50-263

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

July 25, 1997

Mr. Roger O. Anderson, Director
Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: UPDATED ANALYSIS OF DBA CONTAINMENT TEMPERATURE AND PRESSURE RESPONSE AND RELIANCE ON CONTAINMENT PRESSURE TO COMPENSATE FOR POTENTIAL DEFICIENCY IN NPSH FOR ECCS PUMPS DURING DBA (TAC NO. M97781)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 98 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment is in response to your application dated January 23, 1997, as supplemented January 28, March 4, June 19, July 2, July 16 (2 letters), July 21, and July 25, 1997.

The amendment evaluates the apparent unreviewed safety questions associated with (1) the updated analysis of the design-basis accident (DBA) containment temperature and pressure response, and (2) the reliance on containment pressure to compensate for the potential deficiency in net positive suction head (NPSH) for the emergency core cooling system (ECCS) pumps during a DBA with the worst-case scenario assumptions. This amendment also authorizes Northern States Power Company (NSP) to change the Technical Specification bases and the Updated Safety Analysis Report to reflect the reliance on containment pressure to compensate for the potential deficiency in NPSH for the ECCS pumps following a DBA.

As an administrative action by the Commission, which only involves the format of the license and does not authorize any activities outside the scope of your application and supplements, the NRC has amended the license to include a new paragraph 2.C.8, "Additional Conditions," and an Appendix C, which lists additional license conditions that reflect specific commitments made in your submittals. Adding these license conditions to Appendix C was discussed with your staff, and by letter dated July 25, 1997, you agreed to these license conditions.

These additional conditions are as follows:

Prior to starting up the plant from the current maintenance outage, NSP will:

(1) revise the emergency operating procedures (EOPs) to require manual isolation of torus and drywell sprays prior to the point where primary containment pressure would not provide adequate NPSH for the ECCS pumps, change the caution statement regarding NPSH in the Primary Containment Pressure EOP to include the core spray pumps, and add a caution statement regarding NPSH considerations for pressure control while venting to control primary containment pressure.

(2) finalize the additional containment analysis and associated NPSH evaluation which extends the existing long-term case evaluation to the time when the required containment overpressure returns to atmospheric conditions. Changes to the requested long-term containment overpressure, if any, will be promptly reported to the staff prior to startup.

Within 90 days of the date of plant startup from the current maintenance outage, NSP will:

(3) submit the results of the additional containment analysis associated with (2) above.

(4) Update Section 5.2 of the Updated Safety Analysis Report by incorporating Figure E.2 of the NSP submittal dated July 16, 1997.

Within 180 days of the date of plant startup from the current maintenance outage, NSP will:

(5) process a 10 CFR 50.59 evaluation to change the EOP definition of adequate core cooling to 2/3 core height. The corresponding EOP changes and the required operator training will also be completed in this time period. Final implementations will be completed when all the 10 CFR Part 50.59 evaluation requirements are satisfied.

In addition, NSP committed to submit a change to the Boiling Water Reactor Owners Group EOP Committee for evaluation and resolution of the EOP spray limits in regard to adequate NPSH for ECCS pumps.

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,

ORIGINAL SIGNED BY

Tae Kim, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures: 1. Amendment No. 98 to DPR-22
2. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\WPDOCS\MONTICEL\MON97781.AMD *SEE PREVIOUS CONCURRENCES

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:PD31	E	LA:PD31	E	BC:SPLB		BC:EMEB		BC:EELB	
NAME	TJKim		CJamerson		LBMarsh*		RWessman*		JCalvo*	
DATE	7/25/97		7/25/97		7 /24/97		7 /24 /97		7 /24/97	

OFFICE	BC:SCSB		BC:SRXB		OGC		D:PD31	
NAME	CBerlinger*		JLyons*		[illegible]		JHannon	
DATE	7 /25/97		7/24 /97		7/25/97		7/25/97	

OFFICIAL RECORD COPY

Mr. Roger O. Anderson, Director
Northern States Power Company

Monticello Nuclear Generating Plant

cc:

J. E. Silberg, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington DC 20037

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
2807 W. County Road 75
Monticello, Minnesota 55362

Plant Manager
Monticello Nuclear Generating Plant
ATTN: Site Licensing
Northern States Power Company
2807 West County Road 75
Monticello, Minnesota 55362-9637

Robert Nelson, President
Minnesota Environmental Control
Citizens Association (MECCA)
1051 South McKnight Road
St. Paul, Minnesota 55119

Commissioner
Minnesota Pollution Control Agency
520 Lafayette Road
St. Paul, Minnesota 55119

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Commissioner of Health
Minnesota Department of Health
717 Delaware Street, S. E.
Minneapolis, Minnesota 55440

Darla Groshens, Auditor/Treasurer
Wright County Government Center
10 NW Second Street
Buffalo, Minnesota 55313

Kris Sanda, Commissioner
Department of Public Service
121 Seventh Place East
Suite 200
St. Paul, Minnesota 55101-2145

Adonis A. Neblett
Assistant Attorney General
Office of the Attorney General
445 Minnesota Street
Suite 900
St. Paul, Minnesota 55101-2127

DATED: July 25, 1997

AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. DPR-22-MONTICELLO

Docket File
PUBLIC
PDIII-1 Reading
J. Roe
E. Adensam (EGA)
C. Jamerson
T. Kim (2)
OGC
G. Hill (4)
W. Beckner
J. Kudrick
K. Kavanagh
A. Gill
R. Elliott
C. Hammer
D. Nugyen
ACRS
J. M-Barger, DRP/RIII
SEDB (TLH3)

DFol

11

000003

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated January 23, 1997, as supplemented January 28, March 4, June 19, July 2, July 16 (2 letters), July 21, and July 25, 1997, 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 adding paragraph 2.C.8 to Facility Operating License No. DPR-22 as follows:

   2.C.8. Additional Conditions

   The Additional Conditions contained in Appendix C, as revised through Amendment No. 98, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of its date of issuance. Implementation is as specified in Appendix C.

FOR THE NUCLEAR REGULATORY COMMISSION

Tae Kim

Tae Kim, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments: 1. Page 5 of the License*
2. Bases pages 112, 113, and 176

Date of Issuance:

*Page 5 and Appendix C of the license are attached, for convenience, for the composite license to reflect this change.

8. Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 98 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

D. Northern States Power Company shall immediately notify the NRC of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

E. Northern States Power Company shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

F. The licensee shall observe such standards and requirements for the protection of the environment as are validly imposed pursuant to authority established under Federal and State law and as determined by the Commission to be applicable to the facility covered by this facility operating license.

G. This license is effective as of the date of issuance and shall expire at midnight, September 8, 2010.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by: Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Attachments:
1. Appendix A - Technical Specifications
2. Appendix B - (Deleted per Amendment 15, 12/17/82)
3. Appendix C - Additional Conditions

Date of Issuance: January 9, 1981

APPENDIX C

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-22

Northern States Power Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
98	The emergency operating procedures (EOPs) shall be changed to require manual isolation of torus and drywell sprays prior to the point where primary containment pressure would not provide adequate net positive suction head (NPSH) for the emergency core cooling system (ECCS) pumps, change the caution statement regarding NPSH in the Primary Containment Pressure EOP to include the core spray pumps, and add a caution statement regarding NPSH considerations for pressure control while venting to control primary containment pressure.	Prior to starting up from the current maintenance outage, or August 1, 1997, whichever is later.
98	Finalize the additional containment analysis and associated NPSH evaluation which extends the existing long-term case evaluation to the time when the required containment overpressure returns to atmospheric conditions. Changes to the requested long-term containment overpressure, if any, shall be promptly reported to the NRC prior to starting up the unit from the current maintenance outage.	Prior to starting up from the current maintenance outage, or August 1, 1997, whichever is later.
98	Submit the results of the additional containment analysis and associated NPSH evaluation discussed above.	Within 90 days from the date of plant startup from the current maintenance outage, or November 1, 1997, whichever is later.

