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August 4, 1989

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

Subject: McGuire Nuclear Station, Unit 1
Docket No. 50-369
Licensee Event Report 369/89-13

Gentlemen:

Pursuant to 10CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 369/89-13 concerning Unit 1 operation above 100% thermal power because of inappropriate actions and procedural deficiencies. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(i) and (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,

Tony L. McConnell

T.L. McConnell

ROS/U1LER/mgc

Attachment

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MC-815-04
(20)

NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station, Unit 1	0 5 0 0 0 3 6 9	1 OF 0 8

TITLE (4) Unit 1 Operated At Greater Than 100% Thermal Power Because Of Inappropriate Actions And Procedural Deficiencies

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 NAMES	DOCKET NUMBER(S)
0 7	0 5	8 9	8 9	— 0 1 3 —	0 0	0 8	0 4	8 9	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

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)	[ ] 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 Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 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)	
Alan Sipe, Chairman, McGuire Safety Review Group	7 0 4	8 7 5 - 4 1 8 3

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On June 30, 1989, flow indication for Unit 1 Steam Generator (S/G) 'C' failed high. During the subsequent repair of this problem on July 2, 1989, the Instrumentation and Electrical (IAE) technician left the computer points associated with steam flow for S/G 'C' locked at 58% power values. On July 5, 1989, Operations (OPS) personnel began power escalation and at 0715, with indicated power at 86%, received the alarm for 3% deviation between the Nuclear Instrumentation (NI) power indications and the Operator Aid Computer (OAC) Thermal Power Best Estimate indication. This requires recalibration of the NI indications. While performing this calibration, the IAE technician contacted a Reactor Unit Engineer (RE) for a determination of the proper calibration data. The RE instructed the IAE technician to use the OAC Thermal Power Best Estimate figure. This figure was wrong because of the locked computer inputs. The NI indications were adjusted in a non-conservative direction by approximately 6%. OPS personnel resumed power escalation and continued to 95% (actually 101%) indicated power at which point they noticed that the secondary output appeared high and the condensate systems appeared to be unstable. Power escalation was stopped and investigations as to the cause were initiated. While these investigations were ongoing OPS personnel contacted Performance personnel for verification of the Venturi Fouling Correction. The Performance person questioned the unit power versus reactor power anomaly and discovered the locked computer inputs. An immediate power reduction was initiated while the points were returned to service. This event is assigned a cause of Inappropriate Action with a contributory cause of Procedure Deficiency. Training, Programmatic, and Procedural changes will be implemented to prevent recurrence of this event.

TEXT (If more space is required, use additional NRC Form 366A's) (17)

EVALUATION:

Background

Technical Specification Interpretation 1.25, Definition of Rated Thermal Power, was prepared and approved to expand on the definition of rated thermal power. It was based on a memorandum by E.L. Jordan, Assistant Director for Technical Programs, NRC, OIE, dated August 22, 1980, titled Discussion of "Licensed Power Level". Included in the interpretation is the following definition:

> The average power level as indicated by computer heat balance calculations over any eight hour shift should not exceed the "full steady state power level" of 3411 MWT. It is permissible to briefly exceed the "full steady state licensed power level" by as much as 2% for as long as 15 minutes. In no case should 102% full power be exceeded except for a nonrecurring transient situation.

This definition is used wherever "full power", 100% power", or "rated thermal power" are used in the Technical Specifications.

The Thermal Outputs Program on the Operator Aid Computer (OAC) [EIIS:CPU] calculates a heat balance once each minute, and this value is normally displayed on a Control Room [EIIS:NA] video display. The Control Room Operator may select the display format of his choice, i.e. in megawatts or percent, and where on the display the value is shown. It is used by Operations (OPS) personnel to monitor and control the unit power level.

Description of Event

On June 30, 1989, following a Manual Trip of the Unit 1 Main Turbine [EIIS:TRB], because of control problems, flow indication for Steam Generator (S/G) [EIIS:SG] "C" failed high. OPS personnel wrote work request 138042 to correct the problem but it remained in the OPS Shift Supervisor's office unapproved during the recovery and subsequent power escalation to 50%.

On July 2, 1989, the subject work request was reviewed for approval and identified as being more important than originally perceived. This resulted in upgrading the status of the work request to "Emergency". Because of the holiday weekend, Instrumentation and Electrical (IAE) personnel qualified to work on the subject equipment were not on-site. Consequently, at 1430, a qualified IAE technician was called in to perform the necessary repair. This work would normally be performed by two technicians but only one was available on a call-out basis.

