June 11, 2002

Mr. Jeffrey S. Forbes
Site Vice President
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 RELATING TO TECHNICAL SPECIFICATION REVISIONS TO THE REFERENCE POINT FOR REACTOR VESSEL LEVEL SETPOINTS, TO SAFETY LIMITS, AND TO THE BASES (TAC NO. MB2246)

Dear Mr. Forbes:

The Commission has issued the enclosed Amendment No. 128 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 June 18, 2001, as supplemented by letters dated January 30, and March 1, 2002.

The amendment revises (1) the reference point for reactor vessel level instrumentation specifications to use instrument "zero" instead of "top of active fuel;" (2) simplifies the safety limits and limiting safety system settings to eliminate specifications that are unnecessary, outdated, or redundant to other TSs; (3) changes the reactor coolant system pressure safety limit from 1335 psig to 1332 psig to correct a minor calculation error; and (4) makes corresponding TS Bases changes.

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,

/RA by Johnny Eads for/

Samuel Miranda, 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. 128 to DPR-22
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

<u>DISTRIBUTION</u>:

PUBLIC	OGC	RLandry
PDIII-1 Reading	ACRS	RCaruso
LRaghavan	WBeckner	SMiranda
RDennig	GHill(2)	CSchulten
RBouling	BBurgess, RGN-III	EMarinos

* Provided SE input by memo

OFFICE	PDIII-1/PM	PDIII-1/LA	SRXB/SC*	EEIB/SC**	RORP/SC	OGC**	PDIII-1/SC
NAME	SMiranda	THarris for RBouling	RCaruso	EMarinos	RDennig	AHodgdon	LRaghavan
DATE	05/30/02	05/23/02	11/16/01	05/30/02	05/29/02	05/06/02	06/11/02

ADAMS Accession No. ML013390677 OFFICIAL RECORD COPY cc:

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 128
License No. DPR-22

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated June 18, 2001, as supplemented January 30, and March 1, 2002, 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. 128 , 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, 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: June 11, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 128

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
i	i
iv	iv
6 - 12	6 - 12
13 - 16	-
18 - 25	-
28 - 29	28 - 29
36 - 40	36 - 40
49 - 50	49 - 50
52 - 54	52 - 54
59 - 60	59 - 60
60d	60d
64	64
108	108
127	127
150 - 151	150 - 151
217	217
243	243
249b	250

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 128 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 June 18, 2001, as supplemented by letters dated January 30 and March 1, 2002, the Nuclear Management Company, LLC (the licensee), requested changes to the Technical Specifications (TSs) for the Monticello Nuclear Generating Plant. The supplemental letters dated January 30 and March 1, 2002, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 28, 2001 (66 FR 38764).

The proposed changes would:

1) Revise the reference point for reactor vessel level instrumentation specifications to use instrument "zero" instead of "top of active fuel (TAF)." Use of TAF is no longer an unambiguous reference point because of variations in the length of active fuel in modern fuel element designs.

2) Simplify the Safety Limits and Limiting Safety System Settings to eliminate specifications that are unnecessary, outdated, or redundant to portions of TSs Section 3.

3) Change the reactor coolant system pressure Safety Limit from 1335 psig to 1332 psig to correct a minor error in the calculation of this limit.

4) Change the Bases for the upper limit on reactor coolant system safety/relief valve self actuation setpoint to correct the description of this specification.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act of 1954, as amended (the Act) requires applicants for nuclear power plant operating licenses to include the TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36. The regulation requires that the TSs include items in eight specific categories. The categories are: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation;

(3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports. However, the regulation does not specify the particular requirements to be included in a plant's TSs.

The regulation at 10 CFR Section 50.36 specifies four criteria to be used in determining whether a particular structure, system or component is required to be included in a limiting condition for operation (LCO), as follows: (1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; or (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. LCOs and related requirements that fall within or satisfy any of the criteria in the regulation must be retained in the TSs, while those requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents.

The Monticello TSs include detailed information related to system design and operation and also procedural details for meeting TS action and surveillance requirements. Inclusion of such information has been shown to give little or no safety benefit; thus, its removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the industry comments on Standard Technical Specifications (STSs). The NRC staff reviewed generic relaxations contained in STSs and found them acceptable because they are consistent with current licensing practices and the Commission's regulations.

