UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 102
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 July 26, 1996, as supplemented September 5, 1997, as revised December 4, 1997, and as supplemented March 6, March 26, April 8, April 17, April 22, May 5, May 12, May 29, June 15, July 1, July 20, and July 30, 1998, 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 paragraphs 2.C.1, 2.C.2, and 2.C.8 of Facility Operating License No. DPR-22 are hereby amended to read as follows:

C.1 Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1775 megawatts (thermal).

C.2 Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 102, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C.8 Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 102, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of issuance with full implementation within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Page 3 of License No. DPR-22*
2. Changes to the Technical Specifications
3. Pages C-3 and C-4 of Appendix C

Date of Issuance: September 16, 1998

*Page 3 is attached, for convenience, for the composite license to reflect these changes.

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following page of Operating License DPR-22 with the enclosed page.

Remove	Insert
3	3

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
ii	ii
iii	iii
4	4
6	6
14	14
15	15
16	16
19	19
23	23
24	24
28	28
30	30
37	37
38	38
39	39
49	49
50	50
53	53
56	56
64	64
65	65
68	68
104	104
112	112
114	114
150	150

Revise Appendix C - Additional Conditions by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
C-2	C-2
--	C-3
--	C-4

4. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

5. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 1775 megawatts (thermal)

2. Technical Specifications

The Technical Specifications contained in Appendix A,as revised through Amendment No.102,are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Monticello Nuclear Generating Plant Physical Security Plan," with revisions submitted through November 30, 1987; "Monticello Nuclear Generating Plant Guard Training and Qualification Plan," with revisions submitted through February 26, 1986; and "Monticello Nuclear Generating Plant Safeguards Contingency Plan," with revisions submitted through August 20, 1980. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

R. Rated Neutron Flux - Rated flux is the neutron flux that corresponds to a steady-state power level of 1775 thermal megawatts.

S. Rated Thermal Power - Rated thermal power means a steady-state power level of 1775 thermal megawatts.

T. Reactor Coolant System Pressure or Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those existing in the vessel steam space.

U. Refueling Operation and Refueling Outage - Refueling Operation is any operation when the reactor water temperature is less than 212°F and movement of fuel or core components is in progress. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

V. Safety Limit - The safety limits are limits below which the maintenance of the cladding and primary system integrity are assured. Exceeding such a limit is cause for plant shutdown and review by the Commission before resumption of plant operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

W. Secondary Containment Integrity - Secondary Containment Integrity means that the reactor building is closed and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All reactor building ventilation system automatic isolation valves are operable or are secured in the closed position.

X. Sensor Check - A qualitative determination of operability by observation of sensor behavior during operation. This determination shall include, where possible, comparison with other independent sensors measuring the same variable.

2.0 SAFETY LIMITS

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

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

Specification:

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

When the reactor pressure is >800 psia and core flow is >10% of rated, the existence of a minimum critical power ratio (MCPR) less than 1.10*, for two recirculation loop operation, or less than 1.11* for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

* MCPR values are for cycle 19 only.

LIMITING SAFETY SYSTEM SETTINGS

2.3 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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

Specification:

The Limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - The APRM flux scram trip setting shall be:

a. For two recirculation loop operation (TLO):

$$S \leq 0.66W + 65.6\%$$

where

S = Setting in percent of rated thermal power, rated power being 1775 MWt

W = Percent of recirculation drive flow required to produce a core flow of 57.6×10^6 lb/hr

b. For single recirculation loop operation (SLO):

$$S \leq 0.66(W - 5.4) + 65.6\%$$

c. No greater than 120%.

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 1775 MWt. The analyses were based upon plant operation in accordance with the operating map. The licensed maximum power level 1775 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 tranisent results as determined by the current analysis model. Conservatism incorporated into the transient analysis is documented in Reference 1.

Bases Continued:

For analyses of the thermal consequences of the transients, the Operating MCPR Limit (T.S.3.11.C) is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

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

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

A. Neutron Flux Scram The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1775 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Also, the flow biased neutron flux scram (specification 2.3.A.1) provides protection to the fuel safety limit in the unlikely event of a thermal-hydraulic instability.

Bases Continued:

Maximum Extended Load Line Limit Analyses (MELLLA) have been performed to allow operation at higher powers at flows below 87%. The flow referenced scram (and rod block line) have increased (higher slope and y-intercept) for two loop operation (See Core Operating Limits Report). The supporting analyses are discussed in GE NEDC-31849P report (Reference: Letter from NSP to NRC dated September 16, 1992).

