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

March 7, 2001

Mr. James R. Morris
Site General Manager
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: REACTOR WATER CLEANUP (RWCU) SYSTEM AUTOMATIC ISOLATION AND MISCELLANEOUS INSTRUMENTATION SYSTEM TECHNICAL SPECIFICATION CHANGES (TAC NO. MA9605)

Dear Mr. Morris:

The Commission has issued the enclosed Amendment No. 117 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 20, 2000.

The amendment revises the TSs to (1) include the automatic reactor water cleanup (RWCU) system isolation feature, (2) restore the dose equivalent iodine-131 limit to 2 microcuries per gram, (3) change the RWCU reactor water level automatic isolation signal from Low to Low-Low reactor water level and add TSs for the high pressure coolant injection (HPCI) and reactor core isolation cooling low steam line pressure isolation instrumentation, (4) delete the HPCI 150,000 lb/hr low range high flow isolation instrumentation and adds a time delay to the 300,000 lb/hr upper range high flow isolation instrumentation, and (5) change the suppression chamber water allowable water level from volume units to level units.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

CFLyon

Carl F. Lyon, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures: 1. Amendment No. 117 to DPR-22
2. Safety Evaluation

cc w/encls: See next page

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

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 117
License No. DPR-22

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

   A. The application for amendment by the licensee, dated July 20, 2000, 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:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 117 , 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 its date of issuance and shall be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 7, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 117

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
50	50
-	50a
51	51
-	51a
62	62
63	63
63a	63a
64	64
65	65
66	66
67	67
70	70
123	123
148	148
157	157
175a	175a
177	177
183	183

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. <u>Reactor Cleanup System (Group 3)</u>				
a. High Drywell Pressure	≤2 psig	2	2	E
b. Low Low Reactor Water Level**	≥6′ 6″, ≤6′ 10″	2	2	E
c. High RWCU Room Temperature Allowable Value	≤188°F	2	2	E
d. High RWCU System Flow Allowable Value	≤500 gpm with ≤27 second time delay	2	2	E
4. <u>HPCI Steam Lines</u> (Group 4)				
a. HPCI High Steam Flow***	≤300,000 lb/hr with ≤7 second time delay	2(4)	2	F
b. HPCI Steam Line Area High Temp.	≤200°F	16(4)	16	F
c. Low Pressure in HPCI Steam Supply Line	≥85 psig	4(6)	4(6)	F

Table 3.2.1 (Continued)				
Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instru-ment Channels Per Trip System (1, 2)	Required Conditions*
5. RCIC Steam Lines (Group 5)				
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
c. Low Pressure in RCIC Steam Supply Line	≥55 psig	4(7)	4(7)	G
6. Shutdown Cooling Supply Isolation				
a. Reactor Pressure Interlock	≤75 psig at the reactor steam dome	2(4)	2	C

Table 3.2.1 (Continued)

NOTES:

(1) There shall be two operable or tripped trip systems for each function. A channel (a shared channel is considered one channel) may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter.

(2) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated as follows:

(a) With one required instrument channel inoperable in one or more trip functions, place the inoperable channel(s) or trip system in the tripped condition within 12 hours, or

(b) With more than one instrument channel inoperable for one or more trip functions, immediately satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

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

(3) Low pressure in main steam line only need to be available in the RUN position.

(4) All instrument channels are shared by both trip systems.

(5) May be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also, need not be operable when primary containment integrity is not required.

(6) The four pressure switches are arranged in a one-out-of-two-twice logic, the output of the logic providing a trip signal to a single trip system for isolation.

(7) The four pressure switches are arranged in a one-out-of-two-twice logic, the output of the logic providing a trip signal to each of two separate trip systems. Each trip system is able, by itself, to initiate isolation.

