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DATE: March 4, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: McGuire Nuclear Station Unit 1 and 2
Docket No. 50-369
Licensee Event Report 369/96-01, Revision 0
Problem Investigation Process No.: 0-M96-0314

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 369/96-01, Revision 0, concerning a Unit 1 Manual Reactor Trip. This report is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

T.C. McMeekin

JWP/bcb

Attachment

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JE22

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98



| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| McGuire Nuclear Station Unit 1 | 05000 369 | 1 of 7 |

TITLE (4)
A Unit 1 Manual Reactor Trip Was Initiated As A Result Of An Equipment Failure Caused By An Unknown

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAME | DOCKET NUMBER(S) |
| | | | | | | | | | | 05000 |
| 02 | 03 | 96 | 96 | - 001 | - 00 | 03 | 04 | 96 | N/A | 05000 |

| OPERATING MODE (9) | 1 |
|---|---|
| POWER LEVEL (10) | 60% |

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (*Check one or more of the following*) (11)

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | 20.402(b) | | 20.405(c) | X | 50.73(a)(2)(iv) | | 73.71(b) |
| | 20.405(a)(1)(i) | | 50.36(c)(1) | | 50.73(a)(2)(v) | | 73.71(c) |
| | 20.405(a)(1)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(vii) | | OTHER (*Specify in Abstract below and in Text, NRC Form 366A*) |
| | 20.405(a)(1)(iii) | | 50.73(a)(2)(i) | | 50.73(a)(2)(viii)(A) | | |
| | 20.405(a)(1)(iv) | | 50.73(a)(2)(ii) | | 50.73(a)(2)(viii)(B) | | |
| | 20.405(a)(1)(v) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) | | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (AREA CODE) | |
|---|---|---|
| J. W. Pitesa | (704) | 875-4788 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| X | WBD | VALVOP | F130 | YES | | | | | |
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|---|
| | YES (*f yes, complete EXPECTED SUBMISSION DATE*) | X | NO | | | | |

ABSTRACT (*Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines*) (16)

**Unit Status:** Unit 1 - Mode 1 (Power Operations) at 60 percent power.

**Event Description:** On February 3, 1996, during the performance of routine Train 1A Safety Injection Slave Relay Testing, a manual Reactor Trip was initiated when the 1A Reactor Coolant Pump Upper Thrust Bearing reached the administrative limit of 195 degrees. Prior to the trip, Operations personnel had noted a steady increase in the bearing temperature and had begun a rapid power reduction. At approximately 60 percent Reactor Power the administrative trip criteria for the pump was reached and the Reactor was manually tripped as directed by procedure requirements at 1419:15. As soon as the Reactor was tripped, the 1A Reactor Coolant Pump was tripped. All systems responded as required after the trip. Unit 1 was returned to service on February 5, 1996, at 0332.

**Event Cause:** A cause of Equipment Failure due to an unknown cause has been assigned. The increase in 1A Reactor Coolant Pump Upper Motor Bearing temperature was caused when the Component Cooling system flow control valve for the bearing failed in the closed position due to a failure in the associated control loop; however, the exact mode of failure could not be determined.

**Corrective Action:** The associated systems and components were evaluated/tested and determined to be operable.

EVALUATION:

Description of Event

**On February 3, 1996, Unit 1 was in Mode 1 (Power Operation) at 100 percent power. Train A of Unit 1 KC system was being operated in a depressurized alignment with valves [EIIS:V] 1KC-0001A, Engineered Safety Features (ESF) Auxiliary Building (AB)[EIIS:NM] Non-Essential Return Automatic Isolation, and 1KC-0050A, ESF AB Non-Essential Header Automatic Isolation closed due to equipment problems associated with valve 1KC-0001A. Also, the Unit was operating utilizing one Component Cooling (KC) system [EIIS:CC] Pump [EIIS:P] instead of two for a trial period to confirm the feasibility of adopting this as a regular operating alignment for both units. Therefore, only the 1B1 KC Pump was in service at that time.**

