January 27, 1993

Docket No. 50-263

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

Dear Mr. Parker:

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. DPR-22 (TAC NO. M84515)

The Commission has issued the enclosed Amendment No. 84 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated September 16, 1992, as supplemented November 3, 1992.

The amendment permits implementation of an expanded operating domain resulting from maximum extended load line limit analysis (MELLLA) and increased core flow (ICF).

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:

Anthony H. Hsia, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 84 to DPR-22
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD31	PM:PD31	OGC	SRXB	D:PD31
NAME	MShuttleworth	AHsia:dmy		Jones	LMarsh
DATE	1/12/93	1/15/93	1/25/93	1/15/93	1/26/93

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

January 27, 1993

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Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
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Sincerely,

Anthony H. Hsia

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Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 84 to DPR-22
2. Safety Evaluation

cc w/enclosures:
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DATED: January 27, 1993

AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. DPR-22-MONTICELLO

Docket File
NRC & Local PDRs
PDIII-1 Reading
Monticello Plant File
J. Roe
J. Zwolinski
L. Marsh
M. Shuttleworth
A. Hsia
R. Jones
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
ACRS (10)
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OC/LFMB
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cc: Plant Service list

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. 84
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 September 16, 1992, 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. 84 , 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

LBMarsh

Ledyard B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 27, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 84

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 vertical lines indicating the areas of change.

REMOVE	INSERT
6	6
15	15
16	16
28	28
56	56
58	58
249b	249b

2.0 SAFETY LIMITS

2.1 FUEL CLADDING INTEGRITY

Applicability

Applies to the interrelated variables associated with fuel thermal behavior

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Core Thermal Power Limit (Reactor Pressure >800 psia and Core Flow is >10% of Rated)

When the reactor pressure is >800 psia and core flow is >10% of rated, the existence of a minimum critical power ratio (MCPR) less than 1.07, for two recirculation loop operation, or less than 1.08 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

LIMITING SAFETY SYSTEM SETTINGS

2.3 FUEL CLADDING INTEGRITY

Applicability

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

The Limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - The APRM flux scram trip setting shall be:
   a. For two recirculation loop operation (TLO):

      $S \leq 0.66W + 70\%$ where,

      S = Setting in percent of rated thermal power, rated power being 1670 MWT

      W = Percent of the drive flow required to produce a rated core flow of 57.6×10^6 lb/hr
   b. For single recirculation loop operation (SLO):

      $S \leq 0.58(W - 5.4) + 62\%$
   c. No greater than 120%.

Bases Continued:

For analyses of the thermal consequences of the transients, the Operating MCPR Limit (T.S.3.11.C) 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.

Deviations from as-left settings of setpoints are expected due to inherent instrument error, operator setting error, drift of the setpoint, etc. Allowable deviations are assigned to the limiting safety system settings for this reason. The effect of settings being at their allowable deviation extreme is minimal with respect to that of the conservatisms discussed above. Although the operator will set the setpoints within the trip settings specified, the actual values of the various setpoints can vary from the specified trip setting by the allowable deviation.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting or when a sufficient number of devices have been affected by any means such that the automatic function is incapable of preventing a safety limit from being exceeded while in a reactor mode in which the specified function must be operable. Sections 3.1 and 3.2 list the reactor modes in which the functions listed above are required.

A. Neutron Flux Scram The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1670 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

Bases Continued:

Maximum Extended Load Line Limit Analyses have been performed to allow operation at higher powers at flows below 87%. The flow referenced scram (and rod block line) have increased (higher slope and y-intercept) for two loop operation (See Core Operating Limits Report). These analyses have not changed the allowed operation for single loop operation. The supporting analyses are discussed in GE NEDC-31849P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Increased Core Flow analyses have been performed to allow operating at flows above 100% for powers equal to or less than 100% (See Core Operating Limit Report). The supporting analyses are discussed in General Electric NEDC-31778P report (Reference: Letter from NSP to NRC dated September 16, 1992).

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 20% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate 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 procedures. 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% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position and the associated APRM is not downscale. This switch occurs when reactor pressure is greater than 850 psig.

The operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 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.