The staff finds that the licensee in its submittal identified the applicable regulatory requirements. The regulatory requirements on which the staff based its acceptance criteria are 10 CFR Sections 50.36, 50.90, and 50.92.

3.0 EVALUATION

The staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in Exhibit A of the licensee's application. The evaluation below will support the conclusion 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 amendments will not be inimical to the common defense and security or to the health and safety of the public. The evaluation denotes proposed text revisions by striking out the original text with a single line, followed by the proposed replacement in italics.

3.1 Changes that revise the Reference Point for Reactor Vessel Level Specifications From "Top of Active Fuel" to Instrument "Zero"

3.1.1 Statement of Proposed Changes

In several locations in the Monticello TSs, reactor vessel water level is referenced to the TAF. Use of TAF as a reference point can be ambiguous because of the variety of active fuel lengths used in modern core designs. As a practical matter, however, TAF has always been established as 351.5" above the inner clad bottom of the reactor vessel based on the original core fuel design.

It is proposed that all reactor vessel water level TSs currently referenced to TAF be changed to be referenced to reactor vessel instrument "zero." Instrument zero is defined unambiguously as 477.5 inches above the inner clad bottom of the reactor vessel. TAF referenced to instrument "zero" is -126". Also, where both a lower and upper limit are currently specified for a reactor vessel level, it is proposed that the upper limit be dropped. For example, the low-low reactor water level setpoint is currently specified as ≥ 6'6" and ≤ 6'10" above top of active fuel (or some variation of this wording). This setpoint will be changed, wherever appearing, to ≥ -48", where -48" = -126" + 6'6"

Other specifications referenced to TAF are corrected to instrument "zero" in a similar manner. Changes are also proposed to the Bases to define the reference point for reactor vessel level specifications and the region of the vessel where this level is sensed (annulus). The term "annulus," therefore, becomes unnecessary and the licensee proposes it be deleted.

Some proposed changes are deletions of duplicate requirements (i.e., requirements that also appear in other sections of the TS, but not always as the same terms). As these "duplicate" requirements are expressed in terms of level from instrument "zero," they are considered below. Also included are editorial changes (e.g., renumbering of requirements) that result from the deletion of these requirements.

(1) Specification 2.3.C (page 7), Reactor Low Water Level Scram ≥ 10'6" above the top of the active fuel, is deleted.

(2) Item 7 on Table 3.1.1 (page 29), Reactor Low Water Level Scram is changed from ≥ 7 in. (annulus) to ≥ 7 in.

(3) Item 7, Reactor Low Water Level Scram (page 38 Bases), has been changed to read, "The low reactor water level instrumentation is set to trip when reactor water level is ≥ 7" on the instrument.

(4) Specification 2.3.D (page 7), Reactor Low Low Water Level ECCS initiation ≥ 6'6", ≤ 6'10" above the top of the active fuel (TAF), has been deleted.

(5) Item 1.a in Table 3.2.1 (page 49), Reactor Low Low Water Level Main Steam and Recirc sample Line Isolation, is changed from ≥ 6'6", ≤ 6'10" above TAF to ≥ -48".

(6) Item 3.b in Table 3.2.1 (page 50), Reactor Low Low Water Level Reactor Cleanup System Isolation, is changed from ≥ 6'6", ≤ 6'10" above TAF to ≥ -48".

(7) Item A.1 in Table 3.2.2 (page 52), Reactor Low Low Water Level Core Spray and LPCI, is changed from ≥ 6’6", ≤ 6’10" above TAF to ≥ -48".

(8) Item B.2 in Table 3.2.2 (page 53), Reactor Low Low Water Level HPCI System, is changed from ≥ 6’6", ≤ 6’10" above TAF to ≥ -48".

(9) Item C.1 in Table 3.2.2 (page 53), Reactor Low Low Water Level Automatic Depressurization, is changed from ≥ 6’6", ≤ 6’10" above TAF to ≥ -48".

(10) Item D.2 in Table 3.2.2 (page 54), Reactor Low Low Water Level Diesel Generator, is changed from ≥ 6’6", ≤ 6’10" above TAF to ≥ -48". (page 54)

(11) Item 1 in Table 3.2.4 (page 59), Reactor Low Low Water Level Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation, is changed from ≥ 6’6", ≤ 6’10" above TAF to ≥ -48".

