September 28, 198

Docket No. 50-263

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DHagan	BGrimes
MJVirgilio	

Mr. T. M. Parker, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Parker:

SUBJECT: AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-22: RELOCATION OF CYCLE-SPECIFIC THERMAL-HYDRAULIC LIMITS (TAC NO. 74104)

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated July 26, 1989

The amendment relocates cycle-specific thermal-hydraulic limits from the TS to a "Core Operating Limits Report."

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

Sincerely,

original signed by

William O. Long, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 70 to License No. DPR-22
2. Safety Evaluation

DFo1

cc w/enclosures:
See next page

*See previous concurrence

*LA/PD31:DRSP	*PM/PD31:DRSP	*(A)D/PD31:DRSP	*OGC
RIngram	WLong	JThoma	
9/8/89	9/8/89	9/22/89	9/12/89

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WRR

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DCD/JT

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RIngram	WLong	JThoma	MYoung
9/8/89	9/8/89	9/22/89	9/12/89

Ck STATE/SELF Notification re issuance

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

September 28, 1989

Docket No. 50-263

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Northern States Power Company
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See next page

Mr. T. M. Parker, Manager
Northern States Power Company

Monticello Nuclear Generating Plant

cc:

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and
Trowbridge
2300 N Street, NW
Washington, D. C. 20037

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
Box 1200
Monticello, Minnesota 55362

Plant Manager
Monticello Nuclear Generating Plant
Northern States Power Company
Monticello, Minnesota 55362

Russell J. Hatling
Minnesota Environmental Control
Citizens Association (MECCA)
Energy Task Force
144 Melbourne Avenue, S. E.
Minneapolis, Minnesota 55113

Dr. John W. Ferman
Minnesota Pollution Control Agency
520 Lafayette Road
St. Paul, Minnesota 55155-3898

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Commissioner of Health
Minnesota Department of Health
717 Delaware Street, S. E.
Minneapolis, Minnesota 55440

O. J. Arlien, Auditor
Wright County Board of
Commissioners
10 NW Second Street
Buffalo, Minnesota 55313

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.70
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated July 26, 1989, 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 Operating License No. DPR-22 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 70, 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

Dominic C. Di Ianni /JT

John Thoma, Acting Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 28, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 70

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
v	v
vii	vii
2	2
-	5b
14	14
20	20
46	46
57	57
58	58
67	67
69a	69a
211	211
212	212
213	213
214	-
215a	-
215b	-
215c	-
215d	-
216	216
217	217
218	218
-	249b

LIST OF FIGURES

LIST OF TABLES (continued)

F. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, accuracy, and response time to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm or trip. Response time is not part of the routine instrument calibration but will be checked once per cycle.

G. Limiting Conditions for Operation (LCO) - The limiting conditions for operation specify the minimum acceptable levels of system performance necessary to assure safe startup and operation of the facility. When these conditions are met, the plant can be operated safely and abnormal situations can be safely controlled.

H. -Deleted-

I. Limiting Safety System Setting (LSSS) - The limiting safety system settings are settings on instrumentation which initiate the automatic protective action at a level such that the safety limits will not be exceeded. The region between the safety limit and these settings represents margin with normal operation lying below these settings. The margin has been established so that with proper operation of the instrumentation, the safety limits will never be exceeded.

J. Minimum Critical Power Ratio (MCPR) - The minimum critical power ratio is the value of critical power ratio associated with the most limiting assembly in the reactor core. Critical power ratio (CPR) is the ratio of that power in a fuel assembly which is calculated by the GEXL correlation to cause some point in the assembly to experience boiling transition to the actual assembly operating power.

K. Mode - The reactor mode is that which is established by the mode-selector switch.

L. Operable - A system, subsystem, train, component or device shall be Operable or have Operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

AQ. <u>Core Operating Limits Report</u> The Core Operating Limits Report is the unit specific document that provides core operating limits for the current operating reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.7. Plant operation within these operating limits is addressed in individual specifications.

BASES:

2.3 The abnormal operational transients applicable to operation of the Monticello Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power level of 1670 MWt. The analyses were based upon plant operation in accordance with the operating map. The licensed maximum power level 1670 MWt 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 tranisent results as determined by the current analysis model. Conservatism incorporated into the transient analysis is documented in Reference 1.