Table 3.2.1 (Continued)

NOTES: (Continued)

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

A. Group 1 isolation valves closed.

B. Reactor Power on IRM range or below and reactor in startup, refuel, or shutdown mode.

C. Isolation Valves closed for: Shutdown Cooling System, and Reactor Head Cooling Line.

D. Comply with Condition C. above.

E. Isolation Valves closed for: Reactor Cleanup System.

F. HPCI steam line isolated. (See specification 3.5 for additional requirements.)

G. RCIC steam line isolated.

** Function changed from Low Reactor Water Level to Low Low Reactor Water Level following completion of design change.

*** Function changed from $\leq$150,000 lb/hr, $\leq$60 second delay, and $\leq$300,000 lb/hr, instantaneous, isolation to $\leq$300/000 lb/hr, $\leq$7 second delay, isolation following completion of design change.

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None
4. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/12 hours
<u>CONTAINMENT ISOLATION (GROUP 2)</u>			
1. Reactor Low Water Level (Note 10)	-	-	-
2. Drywell High Pressure (Note 10)	-	-	-
<u>RWCU ISOLATION (GROUP 3)</u>		(Note 12)	
1. High RWCU Room Temperature	Once/3 months	Once/Operating Cycle-RTD Input Once/3 months-Trip Unit	Once/12 hours
2. High RWCU System Flow	Once/3 months	Once/Operating Cycle-Transmitter Once/3 months-Trip Unit	Once/12 hours
* 3. Reactor Low Low Water Level (Note 11)	-	-	
4. Drywell High Pressure (Note 10)	-	-	
<u>HPCI (GROUP 4) ISOLATION</u>			
1. Steam Line High Flow	Once/3 months	Once/3 months	None
2. Steam Line High Temperature	Once/3 months	Once/3 months	None
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None
<u>RCIC (GROUP 5) ISOLATION</u>			
1. Steam Line High Flow	Once/3 months	Once/3 months	None
2. Steam Line High Temperature	Once/3 months	Once/3 months	None
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None

* Function changed from Low Reactor Water Level to Low Low Reactor Water Level following completion of design change.

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

Instrument Channel		Test (3)	Calibration (3)	Sensor Check (3)
REACTOR BUILDING VENTILATION & STANDBY GAS TREATMENT				
1.	Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months - Trip Unit	Once/12 hours
2.	Drywell High Pressure (Note 10)	-	-	-
3.	Radiation Monitors (Plenum)	Once/3 months	Once/3 months	Once/day
4.	Radiation Monitors (Refueling Floor)	Once/3 months	Once/3 months	Note 4
RECIRCULATION PUMP TRIP AND ALTERNATE ROD INJECTION				
1.	Reactor High Pressure	Once/3 months (Note 5)	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2.	Reactor Low Low Water Level	Once/3 months (Note 5)	Once/Operating Cycle- Transmitter Once/3 Months-Trip Unit	Once/12 hours
SHUTDOWN COOLING SUPPLY ISOLATION				
1.	Reactor Pressure Interlock	Once/3 months	Once/3 Months	None
SAFEGUARDS BUS VOLTAGE				
1.	Degraded Voltage Protection	Once/month	Quarterly	Not applicable
2.	Loss of Voltage Protection	Once/month	Once/Operating Cycle	Not applicable
SAFETY/RELIEF VALVE LOW-LOW SET LOGIC				
1.	Reactor Scram Sensing	Once/Shutdown (Note 8)	-	-
2.	Reactor Pressure - Opening	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
3.	Reactor Pressure - Closing	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
4.	Discharge Pipe Pressure	Once/3 months (Note 5)	See Table 4.14.1	See Table 4.14.1
5.	Inhibit Timer	Once/3 months (Note 5)	Once/Operating Cycle	-
CONTROL ROOM HABITABILITY PROTECTION				
1.	Radiation	Monthly (Note 5)	18 months	Daily

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

NOTES:

(1) (Deleted)

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per 12 hours and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region.

(3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

(4) Whenever fuel handling is in process, a sensor check shall be performed once per 12 hours.

(5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.

