Docket No. 50-298

Mr. Guy R. Horn
Nuclear Power Group Manager
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68602-0499

Dear Mr. Horn:

SUBJECT: COOPER NUCLEAR STATION - AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. 77112)

The Commission has issued the enclosed Amendment No. 142 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 application dated July 2, 1990, as supplemented by letters dated March 8, and April 19, 1991.

The amendment changes the Technical Specifications to remove cycle-specific reactor physics parameters and incorporate them into a new document called the Core Operating Limits Report. The amendment was requested in response to Generic Letter 88-16.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by

Paul W. O'Connor, Project Manager Project Directorate IV-1 Division of Reactor Projects III, IV, and V Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 142 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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

May 22, 1991

Docket No. 50-298

Mr. Guy R. Horn
Nuclear Power Group Manager
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68602-0499

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Paul W. O'Connor

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1. Amendment No.142 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Guy R. Horn
Nuclear Power Group Manager

Cooper Nuclear Station

cc:

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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. 142
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 July, 2, 1990, as supplemented by letters dated March 8, and April 19, 1991, 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 license 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 Appendix A, as revised through Amendment No. 142 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director Project Directorate IV-1 Division of Reactor Projects III, IV, and V Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 22, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 142

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 PAGES	INSERT PAGES
iv	iv
1	1
-	5c
7	7
8	8
18	18
62	62
102	102
210	210
211	211
211a	-
211b	-
212	212
212b	-
212c	-
212d	-
213	213
214	214
214a	214a
214b	214b
214c	-
232	232
234	234

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1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Thermal Parameters

1. Critical Power Ratio (CPR) - The critical power ratio is the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest as calculated by application of an NRC approved critical power correlation.

2. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

3. Minimum Critical Power Ratio (MCPR) - The minimum critical power ratio corresponding to the most limiting fuel assembly in the core.

4. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type. Design LHGR's for each type of fuel are specified in the Core Operating Limits Report.

5. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

B. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

C. Cold Condition - Reactor coolant temperature equal to or less than 212°F.

D. Design Power - Design power means a steady-state power level of 2486 thermal megawatts. This is 104.4% of Rated Power (105% of rated steam flow).

E. Engineered Safeguard - An engineered safeguard is a safety system the actions of which are essential to a safety action required to maintain the consequences of postulated accidents within acceptable limits.

E.A Dose Equivalent I-131 - The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose if inhaled by an adult as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose equivalent I-131 concentration is calculated by: equiv. I-131 = (I-131) + 0.0096 (I-132) + 0.18 (I-133) + 0.0025 (I-134) + 0.037 (I-135).

E.B Exhaust Ventilation Treatment System - An EXHAUST VENTILATION TREATMENT SYSTEM (EVTS) is a system intended to remove radioiodine or radioactive material in particulate form from gaseous effluent by passing exhaust ventilation air through charcoal absorbers and/or HEPA filters before exhausting the air to the environment. An EVTS is not intended to affect noble gas in gaseous effluent. Engineered Safety Feature (ESF) gaseous treatment systems are not considered to be EVTS. The Standby Gas Treatment System is an ESF and not an EVTS. EVTS are specifically identified in ODAM Figure 3-1.

AA. Core Operating Limits Report

The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.5.1.G. Plant operation within these core operating limits is addressed in individual specifications.

SAFETY LIMITS

1.1 (Cont'd)

D. Cold Shutdown

Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 18 in. above the top of the normal active fuel zone (top of active fuel is defined in Figure 2.1.1).

LIMITING SAFETY SYSTEM SETTINGS

2.1.A.1 (Cont'd)

ΔW = 0 for two recirculation loop operation.

a. In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S \leq (0.66\ W + 54\% - 0.66\ \Delta W)\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD = maximum fraction of limiting power density where the limiting power density for each type of fuel bundle is specified in the Core Operating Limits Report.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

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

When the reactor mode switch is in the REFUEL or STARTUP position, the APRM scram shall be set at less than or equal to 15% of rated power.

c. IRM

The IRM flux scram setting shall be ≤120/125 of scale.