Bases Continued:

that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of the neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

The operator will set this pressure trip at greater than or equal to 825 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 39.

3.0 LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling Subsystems Actuation

When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling subsystems are given in Table 3.2.2.

C. Control Rod Block Actuation

1. SRM, IRM, APRM and Scram Discharge Volume Rod Blocks

   The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.

2. Rod Block Monitor (RBM)

   a. When core thermal power is greater than or equal to 30% of rated and MCPR is below the limits specified in the Core Operating Limits Report, either:

      (1) Both RBM channels shall be operable, or

      (2) With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, or

      (3) With both RBM channels inoperable, control rod withdrawal shall be blocked immediately.

4.0 SURVEILLANCE REQUIREMENTS

C. Control Rod Block Actuation.

During operation requiring RBM operability when only one channel is operable, an instrument functional test of the operable RBM shall be performed within 24 hours prior to withdrawal of control rod(s).

TABLE 3.2.3 - Continued
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes in Which Function Must be Operable or Operating and Allowable Bypass Conditions** Refuel	Startup	Run	Total No. of Instrument Channels per System	Min. No. of Operable or Operating Instrument Channels per Trip System	Required Conditions*
4. RBM							
a. Upscale (power referenced).	(Note 8)	See Section 3.2.C.2			1	See Section 3.2.C.2 (note 5)	See Section 3.2.C.2
b. Downscale	≥94/125 of full scale	See Section 3.2.C.2			1	See Section 3.2.C.2 (note 5)	See Section 3.2.C.2
5. Scram Discharge Volume							
Water Level - High							
a. East	≤40 gal		X	X	1	1 (note 6)	B and D, or A
b. West	≤40 gal		X	X	1	1 (note 6)	B and D, or A

Table 3.2.3 - Continued
Instrumentation That Initiates Rod Block

Notes:

(1) There shall be two operable or operating trip systems for each function. If the minimum number of operable or operating instrument channels cannot be met for one of the two trip systems, this condition may exist up to seven days provided that during this time the operable system is functionally tested immediately and daily thereafter.

(2) "W" is the reactor recirculation driving flow in percent, dw = 0 for two recirculation loop operation, dw = 5.4 for single recirculation loop operation.

(3) Only one of the four SRM channels may be bypassed.

(4) There must be at least one operable or operating IRM channel monitoring each core quadrant.

(5) An RBM channel will be considered inoperable if there are less than half the total number of normal inputs.

(6) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied actions shall be initiated to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition or

(b) Place the plant under the specified required conditions using normal operating procedures.

(7) There must be a total of at least 4 operable or operating APRM channels.

(8) There are 3 upscale trip levels. Only one is applied over a specified operating core thermal power range. All RBM trips are automatically bypassed below 30% thermal power. Trip settings are provided in the Core Operating Limits Report.

Amendment No. ~~29~~, ~~47~~, 70

Bases Continued:

3.2 The HPCI and/or RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 4 and/or Group 5 valves. The trip settings of 200°F and 150% of HPCI and 300% of RCIC design flows and valve closure times are such that the core will not be uncovered and fission product release will not exceed 10 CFR 100 guidelines.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR remains above the Safety Limit (T.S.2.1.A). The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. See Section 7.3 FSAR.

The APRM rod block trip is referenced to flow and prevents operation significantly above the licensing basis power level especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The operator will set the APRM rod block trip settings no greater than that stated in Table 3.2.3. However, the actual setpoint can be as much as 3% greater than that stated in Table 3.2.3 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

The RBM provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is referenced to power. This power signal is provided by the APRMs. A statistical analysis of many single control rod withdrawal errors has been performed and at the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs greater than the Safety Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. With one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked. Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

Bases Continued:

open and instrumentation drift has caused the nominal 80-psi blowdown range to be reduced to 60 psi. Maximum water leg clearing time has been calculated to be less than 6 seconds for the Monticello design. Inhibit timers are provided for each valve to prevent the valve from being manually opened less than 10 seconds following valve closure. Valve opening is sensed by pressure switches in the valve discharge line. Each valve is provided with two trip, or actuation, systems. Each system is provided with two channels of instrumentation for each of the above described functions. A two-out-of-two-once logic scheme ensures that no single failure will defeat the low-low set function and no single failure will cause spurious operation of a safety/relief valve. Allowable deviations are provided for each specified instrument setpoint. Setpoints within the specified allowable deviations provide assurance that subsequent safety/relief valve actuations are sufficiently spaced to allow for discharge line water leg clearing.

