JUN 1976

6/3/96

DISTRIBUTION
Docket
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BScharf (10)
JMcGough
OPA (CMiles)
TBAbernathy
JRBuchanan
ACRS (16)
DRoss

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:

In response to your letters dated January 19 and 26, and March 22, 1976, the Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station.

The amendment consists of changes in the Technical Specifications that change the leak test medium for various primary containment isolation valves from water to air and the correction of two administrative errors in the Technical Specification bases. The amendment also deletes provision no. 2 from the Order for Modification of License No. DPR-46 dated October 31, 1975.

As a result of our review of several reported LPCI valve motor failures at Cooper Nuclear Station, the amendment also includes changes to the Technical Specifications that include calibration and testing requirements for the undervoltage relays associated with the power transfer system for Motor Control Center RB. The appropriate Technical Specifications were submitted in your letter dated January 19, 1976.

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

Sincerely,

Original signed by
Dennis L. Ziemann

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

Enclosures.
1. Amendment No. 25 to License No. DPR-46
2. Safety Evaluation
3. Federal Register Notice

OFFICE	DOR:ORB #2	DOR:ORB #2	OELD	DOR:ORB #2		
SURNAME	MHFletcher:ro	RMDiggs	See att. yellow	DLZiemann		
DATE	6/1/76	6/1/76	5/27/76	6/3/76		

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Sincerely,

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

Enclosures:
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2. Safety Evaluation
3. Federal Register Notice

D.A. – as modified on this copy

OFFICE ➤	OR:ORB #2	OR:ORB #2	OELD	OR:ORB #2		
SURNAME ➤	RMDiggs	MFletcher:ro	Swanson	DLZiemann		
DATE ➤	5/20/76	5/20/76	5/27/76	5/ /76		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

cc w/enclosures:
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Mr. Arthur C. Gehr, Attorney
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Mr. William Siebert, Commissioner
Nemaha County Board of Commissioners
Nebraska County Courtroom
Auburn, Nebraska 68305

cc w/enclosures and NPPD filings
dtd. 1/19 & 26/76 and 3/22/76:
Mr. D. Drain, Director
Department of Environmental Control
Executive Building, Second Floor
Lincoln, Nebraska 68509

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
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 January 26 and March 22, 1976, 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 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. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications,as indicated in the attachment to this license amendment.

3. This license amendment is effective 30 days after the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION
Original signed by
Dennis L. Ziemann
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: JUN 3 1976

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace existing pages 21, 55, 60, 71, 153, 173, 174 and 175 of the Appendix A Technical Specifications with the attached revised pages bearing the same numbers. Changed areas on the revised pages are indicated by marginal lines.

2.1 Bases: (Cont'd)

2. Reactor Water Low Level Scram and Isolation Trip Setting (except MSIV)

The set point for low level scram is above the bottom of the separator skirt. This level has been used in transient analyses dealing with coolant inventory decrease. The results reported in FSAR Subsection 14.5 show that scram at this level adequately protects the fuel and the pressure barrier, because MCPR remains well above 1.06 in all cases, and system pressure does not reach the safety valve settings. The scram setting is approximately 25 in. below the normal operating range and is thus adequate to avoid spurious scrams.

3. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of $\leq$10 percent of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above 1.06 even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

4. Turbine Control Valve Fast Closure Scram Trip Setting

The turbine control valve fast closure scram anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection exceeding the capability of the bypass valves. The reactor protection system initiates a scram when fast closure of the control valves is initiated by the loss of turbine control oil pressure as sensed by pressure switches. This setting and the fact that control valve closure time is approximately twice as long as that for the stop valves means that resulting transients, while similar, are less severe than for stop valve closure. No significant change in MCPR occurs. Relevant transient analyses are presented in Paragraph 14.5.1.1 of the Final Safety Analysis Report.

