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TO: Mr. Victor Stello

FROM: Northern States Power Company
Minneapolis, Minnesota
L. O. Mayer

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ENCLOSURE

Amdt. to ol/change to Appendix A tech specs furnished in exhibits A & B and safety evaluation in support of the change furnished in exhibit C entitled "License Amendment Submittal for Single-Loop Operation".

(37-P)

ACKNOWLEDGED

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

September 7, 1976

REGULATORY DOCKET FILE COPY

DOCKETED USNRC 9/10/76 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Mr Victor Stello, Director
Division of Operating Reactors
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Stello:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request Dated September 7, 1976

Attached are 3 originals and 37 conformed copies of a request for change of the Technical Specifications, Appendix A of the Provisional Operating License for the Monticello Nuclear Generating Plant, Dated September 7, 1976. This request has been reviewed by the Monticello Operations Committee and the Safety Audit Committee.

The proposed change will allow the plant to remain operational at a substantial power level with only one recirculation pump in operation and the equalizer valve closed. Exhibits A and B present the proposed change to the Technical Specifications. Exhibit C presents a safety evaluation in support of the change. Your prompt review of this matter is requested.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager, Nuclear Support Services

Docket # 50-263
Control # 9240
Date Recvd. 9/10/76
Regulatory Docket File

LOM/deb

cc: G Charnoff
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MPCA
Attn: J W Ferman
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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 7, 1976)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B 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 7th day of September, 1976, 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 Branau

DENISE E. BRANAU
NOTARY PUBLIC – MINNESOTA
HENNEPIN COUNTY
My Commission Expires Oct. 10, 1981

Docket # 50-263
Control # 9240
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EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

LICENSE AMENDMENT REQUEST DATED SEPTEMBER 7, 1976

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
APPENDIX A OF PROVISIONAL OPERATING
LICENSE NO. DPR-22

Pursuant to 10CFR 50.59, the holders of provisional operating license DPR-22 hereby propose the following changes to Appendix A Technical Specifications:

Proposed Changes

Incorporate the changes as indicated in the proposed revised pages submitted as Exhibit B.

Reason for Changes

These proposed changes are generally additions to the existing Technical Specifications which are associated with a mode of operation involving only one reactor recirculation pump with the equalizer valves closed. It is desirable to have provisions for this mode of operation because reactor operation can safely continue at a substantial power level when equipment outages exist. The plant was initially designed and licensed to allow operation with only one recirculation pump. The July 9, 1975 ECCS analysis did not address one-pump operation in sufficient detail such that it was restricted by the subsequent License Amendment. An in-depth analysis has now been completed and new, conservative limits are proposed such that the flexibility of one-pump operation can be restored.

Three aspects of these changes not directly related to the one-pump analysis and the reasons for change are discussed individually as follows:

Page 20, Bases - The first complete paragraph on this page is proposed to be replaced. The existing paragraph is part of the initial plant analysis which was made obsolete by the ECCS Interim Acceptance Criteria analysis and never updated. It is presently inconsistent with Specification 3.5.I.

Page 108A, TS 3.5.I.3 - This entire Specification is proposed to be replaced. The first sentence of the existing Specification is redundant to Specifications 3.5.I.1 and 2 and therefore unnecessary. The intent of the second sentence has been incorporated into the proposed Specification.

Page 108A, TS 4.5.I.1 - The spelling of the word "cross" is corrected.

Safety Evaluation

The safety evaluation in support of the proposed changes is included as Exhibit C entitled, "License Amendment Submittal for Single-Loop Operation, Monticello Nuclear Generating Plant Unit I, NEDO-21252".