- Routine Safety Injection Slave Relay testing was in progress on Train 1A using procedure PT/1/A/4200/28A, Train A Slave Relay Test.
- Upon completion of the relay actuation and verification of valve movement, Operations (OPS) Test Group personnel and an OPS Reactor Operator (RO) began the performance of step 12.24.17 of the procedure, which verifies return of the valves associated with the test to the initial position.
- When the OPS Test Group Person began to read the position for valve 1KC-0050A, the OPS RO inadvertently depressed the open push-button associated with the valve prior to completion of the communication stating that the valve should be in the closed position.
- Due to the system alignment at that time (valve 1KC-0056A, Residual Heat Removal (ND) system [EIIS:BP] Heat Exchanger (HX) [EIIS:HX] 1A Automatic Supply, open), a flow and pressure transient occurred in the KC system until valve 1KC-0050A could be returned to the closed position.
- OPS response included starting the 1B2 KC Pump.
- Once valve 1KC-0050A was closed again, the transient to the KC system was over. No further Operator action was necessary, and the 1B2 KC Pump was secured.

- Approximately eight minutes later, at 1354:13, OPS Control Room Personnel received an alarm [EIIS:ALM] for Reactor Coolant (NC) system [EIIS:AB] Pump 1A Upper Thrust Bearing high temperature. The computer indication at that time was noted at 141.2 degrees and increasing.

- The 1B2 KC Pump was started in response to the alarm.

- At 1357:44, the alarm for high vibration on the 1A NC Pump was received, requiring increased monitoring.

- At 1406, in response to these indications, OPS personnel entered procedure AP/1/A/5500/08, Malfunction OF NC Pump, and then AP/1/A/5500/04, Rapid Downpower, in an attempt to rapidly reduce Reactor [EIIS:RCT] power below the P-8 trip setpoint (48 percent Reactor Power) in order to secure the 1A NC Pump.

- While this was underway further actions were taken in an attempt to increase the KC system flow to the 1A NC Pump, but were unsuccessful.

- The 1A NC Pump Upper Thrust Bearing temperature continued to increase and, at 1419:15, reached 195 degrees which is the trip criteria as specified by procedure. At that time OPS personnel manually tripped the Reactor and then manually tripped the 1A NC Pump. An automatic Turbine Generator [EIIS:TG] Trip followed the Reactor Trip.

- OPS personnel entered procedure EP/1/A/5000/E0, Reactor Trip Or Safety Injection, and then entered procedure EP/1/A/5000/ES-0.1, Reactor Trip Response.

- The 4 hour notification to the NRC was made at 1507 in accordance with procedure RP/0/A/5700/10, NRC Immediate Notification Requirements.

- OPS, Maintenance, and Engineering personnel began an investigation to determine the cause of the event. It was discovered that valve 1KC-0488, NC Pump 1A Upper Bearing Outlet Flow, had moved to and remained in the closed position which had stopped KC system flow to the upper bearings.

- After initial investigation and troubleshooting of the components associated with the valve were completed, it was determined that a significant flow and pressure perturbation (not associated with any

water hammer)had been created during the transient causing the control valves for KC system flow associated with all four NC pumps to cycle in an extreme manner. However, after the transient to the KC system was over, valve 1KC-0488 remained in the closed position.

- No exact cause for the failure could be identified. All associated equipment was subsequently tested/verified to be operable.

- The Unit was returned to Mode 1, on February 5, 1996, at 0322.

Conclusion

**This event did not result in any uncontrolled releases of radioactive material, personnel injuries, radiation overexposures. The event is Nuclear Plant Reliability Data System (NPRDS) reportable due to the possible failure of the valve operator for valve 1KC-0488.**

- The failure which caused valve 1KC-0488 to remain in the closed position has been assigned a cause of Equipment Failure due to an Unknown, possible equipment failure in the associated control loop for the valve. The increase in 1A NC Pump Upper Bearing temperature was caused by the loss of KC system flow due to the valve remaining closed. Extensive investigation efforts have been unable to determine the exact mode of failure for the equipment which caused the valve to remain closed. Analysis of the KC system transient revealed that valve 1KC-0488 was required to quickly reposition three times. It failed closed after the third repositioning.

- Further analysis of the failure of the high flow signal to clear has revealed that a possible cause of the failure was blockage of the low pressure side of the flow transmitter manifold block valve by a small amount of debris stirred up from the system transient. The manifold block valves are maintained in a throttled position to limit the effects of pressure oscillations on the flow signal to the control instrumentation. This throttled condition makes the valve seat clearance very small, such that it could be susceptible to blockage by a very small amount of debris.