Amendment No. 25

COOPER NUCLEAR STATION
TABLE 3.2.3 (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	$\geq$2500 gpm	1	A
Break Detection Time Delays	RHR-TDR-K28, A & B	0.25$\leq$T$\leq$0.75 sec.	1	A
	RHR-TDR-K40, A & B	0.25$\leq$T$\leq$0.75 sec.	1	A
	RHR-TDR-K34, A & B	1.5$\leq$T$\leq$2.5 sec.	1	A
	(RHR-TDR-K31, A & B)	9$\leq$T$\leq$11 min.	1	A
	RHR-TDR-K86, A & B	9$\leq$T$\leq$11 min.	1	A
	RHR-TDR-K45, 1A & 1B	4.25$\leq$T$\leq$5.75 min.	1	A
RHR Pump Start Time Delay	RHR-TDR-K75, A & B	4.5$\leq$T$\leq$5.5 sec.	1	A
	RHR-TDR-K70, A & B	$\leq$.5 sec.	1	A
RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	1.8$\leq$T$\leq$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 27 X 2/1G	Loss of Voltage Loss of Voltage	1	B
Pump Discharge Line Low Pressure	CM-PS-266	$\geq$5 psig	(3)	D
Undervoltage Relays	1V, 2V, 3V and LO	410 $\leq$ Transfer $\leq$ 430 (De-energize) 435 $\leq$ Return $\leq$ 455 (Energize)	(5)	B

Amendment No. 25

NOTES FOR TABLE 3.2.B

1. When any ECCS system is required to be operable, there shall be two operable trip systems except as noted. If a requirement of the fourth column is reduced by one, the indicated action shall be taken. If the same function is inoperable in more than one trip system or the fourth column reduced by more than one, action B shall be taken.

   Action:

   A. Repair in 24 hours. If the function is not operable in 24 hours, take action B.

   B. Declare the system or component inoperable.

   C. Immediately take action B until power is verified on the trip system.

   D. The high point vent shall be vented weekly upon failure of PS 73A or B, PS 266, PS 268, PS 269.

   E. Repair as soon as possible. It does not directly effect system operations.

2. In only one trip system.

3. Not considered in a trip system.

4. Requires one channel from each physical location in the steam line space.

5. One relay senses each phase of MCC-S and the LO relay is a transfer permissive relay.

Amendment No. 25

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 Low Water Level	NBI-LIS-72, A,B,C & D, #4	Once/Month (1)	Once/3 Months	Once/Day
3. Reactor Vessel Shroud Level	NBI-LITS-73, A & B #1	Once/Month (1)	Once/3 Months	Once/Day
4. Reactor Low Pressure	RR-PS-128 A & B	Once/Month (1)	Once/3 Months	None
5. Reactor Low Pressure	NBI-PS-52 A & C NBI-PIS-52 B & D	Once/Month (1)	Once/3 Months	None
6. Reactor Low Pressure	NBI-PS-50, A & B	Once/Month (1)	Once/3 Months	None
7. Reactor Low Pressure	NBI-PS-49, A & B	Once/Month (1)	Once/3 Months	None
8. Drywell Press.-Containment Spray	PC-PS-119, A,B,C & D	Once/Month (1)	Once/3 Months	None
9. Riser Differential Pressure	RR-dPIS-129, A,B,C & D	Once/Month (1)	Once/3 Months	None
10. Recirc. Pump Diff. Press.	RR-dPIS-136, A & B	Once/Month (1)	Once/3 Months	None
11. Recirc. Pump Diff. Press.	RR-dPIS-137, A & B	Once/Month (1)	Once/3 Months	None
12. Recirc. Pump Diff. Press.	RR-dPIS-138, A & B	Once/Month (1)	Once/3 Months	None
13. Recirc. Pump Diff. Press.	RR-dPIS-139, A & B	Once/Month (1)	Once/3 Months	None
14. RHR Pump Discharge Press.	RHR-PS-120, A,B,C & D	Once/Month (1)	Once/3 Months	None
15. RHR Pump Discharge Press.	RHR-PS-105, A,B,C & D	Once/Month (1)	Once/3 Months	None
16. RHR Pump Low Flow Switch	RHR-dPIS-125 A & B	Once/Month (1)	Once/3 Months	None
17. Break Detection Time Delay	RHR-TDR-K28, A & B	Once/Month (1)	Once/Oper. Cycle	None
18. Break Detection Time Delay	RHR-TDR-K40, A & B	Once/Month (1)	Once/Oper. Cycle	None
19. Break Detection Time Delay	RHR-TDR-K34, A & B	Once/Month (1)	Once/Oper. Cycle	None
20. Break Detection Time Delay	RHR-TDR-K81, A & B	Once/Month (1)	Once/Oper. Cycle	None
21. Break Detection Time Delay	RHR-TDR-K86, A & B	Once/Month (1)	Once/Oper. Cycle	None
22. Break Detection Time Delay	RHR-TDR-K45, A & B	Once/Month (1)	Once/Oper. Cycle	None
23. RHR Pump Start Time Delay	RHR-TDR-K70, A & B	Once/Month (1)	Once/Oper. Cycle	None
24. RHR Pump Start Time Delay	RHR-TDR-K75, A & B	Once/Month (1)	Once/Oper. Cycle	None
25. RHR Heat Exchanger Bypass T.D.	RHR-TDR-K93, A & B	Once/Month (1)	Once/Oper. Cycle	None
26. RHR Cross Tie Valve Position	RHR-LMS-2	Once/Month (1)	N.A.	
27. Low Voltage Relays	27 X 3/1A	(7)		None
28. Low Voltage Relays	27 X 3/1B	(7)		None
29. Low Voltage Relays	27 X 2/1F, 27 X 2/1G	(7)		None
30. Low Voltage Relays	27 X 1/1F, 27 X(1)1G	(7)		None
31. Pump Disch. Line Press. Low	CM-PS-266	Once/3 Months	Once/3 Months	None
32. Undervoltage Relays	1V, 2V, 3V and 1O	Once/6 Months	Once/Oper. Cycle	None