EXHIBIT B

LICENSE AMENDMENT REQUEST DATED SEPTEMBER 7, 1976

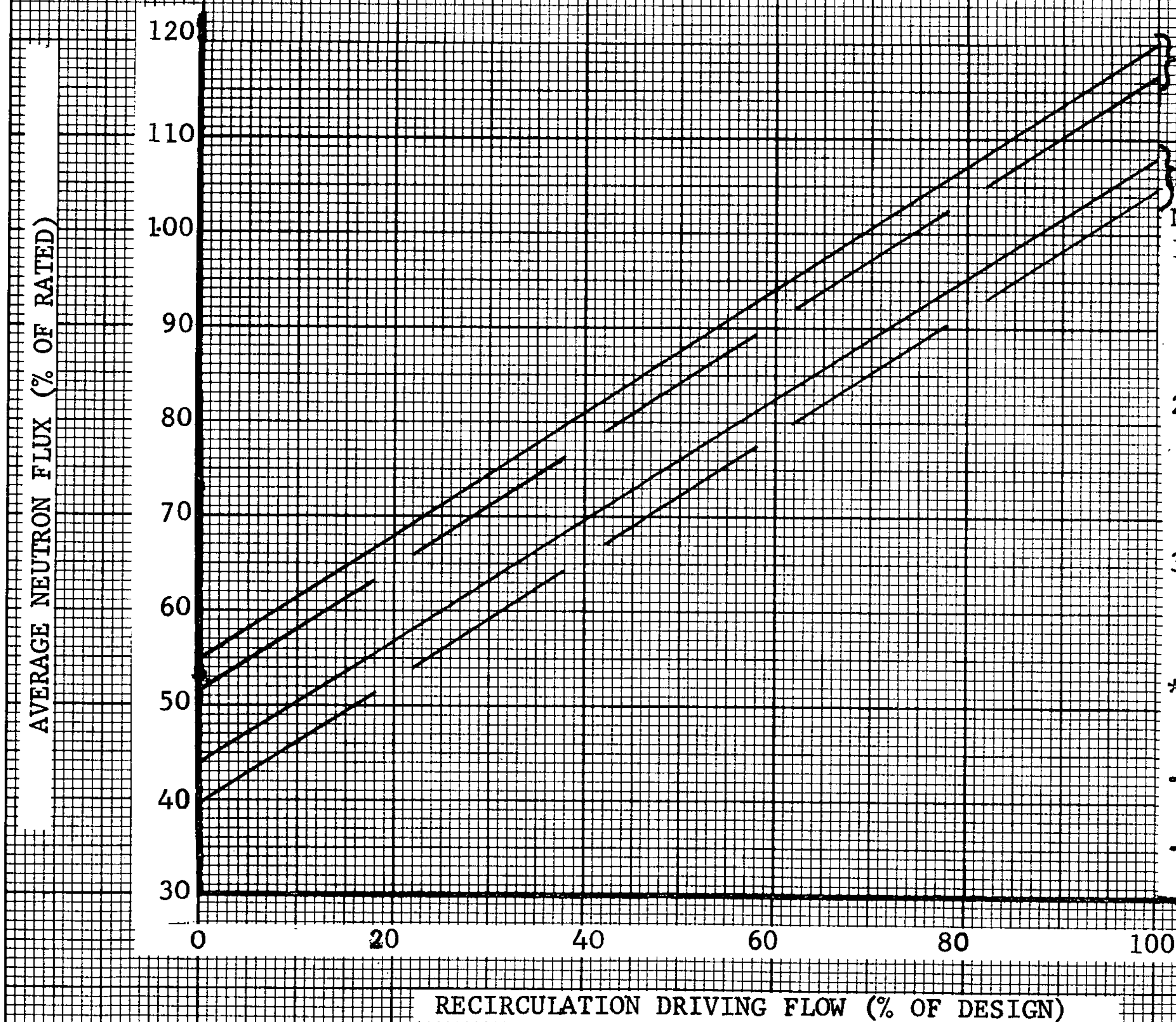
Exhibit B, attached, consists of the following revised pages of the Appendix A Technical Specifications which incorporate the proposed changes:

<u>Pages</u>

11
20
21
57
58
67
108A
113
189B
189E

FIGURE 2.3.1

APRM FLOW REFERENCED SCRAM AND ROD BLOCK TRIP SETTINGS



APRM FLOW REFERENCED SCRAM

APRM FLOW REFERENCED ROD BLOCK

NOTES

1. APRM SCRAM TRIP SETTING SHALL BE:
   $S \leq 0.65W^* + 55.0$ (two pump operation)
   $S \leq 0.65W^* + 51.75$ (one pump operation, equalizer closed)

2. APRM ROD BLOCK TRIP SETTING SHALL BE:
   $RB \leq 0.65W^* + 43.0$ (two pump operation)
   $RB \leq 0.65W^* + 39.75$ (one pump operation, equalizer closed)

3. 100% NEUTRON FLUX IS EQUIVALENT TO 1670 MWt

*W = % OF DESIGN RECIRCULATION DRIVING FLOW

——— two pump operation

— — one pump operation, equalizer closed

Bases Continued:

temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. 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. This switch occurs when reactor pressure is greater than 850 psig.

Operation with one recirculation pump and the equalizer line closed causes flow to bypass the core through idle jet pumps. This loss of core flow is accounted for in the specified setting adjustments for the one-pump, equalizer closed case. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC).

The operator will set the APRM neutron flux trip setting no greater than that shown in Figure 2.3.1. However, the actual setpoint can be as much as 3% greater than that shown on Figure 2.3.1 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 18.

B. APRM Control Rod Block Trips Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the Safety Limit (T.S.2.1.A). This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting,

Bases Continued:

over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. Operation with one recirculation pump and the equalizer line closed causes flow to bypass the core through the idle jet pumps. This loss of core flow is accounted for in the specified setting adjustments for the one-pump, equalizer-closed case. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC). When the maximum total peaking factor exceeds the design value, the rod block setting is adjusted in accordance with the formula in Specification 2.3.B. If the APRM rod block setting should require a change due to an abnormal peaking condition, it will be done by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced rod block curve by the reciprocal of the APRM gain change.

The operator will set the APRM rod block trip settings no greater than that shown in Figure 2.3.1. However, the actual setpoint can be as much as 3% greater than that shown on Figure 2.3.1 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 18.