(6) (Deleted)

(7) (Deleted)

(8) Once/shutdown if not tested during previous 3 month period.

(9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.

(10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.

(11) Uses contacts from Group 1 Isolation logic. Tested and calibrated in accordance with Group 1 Low Low Water Level Instrumentation.

(12) Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

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 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 and Group 3 Primary containment isolation valves, Reference Section 7.7.2.2 FSAR, and also activates the ECC systems and starts the emergency diesel generator.

Bases 3. 2 (Continued):

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, RWCU, and recirc sample 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 3.2 (Continued):

instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel clad temperatures remain less than 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Sections 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "refuel" and "Startup" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 825 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water. Reference License Amendment Request Dated December 1, 1975 from L. O. Mayer (NSP) to R. S. Boyd (USNRC).

The RWCU high flow and temperature instrumentation is provided to detect a break in the RWCU piping. Tripping of this instrumentation results in actuation of the RWCU isolation valves, i.e., Group 3 valves. The trip settings have been established so that the radiological consequences of a high energy line break in this system are bounded by a break in the main steam system. The recirc sample isolation valves, which receive a Group 1 isolation signal, also receive a redundant Group 3 isolation signal.

Bases 3.2 (Continued):

The HPCI and RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI or RCIC piping. The trip settings of 200°F and approximately 300% of HPCI and RCIC design steam flow are such that the core will not be uncovered and the radiological consequences are bounded by the main steam line break accident. The HPCI and RCIC low steam line pressure instrumentation protects the HPCI and RCIC turbines when system operation is no longer useful or possible. Tripping of the high flow, high temperature, or low steam line pressure instrumentation results in actuation of the HPCI or RCIC steam supply valves; i.e., a Group 4 or Group 5 isolation.

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 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. 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 3.2 (Continued):

	Trip Function	Deviation
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Specification 3.2.E.3 and Table 3.2.4	Reactor Building Vent Plenum Monitors	+5 mR/hr
	Refueling Floor Radiation Monitors	+5 mR/hr
	* Low Low Reactor Water Level High Drywell Pressure	-3 inches +1 psi
Primary Containment Isolation Functions Table 3.2.1	* Low Low Water Level	-3 inches
	High Flow in Main Steam Line	+2%
	High Temp. in Main Steam Line Tunnel	+10°F
	Low Pressure in Main Steam Line	-10 psi
	High Drywell Pressure	+1 psi
	* Low Reactor Water Level	-6 inches
	HPCI High Steam Flow	+7,500 lb/hr
	HPCI Steam Line Area High Temp.	+2°F
	HPCI Steam Line Pressure	-5 psi
	RCIC High Steam Flow	+2250 lb/hr
	RCIC Steam Line Area High Temp	+2°F
	RCIC Steam Line Pressure	-5 psi
	Shutdown Cooling Supply ISO	+7 psi

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
4. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head are ≥70°F.	4. When the reactor vessel head studs are under tension and the reactor is in the Cold Shutdown Condition, the reactor vessel shell flange temperature shall be permanently recorded.
C. Coolant Chemistry	C. Coolant Chemistry
1. (a) The steady state radioiodine concentration in the reactor coolant shall not exceed 2.0 microcuries of I-131 dose equivalent per gram of water.	1. (a) A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation.
(b) The steady state radioiodine concentration in the reactor coolant shall not exceed 0.02 microcuries of I-131 dose equivalent per gram of water when the reactor coolant temperature is >212°F, the reactor is not critical, and primary containment integrity has not been established.	(b) A sample of reactor coolant shall be taken and analyzed for radioactive iodines of I-131 through I-135 within 24 hours prior to raising the reactor coolant temperature >212°F, with the reactor not critical, and with primary containment integrity not established.

