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MAY 23 1979

Docket No. 50-298

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

Dear Mr. Pilant:

The Commission has issued the enclosed Amendment No. 57 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications in response to your requests of November 29, 1978 and January 30, 1979.

The amendment modifies the Technical Specifications to permit operation of the facility up to 1% of rated thermal power while performing training startups with the RHR pumps in the shutdown cooling mode. In addition we have made certain administrative modifications in your Technical Specifications according to your November 29, 1978 proposal. For both your November 29, 1978 and your January 30, 1979 proposals we have made changes in your proposed Technical Specifications which we have discussed with you and you have found acceptable.

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

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Contr
CCP

Enclosures:
1. Amendment No. 57 to DPR-46
2. Safety Evaluation
3. Notice

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cc w/enclosures:
see next page

OFFICE	ORB#3	ORB#3	AD/E&PV/DOR	OELD	ORB#3	DOR
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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. 57
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 November 29, 1978 and January 30, 1979, 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.

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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 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. 57, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 23, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 57

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

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

Remove	Insert
65	65
96	96
102	102
115	115
122	122
124	124
128	128
137c	137c
137d	137d
137e	137e
137f	-
137g	-
137h	-
137i	-
167a	167a
171	171

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-TR-23, Ch 1 & 2	0 - 300°F 0 - 400°F	2	A,B,C
Suppression Chamber Water Temperature	PC-TR-21B PC-TR-22, Ch 1 & 2	0 - 300°F 0 - 400°F	2	A,B,C
Suppression Chamber Water Level	PC-LI-10 PC-LR-11	(-4' to +6') (-4' to +6')	2	A,B,C
	PC-LI-12 PC-LI-13	-10" to +10" -10" to +10"	2	A,B,C,E
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
Torus to Drywell Differential Pressure	PC-dPR-20	0 - 2 psid	1	A,B,C,E
Suppression Chamber/ Drywell Pressure (ΔP)	PC-PR-20/513 (2)	0 - 2 psig	1	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.3.B.3 (cont'd)	4.3.B.3.b (cont'd)
e. If Specifications 3.3.B.3a through d cannot be met, the reactor shall not be started, or if the reactor is in the run or startup modes at less than 20% rated power, it shall be brought to a shutdown condition immediately.	1) The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified.
f. The sequence restraints imposed on the control rods may be removed by the use of the individual rod position bypass switches for scram testing only those rods which are fully withdrawn in the 100% to 50% rod density range.	2) The RWM computer on line diagnostic test shall be sucessfully performed.
	3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified.
	4) The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.
	c. When required, the presence of a second licensed operator or other qualified employee to verify the following of the correct rod program shall be verified.
4. Control rods shall not be withdrawn for startup unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup, verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During operation with limiting control rod patterns, as determined by the designated qualified personnel, either:	5. When a limiting control rod pattern exists an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).
a. Both RBM channels shall be operable: or	
b. Control rod withdrawal shall be blocked: or	
c. The operating power level shall be limited so that the MCPR will remain above 1.07 assuming a single error that results in complete withdrawal of any single operable control rod.	

flux. The requirements of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8}% of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

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

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR = 1.07 or LHGR = 18.5kW/ft). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Reactor Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods in other than limiting patterns. Other personnel qualified to perform this function may be designated by the station superintendent.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than 1.07. The limiting power transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram (FSAR Figure III.6.15) with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than 1.07.