Amendment 116 8-5

3.6.G & 4.6.G BASES (cont'd.)

Category J (1)

The more frequent inspections delineated for the Category J (1) pipe welds are to provide additional conservatism in the overall approach of protection against pipe whip which has the potential to breach the containment. A pipe whip protection system has been installed consisting of crushable panels located such that the postulated pipe weld failure will not breach the containment. Additional inspection of critical welds is also included in the inservice inspection program. The J (1) welds listed on Table 4.6.4, are those pipe welds of interest.

An examination of the vessel closure head internal surface will be made at the end of the first ten year period. The rate of corrosion will be evaluated and compared to the design corrosion rate. The results of this study will be used to formulate the requirements for future testing.

Additional information on the CNS Inservice Inspection Program is presented in the FSAR Appendix J.

Amendment No. 25

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 Valve	Water
X-9A	RCIC-AO-22, RCIC-MO-17, and RWCU-15CV, RCIC/RWCU Connection to Feedwater	Water
X-9B	RF-13CV and RF-14CV, Feedwater Check Valves	Water
X-9B	HPCI-AO-18 and HPCI-MO-57, HPCI Connection to Feedwater	Water
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	MV-1305 and MV-1306, ACAD Supply to Drywell	Air
X-26	PC-231MV and PC-246AV, Purge and Vent Exhaust from Drywell	Air

TABLE 3.7.4 (page 2)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-36	CRD-11CV and CRD-12CV, CRD Exhaust Water	
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-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	MV-1303 and MV-1304, ACAD 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-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-220	PC-230MV and PC-245AV, Purge and Vent Exhaust 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

TABLE 3.7.4 (page 3)

PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-223A	CS-MO-26A and CS-MO-5A, Core Spray Test and Minimum Flow	Air
X-223B	CS-MO-26B and CS-MO-5B, Core Spray Test and Minimum Flow	Air
X-225A-D	RHR-MO-13A, RHR-MO-13C, RHR-MO-13B, RHR-MO-13D, RHR Suction From Torus	Air
X-224	RCIC-MO-41, RCIC Suction From Torus	Air
X-226	HPCI-MO-58, HPCI Suction From Torus	Air
X-227A,B.	CS-MO-7A,and CS-MO-7B, Core Spray Suction From Torus	Air

Amendment No. 25

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO DPR-46

COOPER NUCLEAR STATION

DOCKET NO. 50-298

INTRODUCTION

By letters dated January 26 and March 22, 1976, Nebraska Public Power District (NPPD) requested amendments to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS). The requests would (1) change the leak testing medium of various primary containment isolation valves from water to air, (2) correct two administrative errors in the bases to the Appendix A Technical Specifications and (3) delete a monitoring program for detecting instrument tube-fuel channel box interactions required by the Order for Modification of License dated October 31, 1975.

As a result of our investigation of several reported failures of low pressure coolant injection valve motors at CNS, added surveillance requirements for the undervoltage relays associated with the power supplies to Motor Control Center RB (MCC-RB) are considered necessary. Appropriate changes were provided by NPPD's letter of January 19, 1976.

DISCUSSION AND EVALUATION

By letter dated August 5, 1975, the Commission requested NPPD to determine if containment leak testing at CNS was being performed in full compliance with 10 CFR 50 Appendix J. As a result of this request, NPPD has requested to change the leak testing medium of certain containment isolation valves from water to air. The penetrations in question require type C testing as defined in Part II of Appendix J.