Amendment Number	Additional Condition	Implementation Date
98	Update Section 5.2 of the Updated Safety Analysis Report by incorporating Figure E.2 of the NSP submittal dated July 16, 1997.	Within 90 days from the date of plant startup from the current maintenance outage, or November 1, 1997, whichever is later.
98	Process a 10 CFR 50.59 evaluation to change the EOP definition of adequate core cooling to 2/3 core height. The corresponding EOP changes and the required operator training shall also be completed. Final implementations shall be completed when all the 10 CFR 50.59 evaluation requirements are satisfied.	Within 180 days from the date of plant startup from the current maintenance outage, or February 1, 1998, whichever is later.

automatically controls three selected safety-relief valves although the safety analysis only takes credit for two valves. It is therefore appropriate to permit one valve to be out-of-service for up to 7 days without materially reducing system reliability.

B. RHR Intertie Line

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

C. Containment Spray/Cooling Systems

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

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

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

The surveillance requirements provide adequate assurance that the containment spray/cooling system will be operable when required. The head and flow requirements specified for the RHR service water pumps provide assurance that the minimum required service water flow can be supplied to the RHR heat exchangers for the most degraded condition for long-term containment heat removal following the design basis loss of coolant accident.

D. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

The surveillance requirements provide adequate assurance that the RCIC system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

E. Cold Shutdown and Refueling Requirements

The purpose of Specification 3.5.E is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment spray/cooling subsystems may be out of service. This specification allows all core and containment spray/cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.E.2 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may by necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

Bases Continued:

Vent system downcomer submergence is three feet below the minimum specified suppression pool water level. This length has been shown to result in reduced postulated accident loading of the torus while at the same time assuring the downcomers remain submerged under all seismic and accident conditions and possess adequate condensation effectiveness.[3]

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.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the 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 suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

For an initial maximum suppression chamber water temperature of 90°F and conditions which lead to minimum containment pressure, adequate net positive suction head (NPSH) is maintained for the core spray, RHR, and HPCI pumps under loss of coolant accident conditions. Analyses were performed for a broad range of pump combinations and failure modes to define the minimum amount of containment pressure available to provide adequate NPSH in the short and long term. Refer to Section 5.2.3.3 of the USAR for a discussion of these analyses and figures which demonstrate graphically the amount of pressure required and the minimum containment pressure available to supply the required NPSH for the emergency core cooling pumps in the limiting pump combinations evaluated. No pump cavitation will occur over either the short or long term periods under conditions resulting in minimum containment pressure.

(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.

(3) General Electric NEDE-21885-P, "Mark I Containment Program Downcomer Reduced Submergence Functional Assessment Report," June, 1978.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 98

FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated January 23, 1997, as supplemented January 28, March 4, June 19, July 2, July 16 (2 letters), July 21, and July 25, 1997, (Ref. 1-9), Northern States Power Company (NSP, the licensee) submitted a proposed license amendment requesting review and approval of the apparent unreviewed safety questions (USQs) associated with (1) the updated analysis of the design-basis accident (DBA) containment temperature and pressure response, and (2) the reliance on containment pressure to compensate for the potential deficiency in net positive suction head (NPSH) for the emergency core cooling system (ECCS) pumps during a DBA with the worst-case scenario assumptions. This proposed amendment also would authorize the licensee to change the Technical Specification bases and the Updated Safety Analysis Report (USAR)(Ref. 10) to reflect the reliance on containment pressure to compensate for the potential deficiency in NPSH for the ECCS pumps following a DBA.

The June 19, 1997, submittal (Ref. 4) expanded the scope of the initial submittal dated January 23, 1997, and therefore, another proposed no significant hazards considerations determination was issued by the staff based on the June 19, 1997, submittal (62 FR 34086). The July 2, July 16 (2 letters), July 21, and July 25, 1997, submittals provided additional clarifying information within the scope of the application and did not change the NRC staff's proposed no significant hazards considerations determination based on the June 19, 1997, submittal.

2.0 BACKGROUND

During a design-basis reconstitution effort in 1992, the licensee discovered inconsistencies in the assumptions used in General Electric (GE) report NEDO-30485, titled "Monticello Design Basis Accident Containment Pressure and Temperature Response for FSAR Update," December 1983, (Ref. 11) with respect to the most limiting active single-failure criterion. The licensee, through GE, issued a revised report NEDO-32418, "Monticello Nuclear Generating Plant Basis Accident Containment Pressure and Temperature Response for USAR Update,"

9708080136 970725
PDR ADOCK 05000263
P PDR

in December 1994 (Ref. 12). NEDO-32418 demonstrated ample margins to containment design limits for long-term containment heat removal with the correct set of assumptions. The licensee updated Section 5.2.3.3 of the Monticello USAR with the results of NEDO-32418 and reported to the NRC in the periodic report changes, tests and, experiments in accordance with 10 CFR 50.59 on April 20, 1995.

A System Operational Performance Inspection (SOPI) of the Monticello residual heat removal (RHR) system was completed by an NRC Region III inspection team on January 8, 1997. The inspection team identified an apparent USQ related to the containment pressure and temperature analysis in NEDO-32418, the results of which were incorporated in the USAR. The long-term containment heat removal evaluation in NEDO-32418 used the ANS 5.1-1979 decay heat model, without considerations for statistical uncertainties, and the results indicated a slightly higher peak suppression pool temperature relative to the results in the previous analysis, NEDO-30485. NEDO-30485 had been submitted to the NRC in 1986, and it used the May-Witt decay heat model, which the staff considers more conservative.

The inspection team also questioned the meaning of Technical Specification (TS) bases Section 3.5/4.5.C. This TS bases section was interpreted by the inspection team to state that two RHR and two RHRSW [residual heat removal service water] pumps are required to perform the containment spray/cooling function. In the most limiting case, however, only one RHR pump and one RHRSW pump are assumed to be available to perform the containment cooling function in the event of the worst-case single failure for suppression pool cooling (loss of diesel generator with loss of offsite power). In response to this inspection finding, the licensee requested the NRC's review and approval of the revised GE report NEDO-32418, by letter dated January 23, 1997, as supplemented January 28, and March 4, 1997.

The NRC SOPI inspection team further noted that reliance on containment overpressure for NPSH has been the topic of several NRC generic communications. The team reviewed the licensee's previous NPSH analyses and determined that the amount of containment overpressure that may be credited in NPSH evaluations was not clearly established. This was identified as an unresolved item in the inspection report (Ref. 13).

During a series of discussions with the NRC following the inspection, the licensee has maintained that the original design basis of Monticello assumed an elevated pressure in the containment following a postulated DBA for NPSH considerations. Many similar vintage boiling water reactors (BWR) were constructed with ECCS designs that use ECCS pumps and pump locations that do not provide as much NPSH margin as later designs. Monticello is an early vintage plant and the design does not include the additional margin that is available in later designs. However, based on its review of Monticello's licensing basis, the staff determined that the assumption of an elevated post-accident pressure in the suppression chamber was not credited, and therefore, the Monticello's licensing basis does not allow reliance on containment overpressure.

On April 15, 1997, the licensee notified the NRC staff that the NPSH available to the core spray pumps may not meet the required NPSH under all accident conditions. During a review of the ECCS pump NPSH requirements, the licensee calculated a new higher head loss, approximately 3.6 meters (11.7 feet) per 630.9 L/s [liters per second] (10,000 gpm) versus 0.3 meter (1-foot) per 630.9 L/s (10,000 gpm), for clean ECCS suction strainers. The specific scenario of concern involved a failure of the low pressure coolant injection (LPCI) loop select logic to select the intact reactor recirculation loop. On May 9, 1997, the licensee decided to shut down the reactor and replace the ECCS suction strainers.

By letter dated June 19, 1997, as supplemented July 2, July 16 (2 letters), July 21, and July 25, 1997, the licensee requested changes to Monticello's licensing basis to allow credit for a limited amount of containment overpressure to compensate for a slight increase in the NPSH deficiency post-design-basis accident. The following review evaluates the use of containment overpressure with the new suction strainers installed.

