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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated May 30, 1984, as supplemented May 31, 1984, September 6, 1984, and October 17, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   2 Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 29 , 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 16, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Revise the Appendix A Technical Specifications by removing the current pages and inserting the revised pages listed below. The revised areas are identified by vertical lines.

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Amendment No. 29

INTRODUCTION

These Technical Specifications are prepared in accordance with the requirements of 10CFR50.36 and apply to the Monticello Nuclear Generating Plant, Unit No. 1. The bases for these Specifications are included for information and understandability purposes.

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the Specifications may be achieved.

A. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. (Normal operating functions such as control rod movement using the normal drive mechanism, tip scans, SRM and IRM detector movements, etc., are not to be considered core alterations.)

B. Hot Standby - Hot Standby means operation with the reactor critical in the startup mode at a power level just sufficient to maintain reactor pressure and temperature.

C. Fire Suppression Water System - The fire suppression water system consists of: water sources; pumps; and distribution piping with associated sectionalizing isolation valves. Such valves include yard hydrant valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser.

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 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 Control Rod Pattern (LCRP) - A limiting control rod pattern for rod withdrawal error (RWE) exists when: a) Thermal power is below 90% of rated and the MCPR is less than 1.70 or b) Thermal power is 90% of rated or above and the MCPR is less than 1.40.

I. 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.

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, subsystem, train, component or device shall be Operable or have Operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

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 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:

$S \leq 0.58\, W + 62\%$

where,

S = Setting in 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.

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

B. (DELETED)

C. Reactor Low Water Level Scram setting shall be 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

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.

E. Turbine Control Valve Fast Closure Scram shall initiate upon loss of presure 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 intiation of main steamline isolation valve closure shall be $\geq$ 825 psig.

(next page - 10)

Bases:

2.3 The abnormal operational transients applicable to operation of the Monticello Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power level of 1670 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3-1 of Reference 2. The licensed maximum power level 1670 MWt represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. This transient model, evolved over many years, has been substantiated in operation as a conservative tool for evaluating reactor dynamic performance. Results obtained from a General Electric boiling water reactor have been compared with predictions made by the model. The comparisons and results are summarized in Reference 1.

The absolute value of the void reactivity coefficient used in the analysis is conservatively estimated to be about 25% greater than the nominal maximum value expected to occur during the core lifetime. The Doppler reactivity feedback coefficient has conservatively been derated to 90% of the expected value. The scram worth used has been derated to be equivalent to approximately 80% of the total scram worth of the control rods. The scram delay time and rate of rod insertion assumed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by Technical Specificaitons. The effect of scram worth, scram delay time and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. The early portion of the scram stroke accomplishes the desired effect by inserting sufficient negative reactivity to turn the transient around. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

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.

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 operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

B. Deleted

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 39.

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 meeting their criterion. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

The operator will set the low low water level ECCS initiation trip setting ≥6'6" ≤6'10" above the top of the active fuel. However, the actual setpoint can be as much as 3 inches lower than the 6'6" setpoint and 3 inches greater than the 6'10" setpoint due to the deviations discussed on page 39.

E. Turbine Control Valve Fast Closure Scram The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% turbine first stage pressure. This takes into account the possibility of 15% power being passed directly to the condenser through the bypass valves.

F. Turbine Stop Valve Scram The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the Safety Limit (T.S.2.1.A) even during the worst case transient that assumes the turbine bypass is closed. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% turbine first stage pressure. This takes into account the possibility of 15% power being passed directly to the condenser through the bypass valves.

G. Main Steam Line Isolation Valve Closure Scram The main steam line isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation closure. With the scram set at 10% valve closure there is no increase in neutron flux.

H. Main Steam Line Low Pressure Initiates Main Steam Isolation Valve Closure The low pressure isolation of the main steam lines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation at steamline pressures lower than 825 psig requires

Bases Continued:

that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of the neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

The operator will set this pressure trip at greater than or equal to 825 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 39.

References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO-10802, Feb., 1973.

