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SUBJECT:
FORWARDING SUPPLEMENT 1 TO LIC NO DPR-22 APPL FOR AMEND: TECH SPEC PROPOSED CHANGE DTD 03/21/78, CONCERNING USE OF A NEW FUEL TYPE AND ALLOW THE APPL OF A SPECIAL ANALYTICAL TECHNIQUE FOR SUBJECT FACILITY"S CORE...NOTARIZED 08/10/78...W/ATT INFO AND ~~NBEO~~ NEDO-24133

SEE RPT FILE

Rpt dtd July 1978

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

August 10, 1978

Director of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

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

Supplement No. 1 to
License Amendment Request Dated March 21, 1978

On March 21, 1978, Northern States Power Company requested a License Amendment to authorize the use of a new fuel type and allow the application of a special analytical technique for the Monticello core. In particular, the Staff was requested to include the application of General Electric retrofit fuel in the Monticello reactor when performing a generic review of Topical Report NEDO-24011, "Licensing Topical Report, General Electric Boiling Water Reactor, Generic Reload Fuel Application", dated May, 1977. The Staff safety evaluation on this Topical Report was subsequently issued on May 12, 1978. Calculations for the reload core were in progress when we filed our March 21 License Amendment Request and certain specific information was identified to be supplied later. The application of retrofit fuel also involved minor editorial changes to the technical specifications as well as newly calculated technical specification limits based on the analytical techniques presented in the referenced topical report. The partial drilled model was not used for the ECCS analysis of the reload core as we originally proposed in our March 21 Amendment Request. The new fuel type was analyzed more conservatively using the currently approved methods. ECCS limits for fuel types currently in use will remain in effect. It is therefore not necessary at this time to make some of the changes proposed in our March 21 License Amendment Request. However, we continue to request that you approve the use of the partial drilled model for Monticello for future use.

NORTHERN STATES POWER COMPANY

For your convenience, we have prepared this supplement to our initial License Amendment Request as an independent package incorporating the additional information and deleting that which is not applicable at this time, so as to fully replace the March 21, 1978 submittal.

We request that our September 30, 1977 License Amendment Request (Revised MAPLHGR Limits) be reviewed and approved along with the Supplemental document. Since the same pages are subject to change under two amendment requests, it would cause confusion to issue one without the other. We see no technical reason for delaying issuance of an amendment in response to the September 30 request; we implemented these changes about a year ago, based on our agreement at that time. A notice for opportunity for public hearing was issued on November 1, 1977, with no response to our knowledge. The mechanics of the changes are as follows:

1. License Change - Make the change to the Provisional Operating License as discussed in Exhibit A of our September 30, 1977 License Amendment Request.

2. Reasons for Technical Specification Changes and Safety Evaluations - These can be found in Exhibit B of our September 30, 1977 License Amendment Request.

3. Technical Specification Page Revisions - Revised Technical Specification pages are attached as Exhibit B to this Supplement No. 1 to License Amendment Request Dated March 21, 1978. The pages are in the form we request them to be issued; that is, all the September 30, 1977 proposed changes have been incorporated along with the presently requested changes.

Attached are forty copies of Exhibits A, B and C. 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. Exhibit C is a safety evaluation supporting the changes. We ask that this request be reviewed and approved prior to our Fall refueling outage which is presently scheduled to commence on October 1, 1978.

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/MHV/deh

cc: J G Keppler
G Charnoff
MPCA
Attn: J W Ferman

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

(Supplement 1 to License Amendment Request Dated March 21, 1978)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A, Exhibit B, and Exhibit C. 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. Exhibit C is a safety evaluation supporting the 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 10th day of August, 1978, 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

EXHIBIT A

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

SUPPLEMENT NO. 1 TO
LICENSE AMENDMENT REQUEST
DATED MARCH 21, 1978

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

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

1. List of Figures and List of Tables

PROPOSED CHANGES

Delete Figures 2.1-1, 2.3.1, 2.3.2, 3.11.1-a through d and 3.11.2 from Pages vii and viii. Add Table 3.11.1 to Page ix. These proposed changes are incorporated in Exhibit B.

REASON FOR CHANGES

The information from the deleted figures is proposed to be incorporated in either the newly created Table or in the Specifications.

SAFETY EVALUATION

These changes are only in format and require no safety evaluation.

