NBI-LI-85A	Once/6 Months	Each Shift
	NBI-LI-85B	Once/6 Months	Each Shift
Reactor Pressure	RFC-PI-90A	Once/6 Months	Each Shift
	RFC-PI-90B	Once/6 Months	Each Shift
Drywell Pressure	PC-PI-512A	Once/6 Months	Each Shift
	PC-PR-512B	Once/6 Months	
Drywell Temperature	PC-TR-503	Once/6 Months	Each Shift
	PC-TI-505	Once/6 Months	Each Shift
Suppression Chamber Air Temperature	PC-TR-21A	Once/6 Months	Each Shift
	PC-TA-20A,C	Once/6 Months	Each Shift
Suppression Chamber Water Temperature	PC-TR-21B	Once/6 Months	Each Shift
	PC-TA-20B,D	Once/6 Months	Each Shift
Suppression Chamber Water Level	PC-LI-10	Once/6 Months	Each Shift
	PC-LR-11	Once/6 Months	Each Shift
	PC-LI-12	Once/6 Months	Each Shift
Suppression Chamber Pressure	PC-PR-20	Once/6 Months	Each Shift 17
Control Rod Position	N.A.	N.A.	Each Shift
Neutron Monitoring (APRM)	N.A.	Once/Week	Each Shift

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.9.A

4.9.A (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.

Note 1: Failure to synchronize within 30 seconds will not be constructed as a failure, but that particular test will be repeated.

Note 2: Factory tests, which duplicate the conditions of this test, will be included in the total number.

b. Once per operating cycle 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. Once a month the quantity of diesel fuel available shall be logged.

d. 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.

17

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

2. Unit Batteries

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.10.A Refueling Interlocks	4.10.B

17 3. The fuel grapple hoist load switch shall be set at ≤ 650 lbs.

4. If the frame-mounted auxiliary hoist, the monorail-mounted auxiliary hoist, or the service platform hoist is to be used for handling fuel with the head off the reactor vessel, the load limit switch on the hoist to be used shall be set at ≤ 400 lbs.

5. A maximum of two nonadjacent controls rods may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance, provided the following conditions are satisfied:

a. The reactor mode switch shall be locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.

b. A sufficient number of control rods shall be operable so that the core can be made subcritical with the strongest operable control rod fully withdrawn and all other operable control rods fully inserted, or all directional control valves for remaining control rods shall be disarmed electrically and sufficient margin to criticality shall be demonstrated.

c. If maintenance is to be performed on two control rod drives, they must be separated by more than two control cells in any direction.

d. An appropriate number of SRM's are

rod is fully withdrawn. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 980 lbs., in comparison to the load-trip setting of 650 lbs. 17
Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400 lb. load-trip setting on these hoists is adequate to trip the interlock when one of the more than 600 lb. fuel bundles is being handled.

During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. The maintenance is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated or that all remaining control rods have their directional control valves electrically disarmed ensures that inadvertent criticality cannot occur during this maintenance. The adequacy of the shutdown margin is verified by demonstrating that the core is shutdown by a margin of 0.38 percent Δk with the strongest operable control rod fully withdrawn, or that at least 0.38% Δk shutdown margin is available if the remaining control rods have had their directional control valves disarmed. Disarming the directional control valves does not inhibit control rod scram capability.

Specification 3.10.A.6 allows unloading of a significant portion of the reactor core. This operation is performed with the mode switch in the "refuel" position to provide the refueling interlocks normally available during refueling operations. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod provides primary reactivity control for the fuel assemblies in the cell associated with that control rod.

Thus, removal of an entire cell (fuel assemblies plus control rod) results

The staff concluded that a typographical error had been made. The requested change would be consistent with the fact that this instrument is already labeled PC-TA-20A, C on Table 3.2.F. and that instrument PC-TA-20B, D is listed in Table 4.2.F. as indicating Suppression Chamber Water Temperature.

(2) Tables 3.2.B. and 4.2.B. Relabeling of switch designations for the Reactor High Water Level Instrument NBI-LIS-101.

The proposed change would alter switch designations A & C #4 to B & D #2 on page 4 of Table 3.2.B. and switch designations A & C #3 to A & C #2 on page 6 of Table 3.2.B and page 6 of Table 4.2.B. The change would align the Technical Specifications with existing plant wiring diagrams. NPPD states that the current Technical Specification designations were copied from incorrect vendor drawings. NPPD has initiated action to correct the vendor drawings. The staff has concluded that the proposed change represents the correction of a prior administrative error.

(3) Table 3.2.B. The "Action Required When Component Operability Not Assured" for Residual Heat Removal (RHR) Crosstie Valve instrument RHR-LMS-2 would be changed from "D" to "B".