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

Evaluations discussed in NEDC-32546P, July 1996, demonstrated the acceptability of MELLLA and ICF for rerate conditions. In addition, the evaluation demonstrated the acceptability of MELLLA for single loop operation.

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 20% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position and the associated APRM is not downscale. This switch occurs when reactor pressure is greater than 850 psig.

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

B. Deleted

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% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% 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% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% 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:

2.2 The normal operating pressure of the reactor coolant system is approximately 1010 psig. Evaluations have determined that the most severe pressure transient is bounded by the closure of all MSIVs, followed by a reactor scram on high neutron flux (failure of the direct scram associated with MSIV position is assumed). The USAR discusses the analysis of this event. The analysis results demonstrate the safety/relief valve capacity is capable of maintaining pressure below the ASME Code limit of 110% of vessel design pressure (110% X 1250 psig = 1375 psig). The safety limit ensures that the acceptance limit of 1375 psig is met during the design basis event at the vessel location with the highest pressure.

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 Editi , the safety/relief valves must be set to open at a pressure no higher than 105 percent of de pressure, with at least one safety/relief valve set to open at a pressure no greater t . 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 1775 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 remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief valve setpoint is established by the design pressure of the HPCI and RCIC systems. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low-Set SRV system to well below these values.

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 3.1. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1120 psig (1109 psig + 1%) or lower. However, the as-found set point can be as much as 22.3 psi above the 1120 psig indicated set point due to the deviations discussed in the basis of Specification 3.6.

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

TABLE 3.1.1
REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating** Refuel (3)	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Condition*
1. Mode Switch in Shutdown		X	X	X	1	1	A
2. Manual Scram		X	X	X	1	1	A
3. Neutron Flux IRM (See Note 2) a. High-High b. Inoperative	≤ 120/125 of full scale	X	X		4	3	A
4. Flow Referenced Neutron Flux APRM (See Note 5) a. High-High b. Inoperative	See Specifications 2.3A.1			X	3	2	A or B
c. High Flow Clamp	≤ 120 %						
5. High Reactor Pressure (See Note 9)	≤ 1075 psig	X	X(f)	X(f)	2	2	A
6. High Drywell Pressure (See Note 4)	≤ 2 psig	X	X(e,f)	X(e,f)	2	2	A
7. Reactor Low Water Level	≥ 7 in.(annulus)	X	X(f)	X(f)	2	2	A
8. Scram Discharge Volume High Level							
a. East	≤ 56 gal.(8)	X(a)	X(f)	X(f)	2	2	A
b. West	≤ 56 gal.(8)	X(a)	X(f)	X(f)	2	2	A
9. Turbine Condenser Low Vacuum	≥ 22 in. Hg	X(b)	X(b,f)	X(f)	2	2	A or C

Table 3.1.1 - Continued

6. Deleted.

7. Trips upon loss of oil pressure to the acceleration relay.

8. Limited trip setting refers to the volume of water in the discharge volume receiver tank and does not include the volume in the lines to the level switches.

9. High reactor pressure is not required to be operable when the reactor vessel head is unbolted.

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

A. All operable control rods fully inserted within 8 hours.

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

C. Reactor in Startup or Refuel mode and pressure below 600 psig.

D. Reactor power less than 45% (798.75 MWt.).

** Allowable Bypass Conditions

It is permissible to bypass:

a. The scram discharge volume High Water Level scram function in the refuel mode to allow reactor protection system reset. A rod block shall be applied while the bypass is in effect.

b. The Low Condenser vacuum and MSIV closure scram function in the Refuel and Startup modes if reactor pressure is below 600 psig.

c. Deleted.

d. The turbine stop valve closure and fast control valve closure scram functions when the reactor thermal power is ≤ 45% (798.75 MWt).

Bases Continued:

3.1 condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass. Reference FSAR Section 14.5.1.2.2 and supplemental information submitted February 13, 1973. The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 22" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

The main steamline isolation valve closure scram is set to scram when the isolation valves are ≤10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant. Reference Section 14.5.1.3.1 FSAR and supplemental information submitted February 13, 1973.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.7.1 FSAR.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against excessive power levels and short reactor periods in the

Bases Continued:

3.1 start-up and intermediate power ranges. Ref. Section 7.4.4 FSAR. A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions. Ref. Section 7.4.3 FSAR. Thus, the IRM is required in the "Refuel" and "Startup" modes. In the power range the APRM system provides required protection. Ref. Section 7.4.5.2 FSAR. Thus, the IRM system is not required in the "Run" mode. The APRM's cover only the power range, the IRM's provide adequate coverage in the start-up and intermediate range, and therefore, the APRM's are not required for the "Refuel" or "Startup" modes.