Control room habitability protection assures that the control room operators will be adequately protected against the effects of accidental releases of toxic substances and of radioactive leakage which may bypass secondary containment following a loss of coolant accident or radioactive releases from a steam line break accident, thus assuring that the Monticello Nuclear Generating Plant can be operated or shutdown safely. A study conducted by Bechtel Power Corporation concluded that of the onsite and offsite potential toxic chemical hazards, only chlorine required automatic detection and isolation to prevent incapacitation of control room operators. All other chemicals were determined to have at least two minutes between detection and possible incapacitation. Protection for there toxic chemicals is provided through operator training.

Although the operator will set the setpoints within the trip settings specified in Tables 3.2.1 through 3.2.9, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, these deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts:

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3.11 REACTOR FUEL ASSEMBLIES	4.11 REACTOR FUEL ASSEMBLIES

3.11 REACTOR FUEL ASSEMBLIES

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 two recirculation loop power operation, the APLHGR limiting condition for operation for each type of fuel as a function of axial location and average planar exposure shall not exceed limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types as determined by the approved methodology described in NEDE-24011-P-A (GESTAR II). This approval is based on and limited to GESTAR II methodology. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) provided in the Core Operating Limits Report.

During one recirculation loop power operation, the APLHGR limiting condition for operation for each type of fuel shall not exceed the above values multiplied by 0.85.

If at any time during power operation, it is determined that the APLHGR limiting condition for operation is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two hours, reduce thermal power to less than 25% within the next four hours.

4.11 REACTOR FUEL ASSEMBLIES

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 $\geq$25% rated thermal power.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
B. Linear Heat Generation Rate (LHGR) During power operation, the LHGR shall be less than or equal to the limits specified in the Core Operating Limits Report. If at any time during operation it is determined that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 2 hours, reduce thermal power to less than 25% within the next 4 hours.	B. Linear Heat Generation Rate (LHGR) The LHGR shall be checked daily during reactor operation at $\geq$25% of rated thermal power.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
C. Minimum Critical Power Ratio (MCPR)	C. Minimum Critical Power Ratio (MCPR)
The MCPR shall be greater than or equal to the limits provided in the Core Operating Limits Report. The OLMCPR limit for one recirculation loop operation is 0.01 higher than the comparable two loop value. If at any time during operation it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two hours, reduce thermal power to less than 25% within the next four hours.	MCPR shall be determined daily during reactor power operation at $\geq$25% rated thermal power and following any change in power level or distribution which has the potential of bringing the core to its operating MCPR Limit.

The next page is 216

Bases 3.11:

A. Average Planar Linear Heat Generation Rate (APLHGR)

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

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 cladding temperature by less than ±20° 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 at rated conditions conform to 10CFR50.46. The limiting value for APLHGR is given by this specification.

The flow dependent correction factor provided in the Core Operating Limits Report applied to the rated condition's APLHGR limits assures that 1) the 2200°F PCT limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and 2) the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from less than rated core flow conditions. The power dependent correction factor provided in the Core Operating Limits Report applied to the rated conditions APLHGR limits assures that the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from all conditions provided in the Core Operating Limits Report.

Those abnormal operational transients, analyzed in FSAR Section 14.5 which result in an automatic reactor scram are not considered a violation of LCO. Exceeding APHLGR limits in such cases need not be reported.

B. LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of LCO. Exceeding LHGR limits in such cases need not be reported.

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in References 1 and 2 assumed the steady state MCPR prior to the postulated loss-of-coolant accident for all fuel types for rated flow. The Rated

Bases Continued:

MCPR Limit is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_F$ and $MCPR_P$ at the existing core flow and power state. The required MCPR is a function of flow in order to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The MPCRs were calculated such that for the maximum core flow rate and the corresponding thermal power along the 105% of rated power/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 MCPRs were calculated at different points along the 105% of rated power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow ($MCPR_F$) is provided in the Core Operating Limits Report.