3.0 EVALUATION

3.1 Evaluation of the USQ

The proposed license amendment requested review and approval of the apparent USQs associated with (1) the updated analysis of the DBA containment temperature and pressure response, and (2) the reliance on containment pressure to compensate for the potential deficiency in NPSH for the ECCS pumps during a DBA with the worst-case scenario assumptions. This proposed amendment also would authorize the licensee to change the TS bases and the USAR to reflect the reliance of containment pressure to compensate for the potential deficiency in NPSH for the ECCS pumps following a DBA.

As documented in its letters dated July 16, July 21, and July 25, 1997, the licensee has made commitments to revise the emergency operating procedures to address the NPSH considerations during a DBA. The licensee also committed to finalize additional containment analysis and associated NPSH evaluations which extends the existing long-term case evaluations. The licensee's commitments are incorporated into the MNGP operating license as additional license conditions.

3.2 Containment Pressure and Temperature

In its January 23, 1997, submittal the licensee submitted the results and input assumptions of analyses performed with the HXSIZ computer code to determine the long-term containment response contained in the GE report NEDO-32418. In this report, GE used ANS 5.1-1979 decay heat model with no added uncertainty to calculate decay heat. The staff has determined previously that for containment response analyses, a 2-sigma uncertainty should be added to the decay heat calculated by the ANS 5.1-1979 model. The basis for this determination is that the ANS 5.1-1979 model is derived from a best-estimate methodology and thus deviates from the conservative models and methodologies typically required by the staff for DBA analysis. A +2-sigma (i.e., 2 standard deviations) uncertainty corresponds to a 95 percent confidence, i.e., there is a 95 percent statistical confidence that the decay heat calculated by the model will fall within the envelope defined by the calculated decay heat plus 2-sigma.

Because of the staff's determination concerning the use of a +2-sigma uncertainty addition, the licensee submitted the results of additional minimum containment pressure and peak suppression pool temperature analyses at a power level of 102 percent of 1880 MWt [megawatts thermal], which is approximately 115 percent of the currently licensed power level, to provide assurance that the results are conservative. The assumption of reactor operation at 1880 MWt conservatively bounds the calculated core shutdown power that would result from the use of ANS 5.1 decay heat model with a 2-sigma uncertainty adder at 102 percent of 1670 MWt (currently licensed power level).

In its June 19, 1997, submittal, the licensee submitted the results and input assumptions of analyses performed with the SHEX-04 computer code to predict the minimum containment pressure and peak suppression pool temperature resulting from a DBA-LOCA [loss of coolant accident]. Various cases incorporating different degrees of mixing in the containment atmosphere and the effect of containment sprays were analyzed to determine the most limiting cases, regarding NPSH, for the short- and long-term containment response, and to predict the peak suppression pool temperature.

3.2.1 Minimum Containment Pressure/ Maximum Suppression Pool Temperature Analyses

The licensee has analyzed seven cases with varying accident scenarios, five for the long-term and two for the short-term. Based on these analyses, the licensee has requested credit for the following amounts of containment overpressure to satisfy RHR pump and core spray (CS) pump NPSH requirements:

Time Period (seconds)	Containment Overpressure (psig)
10 - 600	2.0
600 - 2,000	2.0
2000 - 10,000	4.0
10,000 - 16,000	5.3
16,000 - 55,000	6.1
55,000 - 69,000	5.6
69,000 - 85,000	5.0
85,000 - 110,000	4.2
110,000 - 140,000	3.3
140,000 - 200,000	2.3
200,000 - 330,000	1.0

The minimum containment pressure analysis conducted by the licensee contains modeling assumptions and input parameters that tend to reduce the predicted post-LOCA containment pressure, thereby providing conservatism in determining how much overpressure can be credited for NPSH.

The short term is defined as the time from the start of the LOCA out to 600 seconds. The long term analysis begins at 600 seconds, the time at which manual operator actions can be credited for throttling ECCS pump flows and initiating containment cooling via drywell/wetwell spray or suppression pool cooling. These analyses varied the degree of thermal mixing between break liquid and containment atmosphere, and also examined different LPCI and CS pump combinations and pump flows, to determine the case that produced the minimum credible containment pressure. The amount of thermal mixing affects the degree of heat removal from the containment atmosphere, while different combinations of pump flows affect the mass and energy released from the break and how much break flow is available for mixing.

In its June 19, 1997, submittal, the licensee listed the input assumptions and parameters common to the SHEX analyses for minimum containment pressure and peak suppression pool temperature. These are as follows:

- The reactor is assumed to be operating at about 115 percent of the rated thermal power to conservatively account for uncertainties in ANS 5.1-1979 decay heat model, except for Case 1 which assumes an initial power of 102 percent of the rated thermal power
- Use of ANS 5.1-1979 decay heat model, without uncertainty additions, to calculate decay heat (the assumption of power operation at about 115 percent of the rated thermal power bounds +2-sigma uncertainty)
- Vessel blowdown flow rates are based upon the Homogeneous Equilibrium Model
- Feedwater flow continues into the reactor until all hot feedwater which maximizes the suppression pool temperature is injected into the vessel
- Thermodynamic equilibrium exists between liquids and gases in the drywell
- The vent system flow to the suppression pool consists of a homogeneous mixture of the fluid in the drywell
- The initial suppression pool volume is at the minimum TS level to maximize the calculated suppression pool temperature
- The initial drywell and suppression chamber pressure are at the minimum expected operating values of 1.0 psig [pounds square inch gauge] and 0 psig, respectively, to minimize containment pressure
- The maximum operating value of the drywell temperature of 150 degrees Fahrenheit and a relative humidity of 100 percent are used to minimize

the initial non-condensible gas mass and to minimize the long-term containment pressure for the NPSH evaluation

- The drywell and torus condensation heat transfer coefficients are based on the Uchida correlation with a 1.2 multiplier
- CS and LPCI/containment cooling system pumps have 100 percent of their horsepower rating converted to a pump heat input added either to the reactor vessel input or suppression pool water
- Containment leakage is not included in the analyses
- The initial suppression pool temperature is at the maximum TS value to maximize the calculated suppression pool temperature
- The initial suppression chamber airspace temperature is at 90 degrees Fahrenheit and the relative humidity is at 100 percent
- The RHRSW temperature is at the maximum allowable value of 90 degrees Fahrenheit to maximize the calculated suppression pool temperature

The case that predicted the minimum containment pressure for the first 600 seconds assumed a postulated break in the recirculation discharge line with all four LPCI pumps and two CS pumps available for vessel injection and with the assumed single failure of the LPCI Loop Select Logic to select the unbroken reactor recirculation loop. In this case, all four LPCI pumps are assumed to be injecting into the broken recirculation loop and subsequently directed into the drywell. The LPCI pumps and the core spray pumps are at a maximum flow condition with no credit for operator action to throttle their flow.

This case resulted in the minimum containment pressure and the maximum suppression pool temperature during the first 10 minutes of an accident when operator actions are not credited. This event is therefore considered to be limiting with respect to NPSH margins for the first 10 minutes of the accident. Two cases were analyzed by the licensee: Case 1 was analyzed with the current rated thermal power of 1670 MWt, and Case 2 was analyzed with a bounding thermal power of 1880 MWt. A 100 percent thermal mixing efficiency between the liquid break flow and the drywell atmosphere was assumed to minimize the suppression chamber airspace pressure (Cases 1 and 2 as identified in the licensee's June 19, 1997, submittal.)

The minimum pressure predicted from the licensee's short-term analysis is 16.65 psia [pounds square inch absolute] for the current power level of 1670 MWt (Case 1) and 16.86 psia for the power level of 1880 MWt (Case 2). The maximum predicted short-term suppression pool temperature is 148.2 and 149.1 degrees Fahrenheit for Cases 1 and 2, respectively, at 600 seconds.