On an early BWR, some degradation of control rod scram performance occurred during plant startup and was determined to be caused by particulate material (probably construction debris) plugging an internal control rod drive filter. The design of the present control rod drive (Model CRDB144B) is grossly improved by the relocation of the filter to a location out of the scram drive path; i.e,, it can no longer interfere with scram performance, even if completely blocked.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.A (cont'd.)	4.5.A (cont'd.)
2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystem and the diesel generators are operable.	2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem, the LPCI subsystem and the diesel generators shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.
3. Both LPCI Subsystems shall be operable: (1) prior to reactor startup from a Cold Condition, except as specified in 3.5.F 7, or (2) when there is irradiated fuel in the vessel and when the reactor vessel pressure is greater than atmospheric pressure, except as specified in 3.5.A.4 and 3.5.A.5 below.	3. LPCI Subsystem Testing shall be as follows: Item — Frequency a. Simulated Automatic Actuation Test — Once/Operating Cycle b. Pump Operability — Once/month c. Motor Operated valve operability — Once/month d. Pump Flow Rate — Once/3 months Each RHR pump shall deliver at least 8400 gpm but no more than 8800 gmp against a reactor vessel pressure at atmospheric conditions and a total system flow of at least 16,000 gpm if two RHR pumps are injecting into the same recirculation loop. e. Recirculation Pump discharge valves shall be tested each refueling outage to verify full open to full closed in $20 \leq t \leq 26$ seconds.
4. From and after the date that one of the RHR (LPCI) pumps is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding thirty days provided that during such thirty days the remaining active components of the LPCI Subsystem and all active components of both core spray subsystems and the diesel generators are operable.	4. When it is determined that one of the RHR (LPCI) pumps is inoperable at a time when it is required to be operating the remaining active components of the LPCI Subsystems, the containment cooling subsystem, both core spray system and the diesel generators shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.F (cont'd)	4.5.F (cont'd)
h. A special flange, capable of sealing a leaking control rod housing, is available for immediate use.	
i. The control rod housing is blanked following the removal of the control rod drive.	
j. No work is being performed in the vessel while the housing is open.	
6. During a refueling outage, refueling operation may continue with one core spray system or the LPCI system inoperable for a period of thirty days.	
7. The LPCI System is required to be operable while performing training startups at atmospheric pressure at power levels less than 1% of rated thermal power with the exception that the RHR system may be aligned in the shutdown cooling mode rather than the LPCI mode.	
G. <u>Maintenance of Filled Discharge Pipe</u>	G. <u>Maintenance of Filled Discharge Pipe</u>
Whenever core spray subsystems. LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.	The following surveillance requirements shall be adhered to, to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:
	1. Whenever the Core Spray, LPCI, HPCI or RCIC systems are made operable, the discharge piping shall be vented from the high point of the system and water flow observed initially and on a monthly basis.
	2. The pressure switches which monitor the LPCI, core spray, HPCI and RCIC lines to ensure they are full shall be functionally tested and calibrated every three months.

3.5.A BASES

Core Spray and LPCI Subsystems

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant analysis included in General Electric Topical Report NEDO-10329 and the sensitivity studies given in Supplement 1 thereto and subsection 6.5 of the FSAR and in accordance with the AEC's "Interim Acceptance Criteria for Emergency Core Cooling Systems" published on June 19, 1971, any of the following cooling systems provides sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel clad temperature to less than 2300°F, to assure that core geometry remains intact, and to limit clad metal-water reaction to less than 1%; the two core spray subsystems; or either of the two core spray subsystems and three RHR pumps operating in the LPCI mode with operable LPCI injection valves.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above. During reactor shutdown when the residual heat removal system is realigned from LPCI to the shutdown cooling mode, the LPCI System is considered operable.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Cooper Nuclear Station, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additionally conservative in that no credit is taken for spray coolant entering the reactor before the internal pressure has fallen to 113 psig.

The LPCI subsystem is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel clad temperature. The LPCI subsystem and the core spray subsystem provide adequate cooling for break areas of approximately 0.2 square feet up to and including the double-ended recirculation line break without assistance from the high pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in reference (1). Using the results developed in this reference, the repair period is found to be 1/2 the test interval. This assumes that the

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(1) Jacobs, I.M., "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", General Electric Co. A.P.E.D., April, 1969 (APED 5736).