B. Deleted

Next Page is 18

TABLE 3.1.1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating** Refuel (3)	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Condition*
1. Mode Switch in Shutdown		X	X	X	1	1	A
2. Manual Scram		X	X	X	1	1	A
3. Neutron Flux IRM (See Note 2) a. High-High b. Inoperative	≤ 120/125 of full scale	X	X		4	3	A
4. Flow Referenced Neutron Flux APRM (See Note 5) a. High-High b. Inoperative c. High Flow Clamp	See Specifications 2.3A.1 ≤ 120 %			X	3	2	A or B
5. High Reactor Pressure (See Note 9)	≤ 1075 psig	X	X(f)	X(f)	2	2	A
6. High Drywell Pressure (See Note 4)	≤ 2 psig	X	X(e,f)	X(e,f)	2	2	A
7. Reactor Low Water Level	≥ 7 in.(6)	X	X(f)	X(f)	2	2	A
8. Scram Discharge Volume High Level							
a. East	≤ 56 gal.(8)	X(a)	X(f)	X(f)	2	2	A
b. West	≤ 56 gal.(8)	X(a)	X(f)	X(f)	2	2	A
9. Turbine Condenser Low Vacuum	≥ 23 in. Hg	X(b)	X(b,f)	X(f)	2	2	A or C

TABLE 3.2.3
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes Which Function Must be Operable or Operating and Allowable Bypass Conditions** Refuel	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels per Trip System	Required Conditions*
1. SRM							
a. Upscale	$\leq 5\times10^5$ cps	X	X(d)		2	1(Note 1, 3, 6)	A or B or C
b. Detector not fully inserted		X(a)	X(a)		2	1(Note 1, 3, 6)	A or B or C
2. IRM							
a. Downscale	≥3/125 full scale	X(b)	X(b)		4	2(Note 1, 4, 6)	A or B or C
b. Upscale	≤108/125 full scale	X	X		4	2(Note 1, 4, 6)	A or B or C
3. APRM							
a. Upscale				X	3	1(Note 1, 6, 7)	D or E
(1) TLO Flow Biased	≤ 0.66W + 58% (Note 2)						
(2) SLO Flow Biased	≤ 0.58(W - 5.4) + 50% (Note 2)						
(3) High Flow Clamp	≤ 108%						
b. Downscale	≥ 3/125 full scale			X	3	1(Note 1, 6, 7)	D or E

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 percent of drive flow required to produce a rated core flow of 57.6×10^6 lb/hr

(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.

7. Core Operating Limits Report

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)

Power to Flow Map
(Bases 2.3.A)

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. 84 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 September 16, 1992, as supplemented November 3, 1992, the Northern States Power Company (the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The proposed amendment approves TS changes to permit implementation of an expanded operating domain resulting from maximum extended load line limit analysis (MELLLA) and increased core flow (ICF).

The November 3, 1992, submittal provided clarifying information that did not change the initial proposed no significant hazards determination.

2.0 EVALUATION

2.1 DESCRIPTION OF PROPOSED TS CHANGES

Average Power Range Monitor (APRM) Flow-Biased Scram Set Point - TS 2.3.A, and Table 3.1.1: The proposed amendment would change the flux scram lines on the power/flow map and establish a new two-loop APRM flux scram line of 0.66W + 70%. The existing two-loop APRM flux scram function is 0.58W + 62%. The single-loop line would remain at 0.58(W-5.4) + 62%. (Note: "W" is the percentage of drive flow required for 100% core flow.) This change would allow higher power operation when core flow is below 87%.

In addition, a high flow clamp of 120% rated power would be added to the APRM scram specifications. The Updated Safety Analysis Report (USAR), Section 7.3.5.2.2 indicates that the 120% clamp instrumentation is currently installed.

Average Power Range Monitor Flow-Biased Rod Block Set Point - Table 3.2.3: The proposed amendment would similarly change the rod block monitor (RBM) set point requirements, establishing a two-loop set point function of 0.66W + 58%, and a high flow clamp of 108% rated power. The present two-loop operation rod block set point function is 0.58W + 50%. The proposed changes maintain the same maximum set points (120% for APRM scram and 108% for rod block) as currently approved. The changes also maintain the same margin (12%) between the APRM scram and rod block set points as provided in the current Technical Specifications. As for the scram function, the single-loop operation requirement would not be changed.

