

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8810250289 DOC. DATE: 88/10/17 NOTARIZED: NO DOCKET #
 FACIL: 50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410
 AUTH. NAME AUTHOR AFFILIATION
 CONWAY, J. Niagara Mohawk Power Corp.
 WILLIS, J. L. Niagara Mohawk Power Corp.
 RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 88-045-00: on 880915, design rated reactor core flow exceeded due to personnel error.

W/B ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 8
 TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR ENCL
	PD1-1 LA	1 1	PD1-1 PD	1 1
	HAUGHEY, M	1 1	BENEDICT, R	1 1
INTERNAL:	ACRS MICHELSON	1 1	ACRS MOELLER	2 2
	ACRS WYLIE	1 1	AEOD/DOA	1 1
	AEOD/DSP/ROAB	2 2	AEOD/DSP/TPAB	1 1
	ARM/DCTS/DAB	1 1	DEDRO	1 1
	NRR/DEST/ADS 7E	1 0	NRR/DEST/CEB 8H	1 1
	NRR/DEST/ESB 8D	1 1	NRR/DEST/ICSB 7	1 1
	NRR/DEST/MEB 9H	1 1	NRR/DEST/MTB 9H	1 1
	NRR/DEST/PSB 8D	1 1	NRR/DEST/RSB 8E	1 1
	NRR/DEST/SGB 8D	1 1	NRR/DLPQ/HFB 10	1 1
	NRR/DLPQ/QAB 10	1 1	NRR/DOEA/EAB 11	1 1
	NRR/DREP/RAB 10	1 1	NRR/DREP/RPB 10	2 2
	NRR/DRIS/SIB 9A	1 1	NUDQCS-ABSTRACT	1 1
	REG FILE 02	1 1	RES TELFORD, J	1 1
	RES/DSIR/EIB	1 1	RES/DSR DEPY	1 1
	RES/DSR/PRAB	1 1	RGN1 FILE 01	1 1
EXTERNAL:	EG&G WILLIAMS, S	4 4	FORD BLDG HOY, A	1 1
	H ST LOBBY WARD	1 1	LPDR	1 1
	NRC PDR	1 1	NSIC HARRIS, J	1 1
	NSIC MAYS, G	1 1		

Cert P967298894

[Handwritten Signature]

TOTAL NUMBER OF COPIES REQUIRED: LTTR 47 ENCL 46

A/O 4



10
10

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Nine Mile Point Unit 2 DOCKET NUMBER (2) 0 5 0 0 0 0 470 1 OF 07

TITLE (4) Design Rated Reactor Core Flow Exceeded due to Personnel Error Results in Plant Operation in an Unanalyzed Condition

EVENT DATE (5) 09 15 88 LER NUMBER (6) 045 REPORT DATE (7) 10 17 88 OTHER FACILITIES INVOLVED (8) N/A

OPERATING MODE (9) 2 POWER LEVEL (10) 002 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) 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 365A) 20.405(a)(1)(iii) X 50.73(a)(2)(ii) 50.73(a)(2)(viii)(A) 20.405(a)(1)(iv) X 50.73(a)(2)(iii) 50.73(a)(2)(viii)(B) 20.405(a)(1)(v) 50.73(a)(2)(ix)

LICENSEE CONTACT FOR THIS LER (12) NAME John Conway, Supervisor Reactor Physics TELEPHONE NUMBER 315 349-2698 AREA CODE 315

Table with 11 columns: CAUSE, SYSTEM, COMPONENT, MANUFACTURER, REPORTABLE TO NPRDS, CAUSE, SYSTEM, COMPONENT, MANUFACTURER, REPORTABLE TO NPRDS. All cells are empty.

SUPPLEMENTAL REPORT EXPECTED (14) YES (If yes, complete EXPECTED SUBMISSION DATE) NO [x] EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

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

On September 15, 1988 at 1225 hours, with the reactor at approximately 2% of rated thermal power and the mode switch in "STARTUP", it was discovered that the Nine Mile Point Unit 2 (NMP2) reactor had operated with greater than 100% of rated core flow. While reviewing documentation associated with jet pump calibration it was discovered that an error had been made when entering flow coefficients into the JR Pump program for the calibrated jet pumps. This error caused the improper calculation of total core flow during startup test N2-SUT-35, "Recirculation System Flow Calibration". As a result of this, the core flow instrumentation was improperly calibrated such that when indicated flow was 100% of rated; actual flow was 104.5% of rated. This placed NMP2 in a condition not specifically addressed in the NMP2 Final Safety Analysis Report (FSAR).