In accomplishing the necessary repairs, procedure IP/0/A/3001/01C, Main Steam Flow Calibration Loop C Channel 1 SMFT5040, was used to calibrate the S/G "C" main steam flow indication. This was the first use of the procedure since a recent procedure upgrade. The technician began work at 1700. During the execution of step 10.1.4 of the subject procedure three OAC analog inputs were locked at 58% power values. A1072 "Steam Generator C Main Steam Flow I", A0867 "Steam Generator C Feedwater

Flow I", and A1119 "Steam Generator C Steam Press I". These points are critical inputs to the OAC Thermal Output Power Calculations and require the use of a password to lock them out. Station Directive 3.1.36 establishes control of the password and procedure for locking out computer points.

The loop calibration was successfully completed with the completion of step 10.6.2 of the procedure. This is also the last step of the procedure which required a signoff. Since all signoffs were accomplished on a separate enclosure of the subject procedure there was nothing to lead the technician back to the remainder of the procedure. Therefore, because of the structure of the procedure, the technician assumed that he was finished and never went back to accomplish steps 10.6.3 through 10.6.8 of the procedure. Step 10.6.5 would have restored the three OAC analog points to service. As a result, the three points were left locked at approximately the 58% power level values.

Unit 1 remained at the 58% power level throughout July 3 and July 4, 1989, because of load dispatch requirements. OPS personnel resumed normal power escalation at 0500 on July 5, 1989.

On July 5, 1989, at 0715, the Nuclear Instrumentation (NI) [EIIS:IG] power indication deviation exceeded the 3% from the OAC Thermal Power Best Estimate indication criteria. Unit 1 was at 86% indicated power level at that time. OPS personnel stopped power escalation as required immediately upon receipt of the 3% deviation alarm. NI indications continued to drift upward however, to the 5.1% to 6.1% deviation range. It should be noted that Control Rod [EIIS:ROD] Bank D adversely impacts the NI power indication because of hardware configuration. This effect is non-conservative during rod withdrawal, i.e. power escalations. Since it was felt that the NI calibrations would require only approximately 15 minutes and both Senior Reactor Operators on duty were aware of the range of deviation, no NI channels were logged inoperable.

OPS personnel requested that IAE personnel perform an NI calibration. There was some confusion at this point because IAE personnel were already in the process of performing a calibration on the Unit 2 source range indications and consequently, the Unit 1 calibration was delayed for approximately two hours. The Unit 1 calibration was initiated using procedure IP/0/A/3007/17, NIS Power Range CAL To Best Estimate TH PR. Section 10.1.5 of this procedure requires an interface with a Reactor Unit Engineer (RE) for the respective unit if the primary and secondary power OAC indications differ by greater than +/- 2%. Unit 1 primary power indicated 85% power level and secondary power indicated 81.1% power level at this time, a difference of 3.9%. The IAE technician made the required interface with the Unit 1 RE using the telephone [EIIS:TEL] and the two different power indications were related to the RE.

The RE contacted at that time was distracted because of his involvement with execution of a complicated Thermal Mixing Test being performed during Unit 2 power descension. Also, the normal inclination was to use the Best Estimate figure in these situations since it has historically been the most accurate. Because of these factors the RE failed to recognize the magnitude and direction of the deviation and erroneously instructed the IAE technician to use the Thermal Power Best Estimate figure for NI calibration.

The IAE technician then proceeded with the calibration using the Thermal Power Best Estimate figure. At the time of the calibration, the Thermal Power Best Estimate figure was indicating approximately 3.9% below the actual power level because of the locked computer points. At approximately 0840, the NI calibration was completed. The NI indications had been adjusted by approximately 6%. The IAE technician reported this to the Control Room Senior Reactor Operator. Since legitimate NI adjustments of this magnitude had been experienced before, no action was deemed necessary.

At 0940, OPS personnel resumed power escalation. At approximately 1100 when at 95% indicated power, OPS personnel noticed that the unit Mw output was higher than expected for the OAC power indications. OPS personnel knew however, that Auxiliary Steam [EIIS:SA] loads were lower and Absolute Back Pressure was lower and therefore did not believe the Mw output was sufficiently out of normal to indicate an overpower condition. Also, anomalies in the Condensate System (CM) [EIIS:KA] i.e. low Condensate Booster Pump [EIIS:P] suction pressure and low Feedwater Pump suction pressure, were noticed but they suspected the problem to be caused by "C" heater [EIIS:HTR] drain dumping to the condenser [EIIS:COND]. Because of these indications, power escalation was stopped and investigations by OPS personnel were initiated. OPS personnel believed that this was a conservative stopping point. OPS personnel located several problems in the CM system that lead them to believe that the problem was in the condensate system. The CM system was however, fully capable of supporting higher loads. Performance (PRF) personnel were not contacted to assist in these investigations or evaluations, but they are not normally called on to solve what are perceived to be condensate system problems.

