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Docket No. 50-298

APR 27 1977

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 requests dated January 21, 1976 and March 4, 1977, the Commission has issued the enclosed Amendment No. 26 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station (CNS).

The amendment modifies the CNS Technical Specification requirements by deleting a temporary restriction regarding the reactor internals vibration test program, altering the Automatic Depressurization System timer setpoint, adding operability and surveillance requirements for recirculation pump discharge and bypass valves, and adding new surveillance requirements for safety/relief valves.

In addition to the above Technical Specification changes which you requested, the amendment corrects various errors in specifications 3.2.D.4 "Main Control Room Ventilation Isolation", 3.5.C "HPCI Subsystem", 3.5.D "Reactor Core Isolation Cooling Subsystem", and 3.5.E "Automatic Depressurization System". These changes have been discussed with your staff.

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

Sincerely,

Original signed by

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. ~~26~~ 30 to License No. DPR-46
2. Safety Evaluation
3. Notice

SAFETY/RELIEF PORTION ONLY

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DATE	4/6/77	4/6/77	4/6/77	4/25/77	4/27/77	

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U. S. Environmental Protection Agency
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Mr. William Siebert, Commissioner
Nemaha County Board of Commissioners
Nebraska County Courtroom
Auburn, Nebraska 68305

cc w/enclosures and cy of NPPD's
filings dtd. 1/21/76 and 3/4/77:
Director, Department of Environmental
Control
Executive Building, 2nd Floor
Lincoln, Nebraska 68509

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. 36
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 21, 1976 and March 4, 1977, 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. 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. 36, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 27, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace existing pages iv, 48, 59, 63, 116, 117, 118, 119, 120, and 136 of the Appendix A portion of the Technical Specifications with the attached revised pages bearing the same number. The changed areas on the revised pages are reflected by a marginal line.

TEMPORARY RESTRICTIONS

1. Drilling of alternate flow path holes in the lower tie plates of unirradiated fuel bundles at the CNS site is permitted provided the procedures of Section 3 of General Electric Document NEDE 21156 are followed and GE personnel, or personnel properly trained by the General Electric Company, perform the drilling.

2. Machining of alternate flow path holes in the lower tie plates of irradiated fuel bundles by means of electrical-discharge machining at the CNS site is permitted provided that the procedures described in letters from G. C. Ross, GE, to D. G. Eisenhut, NRC, dated April 1, and April 23, 1976 are followed and GE personnel, or personnel properly trained by the General Electric Company, perform the machining.

3. The above restrictions apply until removed by written instructions of the NRC staff.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.2.C (cont'd.)	4.2.C
D. Radiation Monitoring Systems - Isolation & Initiation Functions	D. Radiation Monitoring Systems - Isolation & Initiation Functions
1. Steam Jet Air Ejector Off-Gas System	1. Steam Jet Air Ejector Off-Gas System
a. Except as specified in Specification 2.4.3.a.7 of Appendix B, both steam jet air ejector off-gas system radiation monitors shall be operable.	Instrumentation surveillance requirements are given on Table 4.2.D.
b. The time delay setting for closure of the steam jet air ejector isolation valves shall not exceed 15 minutes.	
c. Other limiting conditions for operation are given on Table 3.2.D and Sections 2.4.3.a.6.b and 2.4.3.a.7 of the Environmental Technical Specifications.	
2. Reactor Building Isolation and Standby Gas Treatment Initiation The limiting conditions for operation are given on Table 3.2.D and Section 2.4.3.a of Appendix B.	2. Reactor Building Isolation and Standby Gas Treatment Initiation Instrumentation surveillance requirements are given on Table 4.2.D.
3. Liquid Radwaste Discharge Isolation The limiting conditions for operation are given on Table 3.2.D and Section 2.4.1.b.3 of Appendix B.	3. Liquid Radwaste Discharge Isolation Instrumentation surveillance requirements are given on Table 4.2.D and Section 3.4.1.b.7 of the Environmental Technical Specifications.
4. Main Control Room Ventilation Isolation The limiting conditions for operation are given on Table 3.2.D and the Section entitled "Additional Safety Related Plant Capabilities".	4. Main Control Room Ventilation Isolation The instrument surveillance requirements are given on Table 4.2.D.