ment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. Specification 3.5.F.4 provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation system. Thus, the specification precludes the events which could require core cooling. Specification 3.5.F.5 recognizes that, concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.2.h. In this case, if excessive control rod housing leakage occurred, three levels of protection against loss of core cooling would exist. First, a special flange would be used to stop the leak. Second, sufficient inventory of water is maintained to provide, under worst case leak conditions, approximately 60 minutes of core cooling while attempts to secure the leak are made. This inventory includes water in the reactor well, spent fuel pool, and condensate storage tank. If a leak should occur, manually operated valves in the condensate transfer system can be opened to supply either the core spray system or the spent fuel pool. Third, sufficient inventory of water is maintained to permit the water which has drained from the vessel to fill the torus to a level above the core spray and LPCI suction strainers. These systems could then recycle the water to the vessel. Since the system cannot be pressurized during refueling, the potential need for core flooding only exists and the specified combination of the core spray or the LPCI system can provide this. This specification also provides for the highly unlikely case that both diesel generators are found to be inoperable. The reduction of rated power to 25% will provide a very stable operating condition. The allowable repair time of 24 hours will provide an opportunity to repair the diesel and thereby prevent the necessity of taking the plant down through the less stable shutdown condition. If the necessary repairs cannot be made in the allowed 24 hours, the plant will be shutdown in an orderly fashion. This will be accomplished while the two off-site sources of power required by Specification 3.9.A.1 are available. Specification 3.5.F.7 provides for the performance of training startups without realigning the residual heat removal system from the shutdown cooling mode to the LPCI mode. Power levels during training startups are kept below the level of significant heat addition.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI subsystem, HPCI, and RCIC are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. If a water hammer were to occur at the time at which the system were required, the system would still perform its design functions. However, to minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition.

H. Engineered Safeguards Compartments Cooling

The unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicate that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured.

Table 3.6.1

SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation
AS-S-110	Torus Area	890'9"
AS-S-111	Torus Area	891'9"
AS-S-112	Torus Area	891'9"
AS-S-113	Torus Area	893'
BS-S-1 B&R	Torus	870'8"
BS-S-15	Torus Area	893'4"
CS-S-1	S.E. Quad	918'
CS-S-2	S.E. Quad	929'
CS-S-3	S.E. Quad	946'3"
CS-S-10	Rx Bldg, 931'	946'3"
CS-S-11	Rx Bldg, 931'	946'3"
HP-S-4	S.W. Quad	872'7"
HP-S-11	S.W. Quad	869'11"
HP-S-15	S.W. Quad	874'11"
MS-S-1	S.W. Quad	864'
MS-S-2	S.W. Quad	868'5"
MS-S-3	S.W. Quad	880'4"
MS-S-4	S.W. Quad	873'5"
MS-S-7(2)	S.W. Quad	874'11"
MS-S-8	Torus Area	885'2"
MS-S-10	Torus Area	899'11"
MS-S-11	Torus Area	897'
MS-S-12	Torus Area	888'
MS-S-13	S. RHR Hx Rm	904'10"
MS-S-14	S. RHR Hx Rm	923'
MS-S-15	S. RHR Hx Rm	934'
MS-S-16	Torus Area	885'
MS-S-17	N. RHR Hx Rm	904'10"
MS-S-18	N. RHR Hx Rm	905'6"
MS-S-19	N. RHR Hx Rm	923'
MS-S-20	N. RHR Hx Rm	934'
MS-S-23	Torus Area	898'
MS-S-24	Torus Area	898'
MS-S-25	N.E. Quad	877'6"
MS-S-26	N.E. Quad	879'6"

Table 3.6.1

SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS) (Cont'd)