At 1311, OPS personnel resumed power escalation at a rate of 0.6 Mw/min but stopped again at an indicated power of 96% (actual peak power reached a maximum of 102.4%) because of the same instabilities in the CM system. While the investigations of the problems proceeded, OPS personnel decided to look forward in the Operations procedure for power escalation. This procedure requires validation of the Venturi Fouling correction factor prior to exceeding 98% power level. OPS personnel contacted PRF personnel to perform this required validation.

The PRF person who was contacted had also noted the anomaly between indicated power level and unit output and at this point examined the inputs to the OAC Thermal Power Calculation. At 1730, the PRF person located the locked points and informed OPS personnel. An immediate power reduction was then initiated and the input points returned to service. OPS personnel made the required notification to the NRC.

Conclusion

This event is assigned a cause of Inappropriate Action, because IAE Technician A failed to properly complete procedure IP/0/A/3001/01C, Main Steam Flow Calibration Loop C Channel 1 SMFT5040.

IAE Technician A completed the job after working ten hours and was only assisted by another IAE technician for Independent Verification requirements. He did not stop for a meal or break. He was familiar with the loop calibration and had repaired similar problems before.

The required independent verification steps were completed by roving IAE personnel. After completing step 10.6.2 of the procedure, IAE Technician A never went back to the procedure.

There was a contributing cause associated with this event of Defective Procedure because procedure IP/0/A/3001/01C contained no final signoff to lead IAE Technician A back to the final steps of the procedure. All signoffs were performed on an enclosure which was separate from the body of the procedure. This procedure had also recently been revised and this was the first time it was used since the revision.

A second Inappropriate Action occurred because of a lack of attention to detail by RE A when determining the proper data for use by IAE Technician B when calibrating the NI indications. The intent of procedure IP/0/A/3007/17, NIS Power Range Cal To Best Estimate Th Pr, is to resolve any anomaly prior to adjustment of the NIs. This is accomplished by a required interface with the PRF section RE. Procedural controls and documented analysis have not been used to assess these anomalies in the past, but rather personnel expertise has been relied upon. RE A was greatly distracted at this time because of his involvement with a critical Unit 2 test. His total attention at that time was on this test and therefore, he failed to recognize either the magnitude or direction of the anomaly. Historically the most correct calibration data used in NI power range calibrations has been the Thermal Power Best Estimate figure. Therefore, he instructed the IAE technician to use that figure without proper consideration.

A review of McGuire Licensee Event Reports (LERs) for the previous twelve months revealed eleven events involving Technical Specification (TS) violations because of Inappropriate Actions but none of those event particulars were similar to this event. Those were LERs 370/88-04, 370/88-05, 370/88-06, 370/88-09, 370/88-08, 369/88-26, 370/88-10, 369/88-34, 369/89-05, 369/89-09, and 369/88-11. The corrective actions were specific to those eleven events and would not have prevented this event from occurring, therefore, although the problem of TS violations because of Inappropriate Action has been recurring this event is not considered recurring.

There were also five events, LERs 369/88-16, 369/88-17, 369/88-40, 369/89-03, and 370/89-04, of TS violations that involved a Defective Procedure. However, these events involved plant equipment and were not similar to this event. Therefore, the corrective actions could not have prevented this event from occurring and this event is not recurring.

There were no personnel injuries, radiation overexposures, or releases of radioactivity as a result of this event.

This event is not Nuclear Plant Reliability Data System (NPRDS) reportable.

CORRECTIVE ACTIONS:

Immediate: 1) OPS personnel immediately reduced power when the OAC points were found to be locked.

TEXT (If more space is required, use additional NRC Form 366A's) (17)

2) OAC points A1072, A0867, and A1119 were unlocked and NI recalibration performed.

Subsequent:

1) PRF personnel identified OAC points critical to nuclear safety and provided this information to the Computer Section, OPS, and IAE.

2) PRF personnel reviewed Station Directive 3.1.36 and proposed upgrades to ensure adequate password control and adequate review of lockout summaries.

3) PRF personnel requested upgrade to Thermal Power Best Estimate (TOP) calculations to assure conservative calculation of Best Estimate at lower power levels.

4) IAE supervision has reviewed this event and decided to evaluate work practices to assure the best possible work control in emergency situations.

Planned:

1) IAE supervision will reemphasize to IAE crews through structured training the importance of OAC points and of restoring OAC points that have been locked.

2) IAE personnel will revise IP/0/A/3007/17 to ensure validation of TOP prior to NI Calibration while maintaining timely NI calibrations.

3) Computer Information Services personnel will initiate an OAC program request to alarm OAC critical points not in service or locked.

4) IAE personnel will cover this event with all IAE personnel through structured training or communication.