COOPER NUCLEAR STATION
TABLE 3.2.B (Page 7)
AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) 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
Drywell High Pressure	PC-PS-100, A,B,C & D	$\leq$ 2 psig	2	A
Reactor Low Water Level	NBI-LIS-83, A & B	$\geq$ +12.5" Indicated Level	1	B
	NBI-LIS-72, A,B,C & D	$\geq$ -145.5" Indicated Level	2	A
ADS Timer	MS-TDR-K5, A & B	$\leq$ 120 sec.	1	B

COOPER NUCLEAR STATION

TABLE 3.2.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE SYSTEMS

System	Instrument I.D. No.	Setting Limit	Number of Sensor Channels Provided by Design	Action (1)
Steam Jet Air Ejector Off-Gas System	RMP-RM-150 A & B	(2)	2	A
Reactor Building Isolation and Standby Gas Treatment Initiation	RMP-RM-452 A & B	<100 mr/hr	2	B
Liquid Radwaste Discharge Isolation	RMV-RM-2	(3)	1	C
Main Control Room Ventilation Isolation	(RMV-RM-1)	$4x10^3$ CPM	1	D
Mechnical Vacuum Pump Isolation	RMP-RM-251 A-D	3 times normal full power background. Alarm at 1.5 times normal full power background.	4	E

NOTES FOR TABLE 3.2.D

1. Action required when component operability is not assured.
   - A. Refer to Section 2.4.3.a.7 of the Environmental Technical Specifications.
   - B. Cease refueling operations, isolate secondary containment and start SBGT.
   - C. Refer to Sections 2.4.1.b of the Environmental Technical Specifications
   - D. Refer to Section entitled "Additional Safety Related Plant Capabilities".
   - E. Refer to Section 3.2.D.5 and the requirements for Primary Containment Isolation on high main steam line radiation. Table 3.2.A.

2. Trip setting to correspond to Specification 2.4.3.a.1 of the Environmental Technical Specifications.

3. Trip setting to correspond to Specification 2.4.1.b.1 of the Environmental Technical Specifications.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.A (Cont'd.)	4.5.A (Cont'd.)
5. From and after the date that one LPCI subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days, unless it is sooner made operable, provided that during such 7 days all active components of both core spray subsystems, the containment cooling subsystems (including 2 LPCI pumps) and the diesel generators required for operation of such components shall be operable.	5. When it is determined that the LPCI subsystem is inoperable, both core spray subsystems, the containment cooling subsystem and the diesel generators required for operation of such components if no external source of power were available shall be demonstrated to be operable immediately and daily thereafter.
6. All recirculation pump discharge valves and bypass valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).	6. All recirculation pump discharge and bypass valves shall be tested for operability during any period of Reactor cold shutdown exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.
7. The reactor shall not be started up with the RHR system supplying cooling to the fuel pool.	
8. If the requirements of 3.5.A 1,2,3,4,5,6 or 7 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.	
B. <u>Containment Cooling Subsystem (RHR Service Water)</u>	B. <u>Containment Cooling Subsystem (RHR Service Water)</u>
1. Except as specified in 3.5.B.2, 3.5.B.3, and 3.5.F.3 below both containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, and prior to reactor startup from a Cold Condition.	1. Containment Cooling Subsystem Testing shall be as follows:

Item	Frequency
a. Pump & Valve Operability	Once/3 months
b. Pump Capacity Test. Each RHR service water booster pump shall deliver 4000 gpm.	After pump maintenance and every 3 months
c. Air test on drywell and torus headers and nozzles.	Once/5 years

3.5.B (Cont'd.)

2. From and after the date that any RHR service water booster pump is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days, unless such pump is sooner made operable provided that during such thirty days all other active components of the containment cooling subsystem are operable.

3. From and after the date that one containment cooling subsystem loop is made or found to be inoperable for any reason, continued reactor opertion is permissible only during the succeeding seven days unless such subsystem loop is sooner made operable, provided that all active components of the other containment cooling subsystem loop, including its associated diesel generator, are operable.

