UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50- 263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR-22

(License Amendment Request Dated September 30, 1977)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Operating License and the Technical Specifications as shown on the attachments labeled Exhibit A, Exhibit B, and Exhibit C. Exhibit A describes the proposed amendment to the Operating License along with the reason for the amendment. Exhibit B describes the proposed changes to the Technical Specifications along with reasons for the change. Exhibit C is a set of Technical Specification pages incorporating the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L J Wachter
L J Wachter
Vice President, Power Production & System Operation

On this 30th day of September, 1977, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production & System Operation, and first being duly sworn acknowledged that he is authorized to execute this document in behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Denise E. Halvorson

DENISE E. HALVORSON
NOTARY PUBLIC - MINNESOTA
HENNEPIN COUNTY
My Commission Expires Oct. 10, 1981

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EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263 LICENSE NO. DPR-22

LICENSE AMENDMENT REQUEST
DATED SEPTEMBER 30, 1977

PROPOSED CHANGES TO PROVISIONAL OPERATING LICENSE DPR-22

Pursuant to 10CFR50.59, the holders of Provisional Operating License DPR-22 hereby propose the following change to License DPR-22.

PROPOSED CHANGE

Delete the following provision from the license which was added by a March 11, 1977 Nuclear Regulatory Commission Order for Modification of License:

> "(1) As soon as possible, the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with General Electric Company's Evaluation Model approved by the NRC staff and corrected for the errors described herein and any other corrections in the Model of which the licensee is aware at the time the calculations are performed."

REASON FOR CHANGE

The re-evaluation of ECCS cooling performance has been submitted pursuant to this paragraph, making the referenced paragraph obsolete. The re-evaluation was provided under cover letter dated September 15, 1977 from Mr L O Mayer (NSP) to Mr V Stello (USNRC). The re-evaluation report is entitled, "Loss-of-Coolant Accident Analysis Report for Monticello Nuclear Generating Plant; NEDO-24050, September, 1977."

SAFETY EVALUATION

The safety evaluation for the proposed amendment is embodied in the re-evaluation report referenced above and the numerous references documented therein.

EXHIBIT B

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263 LICENSE NO. DPR-22

LICENSE AMENDMENT REQUEST
DATED SEPTEMBER 30, 1977

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
APPENDIX A OF
PROVISIONAL OPERATING LICENSE DPR-22

Pursuant to 10CFR50.59, the holders of Provisional Operating License DPR-22 hereby propose the following change to the Appendix A Technical Specifications.

1. LIST OF FIGURES (Pages vii and 189I)

PROPOSED CHANGES

Replace the words "Maximum Average Linear Heat Generation Rate versus Average Exposure Monticello 70230 Fuel" with the word "Deleted". Likewise, all information in Figure 3.11.1-B should be replaced by the words "This Figure has been Deleted".

REASON FOR CHANGES

The 70230 fuel type is no longer used in the Monticello reactor and therefore no replacement figure for the thermal limits for that fuel type is presented.

SAFETY EVALUATION

This change only removes obsolete material from the technical specifications.

2. SPECIFICATION 3.11.A and BASIS (Pages 189B, 189C and 189F)

PROPOSED CHANGES

On Page 189B, add the sentence "When core flow is less than 70% of rated flow, the APLHGR shall not exceed 90% of the limiting value shown in Figures 3.11.1." as shown in Exhibit C. Move overflow material to the next page, 189C.

On Page 189F, replace Reference 6 with the following reference, " 'Revision of Low Core Flow Effects on LOCA Analysis for Operating BWR's,' R L Gridley (GE) to D G Eisenhut (USNRC), September 28, 1977."

REASON FOR CHANGE

The recent ECCS re-analysis shows that the most limiting break size is a break smaller than the maximum design break area. The low flow ECCS limitations have likewise been re-analyzed and new limits have been established. The new limits are bounding for the postulated loss of coolant accident over the entire break spectrum.

SAFETY EVALUATION

The document referenced above as proposed Basis Reference 6 is the safety evaluation for this change. Monticello is identified as Plant B in that report.

3. FIGURES 3.11.1-A, C and D and BASIS (Pages 189F, 189H, 189J and 189K)

PROPOSED CHANGES

Replace existing Figures 3.11.1-A, C and D of the Technical Specifications, Pages 189H, 189J and 189K, with the corresponding proposed figures included in Exhibit C. Replace Reference 4 on Page 189F with the reference, " 'Loss-of-Coolant Accident Analysis Report for Monticello Nuclear Generating Plant,' NEDO-24050, September, 1977, L O Mayer (NSP) to V Stello (USNRC) September 15, 1977".