C. Coolant Chemistry

In the event of a main steam line break outside primary containment, calculations show the resultant radiological dose at the exclusion area boundary to be less than 10% of the dose guidelines of 10 CFR 100. This dose was calculated on the basis of the radioiodine concentration limit of 2 μCi of I-131 dose equivalent per gram of water. In the event of a postulated high energy line break in the RWCU system outside the drywell, calculations show the resultant radiological consequences are bounded by the steam line rupture. In the event of a large primary system break in primary containment during a reactor vessel hydrostatic or leakage test with the reactor coolant temperature >212°F, the reactor not critical, and primary containment integrity not established, calculations show the resultant radiological dose at the exclusion area boundary to be conservatively bounded by the dose calculated for a main steam line break outside primary containment. This dose was calculated on the basis of the radioiodine concentration limit of 0.02 μCi of I-131 dose equivalent per gram of water.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.C.1(a) is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam line.

Materials in the primary system are primarily 304 stainless steel and zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. The limit placed on chloride concentration is to prevent stress corrosion cracking of the stainless steel.

3.0 LIMITING CONDITIONS FOR OPERATION

d. During reactor isolation conditions the reactor pressure vessel shall be depressurized to <200 psig at normal cooldown rates if the suppression pool temperature exceeds 120°F.

e. The suppression chamber water level shall be $\geq$ -4.0 and $\leq$ +3.0 inches.

f. Two channels of torus water level instrumentation shall be operable. From and after the date that one channel is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 30 days unless such channel is sooner made operable. If both channels are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding six hours unless at least one channel is sooner made operable.

4.0 SURVEILLANCE REQUIREMENTS

d. Whenever there is indication of relief valve operation with a suppression pool temperature of $\geq$160°F and the primary coolant system pressure >200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.

e. The suppression chamber water level shall be checked once per day.

f. The suppression chamber water level indicators shall be calibrated semiannually.

Bases 3.7/4.7 (Continued):

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the maximum allowable primary containment pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. See USAR Section 5.2.3.2.

The maximum and minimum suppression chamber water levels specified correspond to the values of downcomer submergence assumed in the development of loads for the Monticello Mark I Long Term Program. The minimum specified level corresponds to a downcomer submergence of 3 feet, or an indicated water level of -4 inches. The maximum specified water level corresponds to a downcomer submergence of 3 feet 7 inches, or an indicated water level of +3 inches. The minimum suppression chamber water level also assures an adequate volume of water is available so that, following a loss of coolant accident, the maximum suppression chamber water temperature limits will not be exceeded.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 42 psig which is below the allowable pressure of 62 psig.

Bases 3.7 (Continued):

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor water temperatures above 212°F provides additional margin above that available at 330°F.

The large amount of water that must be added or removed to cause a significant change in the suppression chamber water inventory is not likely to go un-noticed. With a daily check of water level, there is an extremely low probability that a loss of coolant accident will occur simultaneously with water level being outside of the specified range. Two indicators provide redundant readings for comparison (with no automatic action initiation). The provisions allowing one or both indicators out of service are consistent with the need for a redundant indicator and the frequency for checking the level, respectively.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping.

Bases 4.7:

A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident. Daily checks are specified of pool temperature and level to ensure that these parameters are within their allowable ranges.

The interiors of the drywell and suppression chamber are painted to prevent corrosion. The inspection of the paint during each refueling outage, approximately once per year, assures the paint is intact and is not deteriorating. Experience with this type of paint indicates that the inspection interval specified is adequate.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

The design basis loss of coolant accident was analyzed at the primary containment maximum allowable accident leak rate of 1.2% and has been evaluated by the NRC Staff(1). Computed offsite doses are well below the guidelines of 10 CFR Part 100.