(12) Item 2 in Table 3.2.5 (page 60), Reactor Low Low Water Level Recirculation Pump Trip and Alternate Rod Injection, is changed from ≥ 6’6", ≤ 6’10" above TAF to ≥ -48".

(13) Item A.1 in Table 3.2.8 (page 60d), Reactor Low Low Water Level RCIC Initiation, is changed from ≥ 6’6", ≤ 6’10" above TAF to ≥ -48".

(14) Item B.a in Table 3.2.8 (page 60d), High Reactor Level HPCI/RCIC Turbine Shutdown, is redesignated Item B.1.

(15) Item B.1 in Table 3.2.8 (page 60d), High Reactor Level HPCI/RCIC Turbine Shutdown, is changed from ≤ 14’6" above TAF to ≤ -48".

(16) Item C.a in Table 3.2.8 (page 60d), Condensate Storage Tank Low Level Allowable Values, is redesignated Item C.1.

(17) The following changes have been proposed (page 64 Bases):

The low reactor water level instrumentation is set to trip when reactor water level is ~~7"~~ *>7"* on the instrument. This corresponds to a lower water level ~~above the top of active fuel~~ *inside the shroud* 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 active fuel~~ ≥ *-48"*. 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~~ *generators*.

3.1.2 Evaluation of Proposed Changes

The licensee has proposed changing the reference point for the reactor vessel level specification from “Top of Active Fuel” to instrument “Zero.” The original fuel design made the TAF specification an unambiguous physical point. The subsequent fuel loadings have included fuel stacks of various active fuel heights, ranging from 90.0" to 145.24." It has been proposed that a better reference would be the unambiguous location of the measuring instrument itself. The distance from instrument “Zero” to the top of the original, 144", active fuel stack is minus 126". Therefore, a setpoint of 6'-6" above the top of active fuel would translate to minus 48" (or -126"+78") relative to instrument zero location. The revised Technical Specification would simply state that the level setpoint is to be ≥-48". Other level specifications referenced to the top of active fuel are to be corrected accordingly.

It has also been proposed to remove the upper level specification on reactor vessel level to avoid unnecessary automatic actuation of engineered safeguards equipment and to make the specification consistent with the NRC Standard Technical Specifications, NUREG-1433.

These changes do not alter design requirements, but they better define them. Accordingly, the staff agrees with the proposed revisions of level specification.

3.2 Changes that Simplify Safety Limits and Limiting Safety System Settings by Relocating Specifications to other parts of the TS

3.2.1 Statement of Proposed Changes

(1) Revised and relocated Specification 2.1.A (page 6) to Specification 2.1.A.2

	Specification 2.1.A	Specification 2.1.A.2
Steam dome pressure	> 800 psia	≥ 785 psig
Core flow	> 10%	≥ 10%
Min Critical Power Ratio	MCPR < 1.11 for two recirculation loop operation or < 1.12 for one recirculation loop operation is a violation	MCPR shall be ≥ 1.10 for two recirculation loop operation or shall be ≥ 1.12 for single recirculation loop operation

The safety limit for steam dome pressure is changed from > 800 psia to ≥ 785 psig, converting pressure units from psia to psig by subtracting 15 psi, to conform to NUREG-1433.

Also, the specification for the MCPR safety limit is changed from a negative expression (i.e., a definition of what is needed to constitute a violation) to a positive expression (i.e., a definition of the acceptable operating range).

(2) Specification 2.3.A.1 (page 6) is revised and relocated to Table 3.1.1, Item 4

Specification 2.3.A.1

APRM for TLO	S ≤ 0.66W +65.6%
APRM for SLO	S ≤ 0.66(W - 5.4) + 65.6%
APRM	no greater than 120%

where APRM is average power range monitor scram
TLO is two recirculation loop operation
SLO is single recirculation loop operation
S is the setting in % rated thermal power (1775 MWt)
W is % of recirculation drive flow required to produce a core flow of 57.6 million lb/hr

Table 3.1.1, Item 4

Flow Referenced Neutron Flux APRM, High-High and Inoperative	≤ [0.66W + 65.6] %Rated Thermal Power
Flow Referenced Neutron Flux APRM, High-High and Inoperative	≤ [0.66(W - 5.4) + 65.6] %Rated Thermal Power
Flow Referenced Neutron Flux APRM, High Flow Clamp	≤ 120%

The statement of rated thermal power in the current specification, 1775 MWt, has been dropped in the proposed specification, since the rated thermal power is defined as the steady state power level corresponding to 1775 MWt in Section 1.0, Definitions, Item S.