In the notes for Table 3.2.B., the letter D designation for a component directs that "The high point vent shall be vented weekly upon failure of PS 73A or B, PS 266, PS 268, PS 269." This action is appropriate for the failure of one of the pump discharge pressure switches listed in Table 3.2.B. The letter B designation directs "Declare the system or component inoperable." RHR valve MO-20 is equipped with a limit switch valve position indicator. Therefore, letter designation D is incorrect for this valve indication. The appropriate designation would be letter B.

(4) Tables 3.2.F. and 4.2.F. Reference to Suppression Chamber Pressure instrument PC-PR-512B would be eliminated.

Instrument PC-PR-512B was installed in response to question 7.7 of Amendment B to the CNS Facility Operating License. Its purpose is to monitor drywell pressure. In fact, this instrument is already listed in Tables 3.2.F. and 4.2.F. as indicating drywell pressure. The inclusion of this instrument as an indicator of suppression chamber pressure is a clerical error.

(5) Specification 4.9.A.1.d. The diesel fuel quality limits would be changed to meet either ASTM D975-68 standard 1D or 2D.

The present CNS Technical Specification states that fuel shall meet the limits of "Table 1 of ASTM D978-68 for Nos. 1D and 2D." The proposed change would align this specification with the diesel vendor's maintenance manual which requires diesel oil to meet standard 1D or 2D. The requirement to simultaneously meet the limits for Nos. 1D and 2D is not consistent with the normal use of Table 2 of the ASTMD978-68 and is clearly a clerical error. Since the change does not change the original intent of the specification and is consistent with the diesel vendor's requirements, the change is acceptable.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

DATE: OCT 1 7 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 to Facility Operating License No. DPR-46, issued to the Nebraska Public Power District (the licensee), which revised Technical Specifications for operation of the Cooper Nuclear Station (the facility) located in Nemaha County, Nebraska. The amendment is effective as of its date of issuance.

This amendment changes the fuel grapple hoist load switch setting from ≤1200 lbs. to ≤650 lbs. and corrects several clerical errors and oversights in various sections of the Technical Specifications.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated July 3, 1975, (2) Amendment No. 14 to License No. DPR-46, with Change No. 17 and (3) the Commission's concurrently issued Safety Evaluation. All of these items are available for

SAFETY EVALUATION BY OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 TO DPR-46

CHANGE NO. 17 TO TECHNICAL SPECIFICATIONS

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

INTRODUCTION

By letter dated July 3, 1975, Nebraska Public Power District (NPPD) requested changes to the Radiological Technical Specifications appended to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). Three changes were requested. The first and second dealt with Section 3.10 of the Technical Specifications concerning core alterations. The third was intended to correct several clerical errors and oversights in various sections of the Technical Specifications. The first change was the subject of a letter to NPPD, dated September 19, 1975, and could not be reviewed without a more complete safety analysis. This safety evaluation is concerned with the remaining change requests.

DISCUSSION AND EVALUATION

The second proposed change would reduce the "final grapple hoist load switch" setting from ≤ 1200 lbs. to ≤ 650 lbs. This switch senses the combined weight of the fuel grapple and whatever load the grapple carries. It is set to open at the load weight which is lighter than the weight of a single fuel assembly, thus providing positive indication whenever fuel is loaded on the grapple. The switch is part of the logic circuit of the refueling interlocks which reinforce operational procedures that prohibit taking the reactor critical during refueling by restricting movement of control rods and operation of refueling equipment. Since the weight of a fuel assembly of the type used at CNS is approximately 750 lbs. and the combined weight of fuel and grapple is approximately 980 lbs., the NRC staff (the staff) concluded that the present Technical Specification setting of ≤ 1200 lbs. could defeat the purpose of the refueling interlocks if the setpoint were between 980 lbs. and 1200 lbs. The reduction of the setting to ≤ 650 lbs. would ensure that the fuel grapple hoist load switch functioned at all times when a fuel assembly was loaded on the grapple.

The third proposed change would alter several sections of the Technical Specifications as indicated below:

(1) Table 4.2.F. Correction of the Suppression Chamber Air Temperature instrument designation from PC-TA-20B, D to PC-TA-20A, C.

public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Auburn Public Library, 1118 - 15th Street, Auburn, Nebraska 68305. A copy of items (2) and (3) may be obtained upon request addressed to the United States Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 17th day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

OFFICE ➤	RL:ORB #2	RL:ORB #2	OELD	RL:ORB #2		
SURNAME ➤	RMDiggs:ro	MFletcher RDSilver		DLZiemann		
DATE ➤	10/ 8/75	10/ /75	10/ /75	10/ /75		