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(1) Safety Evaluation by the Division of Reactor Licensing, US Atomic Energy Commission, in the Matter of Northern States Power Company Monticello Nuclear Generating Plant, Unit 1, Docket No. 50-263, March 18, 1970, Section 4.1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 117 TO FACILITY OPERATING LICENSE NO. DPR-22

NUCLEAR MANAGEMENT COMPANY, LLC

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By application dated July 20, 2000, the licensee requested changes to the Technical Specifications (TSs) for Monticello Nuclear Generating Plant. The proposed amendment would revise the TSs to (1) include the automatic reactor water cleanup (RWCU) system isolation feature, (2) restore the dose equivalent iodine-131 (DEI) limit to 2 microcuries per gram, (3) change the RWCU reactor water level automatic isolation signal from Low to Low-Low reactor water level and add TSs for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) low steam line pressure isolation instrumentation, (4) delete the HPCI 150,000 lb/hr low range high flow isolation instrumentation and add a time delay to the 300,000 lb/hr upper range high flow isolation instrumentation, and (5) change the suppression chamber water allowable water level from volume units to level units.

2.0 BACKGROUND

By Amendment No. 101, dated August 28, 1998, the NRC approved changes to the TSs governing the performance of the control room emergency filtration system and the limit on reactor coolant DEI. The changes reduced the DEI limit from 5.0 microcuries per gram to 2.0 microcuries per gram based on the licensee's revised analysis for the main steam line break (MSLB) accident and from 2.0 microcuries per gram to 0.25 microcuries per gram to address RWCU high energy line break concerns. At Monticello, the original design of the RWCU system did not have capability to automatically isolate a line break. The reduction of the DEI limit to 0.25 microcuries per gram was an interim measure pending installation of an automatic RWCU line break isolation system, as described in the licensee's letter dated June 10, 1997. The licensee committed to restore the DEI limit to 2.0 microcuries per gram upon completing the proposed RWCU design change. Since they have implemented the automatic RWCU line break isolation system, the licensee proposes in this amendment request to increase the DEI limit from 0.25 microcuries per gram to 2.0 microcuries per gram. In addition, the licensee proposes to include the RWCU automatic isolation requirements and other miscellaneous changes in the TSs.

3.0 EVALUATION

3.1 Radiological Consequences of RWCU Line Break with Automatic Isolation

The licensee calculated the maximum permissible RWCU high-flow setpoint and isolation logic delay time that would ensure that the offsite and control room doses resulting from an RWCU line break are conservatively bounded by the current design-basis MSLB analysis, described in Section 14.7 of the Monticello Updated Safety Analysis Report (USAR). The licensee added additional margin to the proposed TS Table 3.2.1 values for the RWCU high flow isolation setpoint of $\leq$ 500 gal/min with an isolation delay of $\leq$ 27 seconds.

The radiological impact of an RWCU line break or MSLB is directly proportional to the mass of reactor coolant released in the accident. The current Monticello USAR design-basis MSLB dose analysis assumes that 82,200 lbm is released. For the RWCU line break analysis, the licensee calculated a total reactor coolant mass release of 42,200 lbm based on RWCU manual isolation and the proposed RWCU high-flow TS setpoint of 500 gal/min. The methodology and assumptions for the RWCU line break dose analysis are the same as for the USAR design-basis MSLB dose analysis. Both analyses assumed the reactor coolant activity is at the proposed TS limit of 2.0 microcuries per gram DEI. The licensee's RWCU line break dose analysis methodology and assumptions are acceptable. The attached table presents the licensee's results for the RWCU line break calculated in support of this amendment request, as well as the current design basis MSLB in the Monticello USAR.

The regulatory dose limits for these accidents are found in 10 CFR Part 100 for offsite doses and 10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC-19) for the control room. The dose acceptance criteria are 300 rem thyroid and 25 rem whole body for offsite doses, and 30 rem thyroid and 5 rem whole body for the control room. The RWCU line break doses, as calculated by the licensee, are bounded by the current USAR MSLB analysis and are within the acceptance criteria. The proposed changes to TS Table 3.2.1 for the RWCU high-flow isolation instrumentation are acceptable.

3.1.1 Reactor Coolant Chemistry - DEI

Due to RWCU high-energy break concerns, the licensee had previously received staff approval to reduce the TS 3.6.C reactor coolant steady state activity limit to 0.25 microcuries per gram DEI. This was intended to be an interim measure pending installation of an automatic RWCU line break isolation system. The RWCU isolation system is now installed and the licensee has proposed that the TS reactor coolant steady state activity limit be restored to 2.0 microcuries per gram DEI.