The case that predicted maximum suppression pool temperature for the long-term assumed a double-ended break of the LPCI recirculation suction line with no offsite power and the assumed failure of one diesel generator. For this case, (Case 3 as identified in the licensee's June 19, 1997, submittal), there is only one RHR pump and one RHRSW pump available for long term containment cooling. The minimum pressure predicted from the long term analysis is 31.61 psia for the period from 600 seconds to accident termination and 21.13 psia at the maximum predicted suppression pool temperature of 194.2 degrees Fahrenheit for NPSH purposes.

The staff has reviewed the licensee's minimum containment pressure and maximum suppression pool analysis conducted for the purpose of crediting containment overpressure to satisfy NPSH requirements for the LPCI and CS pumps. The staff finds that the licensee has used input and modeling assumptions that minimize the containment pressure and maximize suppression pool temperature and has investigated a sufficient number of cases such that the case that produces the maximum suppression pool temperature concurrent with the limiting NPSH condition has been identified.

In its letters dated July 16 and July 21, 1997, the licensee has made commitments to finalize the additional containment analysis and associated NPSH evaluation which extends the existing long-term case evaluation to the time when the required containment overpressure returns to atmospheric conditions. Changes to the requested long-term containment overpressure, if any, will be promptly reported to the staff prior to startup. In addition, the licensee committed to submit the results of the additional containment analysis.

3.2.2 Containment Sprays

According to the current Monticello emergency operating procedures (EOPs), manual initiation of containment sprays would occur at 12 psig containment pressure, and manual shutoff is directed by the EOPs at 2 psig. Because of concerns with the sprays and the pressure reduction they achieve, by letter dated July 16, 1997, the licensee has committed to change the Monticello EOPs to alert operators to NPSH concerns and to make containment spray operation consistent with the overpressure requirements for NPSH. This will be accomplished by directing operators to terminate containment spray operation at a sufficiently elevated containment pressure such that containment overpressure for NPSH will be present and adequate NPSH margin for ECCS pumps will be ensured. Through training, operators will also be informed of the elevated importance of NPSH, and of the alternate containment spray setpoints. Consideration will also be given to the spray initiation setpoint so that undesirable toggling of the sprays will not occur. The licensee also committed to submit the proposed changes to the BWR Owners Group (BWROG) for evaluation and resolution. The staff concurs with the licensee that the changes to the EOPs increase overall safety.

3.2.3 ANS 5.1-1979 Decay Heat

The current licensing basis calculations for Monticello are based on the use of the May-Witt decay heat model, which is recognized by the staff as conservative and which predicts substantially higher values of decay heat than the ANS 5.1-1979 standard. The staff has determined previously that for containment response analyses, a 2-sigma uncertainty should be added to the decay heat calculated by the ANS 5.1-1979 model. The basis for this determination is that the ANS 5.1-1979 model is derived from a best-estimate methodology, and thus deviates from the conservative models and methodologies typically required by the staff for DBA analysis. A +2-sigma (i.e., 2 standard deviations) uncertainty corresponds to a 95 percent confidence, i.e., there is a 95 percent statistical confidence that the decay heat calculated by the model will fall within the envelope defined by the calculated decay heat plus 2-sigma.

Because of the staff's determination concerning use of a +2-sigma uncertainty addition, the licensee submitted the results of additional minimum containment pressure and peak suppression pool temperature analyses at a power level of 102 percent of 1880 MWt, which is approximately 115 percent of the currently licensed power level, to provide assurance that the results are conservative. The assumption of reactor operation at 1880 MWt conservatively bounds the calculated core shutdown power that would result from the use of ANS 5.1 decay heat model with a 2-sigma uncertainty adder at 102 percent of 1670 MWt (currently licensed power level). It should be noted that the staff has not evaluated the acceptability of the ANS 5.1-1979 decay heat model itself, but rather the acceptability of results from the ANS 5.1-1979 model, with conservative assumptions added to account for statistical uncertainties, when compared to the previously approved values.

3.2.4 SHEX Benchmark

The licensee benchmarked GE's SHEX code against the current DBA-LOCA containment analyses in the USAR using the HXSIZ code. This benchmarking was performed to assess the differences between the USAR and SHEX analytical results produced as a result of the SHEX code and the modeling features inherent to the code. These analyses were provided to the staff in the submittal dated June 19, 1997.

It should be noted that the HXSIZ code has certain limitations which inhibit its use other than for modeling the long-term response for the DBA-LOCA with assumptions that maximize the drywell and suppression chamber airspace pressure. Therefore, the licensee's validation process was intended to demonstrate that the SHEX and HXSIZ codes produce similar results (suppression pool temperature and suppression chamber airspace pressure) for the DBA-LOCA with consistent assumptions which maximize the suppression chamber airspace pressure.

The staff has reviewed the licensee's benchmark analysis for the SHEX code and finds that the long-term suppression pool temperature and suppression chamber airspace pressure responses calculated with the SHEX code are consistent with the HXSIZ results. The comparison also shows that the SHEX code allows a more accurate prediction of the containment pressure and temperature response for the entire event duration. The additional features in the SHEX such as the modeling of vacuum breakers, heat sinks and containment sprays allow for a better prediction capability for a variety of events that could not be modeled with the HXSIZ code.

The peak long-term containment temperature predicted by SHEX/ANS 5.1-1979 was 184.8 degrees Fahrenheit compared to approximately 184 degrees Fahrenheit predicted by HXSIZ/ANS 5.1-1979. The peak long-term containment temperature predicted by SHEX/May-Witt was 196.7 degrees Fahrenheit compared to approximately 195.5 degrees Fahrenheit predicted by HXSIZ/May-Witt.

A comparison of the secondary long-term peak containment pressures shows close comparison (≤ 1 psi) between the results obtained with HXSIZ and SHEX. However, there is a large difference in predicted containment pressure between 600 seconds and approximately 10,000 seconds. The licensee attributed this difference to the more general and simplifying assumptions used in the HXSIZ code. These include, in part, the assumptions that the vessel temperature and drywell temperature are equal and that the drywell and suppression chamber airspace pressures are equal.

The licensee has evaluated the differences in using the SHEX code to analyze minimum containment pressure versus maximum containment pressure in Table A-1 in Exhibit D of the licensee's submittal dated June 19, 1997. Table A-1 provides a comparison between Case A-1, which uses assumptions based on maximizing containment pressure for the DBA-LOCA analysis, and Case 3, which uses assumptions based on minimizing containment pressure for the DBA-LOCA analysis. The assumptions used in Case A-1 are similar to the assumptions used for Case 3. Differences in assumptions between the two cases include (1) initial drywell pressure and initial suppression chamber airspace pressure, (2) initial drywell relative humidity, (3) containment cooling mode, (4) heat and mass transfer between the suppression pool and suppression chamber air space, and (5) thermal mixing efficiency between break flow and drywell atmosphere.

While these differences between the two cases are arguably minor, the staff has raised questions regarding the validity of using the SHEX code to analyze the minimum containment pressure cases. In its submittal dated July 21, 1997, the licensee provided a similarity argument which compared the use of SHEX at Monticello against the use of SHEX at Dresden. By Attachment A to a letter dated February 27, 1997 (Ref. 14), Commonwealth Edison provided a benchmark analysis for the SHEX code for the minimum containment pressure cases. The SHEX code was shown to give conservative results with respect to calculated containment pressure. This benchmarking analysis was subsequently accepted by the staff in a staff safety evaluation dated April 30, 1997 (Ref. 15).

The licensee stated that the same version of the SHEX code (04) that was used for the Dresden analysis was used for the Monticello minimum pressure analysis. Within SHEX, the same spray modelling was used for both Dresden and Monticello. One hundred percent of spray efficiency was assumed for both plant analyses. Although the ECCS configuration at Monticello is somewhat different than Dresden, both plants include a GE Mark I containment, and the containment modelling for both plants is identical with the exception of plant-specific configuration inputs. Containment sprays are assumed in both analyses to be activated at 600 seconds. In addition, the nature of Monticello's containment transient response to spray initiation as shown for Cases 3, 6, and 7 of Exhibit D in the June 19, 1997, submittal, is very similar to that shown in Figure 6 of Attachment A of the February 27, 1997, Commonwealth Edison letter.