2. Power Density Definition

PROPOSED CHANGES

Delete the definition of "Peaking Factor" on Page 3. Add the definition for "Maximum Fraction of Limiting Power Density (MFLPD)" on Page 2. Re-letter the dèfinition on those pages as shown to maintain the alphabetical order.

REASON FOR CHANGES

The concept of peaking factor, as defined, is only meaningful for a core of a single fuel design. The Monticello reactor uses a number of compatible fuel types. The parameter of interest is the relationship between the local power density and the design limit. The concept of peaking factor compares the local surface heat flux to the core average surface heat flux. This involves the heat transfer area which is dependent on the fuel type (Number of fuel pins per assembly, cladding diameter, etc.). The concept of MFLPD is more appropriate for a core of multiple fuel types.

SAFETY EVALUATION

These changes do not affect the safety of the plant, since it simply introduces a new set of units for core monitoring.

3. Fuel Cladding Integrity Safety Limits and Limiting Safety Systems Settings.

PROPOSED CHANGES

Change the MCPR safety limits from 1.06 to 1.07 as shown on Page 6 in Exhibit B.

Change the APRM flux scram and rod block trip settings as shown in Exhibit B, Pages 6, 7 and 8, including the necessary editorial changes on Page 9 to accommodate the overflow information.

Eliminate Figures 2.3.1 and 2.3.2 on Pages 11 and 12.

Make the changes to the Bases shown in Exhibit B, Pages 13, 14, 19, 20 and 21.

REASON FOR CHANGES

The MCPR safety limit change is made necessary by the use of retrofit fuel as discussed in the Safety Evaluation.

The APRM flux scram in rod block trip settings are proposed to be changed for three reasons: First, they implement the definition of MFLPD in place of peaking factor as discussed in Item 2 above. Second, they place the linear equations in the Specification rather than in a Figure for easier reference. Third, the settings are stated in terms of reactor power rather than a heat flux equation which is unique to a fuel type.

Figures 2.3.1 and 2.3.2 are proposed to be deleted because the Specification discussed in the above paragraph incorporates the same information.

The proposed changes to the Bases are to make proper reference to the newly proposed safety limit incorporating the concept of MFLPD in place of peaking factor, and to make correct reference to the Specifications which replace figures.

SAFETY EVALUATION

The Safety Evaluation of the new MCPR safety limit is included as Exhibit C and the Topical Report which Exhibit C supplements, NEDO-24011. The remaining proposed changes are a matter of editing and using different units which require no further safety evaluation.

4. Reactor Fuel Assembly Limiting Condition for Operation.

PROPOSED CHANGES

Revise the APLHGR Specification as shown on Page 189B of Exhibit B to make reference to Table 3.11.1 rather than Figures 3.11.1-a through d.

PROPOSED CHANGES (Continued)

Incorporate the LHGR equation in the Specification as shown on Page 189C of Exhibit B to allow deletion of Figure 3.11.2. Also, the densification penalty at the top of the core has been increased from .021 to .022.

Change the MCPR to the new values supported by this submittal as shown on Page 189D of Exhibit B.

Create a new Table 3.11.1 as shown on Page 189E of Exhibit B, which replaces the existing MAPLHGR figures. This Table adds new fuel types which will be first used in Cycle 7 and eliminates a fuel type which is not expected to be used in the future.

Re-number Pages 189E,F and G to be 189F, G and H as shown in Exhibit B. Insert the newly proposed MCPR operating limit as shown and eliminate the reference to the superseded safety limit.

Delete Figures 3.11.1-a through d and 3.11.2.

REASONS FOR CHANGES

The proposed change to the APLHGR specification is merely a format change which accommodates the use of a Table rather than a Figure. The straight line interpolation referenced is exactly the way data is presently used in deriving the exisiting APLHGR figures.

The proposed change in LHGR is partially editorial to eliminate a figure. The proposed change in the densification penalty is in accordance with the Safety Evaluation.

The operating MCPR limit is proposed to be changed as a result of the transient analysis discussed in the Safety Evaluation. The limit of 1.32 is based on the most limiting abnormal operational transient for Monticello, the turbine trip without bypass. As discussed in Exhibit C, steady state operation with an improperly loaded fuel assembly could, in the worst possible case with a very conservative analytical model, result in a greater delta CPR than a turbine trip without bypass. The May 12, 1978 NRC Staff Safety Evaluation of Topical Report NEDO-24011, "Generic Reload Fuel Application" states that the Staff is still reviewing this matter. When loading fuel in the Monticello reactor, measures are taken to assure that the core is properly loaded, the result being that the plant has never experienced a bundle loading error. We have not seen sufficient evidence to justify a limit so restrictive that full power operation may not be allowed. We therefore propose that in the interim, while the Staff is considering the matter, the licensing basis for Monticello be our past experience and that the turbine trip without bypass remain to be considered the limiting abnormal operational event.

