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Docket No. 50-298

Nebraska Public Power District
ATTN: Mr. J. M. Pilant, Director
Licensing & Quality Assurance
P. O. Box 499
Columbus, Nebraska 68601

Gentlemen:

In response to your requests for license amendment dated November 22, 1976, March 2, 1977, and April 27, 1977 the Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station.

This amendment incorporates provisions into the facility Technical Specifications which establish limiting conditions for operation and surveillance requirements for drywell to suppression chamber differential pressure control and suppression pool water level.

These requirements provide assurance that facility operation will be in accordance with the assumptions utilized in your facility's plant-unique analysis which was performed in conjunction with the Mark I Containment Short Term Program evaluation.

The enclosed license amendment reflects those changes to your original request for license amendment which have been agreed to in dicussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

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

Sincerely,

/S/

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

Enclosures:
see next page

changes noted!

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SURNAME	SSheppard	VRooney:acr	CGrimes	TIppolito		
DATE	6/29/78	6/30/78	6/7/78	7/7/78		

NRC FORM 318 (9-76) NRCM 0240

☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Enclosures:
1. Amendment No. 50 to License No. DPR-46
2. Safety Evaluation
3. Notice

cc: w/enclosures:

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U. S. Environmental Protection Agency
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1735 Baltimore Avenue
Kansas City, Missouri 64108

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. 50
License No. DPR-46

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

   A. The applications for amendment by Nebraska Public Power District dated November 22, 1976, March 2, 1977, and April 27, 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 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. 50, 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: July 10, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Technical Specifications contained in Appendix A of the above indicated license with the attached pages. Changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
65	65
66	66
80	80
184	184

Add page 167a.

COOPER NUCLEAR STATION
TABLE 3.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION

Instrument	Instrument I.D. No.	Range	Minimum Number of Operable Instrument Channels	Action Required When Minimum Condition Not Satisfied (1)
Reactor Water Level	NBI-LI-85A NBI-LI-85B	-150" to +60" -150" to +60"	2	A,B,C
Reactor Pressure	RFC-PI-90A RFC-PI-90B	0 - 1200 psig 0 - 1200 psig	2	A,B,C
Drywell Pressure	PC-PI-512A PC-PR-512B	0 - 80 psia 0 - 80 psia	2	A,B,C
Drywell Temperature	PC-TR-503 PC-TI-505	50 - 170°F 50 - 350°F	2	A,B,C
Suppression Chamber Air Temperature	PC-TR-21A PC-TA-20 A,C	0 - 300°F 0 - 400°F	2	A,B,C
Suppression Chamber Water Temperature	PC-TR-21B PC-TA-20 B,D	0 - 300°F 0 - 400°F	2	A,B,C
Suppression Chamber Water Level	PC-LI-10 PC-LR-11	(-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	

NOTES FOR TABLE 3.2.F

1. The following actions will be taken if the minimum number of operable instrument channels as required are not available.

   A. From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

   B. From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

   C. If the requirements of A and B above cannot be met, an orderly shutdown shall be initiated within 24 hours.

   D. These surveillance instruments are considered to be redundant to each other.

   E. In the event that both channels are inoperable and indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown in six (6) hours and in a Cold Shutdown condition in the following eighteen (18) hours.

2. These instrument channels are utilized as input for the performance of a manual calculation to provide the second Torus to Drywell differential pressure indication.

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

Instrument	Instrument I.D. No.	Calibration Frequency	Instrument Check
Reactor Water Level	NBI-LI-85A NBI-LI-85B	Once/6 Months Once/6 Months	Each Shift Each Shift
Reactor Pressure	RFC-PI-90A RFC-PI-90B	Once/6 Months Once/6 Months	Each Shift Each Shift
Drywell Pressure	PC-PI-512A PC-PR-512B	Once/6 Months Once/6 Months	Each Shift Each Shift
Drywell Temperature	PC-TR-503 PC-TI-505	Once/6 Months Once/6 Months	Each Shift Each Shift
Suppression Chamber Air Temperature	PC-TR-21A PC-TA-20 A,C	Once/6 Months Once/6 Months	Each Shift Each Shift
Suppression Chamber Water Temperature	PC-TR-21B PC-TA-20 B,D	Once/6 Months Once/6 Months	Each Shift Each Shift
Suppression Chamber Water Level	PC-LI-10 PC-LR-11 PC-LI-12 PC-LI-13	Once/6 Months Once/6 Months Once/6 Months Once/6 Months	Each Shift Each Shift Each Shift Each Shift
Suppression Chamber Pressure	PC-PR-20	Once/6 Months	Each Shift
Control Rod Position Neutron Monitoring (APRM)	N.A. N.A.	N.A. Once/Week	Each Shift Each Shift
Torus to Drywell Differential Pressure	PC-dPR-20	Once/6 Months	Each Shift
Suppression Chamber/ Drywell Pressure (ΔP)	PC-PR-20/513 (2)	Once/6 Months	Each Shift

LIMITING CONDITIONS FOR OPERATION

3.7.E CONTAINMENT SYSTEMS

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.

   a. This differential shall be established within 26 hours of achieving operating temperature and pressure.

