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DATE: June 2, 1997

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 369/97-04, Revision 1
Problem Investigation Process No.: PIP 0 M97-1101

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 369/97-04, Revision 1, concerning Main Steam Safety Valve Technical Specifications Inaccuracies. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(v). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Bryan J. Dolan for HB Barron

H. B. Barron

JWP/bcb

Attachment

cc: Mr. L. A. Reyes
U.S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30323

INPO Records Center
700 Galleria Parkway
Atlanta, GA 30339

Mr. Victor Nerses
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. Scott Shaeffer
NRC Resident Inspector
McGuire Nuclear Station



IE22

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

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

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| McGuire Nuclear Station, Unit 1 | 05000 369 | 1 of 4 |

TITLE (4)
Main Steam Safety Valve Technical Specifications Inaccuracies

| 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) |
| 03 | 20 | 97 | 97 | - | 04 | - | 01 | 06 | 02 | 97 | McGuire, Unit 2 | 05000 370 |

| OPERATING MODE (9) | 1 | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (Check one or more of the following) (11) | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| POWER LEVEL (10) | 100 | | 20.402(b) | | 20.405(c) | | 50.73(a)(2)(iv) | | 73.71(b) | | |
| | | | 20.405(a)(1)(i) | | 50.36(c)(1) | x | 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 | | |
| | | | 20.405(a)(1)(iii) | | 50.73(a)(2)(i) | | 50.73(a)(2)(viii)(A) | | Abstract below and | | |
| | | | 20.405(a)(1)(iv) | | 50.73(a)(2)(ii) | | 50.73(a)(2)(viii)(B) | | in Text, NRC Form | | |
| | | | 20.405(a)(1)(v) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) | | 366A) | | |

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 | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | NO | | | | | | | |
| | | | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|---|
| | YES (If 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 was Defueled and Unit 2 was in Mode 1 (Power Operation) at 100 percent power at the time of discovery.

**Event Description:** Technical Specification Table 3.7-1, Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable Steam Line Safety Valves During Four Loop Operation, specifies a non-conservative maximum allowable flux setpoint when applied to one or more inoperable Main Steam Safety Valves on the same loop. McGuire has never operated in a condition requiring plant operation to be restricted due to inoperable main safety valves.

**Event Cause:** This event was caused by an incorrect vendor-supplied algorithm used to initially calculate the Power Range Neutron Flux High Setpoint for inoperable steam line safety valves.

**Corrective Action:** Administrative directions are in place providing plant operation guidelines for operators when one or more steam line safety valves are inoperable. Additionally, a revision to Technical Specification Table 3.7-1 will be submitted to the NRC to change the appropriate setpoint values.

EVALUATION:

Background

The purpose of Technical Specification Table 3.7.1, which specifies maximum reactor power levels with one, two, or three inoperable Main Steam Safety Valves, is to ensure that the secondary coolant system pressure will be limited to within 110% of its design pressure of 1185 psig during the most severe anticipated system operational transient.

Description of Event

**Unit 1 was Defueled and Unit 2 was in Mode 1 (Power Operation) at 100 percent power at the time of discovery.**

- On January 20, 1994, Westinghouse issued a Nuclear Advisory Letter informing utilities that the algorithm used to initially calculate the Power Range Neutron Flux High Setpoint for inoperable Main Steam Safety Valves [EIIS:RV] was not correct.
- McGuire issued an interpretation to Technical Specification Table 3.7.1 to address the concern outlined in the Westinghouse Nuclear Advisory Letter.
- McGuire decided to pursue a proposed Technical Specification change that would allow higher reactor power operation with inoperable Main Steam Valves.
- During the review of the proposed Technical Specification change, Engineering questioned the bases for the existing power level restrictions.
- A notification to the Nuclear Regulatory Commission was conservatively made on March 20, 1997 to report a possible error in the reactor trip setpoint values contained in Technical Specification Table 3.7-1.
- Subsequent plant specific analysis confirmed existing power level restrictions for inoperable Main Steam Safety Valves were non-conservative.

Conclusion

**This event did not result in any uncontrolled releases of radioactive material, personnel injuries, or radiation overexposures. This event is not Nuclear Plant Reliability Data System (NPRDS) reportable.**

This event is assigned a cause of Technical Inaccuracies in written communication resulting from the use of a incorrect vendor supplied algorithm. The algorithm used to initially calculate the Power Range Neutron Flux High Setpoint for inoperable Main Steam Safety Valves resulted in the incorporation of non-conservative data in Technical Specification Table 3.7.1.

A search of the Operating Experience and Problem Investigation Program Database for the previous 24 months did not render any evidence of a similar event. Therefore, this is not considered a recurring event.

CORRECTIVE ACTION:

Immediate:

NONE

Subsequent:

1. Engineering guidelines for plant operation with one or more inoperable main steam safeties inoperable are in place to prevent the plant from operating at a non-conservative power level with inoperable main steam safeties.

Planned:

1. A revision to Technical Specification 3.7.1 to incorporate the appropriate setpoint values for inoperable main steam safety valves will be submitted to the NRC.

SAFETY ANALYSIS:

**The health and safety of the public and plant personnel were not affected as a result of this event. Therefore, this event is not considered significant.**

- The function of the reduced Power Range Neutron Flux High Setpoint values listed in Technical Specification Table 3.7-1 is to enforce a

reduction in core operating power level when MSSVs are inoperable. This reduction is intended to prevent the Steam Generators from reaching 110 percent of its design pressure.

- In a Loss of Load/Turbine Trip transient, the Atmospheric Steam Dump valves and/or Condenser Steam Dump Valves actuate to relieve energy from the steam generators prior to the opening of the Main Steam Safety Valves. The Steam Dump Valves will continue to relieve steam when the Main Steam Safety Valves have opened. Since this is not a safety grade function, no credit is taken for activation of the steam dump valves in the safety analysis. In reality this is the first line of defense in protecting the secondary system against overpressurization.
- Even near the beginning of core life with a positive or zero Moderator Temperature Coefficient (MTC), the primary coolant heatup resulting from such a transient would tend to drive the MTC negative. This would reduce the core power and heat input to the coolant and in turn would result in a lower required Main Steam Safety Valve capacity to prevent secondary overpressurization. The safety analysis does not credit the reduction of MTC during the transient.
- McGuire has not operated with steam line safety valves that would not lift to ensure the main steam pressure did not exceed 110 percent of design pressure.