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Docket No. 50-263

DEC 12 1977

Northern States Power Company
ATTN: Mr. L. O. Mayer, Manager
Nuclear Support Services
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Gentlemen:

RE: MONTICELLO NUCLEAR GENERATING PLANT

In August we sent letters to a number of licensees who operate Boiling Water Reactor (BWR) type reactor facilities regarding surveillance requirements for the relief and safety-relief valves that are installed in the reactor coolant system and/or the automatic depressurization system. The letters requested licensees to propose changes to their Technical Specifications to incorporate a variable frequency test schedule for operability testing of relief and safety-relief valves, and to institute increased inspection of the relief and safety-relief valve line restraints in the torus. Model Technical Specifications were included for guidance in preparing plant specific requirements. This letter was sent to you on August 3, 1977.

From the responses we received from various licensees, it was evident that the Model Technical Specifications were subject to misinterpretation in several areas. Consequently, as a result of comments received from these licensees, and further consideration by ourselves, we have revised the Model Technical Specifications to provide some clarification of the requirements. The major revisions, which are included in the enclosed Model Technical Specifications, are:

1. Clarification that only the number of safety and safety-relief valves that are needed to comply with ASME Code requirements and the plant safety analysis are to be included in Model LCO 3.4.2. The number of valves should not include any installed spares. Therefore, if a licensee's existing Technical Specifications include LCO's for spare safety or safety-relief valves, these LCO's should be modified to exclude these valves. For some plants this represents a relaxation of previously overly restrictive requirements in this area.

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2. Modification of the action statement of Model LCO 3.4.2 to address only the safety valve function of the safety-relief valves. This modification makes the action statement more consistent with the safety objective of the specification which is to assure the preservation of the reactor coolant system pressure boundary.

3. The addition of a footnote to explicitly indicate that, with regard to the action statement of LCO 3.4.2, a safety-relief valve that fails to open as a result of a manual actuation signal need not be considered a failure of the safety valve function of the valve. This addition recognizes that in such cases the safety valve function may still be operable and therefore a plant shutdown is not required.

4. Deletion of the required power level of less than 5%, and revision of the required reactor steam dome pressure to be maintained during manual testing of the safety-relief valves. The required pressure has been changed from nominal operating pressure to any pressure greater than 100 psig. These changes will provide more operating flexibility and will significantly reduce the impact of the testing program on plant capacity factors. Moreover, the reduction in the required pressure will result in a lower incidence of the higher stresses on the relief valve line restraints that can result from testing valves at operating pressure. It will also reduce the likelihood of aggravating pilot seat leakage.

5. A change to the time period for observing inoperable valves for the purpose of determining the initial Next Required Test Interval of Table 4.4-10. This has been changed from the 18 month period beginning September 1, 1977 to the 12 month period beginning March 1, 1978. This will allow additional lead time for licensees to develop and implement improved relief and safety-relief valve maintenance procedures, or other actions to improve valve reliability, prior to implementing the new testing program.

6. The addition of a footnote to Table 4.4-10 indicating that valve testing is required following valve repair, maintenance or replacement. This footnote also further clarifies the treatment of valve failures or successes occuring during such testing. Specifically, it eliminates the potential for unwarranted penalization should valves fail during testing following completion of maintenance activities, and it explicitly gives appropriate credit for test successes obtained from such testing.

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In response to our August letter on this subject, a number of licensees described plans for improving the reliability of one type of safety-relief valve. These plans were developed in consultation with General Electric Company. Some of these approaches were previously described by GE in a meeting with the staff last spring, and are being implemented to varying degrees at several plants. On the whole, we commend these efforts and are appreciative of the fact that the industry recognizes the need to improve the reliability of these valves. However, we cannot agree with the contention advanced by some licensees that the variable frequency testing program would significantly degrade valve reliability and thus should be deleted from the Model Technical Specifications. From a reactor safety standpoint, we have concluded that in-situ testing of these valves is the preferable method for adequately demonstrating their reliability. While we recognize that some small amount of degradation of the valve may occur as a result of testing, we do not believe that the valve operating time associated with the testing program is a significant fraction of the total average yearly operating time that has been historically associated with these valves. Therefore, we do not believe that the testing program will significantly contribute to valve unreliability. On the contrary, we have concluded that the variable frequency test schedule will improve overall pressure relief system reliability because it verifies system operability on a frequency based on demonstrated reliability. Under this scheme, plants with well maintained and reliable valves will not be required to perform any additional testing beyond what is currently required; conversely, plants with valve failures will be required to demonstrate reliability through additional testing. We consider the variable frequency test schedule to be essential to maintaining a more uniform level of reliability for this equipment and have retained this requirement in the Model Technical Specifications.