(3) Revised and relocated Specification 2.1.B (page 7) to Specification 2.1.A.1

	Specification 2.1.B	Specification 2.1.A.1
Core Thermal Power	≤ 25% Rated Thermal Power when reactor steam dome pressure < 800 psia or core flow < 10% rated core flow	≤ 25% Rated Thermal Power when reactor steam dome pressure < 785 psig or core flow < 10% rated core flow

The setpoint is essentially unchanged, since 785 psig is equivalent to 800 psia.

(4) Specification 2.3.A.2 (page 7) is relocated to Table 3.1.1

	Specification 2.3.A.2	Table 3.1.1, Item 3
IRM - Flux Scram	≤ 20% of rated neutron flux	≤ 120/125 of full scale and ≤ 20% of rated neutron flux

IRM - Flux Scram corresponds to Table 3.1.1, Item 3, Neutron Flux IRM, for High-High and Inoperative, which is supplemented by a note indicating that an operable IRM channel must have its detector fully inserted.

(5) Revised and relocated Specification 2.2 (page 21) to Specification 2.1.B

	Specification 2.2	Specification 2.1.B
Reactor Coolant System	Reactor vessel pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel	Reactor steam dome pressure shall be $\leq$ 1332 psig

The licensee has proposed reducing the reactor coolant system pressure safety limit from 1335 psig to 1332 psig due to an incorrect application of the static head correction and a failure to consider the piping attached to the reactor vessel. When these factors are considered, the correct reactor coolant system pressure safety limit is slightly reduced, from 1335 psig to 1332 psig.

(6) Relocated Specification 2.4.B (page 21) to Specification 3.6.E

	Specification 2.4.B	Specification 3.6.E
Reactor Coolant System Safety Relief Valves (SRVs)	at least seven SRVs shall be operable. SRVs shall be set as follows: 8 valves at $\leq$ 1120 psig	SRVs shall be set as follows: 8 valves at $\leq$ 1120 psig

The setpoint remains the same, and the requirement that at least seven SRVs must be operable is present in Specification 3.6.E.

(7) The following changes are proposed (page 108):

3. The reactor may be started and operated, or operation may continue with only one recirculation
loop in operation provided that:

a. The following changes to setpoints and safety limit settings will be made within 24 hours after initiating operation with only one recirculation loop in operation.

1. The Operating Limit MCPR (MCPR) will be changed per Specification 3.1 1.C.

2. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed per Specification 3.1 1.A.

3. The APRM Neutron Flux Scram and APRM Rod Block setpoints will be changed as noted in ~~Specification 2.3.A and Table 3.2.3~~ *Tables 3.1.1 and 3.2.3.*

3.2.2 Evaluation of Proposed Changes

The licensee has proposed simplifying the safety limits and limiting safety system settings such that setpoints for the intermediate range monitor scram, the flow referenced average power range monitor scram, and safety/relief valves will be relocated from Section 2 to Section 3. The requirements are presented in Section 3, and the setpoints therein are basically unchanged.

Specification of the Thermal Power Rating is deleted, since it is defined elsewhere in the TS, and pressure units have been converted from psia to psig, in conformance with NUREG-1433, current Standard Technical Specifications. The licensee has also proposed reducing the reactor coolant system pressure safety limit from 1335 psig to 1332 psig due to an incorrect application of the static head correction and neglect of the design pressure of piping attached to the reactor vessel. When these factors are considered, the correct reactor coolant system pressure safety limit is slightly reduced from 1335 psig to 1332 psig. This reduction is acceptable.

The staff agrees with these simplifications.

3.3 <u>Changes that Simplify Safety Limits and Limiting Safety System Settings and Relocate Requirements to a Licensee-Controlled Document</u>

3.3.1 Statement of Proposed Change

Deleted Specification 2.1.C for power transients (page 7), which states, “To insure that the safety limit established in Specification 2.1.A is not exceeded, each required scram shall be initiated by its primary source signal as indicated by the plant process computer.”