The Bases (2.3.A) and Core Operating Limits Report (COLR) Reporting Requirements (TS 6.7.a): The Bases sections of the TS would be revised to reflect the changes to affected Limiting Safety System Settings and Limiting Conditions for Operation. References to supporting analyses would be included in the revised Bases. Also, reporting requirements for the COLR would be changed to require that the COLR for each reload include a power/flow map.

2.2 EVALUATION

The proposed amendment affects safety system set points for two protective functions, (1) the APRM flow-biased scram, and (2) the RBM trip. Both functions utilize the APRM neutron monitoring system.

The function of the slope and bias circuits of the APRM flow-biased scram is to account for the decreasing margin to fuel damage at a given power level with reduced recirculation flow. The function of the RBM is to prevent rod withdrawal under conditions that could initiate a rod withdrawal error event leading to local fuel damage. Operation with the proposed changes implemented on these protective functions has been analyzed for limiting accidents, transients, and thermal-hydraulic stability.

The results of the Monticello analyses are reported in (a) NEDC-31849P "Maximum Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant Cycle 15," (b) NEDC-31849P-1, "Maximum Extended Load Line Limit Analysis for Monticello Nuclear Generating Plant Cycle 15, Supplement 1," and (c) NEDC-31778P "Safety Review for Monticello Nuclear Generating Plant Increased Core Flow Operation Throughout Cycle 14." These documents provide the results of new analyses and evaluations supporting MELLLA and ICF operation and discussions providing the basis for not reanalyzing certain events. Events addressed in these reports include:

A. Transients:
   (1) Feedwater Controller Failure with Maximum Demand (MELLLA and ICF)
   (2) Turbine Trip with Bypass Failure (MELLLA and ICF)
   (3) Loss of Feedwater Heating (MELLLA)
   (4) Closure of One Turbine Stop Valve at 100% Power and 75% Flow (MELLLA)
   (4) Transfer to Backup Pressure Regulator (MELLLA)
   (5) Slow Flow Runout (MELLLA)
   (6) Main Steam Isolation Valve Closure With Flux Scram (MELLLA and ICF)
   (7) Rod Withdrawal Error Event (MELLLA and ICF)
   (8) Rod Drop Accident (ICF)

B. Design Basis Accident (DBA)
   (1) DBA-LOCA Containment Pressure Response and containment dynamic loads (MELLLA)
   (2) DBA-LOCA Peak Clad Temperature (MELLLA and ICF)

C. Analysis of effect of acoustic loads and flow-induced loads on reactor internals (MELLLA and ICF)

D. Analysis of effects on flow-induced vibration (MELLLA and ICF)

E. Anticipated Transients Without Scram [ATWS] (MELLLA)

Transients: Analysis results for the turbine trip with bypass failure and feedwater controller failure to maximum demand for 100% power and 75% core flow initial conditions resulted in peak vessel pressures, peak neutron flux and heat flux, and delta-Critical Power Ratios (CPRs) less than the corresponding values for the transient with 100% power and 100% core flow initial conditions. The peak vessel pressure for the MSIV closure with flux scram event initiated from 100% power and 75% core flow was slightly less than the corresponding values for the transient with 100% power and 100% core flow initial condition.

For the closure of one turbine stop valve event and for the transfer to backup pressure regulator event, the peak neutron flux values for the 100% power and 75% core flow initial condition were well below the 120% rated power APRM clamp.

The slow flow runout event was analyzed due to the fact that the power increase for this event occurs along a steeper rod line. Reanalysis was performed for various flow conditions down to 30% recirculation pump speed, with each flow condition on the maximum possible rod line. The analysis demonstrated that results are acceptable using current flow dependent minimum critical power ratio ($MCPR_f$) and flow dependent maximum average planar linear heat generation rate factor ($MAPFAC_f$) limit curves.

Rod Withdrawal Error (RWE): The RWE analysis for the Extended Load Line Limit Analysis (ELLLA) region (Amendment 29) bounds the RWE for the MELLLA and ICF regions.