2.1.A.1 (Cont'd)

d. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be:

$S_{RB} \leq 0.66\ W + 42\% - .66\ \Delta W$

where:

S_{RB} = Rod block setting in percent of rated thermal power (2381 MWt)

W and ΔW are defined in Specification 2.1.A.1.a.

In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$$S_{RB} \leq (0.66\ W + 42\% - 0.66\ \Delta W)\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power (2381 MWt)

MFLPD - maximum fraction of limiting power density where the limiting power density for each type of fuel bundle is specified in the Core Operating Limits Report.

The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than the design value of 1.0, in which case the actual operating value will be used.

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

$\geq$ +12.5 in. on vessel level instruments.

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 remains above the safety limit when the transient is initiated from the operating MCPR limit specified in the Core Operating Limits Report.

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 Specification 2.1.A.6.

NOTES FOR TABLE 3.2.C

1. For the startup and run positions of the Reactor Mode Selector Switch, the Control Rod Withdrawal Block Instrumentation trip system shall be operable for each function. The SRM and IRM blocks need not be operable in "Run" mode, and the APRM (flow biased) and RBM rod blocks need not be operable in "Startup" mode. The Control Rod Withdrawal Block Instrumentation trip system is a one out of "n" trip system, and as such requires that only one instrument channel specified in the function column must exceed the Trip Level Setting to cause a rod block. By utilizing the RPS bypass logic (see note 5 below and note 1 of Table 3.1.1) for the Control Rod Withdrawal Block Instrument tion, a sufficient number of instrument channels will always be operable to provide redundant rod withdrawal block protection.

2. W is the two-loop recirculation flow rate in percent of rated. Trip level setting is in percent of rated power (2381 MWt). N is the RBM setpoint selected (in percent) and is calculated in accordance with the methodology of the latest NRC approved version of NEDE-24011-P-A. The Core Operating Limits Report specifies the applicable value for N.

3. IRM downscale is bypassed when it is on its lowest range.

4. This function is bypassed when the count is ≥ 100 cps and IRM above range 2.

5. By design one instrument channel; i.e., one APRM or IRM per RPS trip system may be bypassed. For the APRM's and IRM's, the minimum number of channels specified is that minimum number required in each RPS channel and does not refer to a minimum number required by the control rod block instrumentation trip function. By design only one of two RBM's or one of four SRM's may be bypassed. For the SRM's, the minimum number of channels specified is the minimum number required in each of the two circuit loops of the Control Rod Block Instrumentation Trip System. For the RBM's, the minimum number of channels specified is the minimum number required by the Control Rod Block Instrumentation Trip System as a whole (except when a limiting control rod pattern exists and the requirements of Specification 3.3.B.5 apply).

6. IRM channels A,E,C,G all in range 8 or higher bypasses SRM channels A&C functions. IRM channels B,F,D,H all in range 8 or higher bypasses SRM channels B&D functions.

7. This function is bypassed when IRM is above range 2.

8. This function is bypassed when the mode switch is placed in Run.

9. This function is only active when the mode switch is in Run.

10. The inoperative trips are produced by the following functions:

    a. SRM and IRM

        (1) Mode switch not in operate
        (2) Power supply voltage low
        (3) Circuit boards not in circuit
        (4) Loss of negative supply voltage

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 conditions exists.

   A limiting control rod pattern is a pattern which results in the core being on a thermal hydraulic limit (i.e., MCPR equals the operating limit as specified in the Core Operating Limits Report, and LHGR = as defined in 1.0.A.4). 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 Division Manager of Nuclear Operations.

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 the safety limit. The limiting power transient is defined in Reference 2. Analysis of this transient shows that the negative reactivity rates resulting from the scram provide the required protection, and MCPR remains greater than the safety limit.

