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DESCRIPTION: Ltr rpt on 8-31-72 the failure of PEECO Flow Switch Actuator Paddles.....

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PLANT NAMES: Monticello Plant

ENCLOSURES:

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ACKNOWLEDGED

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NSP

NORTHERN STATES POWER COMPANY

Minneapolis, Minnesota 55401

September 26, 1972

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

Dear Mr. Giambusso:

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

Failure of PEECO Flow Switch Actuator Paddles

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

On Thursday, August 31, 1972, an inspection of the four RHR pump minimum flow protection flow switches for actuator paddle integrity was performed. The inspection revealed that a large piece of the paddle on No. 13 RHR pump flow switch and a small piece of the paddle on No. 11 RHR pump flow switch had broken off and were presumably carried down the associated RHR lines with flow.

Summary Description

As a result of recent actuator paddle integrity problems encountered with PEECO flow switches installed at other sites, a study was conducted to determine which of the PEECO flow switch applications at Monticello should be inspected. The following PEECO flow switch applications were selected on the basis that the flow switches are installed on lines which either directly or indirectly discharge to the reactor vessel:

RHR Pump Minimum Flow Control Switches (4)
Standby Liquid Control System High Flow Switch (1)
Reactor Water Cleanup Low Flow Pump Trip Switches (1)
HPCI Cooling Water Low Flow Alarm Switches (1)

On August 31, the four RHR pump minimum flow control flow switches were inspected. The inspection revealed that a 1 1/8" x 7 1/2" section of the paddle on No. 13 RHR pump flow switch and a 1 1/8" x 3 1/2" section of one lamination on No. 11 RHR pump flow switch paddle were missing and were presumably carried down the associated RHR lines with flow. The paddles of the flow switches on RHR pumps 11, 12, and 14 are

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constructed of a lamination of four 1 1/8" x 8 1/2" stainless steel strips. The paddle of No. 13 RHR pump flow switch was constructed of one piece of 1 1/8" x 8 1/2" stainless steel which was noticeably thicker than any one lamination of the other flow switch paddles. The paddles of the flow switches which still had the paddles attached were noticeably bent in the direction of flow. The deformation was greatest at the paddle-to-actuator arm connection. All paddle mounting screws were found in place.

Paddle failures were observed in the flow switches associated with RHR pumps 11 and 13 only. It is believed that the failures can be correlated to the number of hours of associated pump operation since pumps 11 and 13 have received approximately nine times the operating hours of RHR pumps 12 and 14.

Safety Evaluation

The flow path downstream of the PEECO switches follows a tortuous path through long lengths of piping, elbows, valves and heat exchangers and eventually to several locations, the most important of which is the recirculation system piping.

There are many places for loose pieces to be retained in the RHR system, the most likely of which is the RHR heat exchanger shell side. In this heat exchanger the inlet and outlet lines are 180^{o} apart with the outlet 7 to 10 ft. above the inlet. The tube bundle arrangement is such that passage of any pieces of the switches is virtually impossible.

If, by some means, the loose parts managed to pass through the heat exchanger, they would have to negotiate a twisting path with a direct vertical rise of greater than 60 feet to enter piping systems leading to the reactor. These systems, too, present difficult passage for broken pieces. Nevertheless, for the purpose of determining potential consequences, passage of the missing switch pieces into the reactor plenum is assumed.

Once swept into the lower plenum of the reactor where coolant velocities in most locations are low because of the large clearances for coolant, the chances of a metallic piece moving up into a fuel flow passage are exceedingly small. Continuing hypothetically, however, this is assumed to occur to assess the potential blockage effects.

Each fuel bundle inlet contains an orifice - 1.381" or 2.148" in diameter. Eighty-four Monticello bundles have the smaller size orifice and all are located at the edge of the core, i.e., the very outside row of bundles. The remaining bundles have the larger size orifice. If the missing switch piece could work its way up to the support assembly and lodge across the orifice, total flow blockage would still not occur. Blockage at the orifice is the worst case situation since coolant flow to the bundle is restricted the greatest amount at the orifice. Flow area blockage of 94% in an average channel or 88% in a hot channel would reduce MCHFR to 1.0 (Ref. 1).

A switch piece (maximum possible width of 1 1/8") lodged exactly centered across one of the smaller-diameter orifices would reduce the flow area 90.7%. The same size switch piece lodged exactly centered across the larger diameter orifice will reduce the flow area 63.5%. Since the smaller diameter orificed channels are located only at the very edge of the core, these channels have been determined to

Ref. 1: NEDO-10174, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor", May 1970.

correspond to less than the average channel in the case analyzed. (Ref. 1). In all cases, MCHFR will remain above 1.0.

Pieces small enough to pass through the orifice plates would not be sufficiently large to cause significant flow reduction through the tie plate. (Ref. 1)

If the loose piece was small enough to pass through the 3/8" diameter tie plate holes, it could conceivably become lodged in a fuel coolant passage and cause localized heating. Should a fuel rod fail at the localized hot spot, the radioactivity release would be quite small. Even the inconceivable failure of all 49 rods in a bundle would not result in the release of activity in excess of 10 CFR 20 limits (Ref. 1).

Thus, there are no serious safety problems or potentially serious safety problems associated with reactor operation with the missing switch pieces unlocated.

Corrective Action

One inch triangular shaped paddle segments were left on all four flow switches. The screws which secure the triangular segments were peened over to prevent the screws from backing out due to vibration during operation. The switches were re-installed and calibrated as accurately as possible to RHR loop indicated flow. Flow was cycled through the trip setting several times to verify repeatable operation of each flow switch.

Further failure of the paddles currently installed in the RHR system which were modified as described above is considered unlikely. However; to further reduce the potential for failures, the feasibility of replacing all paddle-type flow switches currently used in critical systems with differential pressure and rotometer flow sensing devices is being evaluated.

The inspection of the HPCI cooling water low flow alarm switch has been completed since the occurrence. The paddle of this switch, which is much shorter than the RHR flowswitch paddles, showed no signs of failure. The switch was re-installed, but will eventually be removed from the system since it is redundant to other more reliable indications of system status.

It is intended that the Reactor Water Cleanup system low flow pump trip switch be inspected when the plant is in a cold shutdown condition. When the inspection is performed, the paddle type flow switch will be removed from the system.

It is intended that the Standby Liquid Control system flow switch be inspected during the next refueling outage. It is considered highly unlikely that the paddle would fail since the flow switch has not been subjected to flow conditions.

Yours very truly,

L O Mayer /DJ

L. O. Mayer, P.E.
Director Nuclear Support Services

LOM/mmm

cc: B H Grier