2. "Average Power Range Monitor, Rod Block Monitor and Technical Specifications Improvement (ARTS) Program for Monticello Nuclear Generating Plant", NEDC-30492-P, April, 1984.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

B. Upon discovery that the requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated to:

1. Satisfy the minimum requirements by placing appropriate devices, channels, or trip systems in the tripped condition, or

2. Place and maintain the plant under the specified required conditions using normal operating procedures.

C. RPS Power Monitoring System

1. Except as specified below, both channels of the power monitoring system for the MG set or alternate source supplying each reactor protection system bus shall be operable with the following setpoints:

			Time Delay
a. Over-voltage	-	$\leq$128 VAC	$\leq$4 seconds
b. Under-voltage	-	$\geq$104 VAC	$\leq$4 seconds
c. Under-frequency	-	$\geq$57 HZ	$\leq$4 seconds

2. With one RPS electric power monitoring channels for the MG set or alternate source supplying each reactor protection system bus inoperable, restore the inoperable channel to Operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

3. With both RPS electric power monitoring channels for the MG set or alternate source supplying each reactor protection system bus inoperable, restore at least one to Operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

B. (DELETED)

C. RPS Power Monitoring System

1. Instrument Functional Tests of each RPS power monitoring channel shall be performed at least once every six months.

2. At least once each Operating Cycle an Instrument Calibration of each RPS power monitoring channel shall be performed to verify over-voltage, under-voltage, and under-frequency setpoints.

B. Emergency Core Cooling Subsystems Actuation

When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling subsystems are given in Table 3.2.2.

C. Control Rod Block Actuation

1. SRM, IRM, APRM and Scram Discharge Volume Rod Blocks

   The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.

2. Rod Block Monitor (RBM)

   a. When core thermal power is greater than or equal to 30% of rated and a Limiting Control Rod Pattern exists, either:

      (1) Both RBM channels shall be operable, or

      (2) With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, or

      (3) With both RBM channels inoperable, control rod withdrawal shall be blocked immediately.

C. Control Rod Block Actuation.

During operation requiring RBM operability when only one channel is operable, an instrument functional test of the operable RBM shall be performed within 24 hours prior to withdrawal of control rod(s).

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. Rod Block Monitor (RBM) (continued) b. RBM Setpoints for control rod block are given in Table 3.2.3. The upscale LTSP shall be applied above 30% and up to 65% of rated thermal power. The upscale ITSP shall be applied at and above 65% and up to 85% of rated thermal power. The upscale HTSP shall be applied at and above 85% of rated thermal power. The RBM Bypass time delay shall be less than or equal to 2.0 seconds. D. (Deleted)	

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**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System	Required Conditions*
		Refuel	Startup	Run			
1. SRM							
a. Upscale	$\leq 5 \times 10^5$ cps	X	X(d)		2	1 (Note 1, 3, 6)	A or B or C
b. Detector not fully inserted		X(a)	X(a)		2	1 (Note 1, 3, 6)	A or B or C
2. IRM							
a. Downscale	$\geq$[illegible]/125 full scale	X(b)	X(b)		4	2 (Note 1, 4, 6)	A or B or C
b. Upscale	$\leq$108/125 full scale	X	X		4	2 (Note 1, 4, 6)	A or B or C
3. APRM							
a. Upscale (flow referenced)	$\leq$.58W + 50% (Note 2)			X	3	1 (Note 1, 6, 7)	D or E
b. Downscale	$\geq$3/125 full scale			X	3	1 (Note 1, 6, 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	Required Conditions*
4. RBM							
a. Upscale (power referenced) (Note 8)		See Section 3.2.C.2			1	See Section 3.2.C.2 (note 5)	See Section 3.2.C.2
1. Low Trip Setpoint (LTSP)	≤115/125 of full scale						
2. Intermediate Trip Setpoint (ITSP)	≤109/125 of full scale						
3. High Trip Setpoint (HTSP)	≤105/125 of full scale						
b. Downscale	≥94/125 of full scale	See Section 3.2.C.2			1	See Section 3.2.C.2 (note 5)	See Section 3.2.C.2
5. Scram Discharge Volume							
Water Level - High							
a. East	≤40 gal		X	X	1	1 (note 6)	B and D, or A
b. West	≤40 gal		X	X	1	1 (note 6)	B and D, or A

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) An RBM channel will be considered inoperable if there are less than half the total number of normal inputs.

(6) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied actions shall be initiated to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(b) Place the plant under the specified required conditions using normal operating procedures.

(7) There must be a total of at least 4 operable or operating APRM channels.

(8) There are 3 upscale trip levels. Only one is applied over a specified operating core thermal power range. All RBM trips are automatically bypassed below 30% thermal power.

Table 3.2.3 - Continued
Instrumentation That Initiates Rod Block

Notes:

*Required conditions when minimum conditions for operation are not satisfied.