The surveillance requirement for scram testing of all the control rods after each refueling outage and 10% of the control rods at 16-week intervals is adequate for determining the operability of the control rod system yet is not so frequent as to cause excessive wear on the control rod system components.

The numerical values assigned to the predicted scram performance are based on the analysis of data from other BWR's with control rod drives the same as those on Cooper Nuclear Station.

The occurrence of scram times within the limits, but significantly longer than the average, should be viewed as an indication of a systematic problem with control rod drives.

In the analytical treatment of the transients which are assumed to scram on high neutron flux, 290 milliseconds are allowed between a neutron sensor reaching the scram point and start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 milliseconds estimated from scram test results. Approximately the first 90 milliseconds of each of these time intervals result from the sensor and circuit delays; at this point, the pilot scram solenoid deenergized. Approximately 120 milliseconds later,

LIMITING CONDITIONS FOR OPERATION

3.11 FUEL RODS

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value specified in the Core Operating Limits Report for two recirculation loop operation. For single-loop operation these values are reduced for each fuel type as specified in the Core Operating Limits Report. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR 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.

B. Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as specified in the Core Operating Limits Report.

SURVEILLANCE REQUIREMENTS

4.11 FUEL RODS

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at ≥25% rated thermal power.

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

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LIMITING CONDITIONS FOR OPERATION

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 then 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 the MCPR for each type of fuel at rated power and flow shall not be lower than the limiting value specified in the Core Operating Limits Reports for two recirculation loop operation. 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 specified in the Core Operating Limits Report.

For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at) 25% rated thermel 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.

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3.11 BASES

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50.46.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50.46 limit. The limiting value for APLHGR for each fuel type is specified in the Core Operating Limits Report.

The APLHGR values are reduced for single loop operation per Reference 10.

B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTPF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern. Pellet densification power spiking in GE fuel has been accounted for in the safety analysis presented in References 1 and 2; thus no adjustment to the LHGR limit for densification effects is required.

3.11 Bases: (Cont'd)

C. Minimum Critical Power Ratio (MCPR)

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.11C are derived from the established fuel cladding integrity Safety Limit and an analysis of abnormal operational transients (Reference 2). For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the more limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The models used in the transient analyses are discussed in Reference 1.

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 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.

The K_f factors as provided in the Core Operating Limits Report 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 as described in Reference 1.

The K_f factors are conservative for Cooper operation because the operating limit MCPR's are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

References for Bases 3.11

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A. (The approved revision at the time the reload analyses are performed.) The approved revision number shall be identified in the Core Operating Limits Report.

2. "Supplemental Reload Licensing Submittal for Cooper Nuclear Station," (applicable reload document).

3-8. Deleted

9. Letter (with attachment), R. H. Buckholz (GE) to P. S. Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

10. "Cooper Nuclear Station Single-Loop Operation," NEDO 24258.

4.11 Bases:

A&B. Average and Local LHGR

The LHGR shall be checked daily to determine if fuel burnup, or control rod movement has caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are moved daily, a daily check of power distribution is adequate.

C. Minimum Critical Power Ratio (MCPR) - (Surveillance Requirement)

At core thermal power levels less than or equal to 25%, the reactor will be operating at less than or equal to minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation was made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin was thus demonstrated such that subsequent MCPR evaluation below this power level was shown to be unnecessary. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

6.5.1.E (Cont'd)

c. Summarized and tabulated results in the format of Table 6.5-1 of analyses of samples required by the radiological environmental monitoring program, and taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

d. A summary description of the radiological environmental monitoring program including any changes; a map of all sampling locations keyed to a table giving distances and directions from the reactor; and the results of participation in the Inter-laboratory Comparison Program, required by Specification 3.21.G.