Table 3.11.1 is proposed to be added to simplify the editing when other fuel types are used. It is also more accurate to use the Table rather than reading from a Figure. These numbers are used in a tabular form in the process computer for core monitoring. The Table has been updated to include those fuel types scheduled to be used in the Monticello Cycle 7 core.

The Bases pages are simply revised to reflect editorial changes in numbering, in stating the revised operating MCPR limit and in deleting the superseded MCPR safety limit.

Figures 3.11.1 a through d are proposed to be deleted because they are replaced by Table 3.11.1 and Specification 3.11.A. Figure 3.11.2 is proposed to be deleted with Specification 3.11.B taking its place.

SAFETY EVALUATION

The Safety Evaluation for these proposed changes is Exhibit C and the Topical Report which Exhibit C supplements.

5. Design Features - Reactor Fuel

PROPOSED CHANGES

Modify the description of the reactor core shown as shown on Page 190, Exhibit B to delete the number of fuel rods per assembly.

REASON FOR CHANGE

The 8x8R fuel has 62 fueled rods which is not within the present scope of Specification 5.2.A. This fuel type has been shown to be fully compatible as can be expected for a number of other fuel designs having different numbers of fuel rods per assembly.

SAFETY EVALUATION

The Safety Evaluation for this proposed change is Exhibit C and the Topical Report which Exhibit C supplements. Two comments should be made in addition to Exhibit C regarding the use of 8x8R fuel.

The generic Topical Report on retrofit fuel, NEDO-24011, considers only standard fuel storage racks. Monticello has been authorized to install a High Density Fuel Storage System (HDFSS) which is not discussed in the Topical Report. However, the April 14, 1978 Staff Safety Evaluation authorizes the installation and use of HDFSS at Monticello for any fuel type containing less than 15.2 grams of U-235 in any axial centimeter of the fuel assembly. All fuel types discussed in NEDO-24011 meet the criteria and are therefore acceptable for use in the HDFSS.

The transient analysis in Exhibit C assumes no recirculation pump trip initiated from ATWS sensors. A March 1, 1978 License Amendment Request for Technical Specifications covering operation of this system is presently before the NRC Staff. It is conservative to assume that the recirculation pump trip is not available with respect to over-pressurization events and operating limit MCPR analyses for Monticello; therefore, the Exhibit C analyses are bounding whether the recirculation pump trip is implemented in Cycle 7 or not.

EXHIBIT B

SUPPLEMENT NO. 1 TO
LICENSE AMENDMENT REQUEST
DATED MARCH 21, 1978

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

vii
viii
ix
2
3
6
7
8
9
10 (DELETED)
11 (DELETED)
12 (DELETED)
13
14
19
20
21
189B
189C
189D
189E
189F
189G
189H
189I (DELETED)
189J (DELETED)
189K (DELETED)
189L (DELETED)
190

LIST OF FIGURES

LIST OF FIGURES

LIST OF TABLES

D. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

E. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the primary sensor to verify the proper instrument channel response, alarm, and/or initiating action.

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 safety controlled.

H. 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.

I. Maximum Fraction of Limiting Power Density (MFLPD) - The maximum fraction of limiting power density is the highest value in the core of the ratio of the existing to the design linear heat generation rate.

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 or component shall be considered operable when it is capable of performing its intended function in its required manner.

M. Operating - Operating means that a system or component is performing its required functions in its required manner.

N. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

O. Power Operation - Power Operation is any operation with the mode switch in the "Start-Up" or "Run" position with the reactor ciritical and above 1% rated thermal power.

P. Primary Containment Integrity - Primary Containment Integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied.

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are ot required to be open during accident conditions are closed.