Snubber No.	Location	Elevation
RCC-S-3	Rx Bldg, 931'	945'11"
RCC-S-4	Rx Bldg, 931'	943'6"
RCC-S-20	Rx Bldg, 931'	953'3"
RCC-S-21	Rx Bldg, 931'	953'3"
RCC-S-22	Rx Bldg, 931'	953'3"
RF-S-1	N.E. Quad	898'6"
RF-S-2	Torus Area	896'
RF-S-3	S.W. Quad	870'
RF-S-4	Torus Area	894'6"
RF-S-5	Torus Area	897'10"
RF-S-6	Torus Area	891'
RH-S-20	Rx Bldg, 903'	912'6"
RH-S-21	Rx Bldg, 903'	911'
RH-S-22	Torus Area	895'9"
RH-S-23	Torus Area	892'
RH-S-24	Torus Area	897'
RH-S-25	N. RHR Hx Rm	927'
RH-S-26	N. RHR Hx Rm	929'
RH-S-29	Rx Bldg, 903'	904'6"
RH-S-30(2)	Torus Area	898'6"
RH-S-32	Torus Area	894'7"
RH-S-33D	Torus	892'3"
RH-S-34	Rx Bldg, 903'	919'6"
RH-S-35	S. RHR Hx Rm	912'
RH-S-36	S. RHR Hx Rm	914'3"
RH-S-37	S. RHR Hx Rm	916'4"
RH-S-38	S. RHR Hx Rm	930'
RH-S-39	S. RHR Hx Rm	927'6"
RH-S-40	S. RHR Hx Rm	915'6"
RH-S-41	S.W. Quad	873'
RH-S-42	S.W. Quad	874'
RH-S-43	Torus Area	897'
RH-S-44	S.W. Quad	884'6"
RH-S-45	S.W. Quad	884'
RH-S-48	N.W. Quad	884'6"
RH-S-49	N.W. Quad	885'
RH-S-51	N. RHR Hx Rm	914'3"
RH-S-52	N. RHR Hx Rm	915'
RH-S-54	N.W. Quad	873'1"
RH-S-55	N.W. Quad	874'
RH-S-56	N. RHR Hx Rm	927'6"
RH-S-57	N. RHR Hx Rm	927'6"
RH-S-58	N. RHR Hx Rm	921'11"
RH-S-59	Torus Area	896'
RH-S-65	S.W. Quad	887'2"
RH-S-66	Rx Bldg, 903'	907'4"

Table 3.6.1

SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS) (Cont'd)

Snubber No.	Location	Elevation
RH-S-76(2)	Torus Area	898'
RH-S-77	Torus Area	890'11"
RH-S-78(2)	Torus Area	897'
RH-S-80	N.W. Quad	889'
RH-S-98	N.W. Quad	891'
SWH-WH-23A	Intake Str.	904'3"
SWH-WH-23B	Intake Str.	904'3"
SWH-WH-23C	Intake Str.	904'3"
SWH-WH-23D	Intake Str.	904'3"

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.7 (cont'd)	4.7 (cont'd)
E. Drywell-Suppression Chamber Differential Pressure	E. Drywell-Suppression Chamber Differential Pressure
1. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.47 psid except as specified in a, b, and c below.	1. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.
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 1.47 psid 24 hours prior to placing mode switch in refuel or shutdown.	
c. This differential may be decreased to less than 1.47 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.	

TABLE 3.7.2

TESTABLE PENETRATIONS WITH DOUBLE O-RING SEALS

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 57 TO FACILITY
OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1. Introduction

Nebraska Public Power District (the licensee) has proposed changes to the Technical Specifications of Facility Operating License No. DPR-46 for Cooper Nuclear Station (CNS). The proposed changes permit operation of the CNS facility with the low pressure coolant injection (LPCI) system residual heat removal (RHR) pumps aligned in the shutdown cooling mode rather than the LPCI mode while performing training startups at atmospheric pressure at power levels less than 1% of rated power.

In addition the licensee has proposed certain administrative changes to the Technical Specifications which we have included in this Amendment.

2. LPCI Pump Alignment Background

It is advantageous to the licensee to be able to perform training startups from the cold shutdown condition while the RHR pumps are aligned for shutdown cooling. Technical Specification 3.5.A.3 requires the LPCI system (a mode of RHR) to be operable prior to performing a reactor startup from cold condition. The definition of operable contained in the Technical Specifications requires a system to be capable of performing its intended function in its required manner. For LPCI mode, the RHR pumps are aligned to take a suction on the torus, whereas in shutdown cooling the pumps are aligned to take a suction on the recirculation loops. Although the realignment of RHR to the LPCI mode is easily accomplished at the main control board, operator action is required. Once RHR is aligned to the LPCI mode, a realignment to shutdown cooling requires the header to be flushed. This places a burden on the radioactive waste disposal system.