F. Semiannual Radioactive Material Release Report

1. A report of radioactive materials released from the Station during the preceding six months shall be submitted to the NRC within 60 days after January 1 and July 1 of each year*.

2. A Semiannual Radioactive Material Release Report shall include the following:

a. A summary by calendar quarter of the quantities of radioactive liquid and gaseous effluents released from the Station. The data should be reported in the format recommended in Regulatory Guide 1.21, Appendix B, Tables 1 and 2.

b. A summary of radioactive solid waste shipped from the Station, including information named in Specification 4.21.E.3.

c. A summary of meteorological data collected during the year shall be included in the Semiannual Report submitted within 60 days after January 1 of each year.

d. A list and brief description of each unplanned release of gaseous or liquid radioactive effluent that causes a limit in Specification 3.21.B.1.a, 3.21.B.2.a, 3.21.C.1.a, 3.21.C.2.a, or 3.21.c.2.a to be exceeded.

e. Calculated offsite dose to humans resulting from the release of effluents and their subsequent dispersion in the atmosphere reported in accordance with Regulatory Guide 1.21.

6.5.1.G <u>Core Operating Limits Report</u>

Core operating limits shall be established and documented in the Core Operating Limits Report prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. The Average Planar Linear Heat Generation Rates (APLHGR) for Specification 3.11.A.

*It should be noted that this data has not normally been available to the District within 60 days and a verbal extension has typically been required from the NRC CNS Project Manager.

Core Operating Limits Report (Continued)

b. The Linear Heat Generation Rate for Specification 3.11.B.

c. The K_f core flow MCPR adjustment factor for Specification 3.11.C.

d. The minimum critical power ratio (MCPR) for Specification 3.11.C.

e. The rod block monitor upscale setpoint for Table 3.2.C of Specification 3.2.C.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel." (The approved revision at the time the reload analyses are performed.) The approved revision number shall be identified in the Core Operating Limits Report.

The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

The Core Operating Limits Report, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.5.2 Reportable Events

A Reportable Event shall be any of those conditions specified in Section 50.73 to 10CFR Part 50. The NRC shall be notified and a report submitted pursuant to the requirements of Section 50.73. Each Reportable Event shall be reviewed by SORC and the results of this review shall be submitted to SRAB and the Nuclear Power Group Manager.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.142 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated July 2, 1990, as supplemented by letters dated March 8, and April 19, 1991, Nebraska Public Power District (the licensee) submitted a request for changes to the Cooper Nuclear Station, Technical Specifications (TS). The requested changes would remove cycle-specific reactor physics parameters from the TS and incorporate them into a new document called the Core Operating Limits Report. The amendment was requested in response to Generic Letter 88-16. The March 8, and April 19, 1991, letters provided clarifications and modifications to various proposed TS pages that did not change the action noticed or affect the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definition section of the TS was modified to include a definition of the Core Operating Limits Report (COLR) that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

   (a) Specification 3.11.A and Bases 3.11.A

   The Average Planar Linear Heat Generation Rate (APLHGR) limits for this specification and for these bases are specified in the COLR.

   (b) Specifications 3.11.B, 2.1.A.1.a and 2.1.A.1.d

   The Linear Heat Generation Rate (LHGR) limits for these specifications are specified in the COLR.

(c) Specification 3.11.C

The Minimum Critical Power Ratio (MCPR) operating limits and the MCPR flow adjustment factor (k_f) for this specification are specified in the COLR.

(d) Specification Table 3.2.C

The scoop tube set point (N) for the Rod Block Monitor (RBM) upscale (flow bias) trip level setting is specified in the COLR.

(3) Specification 6.5.1.G was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(a) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel,"

(The approved revision at the time the reload analyses are performed.) The approved revision number shall be identified in the COLR.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

3.0 STATE CONSULATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comment.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant chnage in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or comulative radiation exposure. These amendments also involve changes in recordkeeping or reporting requirements. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (56 FR 6874). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: May 22, 1991