C. Reactor Low Water Level Scram The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained.

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual setpoint can be as much as 6 inches lower due to the deviations discussed on page 18.

D. Reactor Low Low Water Level ECCS Initiation Trip Point The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could prevent the ECCS components from

Table 3.2.3
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 Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
1. **SRM**							
a. Upscale	≤5 x 10^5 cps	X	X(d)		2	1 (Note 3)	A or B or C
b. Detector not fully inserted		X(a)	X(a)		2	1(Note 3)	A or B or C
2. **IRM**							
a. Downscale	≥3/125 full scale	X(b)	X(b)		4	2(Note 4)	A or B or C
b. Upscale	≤108/125 full scale	X	X		4	2 (Note 4)	A or B or C
3. **APRM**							
a. Upscale (flow referenced)	Reference Specification 2.3.B			X	3	1 (Note 7)	D or E
b. Downscale	≥ 3/125 full scale			X	3	1 (Note 7)	D or E

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 Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
4. RBM							
a. Upscale (flow referenced)	Two-pump operation: ≤.65W + 43 One-pump operation, equalizer closed: ≤.65W + 39.75 (Note 2)			X(c)	1	1 (Note 5)	D or E
b. Downscale	≥3/125 full			X(c)	1	1 (Note 5)	D or E

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.

(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) One of the two RBMs may be bypassed for maintenance and/or testing for periods not in excess of 24 hours in any 30 day period. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

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 $200^{\circ}F$ 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 a significant reduction in MCPR 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 trips are set so that MCPR is maintained greater than the Safety Limit.

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 flow. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked at MCPR greater than the Safety Limit, thus allowing adequate margin. Below 60% power, MCPR remains above the Safety Limit for the worst case withdrawal of a single control rod without rod block action, thus below this level it is not required. This subject is discussed in General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application, NEDO-10958. Requiring at least half of the normal LPRM inputs from each level to be operable assures that the RBM response will be adequate to prevent rod withdrawal errors. Operation with one recirculation pump and the equalizer line closed causes flow to bypass the core through idle jet pumps. This loss of core flow is accounted for in the specified setting adjustments for the one-pump, equalizer closed case. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC).

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the Safety Limit (T.S.2.1.A).

A downscale indication of an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 3/125 of full scale.

I. Recirculation System

1. Except as specified in 3.5.I.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.

2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.

3. Operation with one recirculation pump (equalizer valve closed) is permitted provided that within 24 hours from the time one pump operation commences the appropriate adjustments to limits specified in Specifications 2.3.A.1, 2.3.B, 3.2.C and 3.11.A are incorporated. If the settings cannot be adjusted or two-pump operation restored by the end of 24 hours, an orderly reactor shutdown shall be initiated.

I. Recirculation System

1. Once per month, when irriated fuel is in the reactor with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be demonstrated to be operable by verifying that the cross tie valves cannot be opened using the normal control switch.

2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily.

G. Emergency Cooling Availability

The purpose of Specification G is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment cooling subsystems may be out of service. Specification 3.5.G.3 allows all core and containment cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.G.4 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

H. Deleted

I. Recirculation System

The capacity of the Emergency Core Coolant System is based on the potential consequences of a double ended recirculation line break. Such a break involves 3.9 sq. ft. when the cross tie valves are closed and 5.3 sq. ft. when the cross tie valves are open. Specification 3.11.A is based on an ECCS evaluation assuming a break area of 3.9 sq. ft.; the limitations of 3.11.A do not apply to the larger break area. Therefore, at least one cross tie valve must remain closed at all times to reduce the potential break area as required by Specifications 3.5.I.1 and 2.

An analysis of one-pump operation (equalizer valve closed) identifies certain limitations peculiar to that mode of operation. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC). Operation with only one pump is not a normal mode; it will generally involve a forced outage of equipment. There may be insufficient time to make adjustments to the RBM and APRM flow referenced rod block and scram prior to commencing one-pump operation. The reduction in power with the reduced core flow will cause the APLHGR to reduce accordingly, naturally moving in the direction of the new limit. Specification 3.5.I.3 allows 24 hours before these new limits are required to be implemented.

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-pump 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; during one-pump operation (equalizer valve closed) the respective limit shall be 0.86 times the value shown in these figures. If at any time during operation it is determined that the limiting value for APLHGR 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 (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 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.

Bases 3.11

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 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 $\pm$ 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 Appendix K limit. The limiting value for APLHGR is given by this specification.

Operation with one reactor recirculation pump and the equalizer line closed causes flow to bypass the core through the idle jet pumps. This loss of core flow results in an increased peak clad temperature during a LOCA. The limits on APLHGR are reduced correspondingly. Reference the September 7, 1976 License Amendment Request from L O Mayer (NSP) to Victor Stello (USNRC).

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 APLHGR 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 if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation and axial gaps between core bottom and top and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

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 LHGR limits in such cases need not be reported.