Enclosure

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REQUEST FOR RELIEF FROM THE APPROVED INSERVICE TESTING PROGRAM

FOR

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NUMBER 50-263

1.0 INTRODUCTION

The Code of Federal Regulations, 10 CFR 50.55a, requires that inservice testing (IST) of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda, except where relief has been requested and granted or proposed alternatives have been authorized by the Commission pursuant to 10 CFR 50.55a(f)(6)(i), (a)(3)(i), or (a)(3)(ii). In order to obtain authorization or relief, the licensee must demonstrate that: (1) conformance is impractical for its facility; (2) the proposed alternative provides an acceptable level of quality and safety; or (3) compliance would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Section 50.55a(f)(4)(iv) provides that inservice tests of pumps and valves may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed, and subject to Commission approval. NRC guidance contained in Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," provided alternatives to the code requirements determined to be acceptable to the staff and authorized the use of the alternatives in Positions 1, 2, 6, 7, 9, and 10 provided the licensee follows the guidance delineated in the applicable position. When an alternative is proposed which is in accordance with GL 89-04 guidance and is documented in the IST program, no further evaluation is required; however, implementation of the alternative is subject to NRC inspection.

Section 50.55a authorizes the Commission to grant relief from ASME Code requirements or to approve proposed alternatives upon making the necessary findings. The NRC staff's findings with respect to granting or not granting the relief requested or authorizing the proposed alternative as part of the licensee's IST program are contained in this safety evaluation.

By letter dated March 13, 1995, Northern States Power Company, the licensee for the Monticello Nuclear Generating Plant, submitted two relief requests pertaining to its IST program involving the exercising requirements for the main steam safety relief valves and residual heat removal service water (RHRSW) control valves.

The current 10-year interval for the Monticello Nuclear Generating Plant began May 31, 1992, and ends May 31, 2002. The current IST program is based on the requirements of the 1986 Edition of the ASME Code.

2.0 RELIEF REQUEST RSW-1

Relief Request RSW-1 involves two control valves located on the outlet line of the residual heat removal (RHR) heat exchangers. The control valves open as a function of the differential pressure developed between the inlet of the RHR process stream and the outlet of the RHRSW process stream. The differential pressure modulates the control valves open as necessary to provide a flow path for the RHRSW through the RHR heat exchanger. The licensee has requested relief from ASME Code, Section XI, paragraph IWV-3413 pertaining to the full-stroke time testing of power-operated valves, and Section XI, paragraph IWV-3417 which provides the corrective actions required for power-operated valves that have exceeded the limiting value of full-stroke time designated by the owner.

2.1 Licensee's Basis for Requesting Relief

The licensee states the following in support of Relief Request RSW-1:

> IWV-3413 requires that a limiting value of full stroke time be established for a power operated valve and that the stroke time be measured whenever such a valve is stroke tested. Performing full stroke time testing of these valves is impractical based on the control scheme design of the valves, adverse plant impact, and the functional requirements of the valves.
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> IWV-3413 states that *full stroke time* is that time interval from initiation of the actuating signal to the end of the actuating cycle. The control scheme design of these valves does not receive an actuation signal (neither by manual handswitch nor by automatic logic) to stroke to the position required to fulfill their safety function. RHRSW valves CV-1728 and CV-1729 are air operated control valves on the outlet line of the RHRSW side of the "A" and "B" RHR heat exchangers, respectively. These control valves maintain a differential pressure between the RHRSW process stream and the RHR process stream during RHRSW system operation. The valves are controlled by a positioner, controlled by a differential pressure indicating controller (DPIC). The DPIC senses pressure on the RHRSW discharge line and the RHR inlet line to the RHR heat exchanger. The desired differential pressure control point, and thus the desired valve position for system flow, is manually set by the operator by manual adjustment of the DPIC setpoint. The valve positioner positions the valve and modulates the valve position as necessary to maintain this control point. Stroke time testing of these valves on a quarterly basis is not consistent with the design of the valve's control scheme and is not in the interest of plant safety.
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> These valves are interlocked to receive a closed signal when the Residual Heat Removal Service Water (RHRSW) pumps are de-energized.

This interlock is provided to ensure that system water inventory is not lost during system shutdown. Stroke time testing of valves CV-1728 and CV-1729 when the RHRSW pumps are de-energized would result in the loss of liquid fill for a significant portion of the RHRSW system as well as require the bypassing of an interlock designed to minimize the potential for water hammer. Such testing increases the possibility of an adverse water hammer during startup of the RHRSW system as well as requires filling and venting of the system following the stroke time testing. In addition to the adverse impact on plant operation, such testing results in an undesirable burden on plant resources via the expenditure of person-hours and person-rem to perform system filling and venting.