- The high and low pressure impulse lines of all four upper bearing cooling water flow loops were blown down in order to remove any potential debris and to help prevent future blockage. No evidence of any debris were found during this blowdown.

- Three other potential failure modes were identified. These were temporary backchecking of the low pressure manifold block valve against the valve seat, temporary blockage of the pneumatic controller setpoint restriction orifice [EIIS:OR], and temporary blockage of the instrument air associated with the flow transmitter [EIIS:FT] nozzle [EIIS:NZL]. Each of these was eliminated due to low probability.

A review of the Operating Experience Program (OEP) and Problem Investigation Process (PIP) data bases for the past 24 months revealed no similar reportable events. This event is not considered to be recurring.

CORRECTIVE ACTION:

Immediate:

1. OPS personnel entered procedure AP/1/A/5500/08, Malfunction OF NC Pump, and then AP/1/A/5500/04, Rapid Downpower.

2. OPS personnel attempted to increase the KC system flow to the 1A NC Pump.

3. OPS personnel initiated a Manual Reactor Trip.

4. OPS personnel entered procedure EP/2/A/5500/E0, Reactor Trip Or Safety Injection, and then entered procedure EP/1/A/5000/ES-0.1, Reactor Trip Response.

Subsequent:

1. OPS, Maintenance, and Engineering personnel began an investigation to determine the cause of the event.

2. The KC system, 1B1 KC Pump, 1A NC Pump, valve 1KC-0488A, and all associated equipment were verified to be operable.

3. The high and low pressure impulse lines of all four upper bearing cooling water flow loops were blown down in order to insure clear instrumentation tubing.

Planned:

1. The details of this event will be covered with appropriate Operations personnel.

2. Engineering personnel will evaluate the design of the control function and associated instrumentation for controlling KC flow to the NC Pumps.

SAFETY ANALYSIS:

**Based on this analysis, this event is not considered to be significant. At no time were the health and safety of the public or plant personnel affected by this event.**

- The Safety Related portion of the KC system is designed to provide cooling to the redundant trains of the ND system, via the ND system HXs, during the Containment Sump Recirculation phase of a Loss Of Coolant Accident (LOCA). This Design Basis function is achieved by isolating flow from the normally in-service Auxiliary Building Non-Essential Header on a Safety Injection Signal, as well as isolating flow from the normally in-service Reactor Building Non-Essential Header on a High-High Containment Pressure Signal.

- Isolation of these headers ensures that adequate flow is diverted to the respective ND system HXs. However, flow to the Reactor Building Header is not isolated on a Safety Injection Signal so that the NC Pump Thermal Barrier HXs continue to receive cooling in non-LOCA events.

- The transient associated with this event had no effect on the ability of the KC system to provide the above Design Basis function.

- Post transient monitoring of the 1B1 KC pump revealed no increase in pump vibration. Additional Engineering evaluation has concluded that long term reliability of the 1B1 pump was not affected by this transient.

- The flow path which produced the transient would have been isolated had a Safety Injection Signal been initiated during the transient.

- Closure of valves by a Safety Injection Signal would have isolated the discharge headers of A and B train, eliminating the cross train flow, and restoring NPSH to the B train pumps. The ability of these valves to close was unaffected by the transient. The required flow would therefore have been available to provide cooling of the ND system HXs.

- The Main Feedwater (CF) system [EIIS:SJ] was available after the trip and continued to provide feedwater flow.

- The resulting secondary side transient as a result of the trip caused the B Steam Generator (SG) [EIIS:SG] level to reach the Low Low Reactor Trip Logic setpoint and the Auxiliary Feedwater (CA) system [EIIS:BA] Motor [EIIS:MO] Driven Pumps started as designed and operated properly to assist in returning SG levels to normal.

- The Main Steam Line Code Safety Valves [EIIS:RV] and SG Power Operated Relief Valves did not operate nor were they challenged. All Steam Dump To Condenser Valves operated properly. No Atmospheric Dump Valves opened. No Pressurizer [EIIS:PZR] Code Safety or Pressurizer Power Operated Relief Valves opened, nor were the setpoints for these valves reached.

- The Primary and Secondary plant parameters were stabilized at no load conditions within 30 minutes following the trip and all plant equipment responded as expected.