The radiological analysis of the MSLB currently in the Monticello USAR assumes the reactor coolant activity is at the proposed TS limit of 2.0 microcuries per gram DEI. The licensee did not revise the MSLB analysis for this license amendment request. The staff has reviewed the licensee's RWCU line break analysis and agrees with the licensee that the radiological consequences of the RWCU line break will not exceed the consequences of the design-basis MSLB now that the RWCU system has automatic isolation capability. Therefore, the proposed change to restore the TS 3.6.C steady-state radioiodine concentration limit to 2.0 microcuries per gram DEI is acceptable.

3.2 <u>RWCU System Automatic Isolation Instrumentation</u>

The licensee stated that it has completed modifications to the RWCU system consisting of the addition of high RWCU room temperature automatic isolation instrumentation and high RWCU system flow automatic isolation instrumentation. The licensee has proposed to revise TS Table 3.2.1 to include setpoints for the high RWCU room temperature isolation and a high RWCU system flow isolation.

The licensee proposes to (1) add Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) for the RWCU system automatic high energy line break isolation instrumentation to TS Tables 3.2.1 and 4.2.1; and (2) revise TS Bases Section 3.2 to add a new paragraph to describe the automatic RWCU isolation logic and include a discussion of the basis for the redundant Group 3 closure signal provided for the recirculation sample isolation valves.

The design change for auto isolation of a pipe break revises the valve closing control logic for the RWCU line valves by interlocking the circuit with signals from flow and temperature instruments. Leaked mass from a pipe break will increase the area temperature. A break between the containment isolation valve and the flow element will be sensed and isolated by a signal from the RWCU room temperature instruments by closing isolation valves at or above 188° F. Normal RWCU flow is approximately 140 gpm and excessive flow is another indication of a pipe-break. RWCU flow is sensed by a flow element installed in the piping as close as practical to the outboard containment isolation valve and penetration. The differential pressure signal from the flow element is processed by four differential transmitters and trip units using a one-out-of-two-taken-twice logic scheme and is set to initiate isolation at a trip setting of ≤500 gpm with a time delay of ≤27 seconds. The time delay is incorporated to prevent spurious trips caused by momentary flow disturbances. A reversal of the differential pressure signal will indicate a break in the flow element sensing line and will also generate an isolation signal. In its submittal, the licensee stated that selection of actuation setpoints and the duration of the time delay ensured that the radiological effects of an RWCU break are conservatively bounded by the MSLB accident.

The proposed changes are consistent with the Standard Technical Specifications (NUREG-1433). The licensee stated that it used the General Electric (GE) Setpoint Methodology (NEDC-31336), as noted in TS Bases Section 3.1, in the creation of calculations supporting the RWCU flow and temperature TS values that were submitted. The NRC has approved the methodology, as documented in a safety evaluation report dated November 6, 1995. Therefore, the proposed TS requirements associated with the implemented design-change providing interlocks between RWCU high-flow, RWCU room high-temperature, and time delay instrumentation to automatically isolate an RWCU pipe break and provide indication of broken flow element sensing lines are acceptable. Also, the staff has no objection to the proposed changes to the TS Bases.

3.3 <u>RWCU Automatic Isolation Reactor Water Level Setpoint</u>

The licensee proposes to change the RWCU reactor water level automatic isolation setpoint from "low reactor water level" to "low-low reactor water level."