Based on the above justifications provided by the licensee, the staff accepts the licensee's conclusion that it is reasonable to assume that similar results would be obtained for the Monticello plant and the Dresden plant in regard to the minimum pressure case and that it is reasonable to conclude that the benchmarking analysis is also valid for Monticello.

3.3 LPCI and CS NPSH Calculations

The licensee provided evaluations of post-LOCA NPSH for CS and LPCI pumps. The evaluations were divided into two portions as follows:

Short-Term: 0 to 600 seconds (10 minutes), no operator action credited, vessel injection phase

Long-Term: 600 seconds to completion of event, operator actions credited, containment cooling phase

Section 5.2.3.3 in the USAR established the 600-second mark for operator action and the time at which credit for manual initiation of containment cooling can be taken. Therefore, for the long-term case, operator action is credited at the 600-second mark.

3.3.1 Short-Term NPSH Requirements

The bounding NPSH case for LPCI and CS pumps for short-term evaluation was determined to be four LPCI and two CS pumps at runout conditions, with the LPCI pumps injecting into a broken reactor recirculation suction loop. Only CS flow is injecting into the reactor. This event was described in GE Service Information Letter (SIL) 151 that postulates a failure of the LPCI Loop Select logic. This SIL primarily focused on the potential for loss of long-term containment cooling due to damage to the LPCI pumps under single-failure assumptions. The concern was that operation in cavitation conditions could cause loss of the LPCI pumps and subsequent loss of the containment heat removal function.

The licensee stated that the head loss across the clean strainers was restored to 0.3 meter (1 foot) per 630.9 L/s (10,000 gpm) by installing the new suction strainers. With the bounding event described above, the licensee determined that a CS system flow of 8740 gpm (4370 gpm per pump) should be available at runout conditions. In subsequent calls with the licensee, the licensee stated that the 10 CFR 50.46 analysis, SAFER/GESTR model, assumes a total CS flow of 7080 gpm (3540 gpm per pump) which limits the PCT [peak cladding temperature] to under 2200 degrees Fahrenheit post accident. In order to ensure that the total required CS flow is met, and to ensure that potential cavitation of the CS pumps does not occur, the licensee has requested that the current licensing basis be changed to credit the following containment overpressure for the specified time period.

Time Period (seconds)	Containment Overpressure (psig)
10 - 600	2.0

As shown on Figure E.1 of the licensee's July 16, 1997, submittal, 2.0 psig is equivalent to 16.26 psia. The staff notes that atmospheric pressure used in the calculations is based on 14.26 psia which is the minimum expected operating pressure and is based on historical minimum average local pressure conditions at Monticello. The licensee stated that the requested pressure is below the minimum pressure available and above the pressure required for adequate NPSH. The licensee anticipates that even though the requested overpressure is more than required for current licensing conditions, the requested amount should sufficiently bound the containment overpressure required to account for head loss associated with debris loading per NRC Bulletin 96-03 (Ref. 16). The staff has reviewed the licensee's minimum pressure analysis, which demonstrates the existence of 2.0 psig containment overpressure, and finds it acceptable. Based on the minimum pressure analysis, the following assumptions were made:

1. LPCI and CS pump friction losses were developed using clean, commercial steel pipe, and were increased by 15 percent to account for the effects of aging.

2. One of the four torus strainer assemblies was assumed to be 100 percent blocked while the others remained clean. This is consistent with Monticello's current licensing basis. The "A" suction strainer assembly was assumed blocked because it was calculated that the "A" strainer assembly passes the largest amount of flow.

3. A suppression pool pressure of 2.0 psig was assumed to exist from 10 to 600 seconds. As discussed above, the containment analysis has shown that the suppression pool pressure credited will be present during the first 600 seconds post accident.

4. The initial suppression pool temperature is assumed to be 90 degrees Fahrenheit per TS 3.7.A. The corresponding suppression pool temperature at 600 seconds is 149.3 degrees Fahrenheit.

5. The maximum LPCI and CS flow were assumed to be 3875 gpm (15,500 gpm total) and 4370 gpm (8740 gpm total), respectively, at the beginning of the event.

Based on the above assumptions, the licensee evaluated the NPSH Available (NPSHA) using the following equation.

$$NPSHA=Hb/\gamma-Hva/\gamma+Ps/\gamma+Z+Vs^2/2g$$

where:
- Hb = atmospheric pressure, 2053.44 lb/ft^2
- Hva = vapor pressure at fluid temp, lb/ft^2
- Ps = fluid pressure at pump suction, lb/ft^2

γ = specific weight of fluid, lb/ft^3
Vs = average velocity of fluid at pump suction, ft/s
Z = vertical distance between center line of pump and indication of Ps = 0.0 ft
g = $32.2\ ft/s^2$

The licensee's analysis, V75100.NSP97.00501, Case 1 (Ref. 17), demonstrated that with all six ECCS pumps running and credit for the containment overpressure specified above, no NPSH deficit exists for the LPCI and CS at the 600-seconds mark. The staff notes that the NPSH Required (NPSHR) for the CS pumps used in this calculation was 33 feet. However, Figure E.1, from the licensee's supplemental submittal dated July 16, 1997, is based on an NPSHR of 27 feet for the CS pumps. The use of 27 feet for NPSHR for the CS pumps for the short-term case was found acceptable by the staff as discussed in Section 3.4 of this safety evaluation. Therefore, the licensee adjusted its calculation, V75100.NSP97.00501, for an NPSHR of 27 feet and provided the results on Figure E.1. The staff notes that a revised calculation using the NPSHR of 27 feet for CS was not provided on the licensee's docket. However, the staff did perform its own calculations using an NPSHR of 27 feet and confirmed the data presented on Figure E.1.

Based on the above analysis, the staff finds that with credit for containment overpressure of 2.0 psig from 10 to 600 seconds, NPSH for the ECCS pumps will be available to meet the short-term worst-case scenario. This four LPCI/two CS pump case is shown on Figure E.1. The licensee intends to add this figure to the Monticello USAR. The staff concludes that there is reasonable assurance that plant operation in this manner poses no undue risk to the health and safety of the public.

3.3.2 Long-Term NPSH Requirements

The bounding NPSH case for LPCI and CS pumps for long-term evaluation was determined to be a DBA LOCA with no offsite power and failure of one diesel generator. For this case, Case 3 in the licensee's submittal of June 19, 1997, there is one division with one RHR heat exchanger, one RHR pump, and one RHRSW pump for long-term containment cooling. Case 3 also assumes that at 600 seconds post-LOCA, one of the RHR pumps is turned off to allow the start of an RHRSW pump. This scenario produces the worse-case for containment cooling, peak suppression pool temperature, and ECCS NPSH. The evaluation performed was time and temperature dependent beginning at 742.7 seconds post-DBA. The licensee's calculation, GE-NE-T2300731-2 (Ref. 18), demonstrates that the peak suppression pool temperature of 194.2 degrees Fahrenheit was reached at the 32,536-seconds mark and maintained at this point for approximately one half hour.

Under this bounding event, the licensee evaluated the long-term NPSH requirements for LPCI and CS crediting operator actions and accounting for the restored head loss of 0.3 meter (1 foot) per 630.9 L/s (10,000 gpm). In order to assure total CS and LPCI flows meet the total required flow, the licensee has requested that the current licensing basis be changed to credit the following containment overpressure for specified time periods.