We request that within 30 days you propose changes to your Technical Specifications that incorporate the requirements of the enclosed revised Model Technical Specifications. If you have any questions, please contact us.

Sincerely,

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Model Technical Specifications

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3/4.4.2 SAFETY VALVES AND SAFETY-RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 At least the following reactor coolant system safety valves and safety-relief valves shall be operable with lift settings within ± 1% of the indicated pressures.

*(2) Safety valves @ (1240) psig
*(3) Safety-relief valves @ (1100) psig
*(3) Safety-relief valves @ (1090) psig
*(3) Safety-relief valves @ (1080) psig

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

With one or more of the above required reactor coolant system safety valves or the safety valve function of one or more of the above required safety-relief valves inoperable, either restore the inoperable valve(s) to operable status or be in at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 In addition to the applicable ASME Boiler and Pressure Vessel Code, Section XI requirements, each safety-relief valve shall be demonstrated operable:

a. At least once per 24 hours, by verifying bellows integrity through instrument indication.

b. Until March 1, 1979, at least once per 18 months by:

1. Manually opening each remotely operated safety-relief# valve with reactor steam dome pressure ≥ 100 psig, and verifying each valve opens by observing that either:

*Number to be consistent with ASME Code requirements and plant safety analyses. Do not include installed spares.

#With regard to the action statement of 3.4.2 above, a failure to open on manual actuation need not be considered a failure of the safety valve function of the valve.

SURVEILLANCE REQUIREMENTS (Continued)

a. The turbine bypass or control valve(s) indicate a compensating valve movement, or

b. The reactor coolant system pressure decreases by an amount equivalent to the valve pressure relieving capacity for the test conditions.

2. Conducting a visual inspection of the safety-relief valve line restraints in the torus to verify structural integrity for continued operation.

c. After March 1, 1979, by performance of the following test program:

1. Manually opening each remotely operated safety-relief valve in accordance with the test schedule of Table 4.4-10 with reactor steam dome pressure $\geq$ 100 psig, and verifying each valve opens by observing that either:

   a. The turbine bypass or control valve(s) indicate a compensating valve movement, or

   b. The reactor coolant system pressure decreases by an amount equivalent to the valve pressure relieving capacity for the test conditions.

2. The initial Next Required Test Interval of Table 4.4-10 shall be determined by the number of remotely operated relief and safety-relief valves found inoperable from March 1, 1978 to March 1, 1979.

3. The initial valve tests of Table 4.4-10 shall be completed by, the earlier of:

   a. The completion of the next refueling outage occurring after March 1, 1979, or

   b. The time period defined by March 1, 1979 plus the initial test interval, determined above.

4. At least once per 18 months, by conducting a visual inspection of the safety-relief valve line restraints in the torus to verify structural integrity for continued operation.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.2.2 Each safety valve and the safety valve function of each safety-relief valve shall be demonstrated operable per the requirements of the ASME Boiler and Pressure Vessel Code (   ) Edition and Addenda through (   ).

TABLE 4.4-10

REMOTELY OPERATED RELIEF AND SAFETY-RELIEF VALVE TEST SCHEDULE#

NUMBER OF REMOTELY OPERATED RELIEF AND SAFETY-RELIEF VALVES FOUND INOPERABLE DURING OPERATION, TESTING OR TEST INTERVAL**	NEXT REQUIRED TEST INTERVAL*
0	18 months $\pm$ 25%
1	184 days $\pm$ 25%
2	92 days $\pm$ 25%
$\geq$ 3	31 days $\pm$ 25%

* The required test interval shall not be lengthened more than one step at a time. Early tests may be performed prior to entering the "next required test interval" (i.e., in advance of the nominal time less the negative 25% tolerance band). Early tests may be used as a new reference point for tests of the same interval, however, they are not acceptable for lengthening the test interval.

**Setpoint drift is not considered to be a valve failure for the purposes of this test schedule.

Each affected remotely operated relief and safety-relief valve shall be demonstrated OPERABLE pursuant to Specifications 4.4.2.1.b.1, 4.4.2.1.c.1, 4.5.2.b.1 and 4.5.2.c.1, as applicable, within 36 hours after exceeding 100 psig, whenever maintenance, repair or replacement work is performed on a valve or its associated actuator. Successful tests performed under this provision may be used to satisfy the test requirements for a "required test interval" provided such tests are performed within the current "required test interval" and its associated tolerance band. Valve failures detected during testing under this provision shall not be considered inoperable valves for the purpose of this table.

3/4.4.2 SAFETY VALVES AND SAFETY RELIEF VALVES

The reactor coolant system safety valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of ______ psig in accordance with the ASME Code. Each spring loaded safety valve is designed to relieve ______ lbs per hour at the valve set point. The capacity of the relief/safety-relief valves is designed to meet the SAR stated requirement that these valves shall function to prevent opening of the spring loaded safety valves. The spring loaded safety valves are not expected to be required to function under the most limiting transient, assuming proper relief/safety-relief valve operation; however an inoperable valve requires shutdown in order to comply with ASME Code requirements.