3.3.2 Evaluation of Proposed Change

The licensee proposes that Specification 2.1.C be deleted. It has been superseded by commitments made to the NRC following the Salem ATWS event.

The guidance for the content of the TS provided by 10 CFR 50.36 contains the subjects to be included in the TS but not the specific requirements of the TS. The Commission provided guidance for the specific contents of the TS in the “Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors” (Final Policy Statement) (58 FR 39132, July 22, 1993). The Commission also concluded that compliance with the Final Policy Statement satisfied Section 182a of the Act. The Final Policy Statement indicates that certain items could be relocated from the TS to licensee-controlled documents. Furthermore, the policy statement encourages licensees to use the improved STS as the basis for complete conversions but also states that “licensees may adopt portions of the improved STS without fully implementing all STS improvements.” To ensure that the safety limit minimum critical power ratio is not exceeded, TS 2.1.C requires confirmation that each scram is initiated by its primary source signal as indicated by the plant process computer. This requirement stems from commitments made to the NRC regarding post-trip review, in response to NRC Generic Letter 83-28, “Required Actions Based on Generic Implications of Salem ATWS Events,” as noted in

USAR Section 14.8.1.2.[1] The requirement does not meet the criteria for inclusion in the TSs established by 10 CFR 50.36, and may be relocated to a licensee-controlled document. In this case, the licensee-controlled document is the USAR, and control is implemented via 10 CFR 50.59 procedures. The staff agrees that the Salem ATWS proposed change is acceptable.

3.4 Changes that Simplify Safety Limits and Limiting Safety System Settings and Delete Duplicate Requirements

3.4.1 Statement of Proposed Changes

(1) Deleted Specification 2.3.E (page 8), which requires that Turbine Control Valve Fast Closure Scram shall initiate upon loss of pressure at the acceleration relay with turbine first stage pressure $\geq$ 30%. Turbine Control Valve Fast Closure Scram is deleted, since it appears in Table 3.1.1, Item 11.

(2) Deleted Specification 2.3.F (page 8), which requires Turbine Stop Valve Scram $\leq$ 10% valve closure from full open with turbine first stage pressure $\geq$ 30%. Turbine Stop Valve Scram is deleted, since it appears in Table 3.1.1, Item 12.

(3) Deleted Specification 2.3.G (page 8), which requires Main Steamline Isolation Valve Closure Scram $\leq$ 10% valve closure from full open. Main Steamline Isolation Valve Closure Scram is deleted, since it appears in Table 3.1.1, Item 10.

(4) Deleted Specification 2.3.H (page 8), which requires Main Steamline Pressure Initiation of Main Steamline Isolation Valve Closure $\geq$ 825 psig. Main Steamline Pressure Initiation of Main Steamline Isolation Valve Closure is deleted, since it appears in Table 3.2.1, Item 1.d. The Trip Setting in Table 3.2.1 is $\geq$ 825 psig, with a note indicating that low pressure in main steam line only needs to be available in the RUN position.

(5) Deleted Specification 2.4.A, which requires Reactor Coolant High Pressure Scram when pressure $\leq$ 1075 psig. Reactor Coolant High Pressure Scram is deleted, since it appears in Table 3.1.1, Item 5. The Limiting Trip Setting in Table 3.1.1 is $\leq$ 1075 psig.

(6) Deleted reference to Bases Section 2.2 (page 217)

MCPR Limit is determined from the analysis of transients discussed in Bases Section~~s~~ 2.1 ~~and 2.2~~. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

[1] Commitments were made by the licensee in a letter from D. Musolf to the NRC dated November 14, 1983.

(7) Deleted references to Sections 3.2 and 3.5, and to Bases 2.2 (page 151)

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/ Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation). ~~Low Low Set and ADS functions are discussed further in Sections 3.2 and 3.5.~~

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III Nuclear Vessels requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

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, t~~*T*he 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

(8) In Specification 6.7.A.7.a (page 250), reference to Bases Specification 2.3.A is changed to reference Bases Specification 3.1.

3.4.2 Evaluation of Proposed Changes

The licensee has proposed to eliminate those settings, in Section 2, which are also specified in Section 3. The staff agrees with the specified elimination of duplicate requirements, since these are simplifications of the TS.