2. At least one door in the airlock is closed and sealed.

3. All automatic containment isolaiton valves are operable or are deactivated in the closed position or at least one valve in each line having an in-operable valve is closed

4. All blind flanges and manways are closed.

Q Protective Instrumentation Logic Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system, a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate a protection action. A trip system may require one or more instrument channel trip signals related to one or more plant parameters to initiate trip system action. Initiation of the protective function may require tripping of a single trip system (e.g., HPCI system isolation, off-gas system isolation, reactor building isolation and standby gas treatment initiation, and rod block), or the coincident tripping of two trip systems (e.g., initiation of scram, reactor isolation, and primary containment isolation).

3. Protective Action - An action initiated by the protection system when a limit is exceeded. A protective action can be at channel or system level.

2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

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 8x8 fuel and less than 1.07 for 8x8R fuel shall constitute violation of the fuel cladding integrity safety limit

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:

$$S \leq 0.65 W + 55\%$$

where,

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

W = recirculation drive flow in percent

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 Psia or Core Flow $\leq$ 10% of Rated)

When the reactor pressure is $\leq$ 800 psia or core flow is $\leq$ 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transients

To insure that the safety limit established in Specification 2.1.A is not exceeded, each required scram shall be initiated by its primary source signal as indicated by the plant process computer

except in the event of operation with a maximum fraction of limiting power density for any fuel type in the core greater than the fraction of rated power, when the setting shall be modified as follows:

$$S \leq (0.65\ W + 55\%)\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power, rated power being 1670 MWt

MFLPD = maximum fraction of limiting power density for any fuel type in the core.

2. IRM - Flux Scram setting shall be $\leq$ 20% of rated neutron flux

B. APRM Rod Block - The APRM rod block setting shall be:

$$S \leq 0.65\ W + 43\%$$

where,

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

W = recirculation drive flow in percent

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core. This level shall be continuously monitored whenever the recirculation pumps are not operating.

except in the event of operation with a maximum fraction of limiting power density for any fuel type in the core greater than the fraction of rated power, when the setting shall be modified as follows:

$$S \leq (0.65\ W + 43\%)\frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power, rated power being 1670 MWt

MFLPD = maximum fraction of limiting power density for any fuel type in the core.

C. Reactor Low Water Level Scram setting shall be $\geq$ 10′6"above the top of the active fuel.

D. Reactor Low Low Water Level ECCS initiation shall be $\geq$ 6′6" $\leq$ 6′10" above the top of the active fuel.

2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS
	E. Turbine Control Valve Fast Closure Scram shall initiate upon loss of pressure at the acceleration relay with turbine first stage pressure $\geq$ 30%. F. Turbine Stop Valve Scram shall be $\leq$ 10% valve closure from full open with turbine first stage pressure $\geq$ 30%. G. Main Steamline Isolation Valve Closure Scram shall be $\leq$ 10% valve closure from full open. H. Main Steamline Pressure initiation of main steamline isolation valve closure shall be $\geq$ 825 psig.

NEXT PAGE IS 13

Bases:

2.1 The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is no less than 1.07. This limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling. (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The concept of MCPR, as used in the GETAB/GEXL critical power analysis, is discussed in Reference 1.

A. Core Thermal Power Limit (Reactor Pressure > 800 psia and Core Flow > 10% of Rated.) Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. The Safety Limit (T.S.2.1.A) has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the Operating MCPR Limit (T.S.3.11.C) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit

Bases Continued:

is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are provided at the beginning of each fuel cycle.

Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confindence that operation of a fuel assembly at the MCPR Safety Limit would not produce boiling transition. Thus, although it is not required to establish the Safety Limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximatley 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Monticello operated above the boiling transition for a significant period of time (30 minutes) without clad perforation.

If reactor pressure should ever exceed 1400 psia during normal power operating (the limit of appalicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

In addition to the MCPR Safety Limit, operation is constrained to a maximum design linear heat generation rate for any fuel type in the core.

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia or Core Flow $\leq$ 10% of Rated) At pressure below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and all core flows, this pressure differential is maintained in the bypass region of the core.

Bases Continued:

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.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams. Therefore, it is intended to ultimately replace (with prior NRC approval) the automatic flow referenced scram with a fixed 120 percent scram setting.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.3.A.1, when the maximum fraction of limiting power density is greater than the fraction of rated power. If the APRM scram 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 scram curve by the reciprocal of the APRM gain change. Analyses of the limiting transients show that no scram adjustment is required to assure that the MCPR Safety Limit (T.S.2.1.A) is not exceeded when the transient is initiated from the Operating MCPR Limit (T.S.3.11.C).