4. If the requirements of 3.5.B.1, 3.5.B.2 or 3.5.B.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

C. HPCI Subsystem

1. The HPCI Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 113 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.C.2 and 3.5.C.3 below.

4.5.B (Cont'd.)

2. When it is determined that any RHR service water booster pump is inoperable, the remaining active components of the containment cooling subsystems shall be demonstrated to be operable immediatly and weekly thereafter.

3. When one containment cooling subsystem loop becomes inoperable, the operable subsystem loop and its associated diesel-generator shall be demonstrated to be operable immediately and the operable containment cooling subsystem loop daily thereafter.

C. HPCI Subsystem

1. HPCI subsystem testing shall be performed as follows:

	Item	Frequency
a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3.5.C <u>HPCI Subsystem</u> (cont'd.)

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, providing that during such seven days all active components of the ADS subsystem, the RCIC system, the LPCI subsystem and both core spray subsystems are operable.

3. With the surveillance requirements of 4.5.C not performed at the required intervals due to reactor shutdown, a reactor startup may be conducted provided the appropriate surveillance is performed within 48 hours of achieving 150 psig reactor steam pressure.

4. If the requirements of 3.5.C.1 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 113 psig or less within 24 hours.

4.5.C <u>HPCI Subsystem</u> (cont'd.)

	Item	Frequency
d.	Flow Rate at approximately 1000 psig Steam Press.	Once/3 months
e.	Flow Rate at approximately 150 psig Steam Press.	Once/operating cycle

The HPCI pump shall be demonstrated to be capable of delivering at least 4250 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

2. When it is determined that the HPCI Subsystem is inoperable the RCIC, the LPCI subsystem, both core spray subsystems, and the ADS subsystem actuaation logic shall be demonstrated to be operable immediately. The RCIC system and ADS subsystem logic shall be demonstrated to be operable daily thereafter.

D. <u>Reactor Core Isolation Cooling (RCIC) Subsystem</u>

1. The RCIC Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor pressure is greater than 113 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 and 3.5.D.3 below.

D. <u>Reactor Core Isolation Cooling (RCIC) Subsystem</u>

1. RCIC Subsystem testing shall be performed as follows:

	Item	Frequency
a.	Simulated Automatic Actuation Test	Once/operating cycle

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.D (cont'd.)	4.5.D (cont'd.)

	Item	Frequency
b.	Pump Operability	Once/month
c.	Motor Operated Valve Operability	Once/month
d.	Flow Rate at approximately 1000 psig Steam Pressure	Once/3 months
e.	Flow Rate at approximately 150 psig Steam Pressure	Once/operating cycle

The RCIC pump shall be demonstrated to be capable of delivering at least 400 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

2. When it is determined that the RCIC subsystem is inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.

3.5.D (cont'd.)

2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.

3. With the surveillance requirements of 4.5.D not performed at the required intervals due to reactor shutdown, a reactor startup may be conducted provided the appropriate surveillance is performed within 48 hours of achieving 150 psig reactor steam pressure.

4. If the requirements of 3.5.D 1 & 2 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 113 psig or less within 24 hours.

E. Automatic Depressurization System (ADS)

1. The Automatic Depressurization Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 113 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 and 3.5.E.3 below.

E. Automatic Depressurization System (ADS) (Surveillance Requirement)

1. During each operating cycle the following tests shall be performed on the ADS:

   A simulated automatic actuation test shall be performed prior to startup after each refueling outage.

3.5.E (cont'd.)

2. From and after the date that one valve in the automatic depressurization subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such valve is sooner made operable, provided that during such seven days the HPCI subsystem is operable.

3. With the surveillance requirements of 4.6.D.5 not performed at the required intervals due to reactor shutdown, a reactor startup may be conducted provided the appropriate surveillance is performed within 12 hours of achieving 113 psig reactor steam pressure.

4. If the requirements of 3.5.E.1 or 3.5.E.2 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 113 psig within 24 hours.

F. <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u>

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that all of the low pressure core and containment cooling subsystems and the remaining diesel generator shall be operable and the requirements of 3.9.A.1 are met. If this requirement cannot be met, the requirements of 3.5.F.2 shall be met.