REASON FOR CHANGE

The ECCS thermal limitations for the fuel types in service at Monticello have been re-evaluated using a newly approved model. The thermal limits are being adjusted and documented accordingly.

SAFETY EVALUATION

The document reference above as a proposed Basis Reference 4 is the safety evaluation for this change.

EXHIBIT C

LICENSE AMENDMENT REQUEST
DATED SEPTEMBER 30, 1977

This exhibit consists of the following pages revised to incorporate all of the proposed Technical Specification changes:

vii
189B
189C
189F
189H
189I
189J
189K

<u>LIST OF FIGURES</u>

3.0 LIMITING CONDITIONS FOR OPERATION

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 power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.11.1. When core flow is less than 90% of rated core flow, the APLHGR shall not exceed 95% of the limiting value shown in Figures 3.11.1. When core flow is less than 70% of rated core flow, the APLHGR shall not exceed 90% of the limiting value shown in Figures 3.11.1. If any time during operation it is determined that the limit for APLHGR is being exceeded, action shall be initiated within 15

4.0 SURVEILLANCE REQUIREMENTS

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.

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 (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours.

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR as a function of core height shall not exceed the limiting value shown in Figure 3.11.2. 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 two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours.

B. Linear Heat Generation Rate (LHGR)

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

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 assumed the steady state MCPR prior to the postulated loss-of-coolant accident to be 1.18 for all fuel types. In addition, the ECCS analysis presented in Reference 6 assumed an initial MCPR of 1.24 for reduced flow conditions. The Operating MCPR Limit of 1.38 for 8x8 fuel and 1.29 for 7x7 fuel 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 of 1.06 (T.S.2.1.A) applicable to all fuel types is maintained in the event of the most limiting abnormal operational transient.

For operation with less than rated core flow the Operating MCPR Limit is adjusted by multiplying the above limit by K_f. Reference 5 discusses how the transient analysis done at rated conditions encompasses the reduced flow situation when the proper K_f factor is applied.

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. "Fuel Densification Effects in General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff)

3. Communication: VAMoore to IS Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

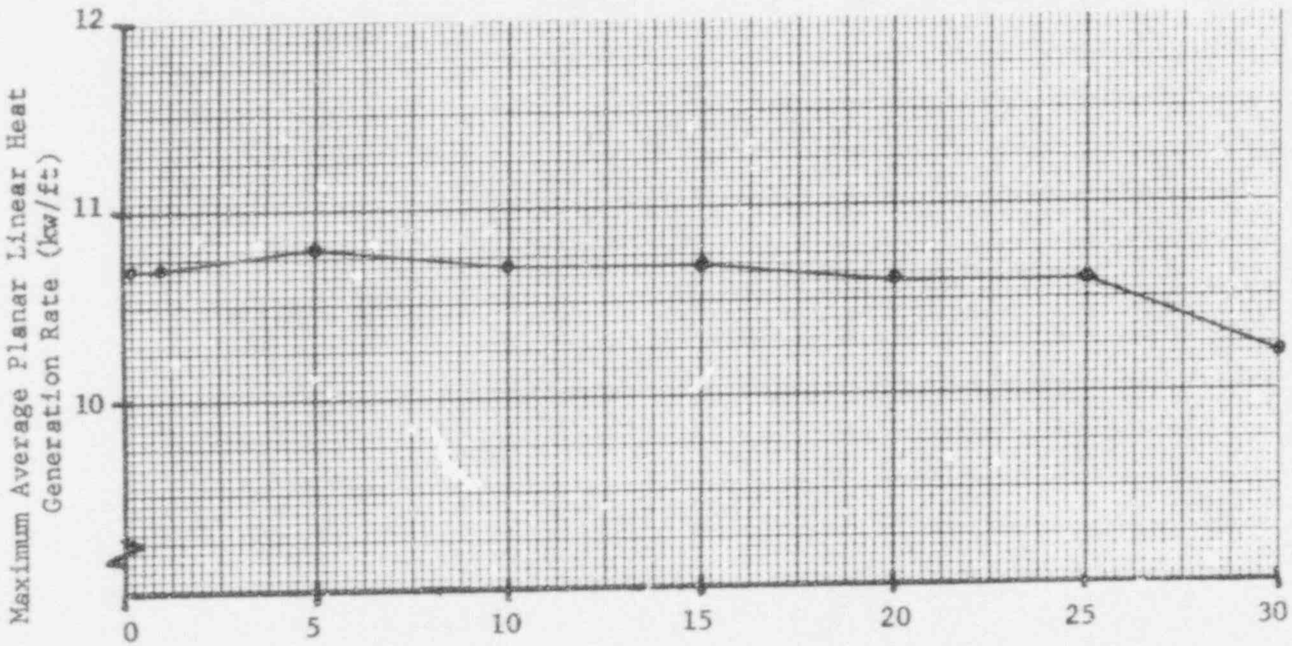
4. "Loss-of-coolant Accident Analysis Report for the Monticello Nuclear Generating Plant," NEDO-24050, September 1977, L O Mayer (NSP) to V Stello (USNRC), September 15, 1977.