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

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.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be open. Analysis of transients from this operating condition are less severe than the same transients from the two pump operation.

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 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, asuming a steady-state operation at the trip setting, 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 the maximum fraction of limiting power density exceeds the fraction of rated thermal reactor power, 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 stated in Specification 2.3.B. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.B 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

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

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 given in Table 3.11.1 based on a straight line interpolation between data points. When core flow is less than 90% of rated core flow, the APLHGR shall not exceed 95% of the limiting value given in Table 3.11.1. When core flow is less than 70% of rated core flow, the APLHGR shall not exceed 90% of the limiting value given in Table 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

4.11 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 ≥ 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 be limited to:

$$LHGR \leq 3.4(1 - .022\ X/L)$$

where,

X = Elevation from the bottom of the core
L = Fuel Column Length

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 $\geq$ 25% of rated thermal power.

C. Minimum Critical Power Ratio (MCPR)

During power operation, the Operating MCPR Limit shall be ≥ 1.32 for 8x8 fuel and ≥ 1.32 for 8x8R fuel at rated power and flow. 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 (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. For core flows other than rated the Operating MCPR Limit shall be the above applicable MCPR value times K_f where K_f is as shown in Figure 3.11.3.

C. Minimum Critical Power Ratio (MCPR)

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.

TABLE 3.11.1

MAXIMUM AVERAGE PLANAR LINEAR HEAT

GENERATION RATE

vs. EXPOSURE

Exposure	MAPLHGR FOR EACH FUEL TYPE (kw/ft)				
MWD/STU	8DB262	8DB250	8DB219L	8DRB265	8DRB282
200	10.6	10.6	10.7	10.4	10.3
1,000	10.7	10.7	10.7	10.4	10.4
5,000	10.7	10.7	10.8	10.4	10.4
10,000	10.8	10.8	10.7	10.5	10.5
15,000	10.7	10.7	10.7	10.5	10.5
20,000	10.7	10.6	10.6	10.4	10.4
25,000	10.6	10.6	10.6	10.3	10.3
30,000	10.6	10.6	10.2	10.3	10.3

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 $\pm 20^{o}$ 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.

Reference 6 demonstrates that for lower initial core flow rates the potential exists for earlier DNB during postulated LOCA's. Therefore a more restrictive limit for APLHGR is required during reduced flow conditions.

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.

Bases Continued:

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 1.32 for 8x8 fuel and 1.32 for 8x8R 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 (T.S. 2.1.A) 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: VA Moore 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 8x8 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 28, 1977.

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.

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5.0 DESIGN FEATURES

5.1 Site

A. The reactor center line is located at approximately 850,810 feet North and 2,038,920 feet East as determined on the Minnesota State Grid, South Zone. The nearest site boundary is approximately 1630 feet S 30° W of the reactor center line and the exclusion area is defined by the minimum fenced area shown in FSAR Figure 2.2.2a. Due to the prevailing wind pattern, the direction of maximum integrated dosage is SSE. The southern property line follows the northern boundary of the right-of-way for the Burlington Northern Railway.

5.2 Reactor

A. The reactor core shall consist of not more than 484 fuel assemblies.

B. The reactor core shall contain 121 cruciform-shaped control rods. The control rod material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

A. The pressure vessel shall be designed for a pressure of 1250 psig and a temperature of 575°F. The coolant recirculation system shall be designed for a pressure of 1148 psig on suction side of pump and 1248 psig at pump discharge. Both the pressure vessel and recirculation system shall be designed in accordance with the ASME Boiler and Pressure Vessel Code Sections III and IX.

5.4 Containment

A. The primary containment shall be of the pressure suppression type having a drywell and an absorption chamber constructed of steel. The drywell shall have a volume of approximately 134,200 ft^3 and is designed to conform to ASME Boiler and Pressure Vessel Code Section III Class B for an internal pressure of 56 psig at 281°F and an external pressure of 2 psig at 281°F. The absorption chamber shall have a total volume of approximately 176,250 ft^3.

EXHIBIT C

SUPPLEMENT NO. 1 TO

LICENSE AMENDMENT REQUEST

DATED MARCH 21, 1978

This exhibit consists of the report NEDO-24133 entitled, "Supplemental Reload Licensing Submittal for Monticello Nuclear Generating Plant Reload 6".