The high reactor pressure, high drywell pressure, and reactor low water level scrams are required for all modes of plant operation unless the reactor is subcritical and depressurized. They are, therefore, required to be operational for all modes of reactor operation except in the "Refuel" mode with the reactor subcritical and reactor temperature less than 212°F as allowed by Note 3.

The scram discharge volume high level trip function is required for all modes with the exception that it may be bypassed in the "Refuel Mode" under the provisions of Table 3.1.1, allowable by-pass condition (a). In order to reset the safety system after a scram condition, it is necessary to drain the scram discharge volume to clear this scram input condition. This condition usually follows any scram, no matter what the initial cause might have been. Since all of the control rods are completely inserted following a scram it is permissible to bypass this condition because a control rod block prevents withdrawal as long as the switch is in the bypass condition for this function.

To permit plant operation to generate adequate steam and pressure to establish turbine seals and condenser vacuum at relatively low reactor power, the main condenser vacuum trip is bypassed until 600 psig. This bypass also applies to the main steam isolation valves for the same reason. Ref. Section 7.7.1.2 FSAR.

An automatic bypass of the turbine control valve fast closure scram and turbine stop closure scram is effective below 30% thermal power as indicated by turbine first stage pressure. This insures that reactor thermal power is less than 45% of its rated value. Closure of these valves from such a low initial power level does not constitute a threat to the integrity of any barrier to the release of radioactive material.

Bases Continued:

3.1 The IRMs are calibrated by the heat balance method such that 120/125 of full scale on the highest IRM range is below 20% of rated neutron flux (see Specification 2.3.A.2). The requirement that the IRM detectors be inserted in the core assures that the heat balance calibration is not invalidated by the withdrawal of the detector.

Although the operator will set the set points within the trip settings specified on Table 3.1.1, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. For power rerate, GE setpoint methodology provided in NEDC 31336, "General Electric Setpoint Methodology," is used in establishing setpoints. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, such deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts:

Trip Function	Deviation	Trip Function	Deviation
3. High Flux IRM	+2/125 of scale	*7. Reactor Low Water Level	-6 inches
5. High Reactor Pressure	+10 psi	8. Scram Discharge Volume High Level	+1 gallon
6. High Drywell Pressure	+1 psi	9. Turbine Condenser Low Vacuum	-1/2 in. Hg

* This indication is reactor coolant temperature sensitive. The calibration is thus made for rated conditions. The level error at low pressures and temperatures is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not the indicated level.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or 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 the actions specified in 3.1.B are not initiated as specified.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criterion. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable

Table 3.2.1
Instrumentation That Initiates Primary Containment Isolation Functions

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2)	Required Conditions*
1. Main Steam and Recirc Sample Lines (Group 1)				
a. Low Low Reactor Water Level	≥6'-6" ≤6'10"	2	2	A
b. High Flow in Main Steam Line	≤140% rated	8	8	A
c. High temp. in Main Steam Line Tunnel	≤200°F	8	2 of 4 in each of 2 sets	A
d. Low Pressure in Main Steam Line (3)	≥825 psig	2	2	B
2. RHR System, Head Cooling, Drywell, Sump, TIP (Group 2)				
a. Low Reactor Water Level	≥ 7" (annulus)	2	2	C

TABLE 3.2.1 - Continued

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2)	Required Conditions
b. High Drywell Pressure (5)	≤2 psig	2	2	D
3. Reactor Cleanup System (Group 3)				
a. Low Reactor Water Level	≥7"(annulus)	2	2	E
b. High Drywell Pressure	≤2 psig	2	2	E
4. HPCI Steam Lines				
a. HPCI High Steam Flow	≤150,000 lb/hr with ≤60 second time delay	2(4)	2	F
b. HPCI High Steam Flow	≤300,000 lb/hr	2(4)	2	F
c. HPCI Steam Line Area High Temp.	≤200°F	16(4)	16	F
5. RCIC Steam Lines				
a. RCIC High Steam Flow	≤45,000 lb/hr with 5 ± 2 sec time delay	2(4)	2	G
b. RCIC Steam Line Area	≤200°F	16(4)	16	G
6. Shutdown Cooling Supply Isolation				
a. Reactor Pressure Interlock	≤75 psig at the reactor steam dome	2(4)	2	C