Time Period (seconds)	Containment Overpressure (psig)
10 - 600	2.0
600 - 2,000	2.0
2000 - 10,000	4.0
10,000 - 16,000	5.3
16,000 - 55,000	6.1
55,000 - 69,000	5.6
69,000 - 85,000	5.0
85,000 - 110,000	4.2
110,000 - 140,000	3.3
140,000 - 200,000	2.3
200,000 - 330,000	1.0

As shown on Figure E.2 of the licensee's July 16, 1997, submittal, the requested containment overpressure in psig is based on an atmospheric pressure of 14.26 psia. The licensee stated that the requested pressure is below the minimum pressure available and above the pressure required for adequate NPSH. The licensee anticipates that even though the requested overpressure is more than required for current licensing conditions, the requested amount should sufficiently bound the containment overpressure required to account for head loss associated with debris loading per NRC Bulletin 96-03. The staff has reviewed the licensee's minimum pressure analysis, which demonstrated the existence of the above containment overpressure, and finds it acceptable. Based on this information, the following assumptions were made:

1. LPCI and CS pump friction losses were developed using clean, commercial steel pipe and were increased by 15 percent to account for the effects of aging.

2. One of the four torus strainer assemblies was assumed to be 100 percent blocked while the others remained clean. This is consistent with Monticello's current licensing basis. The "A" suction strainer assembly was assumed blocked since it was calculated that the "A" strainer assembly passes the largest amount of flow.

3. The suppression pool pressure specified above was assumed to exist from 600 to 330,000 seconds. As discussed above, the containment analysis has shown that the suppression pool pressure credited will be present during the specified time period post accident.

4. The initial suppression pool temperature is assumed to be 90 degrees Fahrenheit per TS 3.7.A. The corresponding suppression pool temperature at 32,536 seconds is 194.2 degrees Fahrenheit.

5. The maximum LPCI and CS flows were assumed to be 4000 gpm total and 2700 gpm, respectively, at the 600-seconds mark.

Using the above assumptions, the licensee evaluated the NPSHA required for pump protection using the equation described in Section 3.3.1 above. The licensee's analysis, V75100.NSP97.00501, Case 3, demonstrated that with two ECCS pumps running, one CS pump and one RHR pump, and credit for the containment overpressure specified above, no NPSH deficit exists for the LPCI and CS pumps during the long-term evaluation. This case is shown on Figure E.2 of the licensee's supplemental submittal dated July 16, 1997. The licensee intends to add this figure to the Monticello USAR.

Based on the above analysis, the staff finds that with credit for containment overpressure as specified above, NPSH for the ECCS pumps will be available to meet the long-term worst-case scenario. The staff concludes that there is reasonable assurance that plant operation in this manner poses no undue risk to the health and safety of the public.

3.4 Potential for CS and LPCI Pump Cavitation

The licensee has determined that the CS pumps are more limiting for NPSH than the LPCI pumps, for the limiting DBA-LOCA during both the short-term (i.e., less than 600 seconds following the LOCA) and long-term periods (i.e., after 600 seconds). The licensee has evaluated the NPSH requirements for the CS pumps assuming a short-term flow of 4370 gpm per pump. This is a conservatively high flow for determining required NPSH based on the accident condition system hydraulic resistance. The licensee determined that by using the required NPSH values shown on the pump NPSH curves originally supplied by the manufacturer, Sulzer Bingham Pump Incorporated, the required NPSH at 4370 gpm would be approximately 33 feet.

However, the manufacturer has determined that the original NPSH curve was based on a criterion of a drop in pump head of 1 percent from the maximum value tested, instead of the widely used Hydraulic Institute standards (Ref. 19) for performing pump testing, which recommends that a drop in head of 3 percent be used for determining NPSH requirements. Subsequently, the manufacturer has supplied the licensee with a comparison study results of the plant CS pumps versus the LPCI pumps at the Quad Cities plant in the range of 4000 to 5300 gpm, for which NPSH requirements are based on the 3 percent criterion. Therefore, the licensee has determined that the required NPSH for the CS pumps at 4370 gpm is not 33 feet, but 27 feet, at the same flow and head values, which results in an adequate value of NPSH available to the CS pumps for the credited containment overpressure.

The staff finds that the comparison of the Quad Cities pump test data to the plant CS pumps is acceptable because the pumps are similar in design and operating characteristics, and the licensee's method of determining the short-term NPSH requirement is technically adequate. For the long-term period, after the pumps are throttled to a lesser flow of 2700 gpm, the original NPSH curve values are assumed, which is conservative. Therefore, the licensee has determined that, after assuming credit for containment pressure as discussed in Sections 3.2 and 3.3 of this safety evaluation, there will be adequate NPSH for both the CS and LPCI pumps for the limiting DBA-LOCA conditions, thus assuring no pump cavitation. On this basis, the staff finds the licensee's analysis of the performance of the CS and LPCI pumps to be acceptable.

3.5 Effects of Increase in Peak Suppression Pool Temperature

3.5.1 Torus Attached Piping

The licensee has determined that the maximum suppression pool temperature for the limiting DBA-LOCA conditions would be 194.2 degrees Fahrenheit which is greater than the temperature previously analyzed for torus-attached piping loads. The torus-attached piping was previously analyzed for a temperature of 184 degrees Fahrenheit in 1995. Further, the licensee has determined that the increased thermal loads on the piping are the only loads that are affected due to the change in the LOCA containment response. The postulated hydrodynamic loads such as those associated with LOCA or safety/relief valve (S/RV) discharge remain the same since the reactor pressure and the S/RV setpoints remain unchanged. The licensee reanalyzed the torus-attached piping for a peak suppression pool temperature of 195 degrees Fahrenheit and the concurrent containment hydrodynamic loads, and has determined that all piping stresses, pipe supports, and torus penetrations meet the recommendations of NUREG-0661 (Ref. 20) and requirements of the American Society of Mechanical Engineers Code, Section III (Ref. 21). On this basis, the staff finds that the licensee's actions for addressing the effects of the revised containment response on torus-attached piping are acceptable.

3.5.2 Equipment Qualification

Exhibit H of the licensee's submittal dated June 19, 1997, provides evaluation of the potential impact on environmental qualification (EQ) of equipment inside the containment as a result of the new limiting scenarios for long-term containment heat removal. In this exhibit, the licensee concluded that equipment currently qualified per 10 CFR 50.49 remain qualified to the worst-case bounding conditions. The bounding accident temperature condition in the drywell for the EQ consideration is based on a small break LOCA. The bounding accident pressure conditions in the drywell occur during the DBA-LOCA. Exhibit H states that the EQ equipment inside containment was verified to be qualified to the peak drywell pressure of 42.3 psig and peak temperature of 335 degrees Fahrenheit and will not be changed by the reanalysis of long-term suppression pool temperature.

The staff noted that the licensee's evaluation in Exhibit H did not address the EQ bounding condition for the duration of a postulated LOCA. The staff requested the licensee to confirm that its verification of EQ profile included an evaluation that confirmed that all accident and post-accident temperature and pressure (not just peaks) were bounded for the duration of a postulated LOCA. In addition, the staff requested the licensee to provide a representative sample of EQ test profile curves to demonstrate the EQ test profile still bounds the new containment response profile resulting from the reanalysis.

In its response dated July 16, 1997, the licensee indicated that the SHEX benchmark analyses in Exhibit D compares containment responses using different computer codes and different decay heat models with input assumptions that maximize the containment responses for temperature and pressure. The containment response profiles resulting from this benchmark analysis have been plotted along with the bounding profile for EQ. These plots indicated that not all portions of the containment response are bounded in the EQ profile. In order to evaluate the differences between the accident profile and the EQ profile, the licensee used the Arrhenius methodology to calculate an equivalent integrated temperature profile for EQ equipment in containment. The results from this calculation show that the equivalent temperature exposure time for the EQ temperature profile exceeds the equivalent temperature exposure time for the DBA temperature profile.

The EQ profiles bound the accident profiles (including the peak conditions) with an adequate margin for the first few hours. It is the staff's understanding based on discussions with the licensee that the differences between the EQ profile and the accident profile during the post-LOCA are small. Based on these considerations, the staff concludes that there is reasonable assurance that safety-related electrical equipment in the containment will function as required during the analyzed accident conditions. However, as a separate initiative outside the scope of this evaluation, the staff will revisit the licensee's use of the Arrhenius methodology to calculate an equivalent integrated temperature profile for EQ equipment during the ongoing power uprate review.

3.5.3 Evaluation of RHR Room Temperature During DBA-LOCA

Exhibit G of the licensee's submittal dated June 19, 1997, determined that the maximum RHR room temperature under long-term DBA-LOCA conditions would continue to be less than or equal to the maximum long-term ambient temperature (140 degrees Fahrenheit), as specified in Section 6.2.2.2.1 of the USAR. The licensee's evaluation included three cases with varying input assumptions. The results of the licensee's evaluation showed that the calculated RHR room temperatures would reach the maximum allowable temperature at approximately 1 day and 11.5 days, respectively, after the beginning of the accident, for Cases 2 and 3. The staff notes, however, that Cases 2 and 3 both assume two RHR pumps, two RHRSW pumps, and a CS pump in operation whereas only one RHR pump, one RHRSW pump, and one CS pump would be running at these points in the accident scenario since this time frame in question is well after the time of the peak suppression pool temperature for these two cases.

The staff has reviewed the modeling techniques and assumptions provided by the licensee, NSP Calculation CA 97-157, "RHR Room Temp Response to General Electric Letters GLN 97-017 and GLN 97-019" (Ref. 22). The staff has determined that the modeling techniques and assumptions used were conservative. The staff, however, raised a question regarding the use of the 600-horsepower heat input assumption for the 700-horsepower RHR pump motors in the calculation. In its response dated July 2, 1997, the licensee indicated that two separate studies were conducted to validate this assumption. The actual operating electric horsepower (EHP) for each motor was recently measured, and the resulting brake horsepower (BHP) for each pump was calculated. All operating BHP values were found to be less than the rated value of 600-horsepower. In addition, the rated BHP, which was used in the calculation, is greater than the manufacturer's measured BHP at design operating conditions.

Based on the above, the staff finds that the licensee's evaluation of the RHR room temperature is acceptable.

3.6 Electrical Loading With ECCS Pumps At Runout Flows

Exhibit J of the licensee's submittal dated June 19, 1997, determined that the higher than rated pump flows result in different BHP requirements which are equal to or slightly less than the rated horsepower of the motors. Furthermore, the licensee's evaluation indicated that the electrical input power to the motors of these pumps, when pumping the specified higher than rated pump flows, is less than the values analyzed for in Table 8.4-2 of the USAR for emergency diesel generator system emergency loads for these pump.

Based on the above, the staff concludes that the impact of the higher than rated pump flow on the pump motors is acceptable.

3.7 Changes to the Technical Specifications Bases

The licensee proposed to clarify TS Bases Sections 3.5/4.5.C and 3.7.A as follows:

The Bases for TS 3.5/4.5.C are clarified with respect to the minimum requirements for containment spray/cooling system pumps following a loss of coolant accident. One RHR pump and one RHRSW pump satisfy the minimum requirements for long-term containment heat removal.

The Bases for TS 3.7.A are changed to reflect that there is a dependency on containment overpressure to ensure adequate NPSH for the ECCS pumps in the worst case DBA scenarios.

3.8 Bulletin 96-03

The staff issued NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," (Ref. 16) identifying that the buildup of debris from thermal insulation, corrosion products, and other particulates on ECCS pump strainers is highly likely to occur, creating the potential for a common-cause failure of the ECCS, which could prevent the ECCS from providing long-term cooling following a LOCA. The staff has requested that all BWR licensees implement appropriate measures to ensure the capability of the ECCS to perform its safety function following a LOCA. NRC Bulletin 96-03 also requested all licensees to implement these actions by the end of the first refueling outage starting after January 1, 1997.

This timeframe for implementation was considered appropriate by the staff based on recent cleaning of suppression pools, operator training and appropriate emergency operating procedures (EOPs), alternate water sources, and a low probability of the initiating event. In the case of Monticello, consideration of containment overpressure of 2.0 psig from 10 to 600 seconds restores the ECCS capability to meet the requirements of 10 CFR 50.46(a)(1)(i) with the original licensing basis. The staff notes that this conclusion is based on the licensee's analysis of only one strainer completely blocked and does not take into account the potential for additional blockage as identified in NRC Bulletin 96-03. Appropriate corrective actions, if any, resulting from the licensee's evaluation of NRC Bulletin 96-03 will be implemented in accordance with 10 CFR Part 50, Appendix B. This action will resolve the staff's outstanding questions relative to ECCS performance and will provide long-term assurance that the requirements of 10 CFR 50.46 are met. The resolution of NRC Bulletin 96-03 will be addressed under separate correspondence.

3.9 Qualitative Evaluation of Reliance on Containment Overpressure

Inadequate NPSH to the ECCS pumps could result in a common-mode failure in the inability of the ECCS to provide adequate long-term core cooling and/or the inability of the containment cooling system to maintain the containment pressure and temperature below design limits. Therefore, any reliance on containment overpressure for NPSH considerations is a significant factor both from the safety and risk perspectives.

NRC Regulatory Guide (RG) 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," establishes the regulatory position that ECCS should be designed so that adequate NPSH is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present before any postulated LOCA scenarios. Standard Review Plan (SRP) Section 6.2.2, "Containment Heat Removal Systems," clarifies RG 1.1 by stating that the NPSH analysis should be based on the assumption that the containment pressure equals the vapor pressure of the sump water, to ensure that credit is not taken for containment pressurization during the transient.

Since the issuance of RG 1.1 in 1970, the NRC staff has selectively allowed limited credit for a containment pressure that is above the vapor pressure of the sump fluid (i.e., overpressure) to satisfy NPSH requirements on a case-by-case basis. This is due mainly to the fact that the original design basis for an older plant, such as Monticello, assumed containment overpressure for NPSH considerations. Many similar vintage boiling water reactors (BWRs) were constructed with ECCS designs that use ECCS pumps and pump locations that do not provide as much NPSH margin as later designs.

Although the basis for the staff's approval for crediting a limited amount of containment overpressure is the licensee's analytical results which demonstrate that containment pressure, following the DBA-LOCA with the worst-case scenarios, is greater than the pressure that is credited, the staff also considered the conservatism that exists in the licensing basis DBA-LOCA analysis as well as the plant's ability to mitigate the consequences of the DBA-LOCA without taking credit for containment overpressure.

The analysis for the DBA-LOCA for the limiting ECCS pump NPSH includes assumptions and methodologies that are designed to minimize the amount of containment pressure while maximizing the temperature response of the suppression pool. These assumptions and methodologies are very conservative to the extent that certain conditions assumed or calculated to exist inside containment do not actually reflect any reasonable operating or post-accident conditions. Reducing or eliminating these conservatisms would reduce the calculated amount of containment overpressure needed for NPSH.

In its July 21, 1997, submittal, the licensee provided, in part, the following examples of conservatism in its analysis:

- The assumed decay heat was conservatively based on about 115 percent of the rated thermal power level. This accounts for approximately 2 psig in overpressure.
- Using a typical summer high average daily river water temperature of 82 degrees Fahrenheit instead of 90 degrees Fahrenheit (assumed in the analysis) would reduce the required containment overpressure by about 0.5 psig.
- Conservatism in NPSH calculations would account for about 2.3 psig in containment overpressure.
- Conservatism in the minimum containment pressure calculations would account for about 1.0 psig.

The licensee determined that the cumulative effect of this conservatism, when applied to the limiting ECCS pump, provides reasonable assurance of successful pump operation. Therefore, the credited amount of containment overpressure can be considered as a prudent additional reserve of available pressure such that NPSH considerations would not affect pump operation for the duration of the DBA-LOCA.

The licensee has also analyzed the plant's ability to mitigate the consequences of the DBA-LOCA without taking credit for containment overpressure, as documented in its submittal dated July 21, 1997. For the short term (first 10 minutes following the DBA-LOCA), the worst-case scenario would result in an NPSH deficit of up to 3.16 feet between 85-600 seconds. However, at 189 seconds, the two core spray (CS) pumps will have reflooded the core. This is based on the assumption of 89 percent of the rated flow for the CS pumps consistent with the assumptions used in the NPSH calculations. The ECCS pumps are expected to deliver approximately 90 percent of rated flow with the calculated NPSH deficit, based on test data provided by the pump vendor for a similar pump.