   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 Shutdown in six (6) hours and in a Cold Shutdown condition within the following 18 hours.

SURVEILLANCE REQUIREMENTS

4.7.E CONTAINMENT SYSTEMS

Drywell-Suppression Chamber Differential Pressure

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

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

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

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

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

4. Open vent or drain valve.

   a. Observe flow cessation and any leakage rate.

   b. Reset valve after test completion.

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

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

3.7.E Bases

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed as described in the licensee's letter of October 4, 1976, which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.47 psid and a suppression chamber water level corresponding to a downcomer submergence range of four to five feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 50 TO LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

Introduction

In conjunction with the Short Term Program (STP) evaluation of Boiling Water Reactor facilities with the Mark I containment system, the Nebraska Public Power District (licensee) submitted a Plant Unique Analysis (PUA) for the Cooper Nuclear Station. This analysis was performed to confirm the structural and functional capability of the containment suppression chamber and attached piping, to withstand newly-identified suppression pool hydrodynamic loading conditions which had not been explicitly considered in the original design analysis for the plant. As part of the STP evaluation, specific loading conditions were developed for each Mark I facility, to account for the change in the magnitude of the loads due to plant-specific variations from the reference plant design for which the basic loading conditions were developed.

The results of the NRC staff's review of the hydrodynamic load definition techniques and the Mark I containment plant unique analyses are described in the "Mark I Containment Short Term Program Safety Evaluation Report", NUREG-0408, December 1977. As discussed in this report, the NRC staff has concluded that each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis loss-of-coolant accident (LOCA), and, therefore, that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years, while a methodical, comprehensive Long Term Program is conducted.

As discussed in Section III.C of NUREG-0408, of all of the plant parameters that were considered in the development of the hydrodynamic loads for the STP, only two parameters are expected to vary during normal plant operation; these are (1) the drywell-wetwell differential pressure; and (2) the suppression chamber (torus) water level. Subsequent to the submittal of the PUA, the licensee was requested to submit proposed Technical Specifications which assure that the allowable range of these two parameters during facility operation would be in accordance with the values utilized in the PUA.

The licensee has been operating this facility with differential pressure control to enhance the safety margins of the containment structure since early 1976. This evaluation provides a more detailed bases for establishing the allowable range of drywell-wetwell differential pressure and torus water level, in order to quantify containment safety margins. This amendment incorporates these parameters into the Technical Specifications with the associated limiting conditions for operation and surveillance requirements.

By letters dated November 26, 1976, March 2, 1977, and April 27, 1977, the licensee proposed changes to the facility Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for differential pressure control and torus water level. Our evaluation of these proposed changes follows:

Evaluation

The licensee has proposed certain Technical Specification requirements for the purpose of assuring that the normal plant operating conditions are within the envelope of conditions considered in their PUA. These Technical Specification changes establish (1) limiting condition for operation (LCOs) for drywell to torus differential pressure and torus water level, and (2) associated surveillance requirements. All other initial conditions utilized in the PUA are either presently included in the Technical Specifications or are configurational conditions which have been confirmed by the licensee and will not change during normal operation.

Differential pressure between the drywell and the suppression chamber will result in leakage of the drywell atmosphere to the lower pressure regions of the reactor building and to the torus airspace. This leakage from the drywell will cause a slow decay in the differential pressure. Therefore, surveillance requirements for the differential pressure have been included in the Technical Specifications. Surveillance frequency of once per operating shift for the differential pressure was selected on the basis of previous operating experience.

The torus water level is not expected to vary significantly during normal operation, unless certain systems connected to the suppression pool are activated. The torus water level would normally be monitored whenever such systems are in use. Therefore, we find that inclusion of periodic torus water level surveillance requirements in the Technical Specifications is not required.

We have reviewed the differential pressure and torus water level monitoring instrumentation systems proposed by the licensee with regard to the number of available channels and the instrumentation accuracy. This type of instrumentation is typically calibrated at six-month intervals. To assure proper operation during such intervals, two monitoring channels for both differential pressure and torus water level have been provided, such that a comparison of the readings will indicate when one of the channels is inoperative or drifting. The errors in the instrumentation are sufficiently small relative to the magnitude of the measurement (i.e., maximum differential pressure measurement error of 0.1 psid in a measurement of 1.0 to 2.0 psid and a maximum torus water level measurement error of 10% of the difference between the maximum and minimum torus water level) that they may be neglected, based on the expected load variation with differential pressure and torus water level.

There are certain periods during normal plant operations when the differential pressure control cannot be maintained. Therefore, provisions have been included in the Technical Specifications to relax the differential pressure/control requirements during specified periods. The justification for relaxing the differential pressure control during these specific periods and the basis for selecting the duration of the periods are discussed in detail below.

Startup and Shutdown

During plant startup and shutdown, the drywell atmosphere undergoes significant barometric changes due to the variation in heat loads from the primary and auxiliary systems. In addition, it is during these periods that the drywell is being either inerted with nitrogen gas or deinerted. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have limited the relaxation of the differential pressure control requirements for the startup and shutdown periods to 26 hours following startup and 24 hours prior to a shutdown. These

time periods were selected on a basis similar to that for the inerting requirements, already existing in the Technical Specifications. The postulated design basis accident for the containment assumes that the primary system is at operating pressure and temperature. During the startup and shutdown transients, the primary system is at operating pressure and temperature for only a part of the transient, during which the differential pressure is being established. These time periods have been shown by previous operating experience to be adequate with respect to the startup and shutdown transients, and at the same time sufficiently small in comparison to the duration of the average power run. Since the principal accident event to which differential pressure control is important to assure containment integrity (i.e., with a factor of safety of two) is a large break LOCA, we have considered whether there is a significantly greater probability of a large break LOCA during the startup and shutdown transients. We have concluded that there is not. Further, the operation of the plant systems is monitored more closely than normal during these periods and a finite magnitude of differential pressure will be available during the majority of these periods to mitigate the potential consequences of an accident.

Testing and Maintenance

During normal operation, there are a number of tests which are required to be conducted to demonstrate the continued functional performance of engineered safety features. The testing of certain systems will require, or result in, a reduction in the drywell-torus differential pressure. The operability testing of the drywell-torus vacuum breakers requires the removal of the differential pressure to permit the vacuum breakers to open. For the testing of high-energy systems (e.g., high pressure coolant injection pumps) during normal operation, the discharge flow is routed to the suppression pool. This energy deposition will raise the temperature of the suppression pool, resulting in an increase in torus pressure and a reduction in the differential pressure.

Functional performance testing of engineered safety features is necessary to assure proper maintenance of these systems throughout the life of the plant. Some of these tests (i.e., pump operability and drywell-wetwell vacuum breakers) may require or result in a reduction in the differential pressure. We estimate that not more than four tests will be required each month which will result in a

reduction in differential pressure. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have permitted a relaxation of differential pressure control in order to conduct these tests, limited to a period of up to four hours. Again, we have carefully considered whether the probability of a large LOCA is significantly greater during these testing periods than that during normal operation. We conclude that it is not. Moreover, only the test of the drywell-wetwell vacuum breakers requires complete removal of the differential pressure.

Provisions have also been included in the Technical Specifications for performing maintenance activities on the differential pressure control system and for resolving operational difficulties which may result in an inadvertent reduction in the differential pressure for a short period of time. In certain circumstances, corrective action can be taken without having to attain a cold shutdown condition. To avoid repeated and unnecessary partial cooldown cycles, a restoration period has been incorporated into the action requirements of the LCO for differential pressure control; i.e., in the event that the differential pressure cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. The six hour restoration period was selected on the basis that it represents an adequate minimum period of time during which any short-term malfunctions could be corrected, coupled with the minimum period of time required to conduct a controlled shutdown. The allowable time to conduct a controlled shutdown has been minimized, because the containment transient response is more a function of the primary system pressure than the reactor power level. On this basis, we find the proposed restoration period and action requirement acceptable.

We conclude that the limits imposed on the periods of time during which operation is permitted without the differential pressure control fully effective provides adequate assurance of overall containment in integrity, and the periods of time differential pressure control is completely removed are acceptably small.

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.

Conclusions

The proposed Technical Specifications will provide the necessary assurance that the plant's operating conditions remain within the envelope of the conditions assumed in the Plant Unique Analysis (PUA) performed in conjunction with the Mark I Containment Short Term Program. The PUA supplements the facility's Final Safety Analysis Report (FSAR) in that it demonstrates the plant's capability to withstand the suppression pool hydrodynamic loads which were not explicitly considered in the FSAR. We therefore conclude that the proposed changes to the Technical Specifications are acceptable.

We further conclude, 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 resonable 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: July 10, 1978

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. 50 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 incorporate requirements for establishing and maintaining the drywell to suppression chamber differential pressure and suppression chamber water level, to maintain the margins of safety established in the NRC staff's "Mark I Containment Short Term Program Safety Evaluation", NUREG-0408. Operation in accordance with the conditions specified in NUREG-0408 has been previously authorized in 43 FR 13109, March 29, 1978.

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) applications for amendment dated November 22, 1976, March 2, 1977 and April 27, 1977, (2) Amendment No. 50 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 10 day of July 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

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