5. "General Electric BWR Generic Reload Application for 8 x 8 Fuel," NEDO-20360, Revision 1, November 1974.

6. "Revision of Low Core Flow Effects on LOCA Analysis for Operating BWR's," R L Gridley (GE) to D G Eisenhut (USNRC), September 23, 1977.

Figure 3.11.1-A

Maximum Average Planar Linear
Heat Generation Rate Versus
Planar Average Exposure
Monticello 8D219 Fuel



Tabulation of
Plotted Data

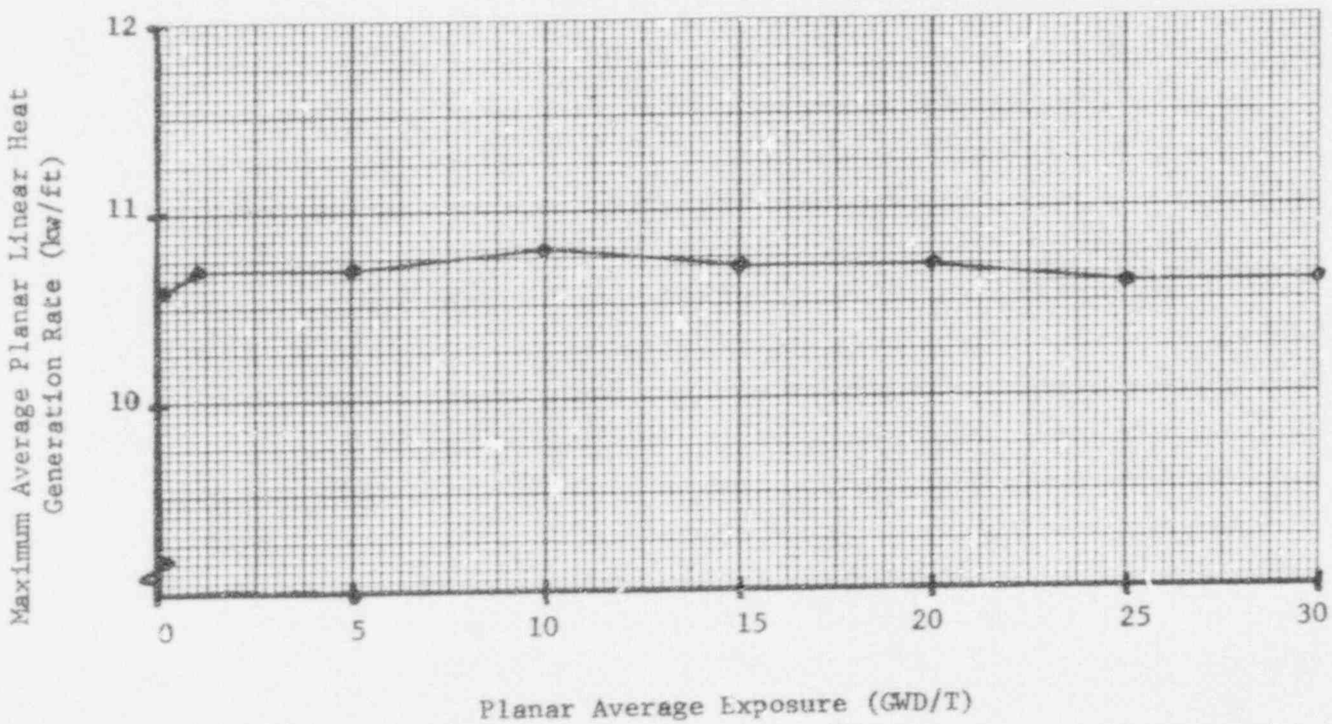
Exposure	MAPLHGR
200	10.7
1,000	1C.7
5,000	10.8
10,000	10.7
15,000	10.7
20,000	10.6
25,000	10.6
30,000	10.2

Figure 3.11.1-B

This figure has been deleted.