A. Reactor in Shutdown mode.

B. No rod withdrawals permitted while in Refuel or Startup mode.

C. Reactor in Run mode.

D. No rod withdrawals permitted while in the Run mode.

E. Power on IRM range or below and reactor in Startup, Refuel, or Shutdown mode.

**Allowable Bypass Conditions

a. SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel count rate is 100 cps or when all IRM range switches are above Position 2.

b. IRM Downscale rod block may be bypassed when the IRM range switch is in the lowest range position.

c. (deleted)

d. SRM Upscale block may be bypassed when associated IRM range switches are above Postion 6.

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°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 operation significantly above the licensing basis power level 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 operator will set the APRM rod block trip settings no greater than that stated in Table 3.2.3. However, the actual setpoint can be as much as 3% greater than that stated in Table 3.2.3 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

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 power. This power signal is provided by the APRMs. A statistical analysis of many single control rod withdrawal errors has been performed and at the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs greater than the Safety Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR of 1.20 prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. With one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked (Reference 1). Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

Bases Continued:

3.2 The RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.

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.

For effective emergency core cooling for the small pipe break the HPCI or Automatic Pressure Relief system must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria is met. Reference Section 6.2.4 and 6.2.6 FSAR. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors measure radioactivity of the reactor building ventilation exhaust and on the refueling floor. Any one upscale trip will cause the desired action. Trip settings for the ventilation exhaust isolation monitors are based upon initiating normal ventilation isolation and Standby Gas Treatment System operation prior to exceeding the maximum release rate limit for the reactor building vent. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The recirculation pump trip description and performance analysis is discussed in Topical Report NEDO-25016, September 1976, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant". (See September 15, 1976 letter from Mr L O Mayer, NSP, to Mr D L Ziemann, USNRC.) The pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction

3.2 increases core voiding, a negative reactivity feedback. High pressure sensors initiate the pump trip in the event of an isolation transient. Low level sensors initiate the trip on loss of feedwater (and the resulting MSIV closure). The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated after reactor isolation in this unlikely postulated event, requiring the trip to be operable only when in the RUN mode is therefore conservative.

Although the operator will set the set points within the trip settings specified in Tables 3.2.1, 3.2.2, 3.2.3, 3.2.4, and 3.2.5, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, these deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts.

References:

1. "Average Power Range Monitor, Rod Block Monitor and Technical Specifications Improvement (ARTS) Program for Monticello Nuclear Generating Plant", NEDC-30492-P, April, 1984.

	Trip Function	Deviation
Instrumentation That Initiates Emergency Core Cooling Systems Table 3.2.2	Low-Low Reactor Water Level	-3 Inches
	Reactor Low Pressure (Pump Start) Permissive	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Pressure (Valve Permissive)	-10 psi
Instrumentation That Initiates Rod Block Table 3.2.3	IRM Downscale	-2/125 of Scale
	IRM Upscale	+2/125 of Scale
	APRM Downscale	-2/125 of Scale
	APRM Upscale	See Basis 3.2
	RBM Downscale	-2/125 of Scale
	RBM Upscale	+2/125 of Scale
	Scram Discharge Volume-High Level	+ 1 gallon
Instrumentation That Initiates Recirculation Pump Trip	High Reactor Pressure	+ 12 psi
	Low Reactor Water Level	-3 Inches

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip settings, or, when a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable or when actions specified are not initiated as specified.

(b) Whenever the reactor is in the startup or run mode below 10% rated thermal power, no control rods shall be moved unless the rod worth minimizer is operable or a second independent operator or engineer verifies that the operator at the reactor console is following the control rod program. The second operator may be used as a substitute for an inoperable rod worth minimizer during a startup only if the rod worth minimizer fails after withdrawal of at least twelve control rods.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

(iv) The rod block function of the rod worth minimizer shall be verified by attempting to withdraw an out-of-sequence control rod beyond the block point.

(b) If the rod worth minimizer is inoperable while the reactor is in the startup or run mode below 10% rated thermal power and the second independent operator or engineer is being used, he shall verify that all rod positions are correct prior to commencing withdrawal or insertion of each rod group.

4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have an observed count rate of at least three counts per second.

consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit (T.S.2.1.A). This requires the negative reactivity insertion in any local region of the core and in the overall core to be equivalent to at least the scram reactivity curve used in the transient analysis. The required average scram times for three control rods in all two by two arrays and the required average scram times for all control rods are based on inserting this amount of negative reactivity at the specified rate locally and in the overall core. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram. The limiting operational transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains above the Safety Limit (T.S.2.1.A).

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 all core locations shall not exceed the appropriate APLHGR limit for those core locations. The APLHGR limit, which is a function of average planar exposure and fuel type, is the appropriate value from Table 3.11.1 (based on a straight line interpolation between data points), multiplied by the smaller of the two MAPFAC factors determined from Figure 3-3 and 3-5 of Reference 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.