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It should be noted that during a reactor cooldown the RHR system must be realigned from LPCI to the shutdown cooling mode. This alignment similarly renders the LPCI system "inoperable" in accordance with the strict Technical Specification definition.

Evaluation

The function of the LPCI system when at less than 1% rated power and at atmospheric pressure, is to makeup cooling water to the reactor coolant system in the event of a LOCA.

In the event of a LOCA it would be possible with BWR piping geometry to drain the core to a maximum of 2/3 core height. Further loss of water level could occur, if losses from steaming were not replenished.

Technical Specification 3.5.A.1(1) requires that both core spray subsystems be operable prior to reactor startup from a cold shutdown. The makeup capacity of both core spray loops is at least 4720 gpm. The flow rate of a single core spray loop is sufficient to makeup much more water than the steaming losses from the reactor core at less than 1% power level. Thus, sufficient makeup water would be available given a single failure.

We have modified the licensee's proposed Technical Specification change so that it would require that the LPCI system remain operable except that the RHR pumps may be valved to draw water from the suction of the recirculation loop.

The recirculation loop would not be a source of makeup water in the event of a LOCA, however if the LPCI system were operable and time permitted, the control room operator could realign the RHR pumps to draw from the torus water supply as an additional source of makeup water.

Although no credit is taken in this review for makeup from the control rod drive cooling water systems and the reactor water cleanup systems, these systems are also normally available as a source of makeup water.

We conclude that operation, at less than 1% rated power and at atmospheric pressure with the RHR pumps in the shutdown cooling mode is acceptable.

3. Administrative Changes

Background

By letter dated November 29, 1978, the licensee requested various Technical Specification changes of an administrative nature which we have addressed in this licensing action. In addition we have taken this opportunity to make two corrections to previously issued Safety Evaluations.

Evaluation

a. Table 3.2 F has been revised so that the instruments for the Suppression Chamber air and water temperatures are changed from an alarm unit to a recorder with an alarm function. This modification increases the accuracy and reliability of the monitoring system, and more accurately describes the equipment.

b. The safety limit MCPR is revised from 1.06 to 1.07 in two places for consistency and to correct an earlier ommission. This change is in a conservative direction.

c. Reference to the count rate requirements during refueling is deleted, to eliminate redundancy. These requirements are fully stated in Technical Specification 3.10.

d. Thirteen drywell penetrations are added to Table 3.7.2 of Testable Penetrations with Double O-Ring Seals. This addition increases safety by adding a testing requirement that did not exist before.

e. Table 3.6.1 is revised to update the listing of safety related hydraulic snubbers accessible for testing. The original listing was based on the licensees determinations, and the revised listing includes corrections in a previous error and revisions to reflect plant modifications.

f. The wording which specifies the time at which drywell-suppression chamber differential pressure must be established is revised for greater precision.

g. The Safety Evaluation for License Amendment No. 52 to License No. DPR-46 issued on September 29, 1978, incorrectly stated at the top of page that "no shielded cask movement will be permitted on the refueling deck prior to the completion of the cask drop analysis review". This sentence should be corrected to state "generic review is completed to the point that cask movement restriction is no longer necessary".

h. The Safety Evaluation for License Amendment No. 43 to License No. DPR-46 issued on April 11, 1978 incorrectly stated at the bottom of the first page "this will be configured in a two-out-of-four coincidence logic per bus". This sentence should be corrected to state "this will be configured in a two-out-of-two coincidence logic per bus".

4. 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 Section 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.

5. 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.

Dated: May 23, 1979

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. 57 to Facility Operating License No. DPR-46, issued to Nebraska Public Power District, which revised the Technical Specifications for operation of the Cooper Nuclear Station, located in Nemaha County, Nebraska. The amendment is effective as of the date of its issuance.

The amendment revised the Technical Specifications to permit operation up to 1% of related thermal power while performing training startups with the RHR pumps in the shutdown cooling mode. The amendment also revised the Technical Specifications to include administrative changes.

The applications for the 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 Section 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 November 29, 1978 and January 30, 1979, (2) Amendment No. 57 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 23rd day of May 1979

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors