Distribution
Docket
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ACRS (16)
OPA (CMiles)
DRoss
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Docket No. 50-298

AUGUST 25 1978

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 51 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 request of August 7, 1978 as supplemented August 16, 1978.

The amendment modifies the Technical Specifications to permit operation of the facility during fuel cycle 4 with a reduced minimum critical power ratio for 7x7 and 8x8 fuel based on revised analysis.

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

Sincerely,

Richard J. Clark for

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

Enclosures:
1. Amendment No. 51 to DPR-46
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

(check to see if there is a typo in Table 2 of the SE...)

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OFFICE	ORB #3	ORB #3	OELD	ORB #3		
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DATE	8/25/78	8/25/78	8/25/78	8/25/78		

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Director
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Lincoln, Nebraska 68509

Mr. William Siebert, Commissioner
Nemaha County Board of Commissioners
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Chief, Energy Systems Analyses
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U. S. Environmental Protection Agency
Region VII
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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. 51
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated August 7, 1978 as supplemented August 16, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   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. 51 , 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

Richard J. Clark for

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 25, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 51

TO THE TECHNICAL SPECIFICATIONS

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
17	17
19	19
212	212
214c	214c
214d	214d

2.1 Bases:

The abnormal operational transients applicable to operation of the CNS Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power condition of 2381 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure III-7-1 of the FSAR. In addition, 2381 MWt is the licensed maximum power level of CNS, and this represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specifications. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greater significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 25% insertion. By the time the rods are 60% inserted, approximately four dollars of negative reactivity have been inserted which strongly turns the transient, and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

For analyses of the Thermal consequences of the transients, a MCPR of 1.24 for 7x7 fuel and 8x8 fuel and 1.23 for 8x8R fuel is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Steady-state operation without forced recirculation will not be permitted, except during startup testing. The analysis to support operation at various

2.1 Bases: (Cont'd)

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.1.a.1.a, when the MFLPD is greater than the fraction of rated power (FRP). This adjustment may be accomplished by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

Analyses of the limiting transients show that no scram adjustment is required to assure MCPR > 1.07 when the transient is initiated from MCPR > 1.24 for 7x7 bundles, 1.24 for 8x8 bundles, and 1.23 for 8x8R bundles.

b. APRM Flux Scram Trip Setting (Refuel or Start & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the APRM scram setting of 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of rated. The margin is adequate to accomodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedure backed up by the rod worth minimizer, and the rod sequences control system. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The 15 percent APRM scram remains active until the mode switch is placed in the RUN position. This change can occur when reactor pressure is greater than 850 psig.

LT = Total core length -12 feet

L = Axial position above bottom of core

G = 18.5 kW/ft for 7x7 fuel bundles
= 13.4 kW/ft for 8x8 fuel bundles

N = 0.038 for 7x7 fuel bundles
= 0.022 for 8x8 fuel bundles

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded action shall than be initiated to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

C. Minimum Critical Power Ratio (MCPR)

During steady state power operation MCPR shall be $\geq$ 1.24 for 7x7 bundles, $\geq$ 1.24 for 8x8 bundles, and $\geq$ 1.23 for 8x8R bundles, at rated power and flow. If, at any time during steady state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met.

For core flows other than rated the MCPR shall be the operating limit at rated flow times K_f, where K_f is as shown in Figure 3.11-2.

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

3.11 Bases (Cont'd)

The limiting transient which determines the required steady state MCPR limit is the rotated bundle loading error for 8x8 and 8x8R bundles and the rod withdrawal error for 7x7 bundles. The transients yield the largest ΔCPRs. When added to the safety limit MCPR of 1.07 the required minimum operating limit MCPR of specification 3.11C are obtained.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4 of NEDO-24011[2] and on core parameters shown in Table A-2 of Reference 5.

The evaluation of a given transient begins with the system initial parameters shown in Table A-2 of Reference 5 that are input to a GE core dynamic behavior transient computer program described in NEDO-10802[3]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566[4]. The principal result of this evaluation is the reduction in MCPR caused by the transient.

D. MCPR Limits for Core Flows Other than Rated

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow, the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of 1.24 for 7x7 and 8x8 and 1.23 for 8x8R will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

3.11 Bases: (Cont'd)

The K_f factor curves shown in Figure 3.11-2 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11-2 are conservative for Cooper operation because the operating limit MCPR of 1.24 for 7x7 and 8x8 and 1.23 for 8x8R are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

References

1. "Cooper Nuclear Station Channel Inspection and Safety Analyses with Bypass Holes Plugged," NEDO-21072, October 1975.

2. Licensing Topical Report, General Electric Boiling Water Reactor, Generic Reload Fuel Application, (NEDE-24011-P) May 1977, Supplement 1 (NEDE-24011-P-1), January 1978.

3. R. B. Linford, Analytical Methods of Plant Transient Evaluations for the GE BWR, February 1973 (NEDO-10802).

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), August 1974.

5. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station Reload 3," January 1978 (NEDO-24093).

6. April 18, 1978 letter from J. M. Pilant (NPPD) to G. E. Lear (NRC).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 51 TO FACILITY
OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 Introduction

Nebraska Public Power District (the licensee) has proposed changes to the Technical Specifications of Facility Operating License DPR-46 for Cooper Nuclear Station (CNS). The proposed changes permit operation of the CNS with a reduced minimum critical power ratio (MCPR) based on revised analysis.

Cooper Nuclear Station (CNS) is now in its fourth cycle of operation. The core loading is a mixture of 7x7, 8x8, and 8x8R fuel types, including twenty-four 8x8 bundles and seventy-six 8x8R bundles which were initially loaded during the last refueling.