Stroke time testing of the valves during RHRSW pump operation negates the loss of system fill concern; however, this testing would also have an adverse impact on plant safety and equipment integrity. Stroke time testing during pump operation would require the valve be initially in the closed position. Establishing the initial test conditions of a closed valve during pump operation would result in an undesirable deadheading of the pump. Subsequent opening of the valve to perform stroke time testing will result in pump runout if a single RHRSW pump is in operation, an undesirable condition which adversely impacts pump integrity and performance. The pump runout concern can be addressed by stroke timing the valve open during operation of both RHRSW pumps; however, this exacerbates the pump deadheading concerns and would result in undesirable transients on the system.

2.2 Alternate Testing

The licensee proposes:

IWV-3412 provides for demonstrating the necessary valve disk movement by observing indirect evidence (such as changes in system pressure, flow rate, level, or temperature), which reflect stem or disk position. The most representative test of the capability of valves CV-1728 and CV-1729 to perform their intended function is performed during inservice testing of the RHRSW pumps. Quarterly testing of the RHRSW pumps verifies the capability of the valves to operate properly to pass the maximum required accident flow as well as the recording of the valve position necessary to achieve required flow conditions. Testing of the valves in this manner demonstrates valve performance capability and provides a means to monitor for valve degradation.

2.3 Evaluation

2.3.1 *Background*

The RHRSW system removes the decay heat removed by the RHR system during normal shutdown and accident operations. Additionally, the RHRSW provides an emergency source of water for the reactor core through the RHR-RHRSW intertie. The function of control valves CV-1728 and CV-1729 is to modulate RHRSW coolant flow according to the pressure differential across the RHRSW discharge line and the RHR inlet line to the RHR heat exchanger. The control valves CV-1728 and CV-1729 are manually positioned from the control room by adjusting the DPIC to establish the desired RHRSW flow conditions. The licensee states that the valves are not required to stroke full open in order to perform their required function. According to the Monticello Updated Final Safety Analysis Report (UFSAR), Section 10.4.2.3, the differential pressure control valve is interlocked with the RHRSW pumps such that the valve actuator instrument air solenoid is energized only when a pump is in service. The valves are closed and the solenoid is deenergized during normal plant operation when the RHR system is not in service.

The primary safety function of CV-1728 and CV-1729 is to remove decay heat from the RHR system when it is in shutdown cooling or containment spray/cooling mode. In the case of an emergency core cooling system (ECCS) initiation signal the RHRSW pumps will automatically trip thereby deenergizing the differential pressure control valve solenoid. Once the ECCS condition has cleared and the reactor water level is being maintained, the RHRSW system must be manually started and the flow established by the operator using the differential pressure indicating controller located in the control room. There is no fixed time requirement for this system to be placed into operation following a design-basis accident loss of coolant accident (LOCA) condition since at least 8 hours are available before the containment design pressure is reached. Therefore, stroke-time testing these valves is not indicative of their ability to perform the safety function described by UFSAR Section 6.2.3.2.1.

2.3.2 *Code Requirements*

The licensee requests relief from Section XI of the ASME "Inservice Testing Requirements for Valves," paragraphs IWV-3413 and IWV-3417, because of impractical test conditions during power operation and because the stroke-time testing of these control valves on a quarterly basis is not consistent with the design of the valves control scheme. IWV-3413 requires that all power-operated valves shall be full-stroke-time tested with a frequency as described in paragraph IWV-3411. Stroke-time testing valves provides a means for monitoring for degrading conditions. IWV-3417 provides the corrective action statements for those valves which exceed the stroke-time acceptance criteria.

2.3.3 *Impracticality of Code Requirements IWV-3413 AND IWV-3417 for RHRSW Control Valves*

The water inventory in the RHRSW piping system remains full when the system is not in operation to maintain a readily available coolant source and to minimize the potential for water hammer during RHRSW pump startup. Stroke-time testing these control valves according to IWV-3413 would require the licensee to fill and vent the system following the test. Filling and venting the RHRSW system requires the expenditure of person-hours and more importantly, person-rem without a significant increase in the licensee's ability to monitor for valve degradation.