In accordance with the current TS, the setpoint for the reactor water level RWCU system isolation function is set at "low," which corresponds to 10 ft. - 4 inches (-in) above the top of active fuel (TAF). The proposed change will revise this setting to "low-low," which is basically a 4-in band from 6 ft. 6-in to 6 ft. 10-in above the TAF. In its submittal, the licensee stated that GE Service Information Letter (SIL) 131, dated April 16, 1975, provides the basis for the proposed change. During full power operation, a reactor scram would result in a reactor vessel water level transient (due to collapsing of voids). During such a transient, if the reactor water-level dips to its current low setting, then a spurious isolation of the RWCU system will occur. The spurious isolation of the RWCU system will create an unnecessary complication because, to recover from the scram, operators must restart the RWCU system immediately following the scram to control the reactor vessel water inventory. The GE SIL concluded that, with the installation of an automatic RWCU break isolation system, the setpoint for the reactor water level RWCU isolation function can be safely changed from its current low (Level 3) setting to a low-low (Level 2) setting and that this change will eliminate the possibility of unnecessary isolation of the RWCU system following a reactor scram. The licensee further added that the low-low setting corresponds to the level at which the emergency core cooling system (ECCS) receives an automatic initiation signal.

In summary, the licensee has proposed to revise the reactor water level RWCU isolation setpoint from Low Reactor Water Level (Level 3) to Low-Low Reactor Water Level (Level 2). The purpose of this change is to eliminate unnecessary isolation of the RWCU system on reactor scrams. The proposed RWCU isolation setpoint is the same reactor water level setpoint used for the initiation of the ECCS and is consistent with NUREG-1433 and with the guidance provided in GE SIL 131. The primary RWCU break isolation protection is now provided by the high flow and temperature sensors of the automatic RWCU break isolation system, as discussed in Section 3.2 above. Therefore, the proposed revision of the RWCU automatic isolation reactor water level setpoint in TS Table 3.2.1 is acceptable.

3.4 High-Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System Low Steam Line Pressure Isolation Instrumentation

The licensee proposes to add LCOs and SRs for the HPCI system and the RCIC system low steam line pressure isolation instrumentation.

Automatic isolation of the HPCI and the RCIC systems on a low steam line pressure condition was part of the original plant design, but LCOs and SRs for instrumentation of these functions are not part of the current TSs. To achieve consistency with NUREG-1433, the licensee committed to the NRC in a letter dated February 3, 1998, to include LCO and SR requirements for these instruments in the TSs. The proposed changes are consistent with the licensee's past commitments and are acceptable.

3.5 Delete the HPCI System Low-Range High-Flow Isolation Signal (150,000 lb/hr) and Add a Time Delay to the Existing High-Range Isolation Signal (300,000 lb/hr)

The licensee has proposed to eliminate the existing low range 150,000 lb/hr delayed HPCI high steam flow trip and add a 7-second time delay to the existing high range 300,000 lb/hr high steam flow trip in TS Table 3.2.1. The purpose of the proposed change is to improve the reliability of the HPCI system by reducing the potential for spurious closure of the steam supply valves on HPCI startup. The licensee stated that the existing 150,000 lb/hr isolation setpoint is too close to the nominal rated HPCI steam flow of 112,000 lb/hr.

In its submittal, the licensee stated that a review of the HPCI high steam flow isolation designs for several plants similar to Monticello indicated that only Monticello has the duel range (i.e., 150,000 lb/hr at low range and 300,000 lb/hr at high range) high-flow isolation logic. A single high-range delayed HPCI steam flow isolation signal, nominally set at 300 percent of the rated flow is the most common logic in other similar plants. Also, the current setpoint of 150,000 lb/hr is too close to the nominal rated steam flow of 112,000 lb/hr. Therefore, the licensee has proposed to delete the 150,000 lb/hr setpoint and add a 7-second time delay to the existing high-range 300,000 lb/hr steam flow isolation signal. The licensee stated that the proposed change is in line with the its efforts to improve the overall reliability of the HPCI system, will be consistent with more recent boiling water reactor protective instrument designs, and will reduce the potential for spurious isolation signals during the HPCI system startup and testing.

