NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

September 8, 1977

Mr D K Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors
c/o Distribution Services Branch, DDC, ADM
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Davis:

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

Safety-Relief Valve Technical Specification Changes

Your letter dated August 3, 1977 requested us to submit a License Amendment Request revising the limiting conditions for operation and surveillance requirements for the safety-relief valves installed at the Monticello Nuclear Generating Plant. We were asked to base our License Amendment Request on model technical specifications enclosed with your letter and to make our submittal within 30 days of receipt of your letter.

We will be unable to meet your schedule for submitting a License Amendment Request. Our review of the NRC model technical specifications has identified a number of areas which require further discussion with you and clarification before a License Amendment Request can be submitted. These areas are summarized in the attached table.

We do not believe that immediate action is required on the proposed technical specification changes for Monticello since they will not significantly improve safety-relief valve reliability. The existing surveillance program provides a high degree of assurance that these valves will function when required. It should also be noted that Monticello has not experienced the large number of safety-relief valve failures noted in your letter. During the last five years of plant operation we have experienced one failure of a safety-relief valve to open, one failure of a redundant automatic depressurization initiation logic circuit, and one instance of premature safety-relief valve actuation. None of these failures caused the safety-relief valve installation to be incapable of performing its safety related functions. Further assurance of relief capability is provided through the installed spare safety-relief valves.

NSP has been cooperating with the General Electric Company in their program for improving the performance of BWR safety-relief valves. This program

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has been described in detail to the Commission's technical staff in meetings with General Electric.

We are ready to discuss this matter with you at any time, either by telephone or through a meeting with you, other BWR utilities, and the General Electric Company. When the areas in your proposed technical specifications needing clarification have been resolved, we will submit a License Amendment Request if the requirement for revised technical specifications still exists. In the interim period, we will continue to perform safety-relief valve testing and accumulate reliability data in accordance with the existing comprehensive surveillance program contained in the technical specifications. A visual inspection of safety-relief valve line restraints, including those in the torus, will be conducted during the Autumn 1977 refueling outage.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

attachment

LOM/DMM/deh

cc: J G Keppler
G Charnoff
MPCA
Attn: J W Ferman

Attachment to L O Mayer letter dated September 8, 1977

Areas of NRC Model Safety-Relief Valve Technical Specifications (dated August 3, 1977) Needing Clarification

Specification No.	Problem Area
3.4.2	The requirement for safety-relief valve operability above 212°F or with the Mode Switch in Run or Startup does not recognize the fact that operability cannot be demonstrated until the reactor is pressurized.
	The model does not recognize the possibility of installed spare safety-relief valves. Spare valves are not explicitly excluded from the Action statement in the event of inoperability.
	The action statement requires going to cold shutdown if operability is not restored within 15 minutes even if the inoperable component is made operable in the course of shutdown and cooldown.
3.5.2	The model requires Automatic Depressurization System (ADS) operability above 212°F or with the Mode Switch in Run or Startup. The system is not required to be functional below 150 psig or if irradiated fuel is not in the reactor vessel.
	Action statement (a) requires going to cold shutdown if operability is not restored within 14 days even if operability is restored in the course of shutdown and cooldown.
4.4.2.1	Increasing surveillance frequency in accordance with Table 4.4-10 may lead to increased valve failure. Experience has shown that increased valve cycles may reduce reliability.
	Table 4.4-10 states a surveillance frequency which depends on number of failures in the past. No recognition is given of the fact that the cause of failure may have been found and corrected. Valve failure from unknown causes results in the same surveillance frequency adjustment that a failure accompanied by a positive correction results in.
	Testing at nominal operating pressure is specified. Valve design suggests that low pressure testing is more demanding. Furthermore, we believe that

Areas of NRC Model Safety-Relief Valve Technical Specifications (dated August 3, 1977) Needing Clarification (Continued)

Specification No.	Problem Area
4.4.2.1 (continued)	valve operability should be verified as soon as possible after reaching plant conditions requiring operability. Current technical specifications require operability prior to reaching 110 psig when temperature is above 345°F. Operability is verified when pressure is sufficient to allow proper functioning of the reactor pressure control system (about 170 psig). Testing at or below five percent rated power is specified. We believe that this restriction should be removed. This power level lies in an ill defined region between the intermediate and power ranges and is difficult to determine with precision. Testing at higher power levels should be permitted to alleviate this problem and to permit more flexibility in scheduling testing.
4.4.2.2	The applicable ASME Code and Addenda are referenced. This should be replaced with a general reference to 10 CFR 50, Section 50.55a(g) to eliminate the requirement for a technical specification revision each time the program of inservice inspection and testing is revised.
4.5.2	See comments above on Section 4.4.2.1 and 4.4.2.2.

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U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50 - 263

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TO: Mr. D. K. Davis

FROM: NSP
Minneapolis, Minnesota 55401
L. O. Mayer

DATE OF DOCUMENT 09/08/77

DATE RECEIVED 09/13/77

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DESCRIPTION Advising that NSP is unable to meet NRC's schedule for submitting a License Amend Request revising the limiting conditions for operation and surveillance requirements for the safety-relief valves installed at Monticello due to lack of clarification in NRC's model tech specs. w/att list of areas needing clarification...

2p + 2p

ENCLOSURE

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PLANT NAME: MONTICELLO
jcm 09/14/77

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