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Mr. Giambusso				

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DESCRIPTION:

Ltr reporting an incident on 12-19-72, in which three primary containment isolation valves were found to have seat leakage greater than the Tech Spec limit of 17.2scfh at 41 psig.........

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PLANT NAMES: Monticello

ENCLOSURES:

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FOR ACTION/INFORMATION 1-29-73 AB

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Regulatory File Cy.

NSP

NORTHERN STATES POWER COMPANY
Minneapolis, Minnesota 55401

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RECEIVED JAN 29 1973 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

January 25, 1973

Mr. A. Giambusso
Deputy Director for Reactor Projects
United States Atomic Energy Commission
Washington, D C 20545

Dear Mr. Giambusso:

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

Primary Containment Isolation Valve Leakage

A condition occurred at the Monticello Nuclear Generating Plant recently which we are reporting to your office in accordance with provisions of Section 6.6.B.3 of Appendix A, Technical Specifications, of the Provisional Operating License DPR-22.

Three primary containment isolation valves were found to have seat leakage greater than the Technical Specification limit of 17.2 scfh at 41 psig. The leaking valves were AO-2378 (Suppression chamber purge inlet valve), AO-2380 (Reactor building to suppression chamber vacuum breaker valve), and AO-2383 (Suppression chamber purge outlet valve). These valves are 18" and 20" offset disc butterfly valves with a "T" shaped inflatable seal for positive seat sealing. In each case a redundant primary containment isolation valve on the same line was found to be within the Technical Specification limit.

On December 19, 1972, while pressurizing the drywell through the 18" air purge supply line for the purpose of leak testing the torus to drywell vacuum breaker valves, personnel inside the torus detected leakage through AO-2378. A local leak rate test was attempted on the section of the piping bounded by AO-2378, AO-2381 (the drywell purge inlet valve) and AO-2377 (the purge line isolation valve outboard of AO-2378 and AO-2381). Due to the extent of the leakage, the piping section could not be pressurized to the local leak test pressure required by the Technical Specifications (41 psig). AO-2378 was cycled several times but proper disc sealing could not be obtained.

It was postulated that the excessive leakage might be due to either improper adjustment of the valve linkage or degradation of the resilient seal. The valve actuator to stem linkage was adjusted to position the valve disc on a new area of the seal. A subsequent leak rate test indicated a total leakage of 17.14 scfh past the three valves.

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As a result of the AO-2378 leakage, all similar valves in the plant were tested by pressurizing sections of piping between isolation valves. These tests, which were completed on January 17, 1973, indicated excessive leakage from the sections of piping containing valves AO-2380 and AO-2383. Adjustment of the actuator to stem linkage on AO-2383 decreased the measured leakage to 9.36 scfh. In addition to linkage adjustment, it was necessary to exercise the seal on AO-2380 by alternating applications of pressure and vacuum to obtain acceptable leakage. To assure that the seal was truly operating freely, AO-2380 was cycled 25 times and then leak tested with acceptable results after 3 additional cycles. The final leakage measurement was 7.24 scfh.

The problem will be investigated further during the refueling outage planned for March, 1973 and the results of this investigation will dictate what further action is required on these three valves as well as similarly constructed valves. At present, all suppression chamber and drywell purge inlet and outlet valves and the reactor building to suppression chamber valves are closed and sealed properly. To verify that this sealing capability is maintained between now and the refueling outage, a leak rate test will be performed following each cycling of these valves.

In the initial integrated leak rate test of the primary containment, the total leakage was determined to be 0.437 weight percent of the contained air during a 24 hour period. Taking the conservative approach that the present leak rates are attributable to the primary containment valve on each line that was not adjusted, and adding 17.14 scfh, 9.36 scfh, and 7.24 scfh to the previous total leakage, the total leakage becomes .537 weight percent. This leakage is within the Technical Specification limit of 1.2 weight percent.

An Abnormal Occurrence report will be available at the site for the Regulatory Operations inspector.

Very truly yours,

L. O. Mayer

L. O. Mayer, PE
Director of Nuclear Support Services

LOM/kik

cc: B H Grier
G Charnoff
Minnesota Pollution Control Agency, Attn: K. Dzugan