5) IAE personnel will discuss the importance of timeliness in NI calibrations with IAE crews.

6) IAE personnel will change all Periodic Tests and Instrument Procedures to require independent verification or restoration of critical TOP points and evaluate the need to do so regarding other critical OAC points.

7) IAE personnel will modify their procedure standards for new upgraded instrument procedures to assure completion of Section 10 steps and restoration to service has been completed.

8) OPS personnel will provide training to OPS personnel during requalification regarding this event.

9) OPS and PRF personnel will add TOP verification as a requirement to power escalation in 90-95% power level range.

10) Computer Information Services personnel will review control and update of OAC manuals in the Computer Room and upgrade them as necessary.

11) PRF personnel will evaluate and upgrade documentation for the OAC flow calculator.

SAFETY ANALYSIS:

The adjustment of the power range (P/R) NI to correct a mismatch error with the Best Estimate thermal power was determined to have resulted in an indicated power level (by the NI) which was approximately 6% lower than actual. The safety concerns associated with operation in this condition involve actual operation at a power level in excess of full steady state licensed power level and the potential effect on automatic safety functions which are dependent on power level setpoints as monitored by the NI.

The highest value of average primary power level attained during this event was determined to have been approximatley 101.4% full power. Since the initial conditions for accident analysis as described in the Final Safety Analysis Report (FSAR) include the assumption of an initial power level of 102%, this event did not cause any plant parameters to reach levels beyond the scope of the FSAR accident analysis. The use of the value of 102% power is the result of a ± 2% allowance for core power calorimetric error. Also, administrative controls, established through an interpretation of the Definition of Rated Thermal Power (Tech. Spec. 1.25), allow operation up to 102% power for a time interval of 15 minutes. The limits of this guideline were not exceeded during this event.

The safety functions which are dependent on direct input from the NI would have been affected in such a way that setpoints were essentially raised by 5%. Actually, the setpoints were unchanged, but rather would be attained at a value which is 6% above the setpoint. The incorrectly set condition of the NI was in effect from approximately 86% (power indication before the adjustment was made) to 101.4% (but represented as 96%). For operation in this power range, only the P/R Hi Neutron Flux Reactor Trip function which occurs at 109% full power is of concern. With the subject discrepancy present, the 'effective' setpoint would have been approximately 115% power. (The control rod withdrawal interlock [EIIS:IEL] which prevents automatic and manual rod withdrawal above 103% power, would also be affected but is overridden by the assumptions of a postulated Rod Cluster Control Assembly Ejection accident.)

The following is a list of FSAR Chapter 15 transients in which credit is assumed for the P/R Hi Neutron Flux Reactor Trip:

1) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

2) Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power

3) Startup of an Inactive Reactor Coolant Loop

4) Excessive Heat Removal Due to Feedwater System Malfunctions

5) Excessive Load Increase Incident

6) Accidental Depressurization of the Main Steam System [EIIS:SB]

7) Major Secondary System Pipe Rupture

8) Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

From the above list of transients, 1) and 3) are not applicable to this discussion because of the operational status of the unit at the time of the event. Transients 2), 4), 5), 6), and 7) are assumed to result in an Overpower Delta Temperature (OPDT) condition as well as P/R Hi Flux. Transients 6) and 7) will result in safety injection which in turn will trip the reactor. Transient or accident 8), a Rod Cluster Control Assembly Ejection event, is considered the most limiting accident from an upward power excursion point-of-view. In addition to the P/R Hi Neutron Flux trip, the accident analysis for the ejection event also takes credit for reactor protection by the P/R Hi Positive Neutron Flux Rate trip. Protection afforded by the flux positive/negative rate trip would not have been affected by the 5% incorrect setting of the NI.

Table 15.0.7-1 of the FSAR presents the determination of the limiting value of the P/R Hi Neutron Flux trip setpoint to be assumed in the accident analysis. This maximum overpower trip setpoint, assuming all individual errors are simultaneously in the most adverse direction, is determined to be 118% of full power.

Finally, it has been pointed out that actual average power operation at 101.4% did not have any adverse effects on plant parameters or equipment. This level of power was only attained because indications of power were misrepresented, even though a level of 102% is permitted for a limited amount of time. An examination of reactor protective functions provided by setpoints originating directly from the NI has been discussed. From the postulated accidents which take credit in part for a reactor trip by the P/R Hi Neutron Flux parameter, it is concluded that protective features unaffected by the subject discrepancy would have provided the necessary level of reactor protection. In the event that the mitigation of a previously analyzed accident became dependent on the P/R Hi Neutron Flux reactor trip feature, the 'effective' setpoint of 115% is within the scope of the accident analysis.

This event did not affect the health and safety of the public.