Reasonably, the filling and venting would not be necessary if the valves were stroke-time tested during pump operation. However, testing the valve from the fully closed to the fully opened position while the pump(s) is running can create one of two adverse conditions:

(1) If the valve is fully closed when the pump is running, it is likely the pump head will reach its shutoff value. At shutoff head, the maintainable flow rate is reduced to zero and the resistance to flow is greater than the power the centrifugal pump can impart to the fluid. Deadheading the pump in this manner can severely degrade the performance of the pump due to rapid overheating.

(2) Pump runout occurs when the system backpressure is very low and the fluid flows through the pump without absorbing very much energy. Under these conditions, the pumping process is inefficient and the pump undergoes extreme mechanical stresses. Stroking the control valve to the full-open position when only one RHRSW pump is running will result in pump runout which is an undesirable condition.

Pump runout is avoided by running both of the RHRSW pumps when testing the valves, but the shutoff head conditions are worsened by this condition.

The licensee states that the control scheme of the valves is not consistent with the requirements of IWV-3413. The valve position is dialed into the DPIC in the control room and the valve positioner modulates according to the set point. Stroke-time testing these control valves is not indicative of their ability to perform the functions outlined in the UFSAR, Section 10.4.2.3.

2.3.4 *Conclusion*

Paragraph IWV-3412 of the ASME Code states that the valves for which it is impractical to full-stroke exercise during plant operations shall be part-stroke exercised during power operation and full-stroked exercised during cold shutdowns. The licensee proposes to part-stroke test the RHRSW control valves on a quarterly basis concurrent with the RHRSW pump tests and verify the capability of the valves to pass the required accident flow. In addition, the valve disk position will be determined as that which is necessary to achieve the required flow conditions.

In NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," Section 4.2.9, the staff recommends finding an alternative method (when the code requirements for stroke timing valves are impractical) to monitor valve degradation, preferably one that includes stroke timing with acoustic monitors or other non-intrusive methods. The NUREG guidance does however, allow for alternative testing for which the licensee must obtain relief, if stroke-time testing is impractical. In this case, stroke timing is not a practical test due to the obvious adverse effects on the other components in the system. The proposed alternative (which relies on existing code requirements) provides a measure of the valves' capability to perform their safety function, even though degradation is monitored on a more broad basis than if stroke times were measured.

Per a telephone conversation on June 7, 1995, the licensee stated that the control valves are currently in a mechanical maintenance program in which the internals of these valves are inspected every third cycle of plant operation. The valve stem, valve body and the associated piping are visually inspected for signs of degradation. Inspecting the valve in this manner provides another means for the licensee to monitor the valve for degradation.

Relief from the stroke-timing requirements is granted pursuant to 10 CFR 50.55a(f)(6)(i) based on the impracticality of performing testing in accordance with code requirements, and in consideration of the burden on the licensee that could result if the code requirements were imposed on the facility. The proposed alternative testing will provide a level of assurance of the operational readiness of the valves, considering that the code-required testing is impractical.

3.0 RELIEF REQUEST NB-1

The licensee has revised its alternate testing plan for the main steam safety relief valves RV-2-71A, RV-2-71B, RV-2-71C, RV-2-71D, RV-2-71E, RV-2-71F, RV-2-71G and RV-2-71H. These valves can be actuated to rapidly relieve steam pressure and are designated as automatic depressurization system (ADS) valves. Relief from full-stroke-time testing during power operation was granted by letter dated July 6, 1993, based on the licensee's acceptable alternate testing. Subsequently, (January 3 and August 29, 1994), the licensee submitted a Technical Specification (TS) change request (approved as Amendment No. 92) which altered the surveillance requirements for the safety relief valves. Currently, the licensee has resubmitted relief request NB-1 with the revised alternate testing plan for approval.

3.1 Licensee's Basis for Requesting Relief

The licensee states the following in support of relief request NB-1:

> These valves have an active, self actuation safety function to open and relieve an overpressurization condition in the reactor vessel. The valves also have an auxiliary actuating device (as defined in OM-1-1981) that acts to open the valves and depressurize the reactor vessel. This function allows low pressure emergency core cooling systems to inject during a LOCA (ADS) and also controls reactor pressure in certain design transients (low-low set). The ADS automatic actuation provides for depressurizing the reactor vessel to permit low pressure ECCS injection during a loss of coolant accident. The low-low set automatic actuation provides control of the opening and closing setpoint following a scram during pressurization transients.
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> Although these valves have an auxiliary actuating device, they can not be treated and tested as category B, power-operated valves. The requirements to stroke time them quarterly is not within the design capability of the valves or the plant. First of all, these valves have a passive safety function to remain closed and provide reactor coolant

pressure boundary. Exercising them during power operation would cause a severe reactor power and pressure transient that could result in a reactor scram. It also increases the potential for second stage or pilot stage leakage due to seat wear or fouling. If this leakage is high enough, it can cause a spurious SRV [Safety/relief valve] lift and prevent reclosure of the valve. This condition is equivalent to an unisolable small break LOCA event. Such challenges to the ADS function of the main steam safety/relief valves should be minimized. Therefore justification to perform an exercise test on a once-per-cycle frequency is well established.