2. During any period when both diesel generators are inoperable, continued reactor operation is permissible only during the succeeding 24 hours unless one diesel generator is sooner made operable, provided that all the low pressure core & containment cooling subsystems are operable & the reactor power level is reduced to 25% of rated power and the requirements of 3.9.A.1 are met. If this requirement cannot be met, either the requirements shall be met or an orderly shutdown shall be initiated and the reactor placed in the cold shutdown condition within 24 hours.

4.5.E (cont'd.)

2. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the other ADS valves and the HPCI subsystem shall be demonstrated to be operable immediately and at least weekly thereafter.

F. <u>Minimum Low Pressure Cooling and Diesel Generator Availability</u>

1. When it is determined that one diesel generator is inoperable, all low pressure core cooling and containment cooling subsystems shall be demonstrated to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

LIMITING CONDITIONS FOR OPERATION

3.6.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^{o}F, all three safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2

2.

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 cold shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.D Safety and Relief Valves

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. At least one of the relief valves shall be disassembled and inspected each refueling outage.

3. The integrity of the relief safety valve bellows shall be continously monitored.

4. The operability of the bellows monitorin system shall be demonstrated once every three months.

5. Once per operating cycle, with the reactor pressure $\geq$ 100 psig, each relief valve shall be manually opened until the main turbine bypass valves have closed to compensate for relief valve opening.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

INTRODUCTION

By letter dated January 21, 1976, Nebraska Public Power District (NPPD) requested the removal of a temporary restriction from the Appendix A Technical Specifications for Cooper Nuclear Station (CNS). The restriction relates to reactor internals vibration assessment during startup and pre-operational testing.

By letter dated March 4, 1977, NPPD proposed changes to the CNS Technical Specifications to alter the Automatic Depressurization System (ADS) timer setpoint, add operability and surveillance requirements for recirculation pump discharge and bypass valves, and add new surveillance requirements for safety/relief valves.

In addition to the changes proposed by NPPD, other Technical Specification changes are addressed in this evaluation. The first change would correct an erroneous reference to specification 3.11.A in paragraph 3.2.D.4 and Table 3.2.D of the Technical Specifications. The second would remove an inconsistency from the operability testing requirements for the High Pressure Coolant Injection (HPCI), the ADS, and the Reactor Core Isolation Cooling (RCIC) systems.

DISCUSSION/EVALUATION

Reactor Internals Vibration Temporary Restriction

Regulatory Guide 1.20, formerly Safety Guide 20, describes an acceptable method for verifying the structural integrity of reactor internals subjected to flow induced vibrations during preoperational and startup testing. To assure that CNS met the requirements of Regulatory Guide 1.20, two temporary restrictions were included in the Technical Specifications issued with the initial CNS operating license. The first of these restrictions, limited CNS to less than 1% power until the results of the reactor internal vibration cold flow tests could be completed and compared with the cold flow test results from the James A. Fitzpatrick Nuclear Power Plant (Docket No. 50-333). The Fitzpatrick facility had been designated as the prototype plant for reactor internals vibration testing for plants similar to CNS. Upon favorable comparison of cold flow test results, this temporary restriction to 1% power was deleted by Change No. 1 to the CNS Technical Specifications, dated February 28, 1974.

The second temporary restriction required that, upon completion of the startup test program at CNS, the hot (up to full power) vibration results for CNS be compared with the hot vibration results of the Fitzpatrick plant. In the event that the Fitzpatrick tests or results of startup programs and power operation of Browns Ferry Unit No. 1 indicated that potential problems in areas which could not have been detected in CNS because of differences in instrumentation, appropriate corrective action would be required at CNS.

By letter dated January 21, 1976, NPPD submitted data which compares the hot vibration test results for CNS with the results from the Fitzpatrick facility and, based on this comparison, requested that the remaining reactor internals vibration temporary restriction be removed.

The NRC staff has reviewed the data submitted by NPPD and determined that the results of the reactor internals vibration tests at CNS are acceptable and that no further vibration testing or inspections are necessary. Therefore, we conclude that the temporary restriction regarding reactor internals vibration may be removed from the CNS Technical Specifications.