The testing frequency applicable to the relief valve function of the safety-relief valves is provided to ensure operability and demonstrate reliability of the valves. This variable frequency test schedule becomes effective on March 1, 1979. This lead time is intended to permit resolution of the Mark I Safety-Relief Valve Loads and Structural Capability generic concern. The required testing interval varies with observed valve failures. The number of inoperable valves found during both operation and testing of these valves determines the time interval for the next required test of these valves. Early tests may be performed prior to entering the next required test interval (i.e., in advance of the nominal time less the negative 25% tolerance band). Early tests may be used as a new reference point for tests of the same time interval; however, they are not acceptable for lengthening the test interval since there were not performed within the ± 25% tolerance band as required by Table 4.4-10.

Demonstration of the safety valves and safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

EMERGENCY CORE COOLING SYSTEMS

AUTOMATIC DEPRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.5.2 The Automatic Depressurization System (ADS) shall be OPERABLE with at least (6)* OPERABLE ADS valves.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above required ADS valves inoperable, operation may continue provided the actuation logic of the remaining ADS valves is operable and the CSS and LPCI systems are operable, and the HPCI system is demonstrated operable within 4 hours; restore the inoperable ADS valve to operable status within 14 days or be in at least HOT SHUTDOWN within the following 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.2 In addition to the applicable ASME Boiler and Pressure Vessel Code, Section XI requirements, the ADS shall be demonstrated operable:

a. At least once per 18 months by performance of a system functional test which includes simulated automatic actuation through the automatic depressurization sequence, but excluding valve actuation.

b. Until March 1, 1979, at least once per 18 months by:

1. Manually opening each ADS valve with a reactor steam dome pressure $\geq$ 100 psig, and verifying each valve opens by observing that either:

*Number of ADS valves to be consistent with ECCS analysis.

a. The turbine bypass or control valve(s) indicate a compensating valve movement, or

b. The reactor coolant system pressure decreases by an amount equivalent to the valve pressure relieving capacity for the test conditions.

2. Conducting a visual inspection of the safety-relief and relief valve line restraints in the torus to verify structural integrity for continued operation.

c. After March 1, 1979, by performance of the following test program:

1. Manually opening each ADS valve in accordance with the test schedule of Table 4.4-10 with reactor steam dome pressure $\geq$ 100 psig, and verifying each valve opens by observing either:

a. The turbine bypass or control valve(s) indicate a compensating valve movement, or

b. The reactor coolant system pressure decreases by an amount equivalent to the valve pressure relieving capacity for the test conditions.

2. The initial Next Required Test Interval of Table 4.4-10 shall be determined by the number of remotely operated relief and safety-relief valves found inoperable from March 1, 1978 to March 1, 1979.

3. The initial valve tests of Table 4.4-10 shall be completed by, the earlier of:

a. The completion of the next refueling outage occurring after March 1, 1979, or

b. The time period defined by March 1, 1979 plus the initial test interval, determined above.

4. At least once per 18 months by conducting a visual inspection of the safety-relief and relief valve line restraints in the torus to verify structural integrity for continued operation.

BASES

3/4.5.2 AUTOMATIC DEPRESSURIZATION SYSTEM (ADS)

Upon failure of the HPCIS to function properly after a small break loss-of-coolant accident, the ADS automatically causes the safety-relief valves to open, depressurizing the reactor so that flow from the low pressure cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be operable whenever reactor vessel pressure exceeds (150) psig even though low pressure cooling systems provide adequate core cooling up to (350) psig.

ADS automatically controls (7) safety-relief valves although the safety analysis only takes credit for (6). Therefore only (6) ADS valves are required to be OPERABLE.

The testing frequency applicable to ADS valves is provided to ensure operability and demonstrate reliability of the valves. The required testing interval varies with observed valve failures. The number of inoperable valves found during both operation and testing of these valves determines the time interval for the next required test of these valves. Early tests may be performed prior to entering the next required test interval (i.e., in advance of the nominal time less the negative 25% tolerance band). Early tests may be used as a new reference point for tests of the same time interval, however, they are not acceptable for lengthening the test interval since they were not performed within the ±25% tolerance band as required by Table 4.4-10.

TABLE 1.X

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown	$> 212^{o}F$
4. COLD SHUTDOWN	Shutdown	$\leq 212^{o}F$
5. REFUELING*	Refuel or Startup	$\leq 212^{o}F$

*Reactor vessel head unbolted or removed and fuel in the vessel.