The reason the valves can not be stroke timed to provide meaningful data is that there is no direct indication of valve position. Their stroke time is less than one second and can only be measured by indirect means such as changes in SRV discharge line pressure and temperature indication. This results in a high degree of variability in the measured stroke times due to other plant variables such as valve pneumatic supply pressure, reactor pressure, turbine bypass valve condition, measuring instrumentation response time, etc. This prohibits repeatable test conditions without a heavy burden on the licensee to fix these parameters during each test to within a narrow enough range commensurate with a less than one second stroke time. This burden is not offset by an increase in public safety.

3.2 Alternate Testing

The licensee proposes:

These valves will be monitored for degradation by testing them in accordance with all applicable sections of OM-1-1981, including the requirements for auxiliary actuating devices. This testing is in accordance with the Monticello plant technical specifications. The valves will also be tested in accordance with Technical Specification 4.6.E which includes disassembly and inspection of at least two SRVs each refueling outage.

All SRVs will be exercised at reduced reactor pressure, in place, each operating cycle to verify the open and close capability of the valve.

3.3 Evaluation

By letter dated December 7, 1992, the licensee submitted relief request NB-1 for approval as part of "Revision 1 to Third Ten-Year Inservice Testing Program." Relief was granted on an interim basis pending a second submittal outlining the licensee's development of a method to test these valves in a manner that yields meaningful information for monitoring valve degradation. By letter dated July 6, 1993, the Commission granted relief for NB-1 based on an enhanced maintenance plan and testing consistent with the OM-1-1981 Code. The staff issued Amendment 92 to Facility Operating Licensee No. DRP-22 for the Monticello Station by letter dated September 15, 1994. This amendment revised the requirement that a minimum of seven of the eight safety/relief valves be bench checked or replaced with bench-checked valves each refueling outage. The current submittal revises the licensee's alternate testing procedures to be consistent with the amended TS and the applicable sections of OM-1-1981, including the requirements for auxiliary actuating devices.

OM-1-1981 is the standard developed by the ASME Operation and Maintenance Committee which provides the general requirements for performance testing and monitoring of nuclear power plant pressure relief devices. The guidance in OM-1-1981 is the required testing for relief valves per IWV-3511 in the 1986 edition of the ASME Code. For relief valves with auxiliary actuation devices OM-1-1981, paragraph 3.3.1.1 requires the following testing for boiling water reactor pressure relief devices.

> Tests prior to maintenance or set pressure adjustments, or both, shall be performed in the following sequence:
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> (a) visual examination,
> (b) seat tightness determination,
> (c) set pressure determination,
> (d) determination of compliance with the Owner's seat tightness criteria,
> (e) determination of electrical characteristics and pressure integrity of solenoid valve(s),
> (f) determination of pressure integrity and stroke capability of air actuator,
> (g) determination of operation and electrical characteristics of position indicators,
> (h) Determination of operation and electrical characteristics bellows alarm switch,
> (i) determination of actuating pressure of auxiliary actuating device sensing element, where applicable, and electrical continuity.

The staff has concluded that testing the ADS/SRVs of boiling water reactors in accordance with the ASME testing requirements outlined in OM-1-1981 (as listed above) is an acceptable alternative for valve exercise testing as required in ASME, Section XI, subarticle IWV-3410, and that requiring stroke timing in addition to the OM-1 test requirements is not necessary to assure operational readiness of these valves.

3.4 Conclusion

The proposed alternative to the Code requirements is authorized pursuant to 10 CFR 50.55a(a)(3)(i) based on the alternative providing an acceptable level of quality and safety.

4.0 CONCLUSION

Based on the review of Monticello Nuclear Generating Plant IST Program Relief Request RSW-1, the Commission grants relief pursuant to 10 CFR 50.55a(f)(6)(i). The Commission authorizes Relief Request NB-1 pursuant to 10 CFR 50.55a (a)(3)(i). These actions are authorized by law and will not endanger life or property, or the common defense and security and are otherwise in the public interest, giving due consideration to the burden upon the Northern States Power Company that could result if the ASME Code requirements were imposed on the facility.

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Date: July 1995