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DESCRIPTION:

Ltr trans the following....

ENCLOSURES:

Amdt to the OL...consist of prop changes to tech specs....concerning new temperature limitations for the suppression pool....

(40 cys encl rec'd)

PLANT NAME: Monticello

FOR ACTION/INFORMATION 3-27-75 JB

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Regulatory Docket File

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

RECEIVED MAR 26 1975 U.S. ATOMIC ENERGY COMMISSION

March 24, 1975

Mr A Giambusso, Director
Division of Reactor Licensing
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Giambusso:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

License Amendment Request Dated March 24, 1975

Attached are three signed originals and 37 conformed copies of a request for a change of Technical Specifications, Appendix A, of the Provisional Operating License for the Monticello Nuclear Generating Plant. The proposed changes specify new temperature limitations for the suppression pool, which were requested in a letter from Mr D L Ziemann (NRC) to Mr L O Mayer (NSP), dated February 14, 1975.

The temperature limits contained in the attached license amendment request are based on conservative engineering judgments and may be demonstrated to be unnecessarily restrictive by an analysis of the Steam Quenching Vibration Phenomenon specific for the Monticello plant. When the results of this analysis are available, the precise temperature limits for Monticello will be reflected in further requests for changes to the Technical Specifications.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

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3212

Attachments

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

Monticello Nuclear Generating Plant — Docket No. 50-263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR-22

(License Amendment Request Dated March 24, 1975)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L J Wachter
L J Wachter
Vice President, Power Production & System Operation

On this 24th day of March, 1975, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production & System Operation, and first being duly sworn acknowledged that he is authorized to execute this document in behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

John J. Smith
JOHH J. SMITH
Notary Public, Hennepin County, Minnesota
My Commission Expires March 3, 1976

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

AMENDMENT REQUEST DATED MARCH 24, 1975

PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
APPENDIX A OF OPERATING LICENSE NO. DPR-22

Pursuant to 10CFR50.59, the holders of the above mentioned license hereby propose the following changes to Appendix A Technical Specifications:

1. <u>Specification 3.7.A.1</u>

   <u>Proposed Changes</u>

   Change Specification 3.7.A.1 to read:

   1. Whenever primary containment is required, the volume and temperature of the water in the suppression chamber shall be maintained within the following limits:

      (a) Maximum water volume - 77,970 cubic feet.

      (b) Minimum water volume - 68,000 cubic feet.

      (c) Maximum water temperature

         i. During power operation, except as specified in item 3.7.A.1.(c).ii below - 90°F.

         ii. During testing which adds heat to the suppression pool - 100°F.

         iii. Following a scram from a condition where the suppression pool temperature is 90°F or less, the reactor pressure vessel shall be depressurized at normal cooldown rates to less than 200 psig if the suppression pool temperature exceeds 120°F.

         iv. During power operation if the temperature reaches 110°F the reactor shall be scrammed immediately and depressurized at normal cooldown rates to less than 200 psig.

(d) In order to continue power operation after exceeding the limit of 1(c)i, the suppression pool temperature must be reduced to 90°F within 24 hours. If the temperature cannot be reduced within this time limit, normal reactor shutdown procedures shall be initiated.

Reason for Change

To minimize the probability of steam quenching vibration in the suppression pool by precluding the development of elevated temperatures in the pool.

The temperature limits specified in this proposed change were based on conservative engineering judgments by the General Electric Company. An analysis is planned that will establish a definitive set of temperature limits based on the Monticello plant configuration. When this analysis is completed the new limits will be reflected in a further technical specification change request.

2. Specification 4.7.A.1

Proposed Changes

Change Specification 4.7.A.1 to read:

1. Checks and inspections of the suppression chamber structures, water level and temperature shall be conducted as follows:

   (a) Water level and temperature shall be checked once per day.

   (b) The interior painted surfaces above the water level shall be inspected at each refueling outage.

   (c) Whenever there is indication that there was relief valve operation with the temperature of the suppression pool exceeding 160°F and the reactor vessel pressure greater than 200 psig, an external visual examination of the pressure suppression chamber shall be conducted before resuming power operation.

Reason for Change

Specification 4.7.A.1 (c) has been added to provide assurance that no significant damage resulted from the occurrence described in the specification.

3. Bases 3.7.A

Proposed Changes

a. Change the next to last line of the third full paragraph on page 157 to read:

"...restricted by Specification 3.7.A.1(c) by limiting the suppression pool initial temperature and the..."

b. Add the following paragraph after the third paragraph on page 157:

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

Reason for Changes

To provide bases for the proposed changes to Specification 3.7.A.1.