<u>ADS Timer Setpoint</u>

In its March 4, 1977 letter, NPPD requested to change the ADS timer setpoint from the current Technical Specification limit of $115<T<125$ (seconds) to $\leq$120 seconds. To justify this proposed change, NPPD states that "the current technical specification setting limit does not provide adequate tolerance to account for instrument inaccuracies. Therefore, instrument drift has resulted in excessive numbers of 'reportable occurrence' reports. General Electric, on June 3, 1976, issued FDI No. NSGX to change the setpoint from 120 seconds to 109 seconds. An instrument tolerance of 10% places the upper limit of the timer delay at 120 seconds. Since the current CNS ECCS Appendix K Analysis assumes a timer delay of 120 seconds, the proposed setpoint change is consistent with the analysis results".

The NRC staff has determined that, in addition to the secondary benefit of reducing the number of reportable occurrences associated with the ADS timer setpoint, the proposed limit is consistent with the currently approved Emergency Core Cooling System (ECCS) analysis and is, therefore, more appropriate than the current Technical Specification limit. We conclude that the proposed limit is acceptable.

Recirculation Pump Discharge and Bypass Valves

Based on discussions with the NRC staff, NPPD has proposed to add the following operability and surveillance requirements to the CNS Technical Specifications:

> Limiting Condition for Operation 3.5.A.6:
>
> "All recirculation pump discharge valves and bypass valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications)."
>
> Surveillance Requirement 4.5.A.6:
>
> "All recirculation pump discharge and bypass valves shall be tested for operability during any period of reactor cold shutdown exceeding 48 hours, if operability tests have not been performed during the preceding 31 days."

CNS is one of several boiling water reactors on which the low pressure coolant injection system (LPCIS) modification has been completed. This modification included the removal of the loop selection logic feature of the original design and instituted instead the simultaneous opening of both LPCIS injection valves. This was done to insure at least one half of the LPCIS injection capacity could be available after a postulated loss-of-coolant accident (LOCA) (i.e. suction line break). Another essential element of this modification involves the closure of the recirculation pump discharge valves (RPDV) and bypass valves (BV) upon LPCIS initiation following a loss-of-coolant accident. The closure of the RPDV and BV is necessary to isolate a pipe rupture occurring in the recirculation loop suction line and thereby ensure that the LPCIS will not discharge makeup water back through the recirculation pump or bypass line and out of the break. The failure of a RPDV or BV to close upon LPCIS initiation has an adverse affect on core cooling similar to the failure of a LPCIS injection valve to open. The failure of a LPCIS injection valve to open is the limiting single failure in the CNS Emergency Core Cooling System (ECCS) analysis.

Currently, the RPDV's and BV's are tested only during refueling outages which occur every 12 - 18 months. The standard interval for surveillance testing of motor operated ECCS valves is 31 days. We consider it desirable for surveillance to be performed on the RPDV and BV (if installed) with a surveillance frequency similar to that for the LPCIS injection valves. However, unlike the LPCIS injection valves, the RPDV's and BV's cannot be tested during power operation. We have considered the known safety effects on the plant associated with a plant shutdown and cooldown, solely for the purpose of testing these valves and have determined that the increase in reliability that might be gained does not justify an interruption in normal plant operations. Therefore, we have required that the RPDV and BV be tested during periods of reactor cold shutdown in excess of 48 hours if they have not been tested in the previous 31 days. For most operating BWR's today this cold shutdown period would occur, because of the necessity for maintenance and other planned operations, every 3 - 4 months. This expected outage frequency is consistent with the 3 month surveillance interval specified in Section XI of the ASME Boiler and Pressure Vessel Code.

On the basis of the foregoing, we consider the proposed changes to be improvements to overall plant safety and reliability; therefore, the changes are acceptable.

Safety/Relief Valves

By letter dated January 5, 1977, we notified NPPD of a potential deficiency in the method of confirming valve operability during periodic testing of safety/relief valves. This deficiency stems from using the valve temperature indication as a positive method to confirm that a safety/relief valve is open. We have found that an increased temperature indication may be obtained at the safety-relief valve exit with the safety/relief valve closed. This indicated temperature increase is the result of steam vented through the valve actuation mechanism during the surveillance test. In view of this finding, we concluded that a temperature increase at the valve exit, by itself, does not provide a positive means of verification that the safety/relief valve has opened, and we required NPPD to propose revised Technical Specifications to eliminate this deficiency.