Table 3.2.2
Instrumentation That Initiates Emergency Core Cooling Systems

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (3)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (3)	Required Conditions*
B. HPCI System					
1. High Drywell Pressure (1)	≤2 psig	1	4	4	A.
2. Low-Low Reactor Water Level	≥6'6"≤6'10"	1	4	4	A.
C. Automatic Depressurization					
1. Low-Low Reactor Water Level and	≥6'6"≤6'10"	2	2	2	B.
2. Auto Blowdown Timer and	≤120 seconds	2	1	1	B.
3. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	≥60 psig ≤150 psig	2	12(4)	12(4)	B.

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TABLE 3.2.3

Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes Which Function Must be Operable or Operating and Allowable Bypass Conditions** Refuel	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels per Trip System	Required Conditions*
1. SRM							
a. Upscale	≤5x10^5 cps	X	X(d)		2	1(Note 1, 3, 6)	A or B or C
b. Detector not fully inserted		X(a)	X(a)		2	1(Note 1, 3, 6)	A or B or C
2. IRM							
a. Downscale	≥3/125 full scale	X(b)	X(b)		4	2(Note 1, 4, 6)	A or B or C
b. Upscale	≤108/125 full scale	X	X		4	2(Note 1, 4, 6)	A or B or C
3. APRM							
a. Upscale (1) TLO Flow Biased	≤ 0.66W + 53.6% (Note 2)			X	3	1(Note 1, 6, 7)	D or E
(2) SLO Flow Biased	≤ 0.66(W - 5.4) + 53.6% (Note 2)						
(3) High Flow Clamp	≤ 108%						
b. Downscale	≥ 3/125 full scale			X	3	1(Note 1, 6, 7)	D or E

Bases:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminate a single operator error before it results in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, and other safety related functions. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 7" on the instrument. This corresponds to a lower water level above the top of active fuel at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected transient analysis. This trip initiates closure of group 2, and 3 primary containment isolation valves. Reference Section 7.7.2.2 FSAR. The trip setting provides assurance that the valves will be closed before perforation of the clad occurs even for the maximum break in that line and therefore the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is 6'6" above the top of the active fuel. This trip initiates closure of the Group 1 Primary containment isolation valves, Reference Section 7.7.2.2 FSAR, and also activates the ECC systems and starts the emergency diesel generator.

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Bases Continued:

3.2 This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Section 6.2.7 and 14.6.3 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Reference Section 6.2.7 FSAR.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 and Group 3 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 and Group 3 isolation valves include the drywell vent, purge, sump isolation, and RWCU system valves.

Two pressure switches are provided on the discharge of each of the two core spray pumps and each of the four RHR pumps. Two trip systems are provided in the control logic such that either trip system can permit automatic depressurization. Each trip system consists of two trip logic channels such that both trip logic channels are required to permit a system trip.

Division I core spray and RHR pump discharge pressure permissives will interlock one trip system and Division II permissives will interlock the other trip system. One pressure switch on each pump will interlock one of the trip channels and the other pressure switch will interlock the other trip channel within their respective trip system.

The pump pressure permissive control logic is designed such that no single failure (short or open circuit) will prevent auto-blowdown or allow auto-blowdown when not required. The trip setting for the low pressure ECCS pump permissive for ADS is set such that it is less than the pump discharge pressure when a pump is operating in a full flow condition and also high enough to avoid any condition that results in a discharge pressure permissive when the pumps are not operating.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow,

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 rod blocks assure that there will be proper overlap between the neutron monitoring systems and thus, that adequate coverage is provided for all ranges of reactor operation. 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 (two reactor building vent plenum and two refueling floor) are provided which initiate isolation of the reactor building and operation of the standby gas treatment system following a refueling accident. The monitors measure radioactivity in the reactor building ventilation exhaust and on the refueling floor. One upscale trip signal or two downscale/inoperable trip signals, from a pair of monitors performing the same function, will cause the desired action. Trip settings of 100 mR/hr for the reactor building vent plenum monitors and the refueling floor monitors 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 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

3.0 LIMITING CONDITION FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITION FOR OPERATION

C. Containment Spray/Cooling System

1. Except as specified in 3.5.C.2 below, both Containment Spray/Cooling Subsystems shall be operable whenever irradiated fuel is in the reactor vessel and reactor water temperature is greater than 212°F. A containment/spray cooling subsystem consists of the following equipment powered from one division:

   1 RHR Service Water Pump
   1 RHR Heat Exchanger
   1 RHR Pump*
   Valves and piping necess y for:
      Torus Cooling
      Drywell Spray

2. One Containment Spray/Cooling Subsystem may be inoperable for 7 days.

3. If the requirements of 3.5.C.1 or 2 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

C. Containment Spray/Cooling System

1. Demonstrate the RHR Service Water pumps develop 3,500 gpm flow rate against a 500 ft head when tested pursuant to Specification 4.15.B.