For operation above 45% of rated thermal power, the core power dependent MCPR operating limit is the rated MCPR limit, MCPR(100), multiplied by the factor, provided in the Core Operating Limits Report. For operation below 45% of rated thermal power (turbine control valve fast closure and turbine stop valve closure scrams can be bypassed) MCPR limits are provided in the Core Operating Limits Report. This protects the core from plant transients other than core flow increase, including a localized event such as rod withdrawal error.

Bases Continued:

This limit was determined based upon bounding analyses for the limiting transient at the given core power level.

At thermal power levels less than or equal to 25% of rated thermal power, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. MCPR evaluation below this power level is therefore unnecessary. The daily requirement for calculating MCPR above 25% of rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding MCPR limits in such cases need not be reported.

References

1. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", NEDE-20566, November, 1975.

2. "Loss-of-Coolant Accident Analysis Report for the Monticello Nuclear Generating Plant", NEDO-24050-1, December, 1980, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (USNRC), February 6, 1981.

Bases 4.11

The APLHGR, LHGR and MCPR shall be checked daily to determine if fuel burnup, or control rod movement have caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. In addition, the MCPR is checked whenever changes in the core power level or distribution are made which have the potential of bringing the fuel rods to their thermal-hydraulic limits.

7. <u>Core Operating Limits Report</u>

a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

Rod Block Monitor Operability Requirements
(Specification 3.2.C.2a)

Rod Block Monitor Upscale Trip Settings
(Table 3.2.3, Item 4.a)

Maximum Average Planar Linear Heat Generation Rate Limits
(Specification 3.11.A)

Linear Heat Generation Ratio Limits
(Specification 3.11.B)

Minimum Critical Power Ratio Limits
(Specification 3.11.C)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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

NSPNAD-8608-A, "Reload Safety Evaluation Methods for Application to the Monticello Nuclear Generating Plant" (latest approved version)

NSPNAD-8609-A, "Qualification of Reactor Physics Methods for Application to Monticello" (latest approved version)

c. 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.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated July 26, 1989 (Ref. 1), Northern States Power Company (the licensee) proposed changes to the Technical Specifications (TSs) for the Monticello Nuclear Generating Plant (Monticello). The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of the TSs. Guidance on the proposed changes was developed by the NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988 (Ref. 2).

2.0 EVALUATION

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

(1) The Definition section of the TSs was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC-approved methodology that maintains 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 3.2.C.2.a

The Rod Block Monitor operability requirements for this specification are provided in the COLR.

(b) Specification 3.2.C.1 (Item 4.a and Note 8 of Table 3.2.3)

The Rod Block Monitor upscale trip settings for this specification are provided in the COLR.

(c) Specification 3.11.A

The Maximum Average Planar Linear Heat Generation Rate limits for this specification are provided in the COLR.

(d) Specification 3.11.B

The Linear Heat Generation Rate limits for this specification are provided in the COLR.

(e) Specification 3.11.C

The Minimum Critical Power Ratio limits for this specification are provided in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(3) Specification 6.7.A.7 was added to the reporting requirements of the Administrative Controls section of the TSs. 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, this specification requires that the values of these limits be established using NRC-approved methodology and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

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

NSPNAD-8608-A, "Reload Safety Evaluation Methods for Application to the Monticello Nuclear Generating Plant" (latest approved version)

NSPNAD-8609-A, "Qualification of Reactor Physics Methods for Application to Monticello" (latest approved version)

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 the TSs. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC-approved methodology, 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 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in recordkeeping, reporting, or administrative procedures or requirements and changes requirements with respect to the use of facility components located within the restricted are as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types of any effluent that maybe released offsite and that there is no significant increase in individual or cumulative radiation exposure. The staff has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(a) 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 this amendment.

4.0 CONCLUSIONS

We have 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) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter from Thomas M. Parker (NSP) to NRC, dated July 26, 1989.

2. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

Principal Contributor: D. Fieno

Dated: September 28, 1989