The immediate cause of the event was inaccurate core flow indication (i.e. indicated flow was lower than actual). The root cause of this event has been determined to be personnel error due to inattention to detail. Incorrect data was entered into the jet pump calibration computer program. A contributing factor was lack of procedural controls on entering values into the JR Pump program.

Immediate corrective actions included the following: (1) A preliminary evaluation was made on the possible effects on plant safety; (2) Plant operation in Operational Condition 1 was restrained until recalibration of the affected flow instrumentation was completed and appropriate changes were made to the affected surveillance procedures.

8810250289 SS1017 PDR AUOCC 05000410 PDC

1022/11

FACILITY NAME (1) Nine Mile Point Unit 2	DOCKET NUMBER (2) 0 5 0 0 0 410	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		88	045	00	02	OF	07

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

I. DESCRIPTION OF EVENT

On September 15, 1988 at 1225 hours it was discovered that the Nine Mile Point Unit 2 (NMP2) reactor had operated with greater than 100% of rated core flow. At the time of discovery, the reactor was at approximately 2% of rated thermal power with the mode switch in the "STARTUP" position.

While reviewing documentation associated with jet pump calibration it was discovered that an error had been made when entering flow coefficients into the JR Pump program for the calibrated jet pumps. This error caused the improper calculation of total core flow during startup test N2-SUT-35, "Recirculation System Flow Calibration". As a result of this, the core flow instrumentation was improperly calibrated such that when indicated flow was 100% of rated; actual flow was 104.5% of rated.

In addition the value of recirculation drive flow required to reach rated core flow was determined incorrectly due to the erroneous core flow indication. As a result the actual Reactor Coolant System Recirculation Flow upscale rod block setpoints were outside the allowable value stated in the technical specifications.

There were no inoperable and/or out of service components or systems which contributed to this event.

II. CAUSE OF EVENT

The immediate cause of the event was inaccurate core flow indication (i.e. indicated flow was lower than actual). The root cause of this event has been determined to be personnel error due to inattention to detail. Incorrect data was entered into the jet pump calibration computer program by contractor personnel. A contributing factor was lack of procedural controls on entering values into the JR Pump program.

III. ANALYSIS OF EVENT

NMP2 Final Safety Analysis Report (FSAR), Chapter 15, "Accident Analyses", Figure 15.0.2, Power/Flow map represents the acceptable operational constraints for abnormal operational transient evaluations (Attachment 1). The power/flow map places a limit on reactor core flow of 100% of rated for accident analyses.

The error introduced into the reactor core flow measurement due to the improperly calibrated flow instrumentation has been determined to be approximately 4.5%. When indicated reactor core flow exceeded 95.5% of rated, actual core flow exceeded 100% of rated. Therefore, each time core flow exceeded 95.5% of rated, NMP2 was placed in a condition not specifically addressed in the NMP2 Final Safety Analysis Report.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
Nine Mile Point Unit 2.	0 5 0 0 0 410	88	045	00	03	OF 07

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

III. ANALYSIS OF EVENT (Cont'd)

The impact of this condition on plant safety was evaluated with special attention to the following subject areas. The results of that evaluation are as follows:

1. Transient/Accident Analysis

The currently approved transient analyses are based on initial core flow of 100% of rated, however, an Increased Core Flow (ICF) analysis has been performed which demonstrates that the plant can be safely operated at up to 105% of rated flow without requiring any changes to current operating limits.

2. Fuel Thermal Limits Evaluation

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits are not flow dependent. While the power shape calculated by the process computer may be slightly affected, calculations of peak fuel bundle nodal powers used to compare against these limits are not significantly affected by the flow measurement error. The calculated critical bundle power will be lower than actual due to the flow measurement error and will result in calculation of conservative Minimum Critical Power Ratio (MCPR) values.

3. Technical Specification Instrument Setpoints

- a. The setpoints for the Average Power Range Monitors (APRM) Flow Biased Thermal Trip as well as the RBM and APRM Rod Block functions are affected by the calibration error. This is due to the fact that rated loop drive flow was evaluated to be approximately 44,100 gpm. The correct value would have been approximately 41,800 gpm. This results in implementation of trip setpoints lower than Technical Specification requirements (i.e., which is allowed by Technical Specifications) by a factor proportional to total drive flow in the flow biased regions. For example if the reactor operates at 75% of rated core flow (i.e., 81.375 mlbs/hr) the correct APRM scram setpoint would be:

$$\begin{aligned} \text{Trip Setpoint} &= .66 w + 51 \quad \text{at 75\% core flow } w=30.5 \text{ KGPM} \\ &= .66 (30.5/41.8) \times 100 + 51 \\ &= 99.16\% \text{ power} \end{aligned}$$

However at 75% indicated core flow the APRM scram setpoint that would have been experienced was:

$$\begin{aligned} \text{Trip Setpoint} &= .66 w + 51 \\ &= .66 (30.5/44.1) \times 100 + 51 \\ &= 96.65\% \text{ power} \end{aligned}$$

The maximum setpoint implemented when clamping the flow biased setpoint is not affected by the calibration error.



LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Nine Mile Point Unit 2	DOCKET NUMBER (2) 0 5 0 0 0 410	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		88	045	00	04	OF	07

TEXT (If more space is required, use additional NRC Form 368A's) (17)

b. The APRM flow unit upscale Rod Block setpoint (108% nominal-111% allowable) is affected by the calibration in a nonconservative manner. This function would not have occurred until 113.94% flow was reached. This situation does not compromise safety for the following reasons:

- The ICF analysis provides an allowable value for this function of 114%.
- The purpose of this function is to provide a block if the flow unit output fails upscale disabling the flow biased rod blocks. The setpoint selection for this function is somewhat arbitrary. The protective action would still have been provided if an upscale signal failure had occurred.
- No credit is taken for this feature in any FSAR Chapter 15 analysis.

4. Reactor Vessel Internals Vibration

Reactor internals vibration data was recorded and analyzed as part of the startup test program while the reactor was operating at near 104.5% of rated flow. Startup test program acceptance criteria for allowable vibration levels were satisfied under these conditions. In addition, the ICF analysis demonstrated acceptable loading of all applicable components up to 106% of rated core flow.

5. Startup Test Program Results

All test results for startup tests conducted after the calibration error were reviewed. There is some minor impact on the documentation for several of the startup tests conducted after the initial core flow instrumentation calibration. Only a few tests contain calculations or evaluations against acceptance criteria that are affected by the calibration error. No evaluations against Level I acceptance criteria are invalidated by the error except those regarding operation at $\geq 100\%$ of rated core flow. In no instance are the conclusions derived from the test program significantly affected.

6. Summary Conclusion

Previous operation of NMP2 with total core flow in excess of 100% of rated did not adversely affect plant safety. In addition plant response to transients and accidents analyzed in the FSAR would not have been compromised. The event reported in LER 88-22 was reviewed in light of this current information. It has been determined that the event described in LER 88-22 did not compromise plant safety or cause excessive duty on any plant components.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Nine Mile Point Unit 2	DOCKET NUMBER (2) 0 5 0 0 0 410	LER NUMBER (6)			PAGE (3)	
		YEAR 88	SEQUENTIAL NUMBER 045	REVISION NUMBER 00	05	OF 07

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

IV. CORRECTIVE ACTIONS

Corrective actions were as follows:

1. Rerun the Jet Pump Calibration Program using correct input data. Prevent plant operation in operational condition 1 until after completion of the following activities (i.e., completed 9/16/88).
 - *recalibrate jet pump loop flow summer cards
 - *recalibrate total drive flow summer cards
 - *revise affected curves from Surveillance Procedure N2-OSP-LOG-D001, "Daily Checks"

2. Complete the following activities on an as soon as practical basis (i.e., completed 09/19/88):
 - *revise the process computer jet pump flow constants
 - *revise the process computer core flow vs. drive flow arrays
 - *adjust the total drive flow limiter

3. Document potential effects on plant safety with special consideration for the following areas (completed 9/23/88 except for reconciliation of startup test program results documentation: A supplement to the Final Startup Report will be generated and issued prior to plant restart following the outage scheduled to begin 10/1/88):
 - *FSAR Chapter 15 analyses
 - *fuel thermal limits evaluations
 - *instrument setpoints required by the Technical Specifications
 - *loading and vibration of reactor vessel internals
 - *startup test program results

4. Complete the following actions prior to plant restart following the outage scheduled to begin October 1, 1988:
 - *Revise two loop operating map
 - *Revise all Average Power Range Monitor Flow and Jet Pump Operability Curves for N2-OSP-LOG-D001, "Daily Checks"
 - *Issue a permanent plant test procedure for core flow calibration with adequate controls to prevent future errors of a similar nature
 - *Review and upgrade as necessary procedural controls associated with offline evaluation of fuel thermal limits via the Backup Core Limits Evaluation (BUCLE) Program



2

3

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Nine Mile Point Unit 2	DOCKET NUMBER (2) 0 5 0 0 0 410	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	06	OF 07
		88	045	00		

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

V. ADDITIONAL INFORMATION

Identification of Components Referred to in this LER

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
Jet Pumps	P	AD
Average Power Range Monitors	MON	IG
Flow Indicator	FI	AD
Reactor Recirculation System	MA	AD