2. Test the valves in accordance with Specification 4.15.B.

3. Demonstrate the operability of the drywell spray headers and nozzles with an air test during each 10 year period.

* For allowed out of service times for the RHR pumps see Section 3.5.A.

automatically controls three selected safety-relief valves although the safety analysis only takes credit for two valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

B. RHR Intertie Line

An intertie line is provided to connect the RHR suction line with the two RHR loop return lines. This four-inch line is equipped with three isolation valves. The purpose of this line is to reduce the potential for water hammer in the recirculation and RHR systems. The isolation valves are opened during a cooldown to establish recirculation flow through the RHR suction line and return lines, thereby ensuring a uniform cooldown of this piping. The RHR loop return line isolation valves receive a closure signal on LPCI initiation. In the event of an inoperable return line isolation valve, there is a potential for some of the LPCI flow to be diverted to the broken loop during a loss of coolant accident. Surveillance requirements have been established to periodically cycle the RHR intertie line isolation valves. In the event of an inoperable RHR loop return line isolation valve, either the inoperable valve is closed or the other two isolation valves are closed to prevent diversion of LPCI flow. The RHR intertie line flow is not permitted in the Run Mode to eliminate 1) the need to compensate for the small change in jet pump drive flow or 2) a reduction in core flow during a loss of coolant accident.

C. Containment Spray/Cooling Systems

Two containment spray/cooling subsystems of the RHR system are provided to remove heat energy from the containment and control torus and drywell pressure in the event of a loss of coolant accident. A containment spray/cooling subsystem consists of 2 RHR service water pumps, a RHR heat exchanger, 2 RHR pumps, and valves and piping necessary for Torus Cooling and Drywell Spray. Torus Spray is not considered part of a containment spray/cooling subsystem. Placing a containment spray/cooling subsystem into operation following a loss of coolant accident is a manual operation.

The most degraded condition for long term containment heat removal following the design basis loss of coolant accident results from the loss of one diesel generator. Under these conditions, only one RHR pump and one RHR service water pump in the redundant division can be used for containment spray/cooling. The containment temperature and pressure have been analyzed under these conditions assuming service water and initial suppression pool temperature are both 90°F. Acceptable margins to containment design conditions have been demonstrated. Therefore the containment spray/cooling system is more than ample to provide the required heat removal capability. Refer to USAR Sections 5.2.3.3, 6.2.3.2.3, and 8.4.1.3.

During normal plant operation, the containment spray/cooling system provides cooling of the suppression pool water to maintain temperature within the limits specified in Specification 3.7.A.1.

F. Recirculation System

The reactor is designed such that thermal hydraulic oscillations are prevented or can be readily detected and suppressed without exceeding specified fuel design limits. To minimize the likelihood of a thermal-hydraulic instability, a power-flow exclusion region, to be avoided during normal operation, is calculated using the approved methodology as stated in specification 6.7.A.7. Since the exclusion region may change each fuel cycle the limits are contained in the Core Operating Limits Report. Specific directions are provided to avoid operation in this region and to immediately exit upon an entry. Entries into the exclusion region are not part of normal operation. An entry may occur as the result of an abnormal event such as a single recirculation pump trip. In these events, operation in the exclusion region may be needed to prevent equipment damage, but actual time spent inside the exclusion region is minimized. Though operator action can prevent the occurrence and protect the reactor from an instability, the APRM flow biased scram function will suppress oscillations prior to exceeding the fuel safety limit.

Power distribution controls are established to ensure the reactor is operated within the bounds of the stability analysis. With these controls in place, there is confidence that an oscillation will not occur outside of the stability exclusion region. Without these controls, it is theoretically possible to operate the reactor in such a manner as to cause an oscillation outside of the exclusion region. A nominal 5% power-flow buffer region outside of the exclusion region is provided to establish a stability margin to the analytically defined exclusion region. The buffer region may be entered only when the power distribution controls are in place.