C. Minimum Critical Power Ratio (MCPR)

During power operation, the MCPR shall be equal to or greater than the operating limit MCPR (OLMCPR) which is a function of scram time, core power, core flow, and fuel type. The OLMCPR is the greater of:

- the applicable limit determined from Figure 3-4 of Reference 1 or:

1. Thermal power greater than 45% - The applicable limit from Table 3.11.2 multiplied by the K_p factor from Figure 3-2 of Reference 1.

2. Thermal power equal to or less than 45% - The applicable limit from Figure 3-2 of Reference 1.

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 operations 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.

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 3.11.2

Minimum Critical Power Ratio vs Fuel Type			
Fuel Type	Average Scram Insertion Time (τ_{ave})		
	$MCPR_B$ $\tau_{ave} \leq \tau_B$	$\tau_B < \tau_{ave} < 0.9$ sec	$MCPR_A$ $\tau_{ave} = 0.9$ sec
8X8	1.35	*	1.43
P8X8R	1.38	*	1.45

* A linear interpolation between $MCPR_B$ and $MCPR_A$

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 ±20° 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 at rated conditions conform to 10CFR50.46. The limiting value for APLHGR is given by this specification.

The flow dependent correction factor (Figure 3-5, Reference 1) applied to the rated condition's APLHGR limits assures that 1) the 2200°F PCT limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and 2) the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from less than rated core flow conditions. The power dependent correction factor (Figure 3-3, Reference 1) applied to the rated conditions APLHGR limits assures that the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from all conditions.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of 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.

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.

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 and Reference 6 assumed the steady state MCPR prior to the postulated loss-of-coolant accident to be 1.28 for all fuel types for rated flow. The Operating

MCPR Limit 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.

Use of GE's new ODYN code Option B will require average scram time to be a factor in determining the MCPR (Reference 7). In order to increase the operating envelope for MCPR below $MCPR_A$ (ODYN code Option A), the cycle average scram time (τave) must be determined (see Bases 3.3.C). If τave is below the adjusted analysis scram time, the $MCPR_B$ Limit can be used. If $\tau ave > \tau_B$ a linear interpolation must be used to determine the appropriate MCPR. For example:

$$MCPR = MCPR_B + \frac{\tau ave - \tau_B}{0.9 - \tau_B}(MCPR_A - MCPR_B)$$

$MCPR_A$ and $MCPR_B$ have been determined from the most limiting abnormal operational transients analyses.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Section 14.5 that are input to a GE-core dynamic behavior transient computer program described in References 2 and 3.

At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_E$ and $MCPR_P$ at the existing core flow and power state. The required MCPR is a function of flow in order to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The MCPRs were calculated such that for the maximum core flow rate and the corresponding thermal power along the 105% of rated power/flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined in Figure 3-4 of Reference 1.

For operation above 45% of rated thermal power, the core power dependent MCPR operating limit is the rated power-rated flow MCPR operating limit, multiplied by the factor given in Figure 3-2 of Reference 1. For operation below 45% of rated thermal power (turbine control valve fast closure and turbine stop valve closure scrams can be bypassed) absolute MCPR limits are established and the limit is taken directly from Figure 3-2 of Reference 1. This protects the core from plant transients other than core flow increase, including a localized event such as rod withdrawal error.

Bases Continued

This limit was determined based upon bounding analyses for the most limiting transient at the given core power level. Further information on MCPR operating limits for off-rated conditions is presented in NEDC-30492-P.(1)

At thermal power levels less than or equal to 25% of rated thermal power, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. MCPR evaluation below this power level is therefore unnecessary. The daily requirement for calculating MCPR above 25% of rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

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. "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) program for Monticello Nuclear Generating Plant", NEDC-30492-P, April, 1984.

2. "Analytical Methods of Plant Transient Evaluations for the GE BWR", NEDO-10802, February, 1973.

3. "Response to NRC Request for Information on ODYN Computer Code", R H Buchholz to P S Check (USNRC), September 28, 1977.

4. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", NEDE-20566, November 1975.

5. "Revision of Low Core Flow Effects on LOCA Analysis for Operating BWRs", R L Gridley (GE) to D G Eisenhut (USNRC), September 28, 1977.

6. "Loss-of-Coolant Accident Analysis Report for the Monticello Nuclear Generating Plant", NEDO-24050-1, December, 1980, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (USNRC), February 6, 1981.

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.