In its March 4, 1977 letter, NPPD proposed to modify the surveillance requirement for safety/relief valve periodic testing to include a requirement to observe main turbine bypass valve closure, in addition to an increase in valve temperature, to positively demonstrate opening of a safety/relief valve. When a safety/relief valve is passing steam, the turbine bypass valves close to compensate for the reduced steam flow to the main turbine.

In the interest of maintaining consistency between the safety/relief valve specification for CNS and those of other plants and because the observation of turbine bypass valve closure affords a positive means, in itself, of verification of safety/relief valve operation, we have modified the specification proposed by NPPD to delete the requirement to observe an increase in valve temperature.

We conclude that NPPD's proposed Technical Specification for safety/relief valve testing, as modified, fulfills our requirement that a positive means of verification of valve opening exists and, therefore, is acceptable.

HPCI, ADS, and RCIC Testing

The HPCI, ADS, and RCIC systems all require reactor steam to function as designed. The HPCI and RCIC systems utilize steam to power their respective turbine-driven pumps. The ADS relieves reactor steam pressure to effect recirculation system depressurization. The CNS Technical Specifications for these systems require periodic testing of system components to ensure that the system can perform their intended functions if the need arises. In addition, the Technical Specifications require these systems to be capable of performing their intended functions prior to reactor startup from the cold condition (defined as reactor coolant temperature less than or equal to 212°F). These Technical Specification requirements have been shown to be inconsistent. For example, if the reactor remained in the cold shutdown condition for sufficient time to preclude performance of the HPCI flow rate test (required once per 3 months), HPCI operability could not be assured because reactor steam is required to operate the HPCI turbine and pump. However, a reactor startup cannot be performed because the ability of the HPCI pump to develop the specified flow rate, and hence HPCI system operability, have not been demonstrated. As another example, if maintenance is performed on an ADS valve while the reactor is in cold shutdown, a complete test of valve operability could not be conducted because cycling the valve requires reactor steam. But the reactor cannot be started up because operability of the repaired ADS valve cannot be demonstrated.

To eliminate these inconsistencies, the NRC staff compared the CNS Technical Specification requirements for HPCI, ADS, and RCIC to the requirements of the Standard Technical Specifications which we currently apply to new facilities. Based on this comparison, we have determined that a new provision should be added to the CNS Technical Specifications to permit testing these systems, in accordance with the appropriate Technical Specification Surveillance Requirements, within a specified period of time after reactor startup. In accordance with the Standard Technical Specifications, the specified time period would be 48 hours for HPCI and RCIC, and 12 hours for ADS. We have concluded that these changes are consistent with the requirements which we apply to new facilities, are applicable to the CNS Technical Specifications and, therefore, are acceptable.

Technical Specification Correction

Amendment No. 16 to the CNS license, dated October 31, 1975, incorporated a new section 3.11 into the Technical Specifications. This section entitled "Fuel Rods" replaced the section entitled "Additional Safety Related Plant Capabilities" which was renumbered section 3.12. Technical

Specification paragraph 3.2.D.4 and Table 3.2.D which reference "Additional Safety Related Plant Capabilities" were not changed to reflect the renumbering of sections 3.11 and 3.12. To prevent future recurrences of this error paragraph 3.2.D.4 and Table 3.2.D will be changed to reference "Additional Safety Related Plant Capabilities" by title rather than by number.

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 impact statement or negative declaration and 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 amendment 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 amendment 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: April 27, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-298

NEBRASKA PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 36 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.

The amendment deleted a temporary restriction regarding reactor internals vibration testing, altered the Automatic Depressurization System timer setpoint, added operability and surveillance requirements for recirculation pump discharge and bypass valves, added new surveillance requirements for safety/relief valves, and corrected errors in the specifications for Main Control Room Ventilation Isolation, High Pressure Coolant Injection Subsystem, and Automatic Depressurization System.

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 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 impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated January 21, 1976 and March 4, 1977, (2) Amendment No. 36 to License No. DPR-46, and (3) the Commission's 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 single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 27th day of April, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors