Northern States Power Company

Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362

NSP

October 1, 1998

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

NUREG-0737
Supplement 1

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

Proprietary Information Related to
NSP Response to NRC Request for Additional Information Regarding
November 18, 1997 Request for Deviation from Emergency Procedure Guidelines

Ref. Letter from Michael F. Hammer, NSP, to USNRC Document Control Desk, "Request for Deviation From Emergency Procedure Guidelines, Revision 4, NEDO-31331, March 1987," November 18, 1997.

By letter dated November 18, 1997 (Reference 1) NSP requested a deviation from the Boiling Water Reactor (BWR) Owners' Group Emergency Procedures Guidelines (EPGs), Revision 4, NEDO-31331, March 1987. The deviation was requested to recognize 2/3 core height as adequate core cooling following a large break loss of coolant accident.

The NRC Staff subsequently asked five specific questions which were discussed between NSP and NRC representatives in a conference call on September 28, 1998. At the conclusion of that discussion the Staff requested that responses be submitted in writing. These questions and answers are being provided as requested in the attachments to this letter.

Northern States Power Company (NSP), a Minnesota corporation, hereby requests that certain information (Attachments 3 and 4) hereby provided to the Nuclear Regulatory Commission (NRC), be withheld from public disclosure due to its proprietary nature. The details of this request are provided in the attached affidavit.

APO1 //

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Drawing located in Central Files

This submittal contains no new NRC commitments, nor does it modify any prior commitments. Please contact Marcus H. Voth, Project Manager of Licensing, at 612-271-5116 if you require additional information related to this request.

M Hammer

Michael F. Hammer
Plant Manager
Monticello Nuclear Generating Plant

c: Regional Administrator - III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
State of Minnesota, Attn: Kris Sanda
J Silberg

Attachments:

1. NSP Response to NRC Request for Additional Information Regarding November 18, 1997 Request for Deviation from Emergency Procedure Guidelines
2. Affidavit of Michael F. Hammer, Northern States Power
3. Monticello Flowchart, "C.5-1100 RPV Control," Revision 5
4. Monticello Flowchart, "C.5-1100 RPV Control," Revision 6

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

Request to Withhold Proprietary Information from Public Disclosure

AFFIDAVIT

I, Michael F. Hammer, being duly sworn, depose and state as follows:

(1) I am Plant Manager, Monticello Nuclear Generating Plant, Northern States Power Company ("NSP") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld consists of Monticello Nuclear Generating Plant, Emergency Operating Procedure (EOP) Flow Charts, "C.5-1100 RPV Control," Revision 5, and "C.5-1100 RPV Control," Revision 6. This information describes key technical details of NSP's plans for responding to beyond design basis events. The proprietary information is identified by the words "NSP PROPRIETARY INFORMATION" on each page.

(3) The information sought to be withheld is being submitted to the NRC in confidence. The original EOP flow charts were BWR Owners Group proprietary. The attached revisions are NSP proprietary. This information is of a sort customarily held in confidence by NSP, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by NSP, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

(4) Public disclosure of the information sought to be withheld is likely to cause harm to NSP's competitive position and reduce the availability of profit-making opportunities. The research, development, engineering, and analytical costs comprise a substantial investment of time and money by NSP. The value of this information to NSP would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive NSP of the opportunity to exercise its competitive advantage to seek an adequate return on its investment in developing this information.

This letter contains no restricted or other defense information.

By M F Hammer
Michael F. Hammer
Plant Manager
Monticello Nuclear Generating Plant

On this 1st day of October, 1998 before me a notary public in and for said County, personally appeared Michael F Hammer, Plant Manager, Monticello Nuclear Generating Plant, and being first duly sworn acknowledged that he is authorized to execute this document on 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.

Samuel I. Shirey
Samuel I Shirey
Notary Public - Minnesota
Sherburne County
My Commission Expires January 31, 2000

SAMUEL I. SHIREY
NOTARY PUBLIC - MINNESOTA
My Commission Expires Jan. 31, 2000

Attachment 1

NSP Response to NRC Request for Additional Information Regarding
November 18, 1997 Request for Deviation from Emergency Procedure Guidelines

1. *Explain why the MSCRWL (calculated in accordance with the BWROG methodology for EPG Rev 4) is greater than 2/3 core height, given that NEDO-20566A (Section III.A.5) shows the peak clad temperature to be well below 1500F for the most limiting break.*

NEDO-20566A (Reference 1) is the basis for the detailed SAFER evaluation of the design basis loss of coolant accident (DBA-LOCA). The Boiling Water Reactor Owners' Group (BWROG) methodology is a simplified calculation, independent of plant-specific systems or characteristics. As such, the BWROG methodology is very conservative.

General Electric states that the BWROG methodology is based on a simplified, steady state calculation using a single rod model. The BWROG methodology also uses a very conservative axial power shape that has an axial power peak of 2.0 in node 19 of 24 (node 1 is the bottom of the core). A sensitivity study performed by GE determined the axial power shape is the dominant factor in determining the elevation of the minimum steam cooling RPV water level (MSCRWL). Furthermore, the BWROG methodology assumes steam cooling is the only mechanism for bundle cooling above the MSCRWL. References 4 and 5 state the MSCRWL is calculated assuming the reactor has been shutdown from rated power for ten minutes. The assumptions used to calculate the MSCRWL approximate the conditions experienced by the hottest pin in the hottest bundle when the reactor has been reflooded (including level swell) to only the MSCRWL. Using these conservative conditions, peak clad temperatures on the order of 1500 °F are calculated.

In contrast, the clad temperatures reported in Reference 1 are calculated using a single channel, time domain computer model assuming an axial peak of 1.4 near the core mid-plane (as is typical for licensing bases calculations). Figure 6 of Reference 1 shows the peak cladding temperature as a function of time following the design basis accident. The initial temperature transient is terminated when the core is flooded by the accumulation of emergency core cooling system (ECCS) water and decay heat causes the two phase water boundary to swell above the top of the fuel. As bundle decay heat decreases, level swell in that bundle also decreases. A bundle will remain covered as long as the energy stored in that bundle can produce sufficient level swell. When the bundle finally does uncover, the uncovered portion is cooled by the steam generated in the covered portion of the bundle. This results in an increase in the peak cladding temperature. This can occur as early as 10 minutes for the lowest power bundles, but will take several hours for the highest power bundles. Thus the decay heat in the hottest bundle when the upper portion of the bundle is cooled only by steam cooling is significantly less in Reference 1 (two hours) than in the BWROG methodology (ten minutes).

For the reasons given above, the peak clad temperatures calculated in Reference 1 are less than those calculated using the BWROG methodology, even though the MSCRWL used in the BWROG methodology is higher than the 2/3 core height used in the Reference 1 calculation.

2. *Discuss the consequences of maintaining reactor level at 2/3 core height without core spray for an extended period (e.g., while waiting for the TSC to develop recommendations regarding containment flooding). Provide an estimate of the peak cladding temperature and the extent of cladding oxidation under best estimate assumptions.*

Figure 5 of Reference 1 shows the maximum cladding temperature as a function of time using low pressure coolant injection (LPCI) only. This figure shows the peak cladding temperature for long term cooling to be on the order of 950°F. Figure 7 of Reference 1 shows the peak cladding temperature response to a loss of coolant accident with a break area that is in the limiting spectrum of breaks. This figure shows the long-term peak cladding temperature decays below 900 °F within hours of bundle uncovery. Assuming the cladding temperature in the upper three feet of the fuel remained at 900°F for 11 days, and that all rods are at the same temperature as the hottest rod, the amount of metal-water reaction in the total active cladding in the core is calculated to be 0.09% in Reference 1.

3. *The BWROG decided to flood at TAF (or MSCRWL) for several reasons identified in the Monticello submittal dated November 18, 1997 (page 5). Explain why reasons #2 and #3 do not apply to Monticello.*

Reason #2
Flooding of containment and the reactor vessel to above TAF places the reactor into a stable condition for long-term cooling. Reliance on pumps and other active equipment to maintain this condition is minimized. Conditions are stable and required operator action times are long.

Reason #3
Following a LOCA, level will be quickly restored above TAF, and the core will remain covered for long term cooling except for the largest breaks in the recirculation piping. A break of this size will require containment to be flooded anyway for accident recovery.

Reason #2 is concerned with potential additional failures beyond the design basis of the plant and implies early containment flooding is the best method to minimize these hypothetical failures. Requiring containment flooding early in a DBA-LOCA to minimize the consequences of potential additional equipment failure may not provide the best total plant response. For example, requiring containment flooding per the EPG Rev. 4 instructions will require primary containment venting to maintain pressure below the containment design pressure. Venting, however, is not expected to occur until after the hard pipe vent is submerged. Venting would have to be through the standby gas treatment (SBGT) system. At the containment pressures at which venting would occur, damage to the SBGT suction ductwork would probably occur. This could impact the ability of the plant to utilize a filtered, elevated release pathway. The Monticello containment pressure response to the DBA-LOCA (Reference 3, Figure 5.2-15) shows the pressure would be less than 5 psig within 6 days. Deferring containment flooding, and thus deferring the need to vent, greatly reduces the potential for damage to the SBGT ductwork.
As stated on page 6 of the Monticello submittal (Reference 2), Monticello would flood

primary containment if 2/3 core height could not be restored and reliably maintained. If fewer than two ECCS pumps were available for RPV injection (the minimum number expected to be available following the DBA-LOCA), Monticello would flood primary containment if RPV water level could not be restored and maintained above the minimum steam cooling RPV water level.

Reason #3 states that Monticello would need to flood primary containment to recover from a DBA-LOCA. Containment flooding performed during the recovery phase of the DBA-LOCA would be a more controlled evolution than flooding initiated as soon as RPV water level could not be restored and maintained above TAF (or MSCRWL).

4. *Explain the specific benefits of the proposed approach (waiting to flood until directed by the TSC) relative to the BWROG approach of taking these actions immediately. Does the proposed approach reduce the impact on public health and safety when compared to the BWROG approach? Would safety be adversely impacted if the BWROG approach is retained?*

As stated in the Monticello submittal (Reference 2, Page 5), adhering to the EPG Revision 4 (and EPG/SAG) approach of immediately flooding primary containment creates the following conflicts with the licensing basis plant response described in the Monticello Updated Safety Analysis Report (USAR) (Reference 3):

- RHR and RHR service water pumps would not be aligned for long-term suppression pool cooling.
- USAR environmental qualification, shielding, and radiological analyses may no longer be applicable because conditions are different than those originally assumed.
- RPV venting to the condenser (and ultimately to the environment) is required to flood the reactor vessel. This creates a vent path not previously considered in the USAR radiological analysis.

In addition, flooding early in the accident may result in damage to the SBGT system if it used for containment venting. This could result in additional vent paths not previously considered in the USAR radiological analysis and may render vital areas of the plant inaccessible.

The proposed approach is preferred over the BWROG approach since it would enable the Monticello plant to respond to a DBA-LOCA event while remaining within the design basis capabilities of the plant as described in USAR. The proposed approach would also minimize the potential for damage to systems that could be used to recover from the event by deferring venting until primary containment pressure is reduced.

5. *Explain how the transition from the EOPs to the severe accident guidelines would occur if the proposed approach is adopted, since the operators would stay in the EOPs and not flood containment until additional failures resulting in level falling below 2/3 core height. Describe*

the expected progression of events following loss of injection (e.g., timing of core heatup, core relocation, and vessel failure) if actions to flood containment are not initiated until level falls below 2/3 core height.

Monticello ECCS systems are designed to restore and maintain reactor pressure vessel (RPV) water level at 2/3 core height following a DBA-LOCA. No analysis has been performed to determine the timing of core heatup, core damage, or vessel failure if RPV water level falls below 2/3 core height since this would require additional failures beyond the design basis of the plant.

As stated in the response to question 3 above, if fewer than two ECCS pumps are available, Monticello would flood primary containment if RPV water level could not be restored and maintained above the minimum steam cooling RPV water level. When primary containment flooding is initiated the severe accident management guidelines would be entered and decision making responsibility would be transferred to personnel in the Technical Support Center (TSC). The two ECCS pump criteria was chosen because this is the minimum number of ECCS pumps expected to be available for RPV injection following a DBA-LOCA.

If only one ECCS pump were available for injection into the RPV following a DBA-LOCA event, the plant would be in a condition that is beyond the design basis of the facility. Transferring to the severe accident management guidelines under these conditions is the prudent action to be taken, and would occur once the TSC is staffed and operational.

The EPG guidance for RPV level control and alternate RPV level control is provided to the Monticello Operators in flowchart C.5-1100. Revision 5 of this flowchart shows this guidance prior to implementing 2/3 core height as adequate core cooling. Revision 6 (in draft form) shows the proposed changes to implement 2/3 core height as adequate core cooling. Copies of these flowcharts are provided to show how Monticello would transition from the EOPs to the Severe Accident Management Guidelines if the proposed approach is adopted.

REFERENCES:

1. NEDO-20566A, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K, Volume II, Section III

2. Letter form NSP to NRC, "Request for Deviation from Emergency Procedure Guidelines, Revision 4, NEDO-31331, March 1987," November 18, 1997.

3. Monticello Updated Safety Analysis Report.

4. EPG Revision 4 Appendix C Calculations Workshop Notebook, May 1993.

5. BWROG Emergency Procedure and Severe Accident Guidelines, Revision 1, Appendix C, July 1997.