Operating limits on MCPR during Cycle 4 have been based on postulated errors in monitoring critical power ratio (CPR) possible because of undetected fuel misloadings. The staff evaluated the adequacy of these limits during the review of the safety analysis for Cycle 4 operation of CNS, and has found them acceptable (Reference 1). Table 1 summarizes the current MCPR limit and the nature of the most severe fuel misloading for each fuel type. As Table 1 indicates, the current MCPR limits for 7x7 and 8x8 fuel are based on standard GE methods for calculating the consequences of fuel misloadings. The 8x8R limit is based on a new GE method for analysis of rotated bundles which takes into account the fact that a rotated bundle is also tilted away from the vertical.

The Nebraska Public Power District has now proposed (Reference 2) that the CNS Cycle 4 7x7 and 8x8 MCPR limits be reduced to the values indicated in Table 2. They do not propose to change the 8x8R limit. The new limits are based on new analyses of postulated fuel misloadings of fresh 8x8 bundles into sites intended for exposed 7x7 and 8x8 bundles. The new analyses involved application of GE's new method based on a statistically corrected Haling procedure (Reference 3). Because the new GE method predicts less severe CPR reductions than did the standard method, the control rod withdrawal transient becomes the limiting event for 7x7 fuel if the proposed MCPR values are adopted.

Our evaluation of the adequacy of the proposed MCPR limits for the remainder of Cycle 4 are presented in the following section.

2.0 Evaluation

Because the MCPR limit for 8x8R fuel will not be changed there is no need to consider this limit further.

The reduction in the MCPR limits for 7x7 and 8x8 fuel is based on the application of the statistically corrected Haling procedure (Reference 3) to the postulated misloadings. This calculational procedure has been reviewed by the staff and found acceptable for such calculations (Reference 4).

The new analysis predicts less severe errors in monitoring CPR due to misloadings than did the original GE method for calculating fuel loading errors. This is the reason that the reduction in the operating limit is possible. Reduction of the operating limit MCPR does not affect the safety limit MCPR which remains at 1.07.

As indicated in Table 2 the misloading of a fresh 8x8 bundle into a site intended for an exposed 8x8 bundle remains the limiting event for 8x8 fuel. However, for 7x7 fuel the transient reduction in MCPR due to a control rod withdrawal is more severe than the MCPR error due to a misloading. Therefore, the rod withdrawal error is actually the limiting event for 7x7 fuel. As indicated in Reference 1 the analysis of the rod withdrawal event performed in conjunction with the most recent reload is acceptable for determination of MCPR limits.

Because fuel loading errors may result in linear heat generation rates in excess of the design or operating limit we have also considered the LHGR values resulting from misloadings. In References 5 and 6 CNS has reported that GE calculations of the worst misloadings indicate that the LHGR will not exceed the safety limit LHGR for 1% cladding strain. We agree with this conclusion.

Based on our review of the reduced operating MCPR limits which have been proposed for the continuation of CNS Cycle 4, we conclude that these limits are adequate to assure that the safety limit MCPR will not be violated due to anticipated transients or postulated fuel misloadings. A sufficient range of misloadings has been evaluated to assure that the most severe misloadings cannot result in transition boiling during steady state operation. We also conclude that linear heat generation rates will remain below the LHGR safety limit even in the event of anticipated transients or fuel misloadings.

No change in safety limits, increase in probability of an accident or transient, or increase in consequences of an accident or transient will result from the proposed change.

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.

Dated: August 25, 1978

TABLE 1

CURRENT OPERATING LIMIT MCPR VALUES FOR CNS - CYCLE 4

Fuel Type	O. L. MCPR	Limiting Event	Method of Analysis
7x7	1.31	Fresh 8x8 Misloaded Into a 7x7 Position	Original GE Method
8x8	1.31	Fresh 8x8 Misloaded Into an Exposed 8x8 Position	Original GE Method
8x8R	1.23	Misoriented Fresh 8x8R (By 180°)	New GE "Tilted" Bundle Method

TABLE 2

PROPOSED OPERATING LIMIT MCPR VALUES FOR CNS FOR THE REMAINDER OF CYCLE 4

Fuel Type	O. L. MCPR	Limiting Event	Method of Analysis
7x7	1.24	Control Rod Withdrawal	Standard Approved GE Methods
8x8	1.24	Fresh 8x8 Misloaded Into an Exposed 8x8 Position	New GE Method Based on Statistically Corrected Haling Distribution
8x8R	1.23	Misoriented Fresh 8x8R (By 180°)	New GE "Tilted" Bundle Method

5.0 References

1. Amendment No. 46 to Facility Operating License DPR-46, dated May 2, 1978.

2. Letter from J. Pilant of Nebraska Public Power District to T. A. Ippolito of NRC, dated August 7, 1978.

3. Letter from R. E. Engel of GE to D. G. Eisenhut of NRC, dated June 1, 1977.

4. Letter from D. G. Eisenhut of NRC to R. E. Engel of GE, dated May 8, 1978.

5. Letter from J. Pilant of Nebraska Public Power District to George Lear of NRC, dated April 26, 1978.

6. Letter from J. Pilant of Nebraska Public Power District to T. A. Ippolito of NRC, dated August 16, 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. 51 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 revised minimum critical power ratios for 7x7 and 8x8 fuel based on revised analyses.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment 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 application for amendment dated August 7, 1978 and supplemented August 16, 1978, (2) Amendment No. 51 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 25th day of August 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Clark

Richard J. Clark, Acting Chief
Operating Reactors Branch #3
Division of Operating Reactors