Continuous operation with one recirculation loop was analyzed and the adjustments specified in specification 3.5.F.3 were determined by NEDO-24271, June 1980, "Monticello Nuclear Generating Plant Single Loop Operation;" NEDC-30492, April 1984, "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant;" and NEDC-32456P, July 1996. Specification 3.6.A.2 governs the restart of the pump in an idle recirculation loop. Adherence to this specification limits the probability of excessive flux transients and/or thermal stresses.

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D. Coolant Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached. Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. Systems connected to the reactor coolant system boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. As discussed in the Section 2.2 Bases, the 1375 psig Code limit is not exceeded in any case.

APPENDIX C---continued

Amendment Number	Additional Condition	Implementation Date
98	Update Section 5.2 of the Updated Safety Analysis Report by incorporating Figure E.2 of the NSP submittal dated July 16, 1997.	Within 90 days from the date of plant startup from the current maintenance outage, or November 1, 1997, whichever is later.
98	Process a 10 CFR 50.59 evaluation to change the EOP definition of adequate core cooling to 2/3 core height. The corresponding EOP changes and the required operator training shall also be completed. Final implementations shall be completed when all the 10 CFR 50.59 evaluation requirements are satisfied.	Within 180 days from the date of plant startup from the current maintenance outage, or February 1, 1998, whichever is later.
101	Conduct an independent evaluation of the testing methodology and the testing configuration of the EFT [emergency filtration testing] system by HEPA and charcoal filter testing experts. This evaluation shall include review of the exceptions to the ASME N510-1989 testing standard listed in Exhibit F of NSP's June 19, 1998, letter. The evaluation results shall be reported to the NRC.	Within 9 months of the date of issuance of Amendment No. 101.
101	Initiate appropriate modifications to the EFT system to comply with the ASME N510-1989 testing standard or obtain NRC approval for continued use of the exceptions.	Within 24 months of the date of issuance of Amendment No. 101.
102	All affected environmental qualification files, including service life and maintenance intervals if necessary, shall be revised to reflect the new environmental profile changes associated with power uprate.	Prior to implementation of Amendment No. 102 (prior to exceeding 1670 MWt).
102	All affected process computer and SPDS data points shall be changed to reflect uprate operating conditions.	Prior to implementation of Amendment No. 102 (prior to exceeding 1670 MWt).

APPENDIX C---continued

Amendment Number	Additional Condition	Implementation Date
102	Control room simulator changes shall be completed in accordance with ANSI/ANS 3.5-1985 Section 5.4.1, Simulator Performance Testing, and Monticello simulator configuration control procedures.	Prior to implementation of Amendment No. 102 (prior to exceeding 1670 MWt).
102	Classroom and simulator training on new knowledge and abilities associated with the power uprate shall be provided in accordance with Monticello Training Center procedures.	Prior to implementation of Amendment No. 102 (prior to exceeding 1670 MWt).
102	NSP shall monitor plant operational parameters for uprate impacts on the PRA models.	During and after the power uprate ascension test program.
102	Control room simulator changes shall be verified against actual plant startup data.	Within 3 months of completion of the power uprate ascension test program.
102	The applicable training programs and the simulator shall be modified, or appropriate compensatory actions shall be taken, in accordance with the Monticello Training Center procedures to reflect issues and discrepancies identified during startup testing.	Within 6 months of completion of the power uprate ascension test program.
102	The MNGP USAR shall be updated to reflect the changes associated with power uprate operation. This update shall not include credit for suppression pool scrubbing in the MSIV leakage pathway in the revised LOCA analysis.	Within 9 months of completion of the power uprate ascension test program.

APPENDIX C---continued

Amendment Number	Additional Condition	Implementation Date
102	NSP shall evaluate whether MO-2034 and MO-4229 are capable of allowing a subsequent operation after the required isolation safety functions are completed. This evaluation may include an examination of assumptions and methodologies, additional administrative controls, and modifications. The evaluation shall be completed in order to institute the corrective actions, if any, by the end of the next scheduled refueling outage.	By the end of the next scheduled refueling outage from the date of Amendment No. 102.
102	NSP shall evaluate the capacity margins of MO-2398 and MO-2034. This evaluation may include an examination of assumptions and methodologies, additional administrative controls, and modifications. The evaluation shall be completed in order to institute the corrective actions.	By the end of the next scheduled refueling outage from the date of Amendment No. 102.