For the long term, following the first 10 minutes into the DBA-LOCA, operator response is assumed. Since it is expected that the operators cannot restore and maintain level above the top of the active fuel, the EOP C.5-2004, "Drywell Flooding," would be entered. This procedure will direct the operators to flood the drywell with all available systems. Operators are directed to keep one loop of core spray aligned to the torus and to align the remaining ECCS pumps to the condensate storage tanks (CSTs). These actions provide the following benefits:

- Relatively cool water is now being added to the reactor core from the LPCI and the remaining CS system which are aligned to the CST.
- Torus water level will increase. This will add available NPSH to the operating CS pump.
- The cooler water and increased elevation head together with less friction head loss, as the number of pumps is reduced, would likely allow for continued operation of the CS pump regardless of the torus pressure.
- The ECCS pump NPSH concerns related to containment pressure are eliminated while the pumps are aligned and operated from the CST.
- The CST suction source would provide core cooling for approximately 40 minutes assuming 8000 gpm through two RHR pumps.

In the DBA-LOCA scenarios, the CST suction source is not credited since the CSTs are not seismically qualified. However, EOP C.5-3203, "Use of Alternate Injection Systems for RPV [reactor pressure vessel] Makeup," directs the use of the following safety-grade systems as a means to flood the drywell:

- When the LPCI pumps have exhausted the CST inventory, the LPCI pumps would be secured and the RHRSW pumps would be used to provide an inexhaustible supply of cold river water to the reactor vessel via the LPCI piping.
- Another available and inexhaustible source that utilizes river water is the fire protection system, which utilizes either an electric fire pump or the diesel fire pump, that can be aligned to inject to the reactor vessel via the LPCI piping.

Based on these factors, the licensee concluded that containment pressure above atmospheric levels to support NPSH requirement is not necessary to successfully mitigate a design-basis LOCA at Monticello. Although the primary ECCS may be degraded when post-accident increases in torus temperature may result in reduced NPSH, sufficient methods are available to maintain adequate core cooling and containment integrity even if containment pressure is artificially held to atmospheric levels. These methods utilize systems that have capacities well in excess of that required to sufficiently remove core decay heat. These methods are implemented using existing procedures on which the operators are continually trained.

During its review of the EOPs, the licensee identified a potential discrepancy between a EOP definition and the expected plant condition regarding the core geometry following the DBA-LOCA. The staff notes that the licensee has committed to process a 10 CFR 50.59 evaluation to change the EOP definition of adequate core cooling to 2/3 height to be consistent with the expected plant condition during the DBA-LOCA.

In addition, the staff has determined that the licensee's analyzed suppression pool temperature responses to the DBA-LOCA scenarios submitted on June 19, 1997, will remain virtually unchanged if a loss of containment integrity were assumed. Since the calculated suppression pool temperature will be below 212 degrees Fahrenheit, and the containment pressure will be at atmospheric pressure, no flashing will occur in the suppression pool. Therefore, it is reasonable to assume that the temperature transient analyses will remain valid.

4.0 SUMMARY

Based on the above evaluation, the staff finds it acceptable to rely on a limited amount of containment overpressure, for the time periods designated above, to compensate for a slight increase in the amount of NPSH deficiency during the worst-case DBA scenarios. In addition, the staff finds it acceptable for the licensee to change the USAR to reflect the new NPSH and containment pressure/temperature conditions addressed by this safety evaluation.

The staff also finds the analysis that evaluated the consequences of the increase in the peak suppression pool temperature, to 194.2 degrees Fahrenheit, acceptable.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component found within the restricted area as defined in 10 CFR Part 20. The NRC 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 two proposed findings that the amendment involves no significant hazards consideration, and there has been no public comment on such findings (62 FR 6576) and (62 FR 34086). 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.

7.0 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.

Principal Contributors: K. Kavanagh
A. Gill
C. Hammer
J. Kudrick
R. Elliott
D. Nugyen
J. Raval
T. Kim

Date: July 25, 1997

8.0 REFERENCES

1. Hill, William J., Northern States Power Company (NSP), to USNRC, "License Amendment Request Dated January 23, 1997, Update of Design Basis Accident Containment Temperature and Pressure Response," January 23, 1997.

2. Hill, William J., NSP, to USNRC, "Revision No. 1 to License Amendment Request Dated January 23, 1997, Update of Design Basis Accident Containment Temperature and Pressure Response," January 28, 1997.

3. Hill, William J., NSP, to USNRC, "Supplement No. 1 to License Amendment Request Dated January 23, 1997, Update of Design Basis Accident Containment Temperature and Pressure Response," March 4, 1997.

4. Hill, William J., NSP, to USNRC, "Revision No. 2 to License Amendment Request Dated January 23, 1997, Update of Design Basis Accident Containment Temperature and Pressure Response," June 19, 1997.

5. Hill, William J., NSP, to USNRC, "Response to Request for Additional Information Regarding Revision 2 to MNGP License Amendment Dated January 23, 1997 (TAC No. 97781)," July 2, 1997.

6. Hill, William J., NSP, to USNRC, "Response to Request for Additional Information Regarding MNGP License Amendment Dated June 19, 1997 (TAC No. 97781)," July 16, 1997.

7. Hill, William J., NSP, to USNRC, "Request for Information Regarding MNGP License Amendment Dated June 19, 1997 (TAC NO. 97781)," July 16, 1997.

8. Hill, William J., NSP, to USNRC, "Response to Staff Questions Regarding NSP Letter of July 16, 1997 (TAC No. 97781)," July 21, 1997.

9. Hill, William J., NSP, to USNRC, "Treatment of Commitments as License Conditions (TAC No. 97781)," July 25, 1997.

10. Northern States Power Company, Monticello Nuclear Generating Plant, Updated Safety Analysis Report.

11. Generic Electric Company (GE), "Monticello Design Basis Accident Containment Pressure and Temperature Response for FSAR Update," NEDO-30485, December 1983.

12. GE, "Monticello Nuclear Generating Plant Design Basis Accident Containment Pressure and Temperature Response for USAR Update," NEDO-32418, December 1994.

13. Grant, Geoffrey E., NRC, to NSP, "NRC System Operational Performance Inspection (SOPI) Report 50-263/96009(DRS)," February 20, 1997.

14. Perry, J. S., Commonwealth Edison, to USNRC, "Dresden Nuclear Power Station Units 2 and 3, Additional Information Regarding Application for Amendment to Facility Operating Licenses DPR-19 and DPR-25, Appendix A, Technical Specifications, Section 3/4.7.K, Suppression Chamber, and Section 3/4.8.C, Ultimate Heat Sink," February 27, 1997.

15. Stang, J. F., USNRC to Commonwealth Edison, "Issuance of Amendments (TAC Nos. M97983 AND M97984)," Docket Numbers 50-237/50-249, April 30, 1997.

16. NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers By Debris in Boiling-Water Reactors," May 6, 1996.

17. Duke Engineering and Services, "Determination of Containment Overpressure Required for Adequate NPSH of the Low Pressure ECCS Pumps," V75100.NSP97.00501, June 18, 1997.

18. GE Nuclear Energy, "Monticello Nuclear Generating Plant LOCA Containment Analyses For Use in Evaluation of NPSH for the RHR and Core Spray Pumps," GE-NE-T2300731-2, June 1997.

19. Hydraulic Institute, "Pump Standards," ANSI/H1 1.6-1994, Parsippany, NJ.

20. USNRC, "Safety Evaluation Report 'Mark I Containment Long-Term Program'," NUREG-0661, July 1980.

21. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 1977 Edition, Section III, New York.

22. NSP, Calculation Number CA-97-157, "RHR Room Temp. Response To General Electric Letters GLN-97-017 and GLN-97-019," June 13, 1997.