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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 37
License No. DPR-22

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

   A. The application for amendment by the Northern States Power Company (the licensee), dated March 21, 1978, as supplemented August 10 and September 28, 1978 and applications dated September 30, 1977 and August 16, 1978, comply 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 applications, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Provisional License No. DPR-22 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 37, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 6, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 37

PROVISIONAL OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages. The changed areas on the revised pages are reflected by a marginal line.

i thru vi	
vii	(DELETED)
viii	(DELETED)
ix	(DELETED)
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D. <u>Immediate</u> - 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. <u>Instrument Functional Test</u> - An instrum[illegible] test means the injection of a simulated signal into the primary sensor to verify [illegible]strument channel response, alarm, and/or initiating action.

F. <u>Instrument Calibration</u> - 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. <u>Limiting Conditions for Operation</u> (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. <u>Limiting Safety System Setting</u> (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. <u>Maximum Fraction of Limiting Power Density</u> (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. <u>Minimum Critical Power Ratio</u> (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. <u>Mode</u> - The reactor mode is that which is established by the mode-selector switch.

L. <u>Operable</u> - A system or component shall be considered operable when it is capable of performing its intended function in its required manner.

M. <u>Operating</u> - 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 critical 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 not 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 isolation valves are operable or are deactivated in the closed position or at least one valve in each line having an inoperable 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.

Amendment No. 37

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

2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

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

2.0 SAFETY LIMITS

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.

LIMITING SAFETY SYSTEM SETTINGS

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 steam-line isolation valve closure shall be $\geq$ 825 psig.

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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 oper ion 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 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 confidence 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 $1100^{\circ}F$ 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 operation (the limit of applicability 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.

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

Bases Continued:

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 flo 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, 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 by the in-core LPRM system. When 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 lev l 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

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor vessel pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

2.4 REACTOR COOLANT SYSTEM

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:

A. Reactor Coolant High Pressure Scram shall be $\leq$ 1075 psig.

B. The self-actuation function of at least seven Reactor Coolant System safety relief valves shall be operable. Valves shall be set as follows:

   8 valves at $\leq$ 1108 psig.

Bases Continued:

2.2 The normal operating pressure of the reactor coolant system is approximately 1025 psig. The turbine trip with failure of the bypass system represents the most severe primary system pressure increase resulting from an abnormal operational transient. The peak pressure in this transient is limited to 1207 psig. The safety/relief valves are sized assuming no direct scram during MSIV closure. The only scram assumed is from an indirect means (high flux) and the pressure at the bottom of the vessel is limited to 1248 psig in this case. The analysis assumed that only seven of the eight valves are operable and that they open at 1% over their setpoint with a 0.4 second delay. Reactor pressure is continuously monitored in the control room during operation on a 1500 psig full scale pressure recorder.

Bases:

2.4 The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1670 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure (at the bottom of the pressure vessel) would be about 1248 psig. Only seven of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 1% above their setpoint with a 0.4 second delay.

The operator will set the reactor coolant high pressure scram trip setting at 1075 psig or lower. However, the actual setpoint can be as much as 10 psi above the 1075 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1108 psig or lower. However, the actual set point can be as much as 11.1 psi above the 1108 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18.

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

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Safety/Relief Valves	E. Safety/Relief Valves

3.0 LIMITING CONDITIONS FOR OPERATION

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F.

   a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.

   b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.

4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1. a. A minimum of seven safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The nominal setpoint of all operational safety/relief valves shall be 1108 psig.

   b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such a leakage was comning from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

E. Safety/Relief Valves

Testing of all required safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1108 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the main steam flow stoppage resulting from a postulated MSIV Closure from a power of 1670 Mwt. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 83.2% of the full power steam generation rate.

3.0 LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value 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

4.0 SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours.

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR as a function of core height shall be limited to:

$LHGR \leq 13.4(1 - .022\ X/L)$

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)

1. During power operation, the Operating MCPR Limit shall be $\geq$1.33 for 8x8 fuel and $\geq$1.33 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.

2. If the gross radioactivity release rate of noble gases at the steam jet air ejector monitors exceeds, for a period greater than 15 minutes, the equivalent of 236,000 uCi/sec following a 30-minute decay, the Operating MCPR Limits specified in 3.11.C.1 shall be adjusted to $\geq$ 1.45 for 8x8 fuel and $\geq$ 1.40 for 8x8R fuel, times the appropriate K_f. Subsequent operation with the adjusted MCPR values shall be per paragraph 3.11.C.1.

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

Amendment Nos. ~~20~~,~~27~~, 37

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 $\pm 20^{c}$ 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 1.33 for 8x8 fuel and 1.33 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.

Noble gas activity levels above that stated in 3.11.C.1 are indicative of fuel failure. Since the failure mode cannot be positively identified, a more conservative Operating MCPR Limit must be applied to account for a possible fuel loading error.

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.