3.5 <u>Other changes</u>

3.5.1 Statement of Proposed Changes

(1) Change “design pressure” to “operating limit” (page, as follows:

E. Safety/Relief Valves
The reactor coolant system safety/relief valves .... The upper limit on safety/relief valve setpoint is established by the ~~design pressure~~ *operating limit* of the HPCI and RCIC systems *of 1120 psig*. The design capability of the HPCI and RCIC systems ....

(2) Replace Specification 6.4 with new Specification 2.2

	Specification 6.4	Specification 2.2
Action to be Taken if a Safety Limit is Exceeded	If a Safety Limit is exceeded, the reactor shall be shut down immediately. An immediate report shall be made to the Commission and to the corporate officer with direct responsibility for the plant or his designated alternate in his absence. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations by the Operations Committee, shall also be prepared. This report shall be submitted to the Commission, to the corporate officer with direct responsibility for the plant, and the Chairman of the Safety Audit Committee within 14 days of the occurrence. Reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission.	With any Safety Limit violation, the following actions shall be completed within 2 hours: A. Restore compliance with all Safety Limits, and B. Insert all insertable control rods

(3) Revise Specification 2.1.D (page 8) and Relocate to Specification 2.1.A.3

Specification 2.1.D	Specification 2.1.A.3
Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core. This level shall be continuously monitored whenever the recirculation pumps are not operating.	Reactor vessel water level shall be greater than the top of active irradiated fuel

3.5.2 Evaluation of Proposed Changes

The licensee has proposed correcting the basis for the upper limit on safety/relief valve setpoint based on the design pressure of the HPCI and RCIC systems to read “operating limit” rather than “design pressure.” This change will be consistent with the terminology used in the Monticello USAR. The staff agrees with the correction of the terminology used in the basis for specifying the upper limit on reactor coolant system safety/relief valve self actuation setpoint.

Section 6.4 is being replaced by new Section 2.2 and its associated bases. Administrative requirements contained in Section 6.4, which are covered in the Operational Quality Assurance Program, are deleted. In replacing Section 6.4, “Action to Be Taken if a Safety Limit is Exceeded,” with proposed new Section 2.2 and its associated Bases, the licensee proposes to delete those administrative actions in Section 6.4 that are redundant to the Operational Quality Assurance Program regarding the actions to be taken by corporate officers, the Operations

Committee, and the Safety Audit Committee when a safety limit is exceeded. The proposed change is consistent with the guidance of NRC Administrative Letter 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," and, therefore, is acceptable.

The reactor water level safety limit is lowered from 12 inches above the top of the active fuel when it is seated in the core to greater than the top of active irradiated fuel. During shutdown or refueling operations, decay heat is removed via the Residual Heat Removal (RHR) System, which takes suction from a reactor recirculation loop, which is located at an elevation that is lower than the top of the core. The RHR system can remove the decay heat as long as all of the active fuel elements in the core are covered with water. If reactor water level falls below the top of the active fuel during shutdown or refueling operations (e.g., during certain in-vessel inspection and maintenance activities), the RHR system's cooling capability decreases. The requirement for continuous level monitoring, whenever the recirculation pumps are not operating, is deleted, since this is in the requirements for operability of reactor vessel level instrumentation in Section 3 of the TSs. It is also in the Bases for Section 2.1, which states, "During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. Establishment of the safety limit above the top of the fuel provides adequate margin. The level will be continuously monitored whenever the recirculation pumps are not operating. "

During shutdown or refueling operations, a reactor vessel safety limit based upon covering all of the active fuel with water would assure that there is adequate core cooling, which would prevent fuel damage. The staff finds this revision acceptable, since a reactor vessel Safety Limit, based on covering all of the active fuel with coolant, would be sufficient to ensure the core is adequately cooled and fuel damage does not occur.

3.6 Conclusion

The licensee has proposed revisions to the TSs for Monticello to (1) orient level specifications to a reference that is less ambiguous than "top of active fuel," (2) delete duplicate requirements, (3) be more consistent with the NRC STS, and (4) simplify the TSs. The NRC staff finds the changes proposed to be 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 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. 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 (66 FR 38764). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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.

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