Design Basis Accidents - Peak Clad Temperature (PCT): SAFER/GESTR and 10 CFR Part 50, Appendix K calculations for a 102% power and 80% core flow initial condition indicate a PCT of 1623°F. This is less than the 1769°F PCT calculated based on the 102% power and 105% initial flow condition. This indicates that for a 75% flow initial condition, the PCT would be expected to be close to that calculated for the 80% flow initial condition, considerably under the 2200°F limit. For ICF operation, the initial core heatup will be lessened, but the core uncovery duration is increased resulting in a slightly increased PCT, on the order of 10°F, which is insignificant.

Design Basis Accidents - Containment Pressure Response: The DBA-LOCA short-term containment response for initial conditions in the MELLLA region was found to be bounded by the current USAR analysis limiting value of 42 psig which is below the containment design value of 56 psig. Long-term containment response is unaffected by initial flow. Containment dynamic loads, including pool swell, condensation oscillation, chugging, and vent clearing are also not impacted by operation in the MELLLA region.

Design Basis Accidents - Effect on Reactor Vessel Internals: Pressure differential forces on fuel assemblies and vessel internals for DBAs (the main steam line (MSL) break upstream of flow restrictors is the worst case) initiated from the MELLLA region are bounded by the values for higher flow operation. Reanalysis was performed for ICF operation and stresses were found to be within allowable values. For the MELLLA region and ICF, flow-induced vibrations and acoustic loads are evaluated with particular attention on the shroud, separator assembly, jet pumps, and incore housing and control rod guide tubes. It was concluded that balanced recirculation loop operations in the MELLLA region and ICF operations up to 105% rated core flow are acceptable.

(Note: The licensee is currently evaluating test results from a series of jet pump tests which were conducted as a result of cracking in a jet pump upper riser brace. Data indicates that riser brace harmonic frequencies may exist at certain pump speeds. Pending completion of full scale tests and analysis of findings, the licensee is restricting operation of the recirculation pumps in the area of concern. The lower riser braces at Monticello are capable of providing the required jet pump support and the upper braces are considered unnecessary. The potential effects of loose parts resulting from upper riser brace failure have been analyzed by the vendor and licensee, and found to result in no safety concern.)

Thermal-Hydraulic Stability: Thermal-hydraulic stability reanalysis was not necessary in support of MELLLA/ICF. Stability performance of GE fuel, which is used exclusively at Monticello, is generically demonstrated in NEDE-22277-P-1 "Compliance of GE BWR Fuel Designs to Stability Licensing Criteria." This NEDE-22277-P-1 has been incorporated as part of the GESTAR II methodology. This methodology utilizes calculations to show that individual channels are as stable or more stable than the core, thus indicating the core-wide limit cycle oscillations will not occur.

ATWS: The limiting ATWS event (MSIV Closure with one safety relief valve out of service, recirculating pumps trip and alternate rod insertion) was reanalyzed for the 100% power and 75% flow initial condition in the MELLLA region. Analysis results of peak vessel pressure, peak containment pressure, and peak suppression pool bulk temperature were within applicable design guidelines.

Summary: The new and earlier analyses encompass and bound the range of effects on accidents and transients resulting from the expanded MELLLA/ICF operating regime. The areas examined are the same areas examined in previous MELLLA-ICF reviews such as that for Fermi-2 reported in the staff's Safety Evaluation of May 15, 1991, supporting Amendment 69 to Operating License NPF-43. Based on the results of the analyses and evaluations described in the report, maximum extended load line operation will not result in significantly reduced fuel thermal margins and will not compromise the structural integrity of the containment or of the reactor vessel or its internals. Based on the findings, the proposed amendment is acceptable.

The analyses conducted in support of MELLLA and ICF operation are cycle specific and will be reanalyzed for each reload as necessary. However, the associated TS changes encompass anticipated future reloads.

The addition of the cycle dependent Power to Flow Map to COLR as described in TS 6.7.a is a clarification and therefore, the staff finds it acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 48823). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The staff 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.

Principal Contributors: William Long, A. H. Hsia

Date: January 27, 1993