4. Bases 4.7.A

Proposed Change

Add the following paragraph to page 161:

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.

Reason for Change

To provide bases for the proposed changes to Specification 4.7.A.1.

EXHIBIT B

Exhibit B attached, consists of newly prepared pages of Appendix A Technical Specifications as listed below. These pages incorporate the proposed changes contained in Exhibit A.

<u>Pages</u>

139
140
140A (New Page)
157
157A (New Page)
161

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment.

1. Whenever primary containment is required, the volume and temperature of the water in the suppression chamber shall be maintained within the following limits:

   (a) Maximum water volume - 77,970 cubic feet.

   (b) Minimum water volume - 68,000 cubic feet.

   (c) Maximum water temperature

      i. During power operation, except as specified in item 1.(c).ii below, 90°F.

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment.

1. Checks and inspections of the suppression structures, water level and temperature shall be conducted as follows:

   (a) Water level and temperature shall be checked once per day.

   (b) The interior painted surfaces above the water level shall be inspected at each refueling outage.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

ii. During testing which adds heat to the suppression pool - 100° F.

iii. Following a scram from a condition where the suppression pool temperature is 90°F or less, the reactor pressure vessel shall be depressurized at normal cooldown rates to less than 200 psig if the suppression pool temperature exceeds 120°F.

iv. During power operation if the temperature reaches 110°F the reactor shall be scrammed immediately and depressurized at normal cooldown rates to less than 200 psig.

(d) In order to continue power operation after exceeding the limit of 1.(c).i, the suppression pool temperature must be reduced to 90°F within 24 hours. If the temperature cannot be reduced within this time limit, normal reactor shutdown procedures shall be initiated.

(c) Whenever there is indication that there was relief valve operation with the temperature of the suppression pool exceeding 160°F and the reactor vessel pressure greater than 200 psig, an external visual examination of the pressure suppression chamber shall be conducted before resuming power operation.

2. Primary containment integrity as defined in the Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212° F and fuel is in the reactor vessel except while performing low power physics test at atmospheric pressure during or after refueling at power levels not to exceed 5 Mw(t).

2. The primary containment integrity shall be demonstrated as follows:

   (a) Integrated Primary Containment Leak Test (IPCLT)

      (1) An integrated leak rate test shall be performed prior to initial unit operation at an initial test pressure (Pt) of 41 psig.

      (2) Subsequent leak rate tests shall be performed without preliminary leak detection surveys or leak repairs immediately prior to or during the test, at an initial pressure of approximately 41 psig.

      (3) Leak repairs, if necesssary to permit integrated leak rate testing, shall be preceded by local leak rate measurements where possible. The leak rate differ-

Bases Continued:

3.7 A. Primary Containment

length of four feet, which resulted in complete condensation. Thus with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humboldt Bay [1] and Bodega Bay[2] tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

In order not to exceed 170°F, immediately following a hypothetical loss of coolant accident, the maximum suppression pool temperature must not exceed 130°F during RCIC operation, assuming the minimum suppression pool mass permitted by these tech specs. Normally the pool temperature will be less than 90°F and will only rise during RCIC operation or testing of the HPCI system.

For an initial maximum suppression chamber water temperature of 90°F and assuming the normal complement of containment cooling pumps (2 LPCI pumps and 2 containment cooling service water pumps) containment pressure is not required to maintain adequate net positive suction head (NPSH) for the core spray, LPCI and HPCI pumps. However, during an approximately one-day period starting a few hours after a loss-of-coolant accident, should one RHR loop be inoperable and should the containment pressure be reduced to atmospheric pressure through any means, adequate NPSH would not be available. Since an extremely degraded condition must exist, the period of vulnerability to this event is restricted by Specification 3.7.A.1.c by limiting the suppression pool initial temperature and the period of operation with one inoperable RHR loop.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the pressure suppression pool is maintained below 160° F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

(1) Robbins, C. H., "Tests of Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

Bases Continued

3.7 A. Primary Containment

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.

Bases:

4.7 A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a weekly check of the temperature and volume is adequate to assure that adequate heat removal capability is present. For additional margin, these will be checked once per day.

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.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major 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 is adequate.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss of coolant accident. The peak drywell pressure would be about 41 psig, which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay. See Section 5.2.3 FSAR.

The design pressure of the drywell and absorption chamber is 56 psig. See Section 5.2.3 FSAR. The design leak rate is 0.5%/day at a pressure of 56 psig. as indicated above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 41 psig. The analysis showed that with this leak