Figure 3.11.1-C

Maximum Average Planar Linear
Heat Generation Rate Versus
Planar Average Exposure
Monticello 8D262 Fuel

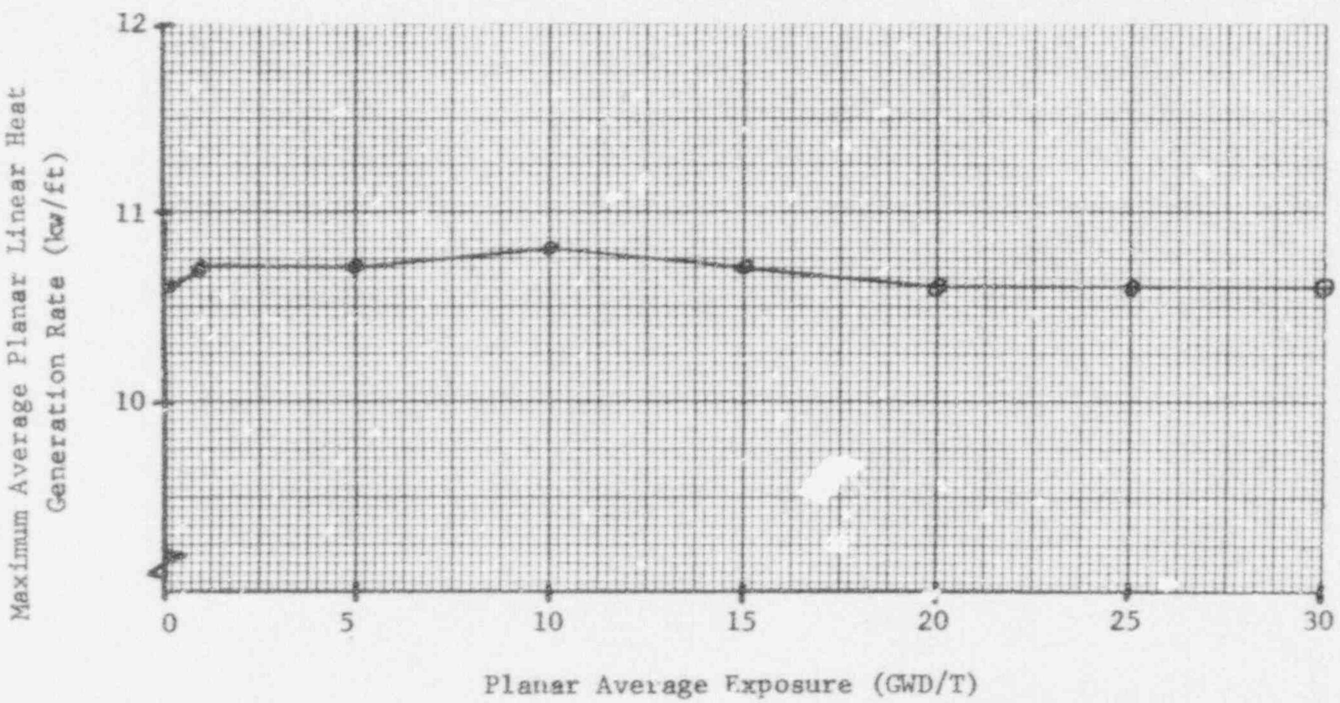


Tabulation of Plotted Data

Exposure	MAPLHGR
200	10.6
1,000	10.7
5,000	10.7
10,000	10.8
15,000	10.7
20,000	10.7
25,000	10.6
30,000	10.6

Figure 3.11.1-D

Maximum Average Planar Linear
Heat Generation Rate Versus
Planar Average Exposure
Monticello 8D250 Fuel



Tabulation of Plotted Data

Exposure	MAPLHGR
200	10.6
1,000	10.7
5,000	10.7
10,000	10.8
15,000	10.7
20,000	10.6
25,000	10.6
30,000	10.6

NRC FORM 195
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U.S. NUCLEAR REGULATORY COMMISSION

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L O Mayer

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ENCLOSURE

License # DPR-22 Amend: proposed change to tech specs concerning incorporation of ECCS re-analysis..................................

12p

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40 ENCL

SAFETY

FOR ACTION/INFORMATION

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