The NRC staff has concluded that the requested change of testing medium from water to air for the type C containment penetrations listed in the NPPD January 26 submittal would be in conformance with the testing procedures of 10 CFR 50 Appendix J. Specifically, paragraph C.2.(a) of Part III of Appendix J requires type C isolation valves to be tested by pressurizing with air or nitrogen. On this basis the requested change is acceptable.

As part of the remedial action for preventing flow induced impact of instrument tubes against fuel assembly channel boxes in the CNS core, the Commission, on October 8, 1975, issued an Order for Modification of License which authorized NPPD to install plugs in the lower core plate bypass holes. To determine if plugging the bypass holes prevented significant impact, the Order for Modification of License, dated October 31, 1975, required NPPD to conduct a monitoring program using Local Power Range Monitor (LPRM) and Traveling Incore Probe (TIP) traces and accelerometers on incore instrument guide tubes for the purpose of detecting any instrument tube-channel box interaction. In accordance with the monitoring program, NPPD has taken data at various core flow rates up to 100% flow. The data was submitted with the March 22 request.

We have reviewed the results of the LPRM, TIP and accelerometer monitoring program at CNS and concluded that the installation of lower core plate plugs has eliminated significant in-core vibration. This monitoring program has accomplished its intended objectives and therefore may be discontinued. Performance of LPRM and TIP traces will revert to the frequency required by the Technical Specifications.

CNS Abnormal Occurrence (AO) Reports 74-30, 74-31, 74-49, and 74-57 described incidents which resulted in failure of four motor operators on Low Pressure Coolant Injection (LPCI) valves during surveillance testing. Concern that a continuing high failure rate of motor operators could degrade the LPCI safety function prompted a review by the NRC staff of the causes of the failures and corrective actions taken to prevent further failures. The four motor failures that occurred involved the valve operator motor of inboard LPCI injection valves RHR-MO-25A and RHR-MO-25B. The cause of failure of these motors was the attempted operation of valves with a differential pressure across the valve disc which exceeded the design capability of the valve operator. No motor overload protection is provided for safety related valves in the Cooper design. Therefore, an overload condition results in failure of the motor. During normal automatic operation, pressure interlocks are provided to prevent operation of these valves during existence of a high differential pressure condition. Information concerning the motor operator failures has been submitted by NPPD by letters dated October 3, 1974, February 4, 1975, and January 19, 1976.

We have reviewed the information supplied by NPPD relative to the LPCI valve failures and concluded that the corrective actions which have been taken will significantly reduce future valve failures. System modifications made by NPPD have been in accordance with the rules and regulations of the Commission.

AO-74-57 also described a failure of the power supply transfer system associated with MCC-RB. A transfer attempt was initiated by the voltage transient resulting from failure of the motor of valve RHR-25A which is powered from MCC-RB. Under normal conditions, a low voltage transient on the normal power supply to MCC-RB would be sensed by undervoltage (UV) relays. The UV relays would then transfer MCC-RB to its alternate power supply. Investigation of the failure to transfer indicated that the input voltage to the UV relay was set too low. To prevent further failures of this type, NPPD has written maintenance procedures for testing UV relays. However, the procedures are not incorporated into the CNS Technical Specifications. In a letter dated January 19, 1976, NPPD presented Technical Specifications for calibration and functional testing of UV relays.

We have concluded that the design of the MCC-RB power transfer system is acceptable. However, to assure proper operation of the system, the proposed specifications for UV relays should be incorporated into the CNS Technical Specifications. We have reviewed the proposed specifications and find that they are acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do 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: **JUN 3 1976**

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 25 to Facility Operating License No. DPR-46, issued to Nebraska Public Power District, which revised Technical Specifications for operation at the Cooper Nuclear Station (the facility) located in Nemaha County, Nebraska. The amendment becomes effective 30 days after the date of issuance.

This amendment revises the Technical Specifications for the facility to change the leak test medium for various primary containment isolation valves from water to air and to include calibration and testing requirements for the undervoltage relays associated with the power transfer system for Motor Control Center RB. The amendment also deletes provision no. 2 from the Order for Modification of License dated October 31, 1975. This provision required the performance of a monitoring program to detect instrument tube-channel box interaction in the facility core.

The applications for amendment comply 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

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amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration. The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the applications for amendment dated January 19, January 26, and March 22, 1976, (2) Amendment No. 25 to License No. DPR-46, and (3) the Commission's concurrently issued related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Auburn Public Library, 118 - 15th Street, Auburn, Nebraska 68305. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attnetion: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 3rd day of June, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

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

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