Failed Components - None

Previous Similar Events - No previous similar events

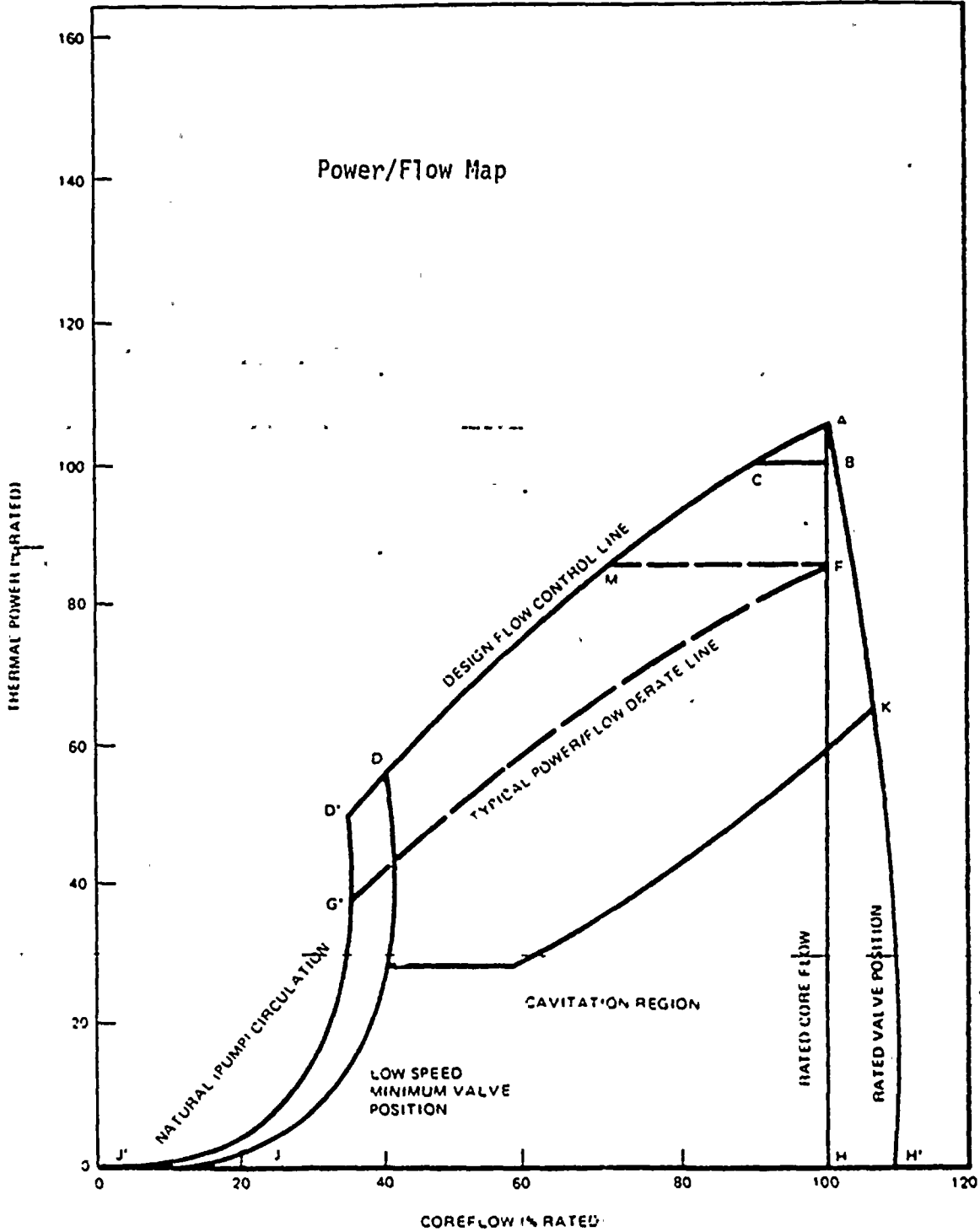


12

13

FACILITY NAME (1) Nine Mile Point Unit 2	DOCKET NUMBER (2) 0 5 0 0 0 410	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		88	045	00	07	OF	07

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





October 17, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 88-45

Gentlemen:


In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

- LER 88-45 Is being submitted in accordance with 10 CFR 50.73 (a) (2) (ii), "Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being:
- (A) In an unanalyzed condition that significantly compromised plant safety;
 - (B) In a condition that was outside the design basis of the plant; or
 - (C) In a condition not covered by the plant's operating and emergency procedures."

A 10CFR50.72 report was made at 1315 hours on September 15, 1988.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,


J. L. Willis
General Superintendent
Nuclear Generation

JLW/JMT/mjd

Attachments

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W. A. Cook

Cert No.
P967928894

IK22
1/1