In support of the proposed changes to the HPCI high steam flow isolation, the licensee performed a HPCI line break mass release analysis, Monticello calculation CA-00-082. The calculation supports that the proposed HPCI isolation and valve closure times are such that the core will not be uncovered. The staff has reviewed the licensee's submittal and supporting calculation. The results of the calculation show that all applicable acceptance criteria are met with the proposed HPCI high steam flow setpoint and time delay.

The staff also reviewed the licensee's analysis of the radiological consequences of an HPCI line break with the proposed changes. To ensure that the MSLB remains the bounding accident, the licensee calculated the maximum permissible HPCI isolation logic delay time, then added some margin. The calculation assumed the maximum total coolant mass release of 82,200 lbm from the design-basis MSLB and the maximum coolant mass flow rate of 767 lbm/sec resulting from a break in the HPCI system. This resulted in the licensee proposing a TS HPCI high-steam flow isolation logic delay of ≤ 7 seconds. The licensee calculated that 53,300 lbm reactor coolant is released due to an HPCI line break, assuming the TS flow setpoint of 300,000 lb/hr and manual isolation of the HPCI system. The methodology and assumptions for the HPCI line break dose analysis are the same as for the design basis MSLB dose analysis described in Section 14.7 of the Monticello USAR. The staff finds the licensee's assumptions acceptable. The radiological consequences of an HPCI break with a total mass release of 53,300 lbm of reactor coolant at 2.0 microcuries per gram DEI, as calculated by the licensee, are presented in the attached table. The resulting doses are within the acceptance criteria given in 10 CFR Part 100 and GDC-19, and the MSLB remains bounding.

Since the results of the calculations show that all applicable acceptance criteria are met with the proposed HPCI high steam flow setpoint and time delay, the resulting doses are within the acceptance criteria given in 10 CFR Part 100 and GDC-19, and the MSLB remains bounding, the proposed changes to TS Table 3.2.1 for the HPCI high steam flow isolation instrumentation are acceptable.

3.6 <u>For Suppression Pool Water Inventory LCOs and SRs, Change Measurement Units From Volumetric (i.e., Cubic Feet) to Equivalent Level Measurement (i.e., Water Level in Inches)</u>

In its submittal, the licensee stated that establishing suppression inventory limits in inches of water level is preferable because it is unambiguous and it allows direct level measurment. Expressing pool inventory directly in terms of level is also consistent with NUREG-1433. The staff considers the proposed change to be administrative and, therefore, finds it acceptable.

3.7 Conclusion

On the basis of this evaluation, the staff agrees with the licensee's conclusions that the proposed changes will not alter assumptions relative to the mitigation of an accident or transient event and will not adversely affect normal plant operation and testing. Also, the licensee's in-house setpoint calculation methodology is based on the guidance provided by the GE Setpoint Methodology (NEDC-31336), which was previously approved by the staff. Therefore, the staff finds the proposed changes acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes administrative requirements, or a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (65 *FR* 51361). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: Table

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Date: March 7, 2001

Table
Results of Radiological Analyses

RWCU Line Break Radiological Consequences

Location	Whole Body Dose (rem)	Whole Body Criterion (rem)	Thyroid Dose (rem)	Thyroid Criterion (rem)
Exclusion Area Boundary (EAB)	0.204	25	12.4	300
Low Population Zone (LPZ)	0.018	25	1.07	300
Control Room	0.002	5	5.26	30

HPCI Line Break Radiological Consequences

Location	Whole Body Dose (rem)	Whole Body Criterion (rem)	Thyroid Dose (rem)	Thyroid Criterion (rem)
EAB	0.258	25	15.7	300
LPZ	0.022	25	1.35	300
Control Room	0.003	5	6.65	30

MSLB Radiological Consequences (from USAR Tables 14.7-18 & 14.7-19)

Location	Whole Body Dose (rem)	Whole Body Criterion (rem)	Thyroid Dose (rem)	Thyroid Criterion (rem)
EAB	0.40	25	24.2	300
LPZ	0.03	25	2.08	300
Control Room	0.004	5	10.1	30