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ACCESSION NBR:8806060108 DOC.DATE: 88/05/27 NOTARIZED: NO DOCKET #
FACIL:50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
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SUBJECT: LER 88-006-00:on 880430,ECCS flow imbalance caused by normal sys fluctuations.
W/8 ltr.

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TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

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	NRR/DREP/RAB 10	1	1	NRR/DREP/RPB 10	2	2
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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)
D. C., Cook Nuclear Plant - Unit 2	0 5 0 0 0 3 1 6	1 OF 0 5

TITLE (4)
ECCS Flow Imbalance Caused by Normal System Fluctuations

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 4	3 0	8 8	8 8	— 0 0 6 —	0 0	0 5	2 7	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5
POWER LEVEL (10): 0 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)	[ ] 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	
T. K. Postlewait - Technical Engineering Superintendent	6 1 6	4 6 5 - 5 9 0 1

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)		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)

On April 30, 1988, at 1840 hours, with Unit 2 in Mode 5 (Cold Shutdown), as-found data obtained while performing a routine surveillance on the Boric Acid Injection System reflected an out-of-tolerance flow rate and distribution as compared to Technical Specification (T/S) 4.5.2h. No cause other than normal system fluctuation, combined with standard instrumentation/measurement error, could be identified as the reason the T/S limits could not be met. The system will be balanced prior to Unit startup after the completion of the current outage. Relaxation in flow balance tolerance is currently being analyzed by Westinghouse as part of the program to allow operation at reduced temperature and pressure.

An Engineering Analysis indicates that the Boric Acid Injection System would have functioned as designed during an accident. Adequate cooling was available to cool the core. Pump cavitation would only be a concern during large break LOCA conditions and it is believed that the high flow rate would not be sufficient to cause a cavitation problem. Furthermore, the large break LOCA result is not highly dependent on charging pump flow capacity due to the rapid depressurization to Accumulator and Residual Heat Removal Pump actuation pressures. It was concluded that no significant risk to public health and safety existed.

IE22 '1,

8806060108 880527
PDR ADOCK 05000316
S DCD

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

Conditions Prior to Occurrence

Unit 2 in Mode 5 (Cold Shutdown)

Description of Event

On April 30, 1988, at 1840 hours, while conducting an 18 month surveillance on the flow rates for the Boron Injection (BI) System (EIIS/BQ), it was determined that the system flow rate and distribution was not in compliance with the requirements of Technical Specification (T/S) 4.5.2h. The BI system was found with a total combined flowrate for all loops in excess of the maximum 470 gpm allowable T/S value, and the highest and lowest single loop flowrates varied by more than 10 gpm as allowed by T/S 4.5.2h (see ATTACHMENT 1 for as-found data).

There were no inoperable components, systems or structures that contributed to this event.

Cause of Event

No cause other than normal system fluctuations, combined with standard instrument/measurement error, could be identified. The BI system flow is balanced by adjusting the throttle valve positions within each system until an acceptable flow distribution is obtained. The throttle valves are then locked into position. An inspection of the locking devices indicated that the valve positions have not changed since the last 18 month Surveillance Test, at which time the flow distribution was left within T/S limits (see Attachment 1 for this previous as-left data). The Technical Specification limits are very narrow and the system fluctuations, due to such factors as internal hydraulics and instrument tolerances, are felt to be responsible for the small deviations from one surveillance period to another.

Analysis of Event

This event is considered reportable under the provisions of 10CFR50.73.a.2.i.b, Operations or Condition Prohibited by T/S.

Although the flows were out of Technical Specification limits, it is believed that no significant risk to public health and safety existed. This is based on a review of various FSAR accidents impacted by the boron injection flows and the effect of high flow on the potential for centrifugal charging pump cavitation.

Small Break Loss of Coolant Accident (LOCA) - The combined flow of the three lowest loops exceeded the Technical Specification minimum requirement. Thus, sufficient water would be available to provide the flows assumed in the FSAR small break LOCA analysis.

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Analysis of Event (con't)

Large Break LOCA, Maximum Safeguards Analysis - The limiting large break LOCA case for Unit 2 of the Cook Nuclear Plant is the maximum safeguards case; i.e., no Emergency Core Cooling System (ECCS) pump failures. This is because maximum safeguards assumptions result in a prediction of lower containment pressure, which in turn results in a slower core level recovery rate. Although the flows were higher than assumed in the analysis, no significant impact on public health and safety is believed to have existed. Sensitivity studies, performed by Westinghouse Electric Corp. and Advanced Nuclear Fuels, have indicated that peak clad temperature is a strong function of power level. Since Unit 2 had been operating at an administrative power limit of essentially 80%, due to steam generator degradation, peak clad temperatures, had a large break LOCA occurred, would have been expected to remain below 10 CFR 50.46 limits.

Large Break LOCA, Pump Cavitation - Pump cavitation, caused by high flow, is of concern only when the back pressure against the centrifugal charging pumps is low (a situation which is only expected to occur under large break LOCA conditions). It is believed that the high flows measured for the boron injection lines would not be sufficient to cause a pump cavitation problem. Westinghouse Electric Corp. has indicated that the large break LOCA result is not highly dependent on charging pump flow capability (due to the rapid depressurization to accumulator actuation pressure and the continued rapid depressurization to Residual Heat Removal Pump actuation pressure). Thus, even if the centrifugal charging pumps were lost, due to cavitation, the large break LOCA results would not be significantly impacted.

Steam Line Break - All boron injection line flows were in excess of the Technical Specification limits, rather than below. Thus, a sufficient quantity of highly borated water would have been delivered to the reactor coolant system to mitigate the power increase associated with a steam line break.

Based on the review detailed above, it is concluded that no significant risk to the public health and safety existed.

Corrective Actions

The BI system flow imbalance will be reconciled by adjusting the throttle valve positions, to achieve an acceptable flow distribution, prior to Unit startup (after the completion of the current refueling and Steam Generator Repair Outage), tentatively scheduled for January 1989. Relaxation in flow balance tolerance is currently being analyzed by Westinghouse as part of the program to allow operation at reduced temperature and pressure.

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

Failed Component Identification

None

Previous Similar Events

50-315/82-075-0
50-315/83-090-0
50-316/86-012-0
50-315/87-016-0

★U.S.GPO:1986-0-624-538/455

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)

ATTACHMENT 1

Unit 2 BI Flow Balance Data

Data Description	Parameter	Flows (gpm) East Train	Flows (gpm) West Train	Technical Specification 4.5.2h Criteria (gpm)
As-found 04/30/88	Loop 1	136.4	136.4	Nominal 117.5
	Loop 2	119.0	116.4	Nominal 117.5
	Loop 3	117.6	117.1	Nominal 117.5
	Loop 4	123.8	118.9	Nominal 117.5
	Δ flow (highest - to - lowest loop)	18.8	20.1	≤ 10.0
	Total combined flow (all loops)	496.8	488.8	≤ 470
	Minimum combined flow (lowest 3 branches)	360.4	352.4	≥ 345.8
Previous As-left 09/03/87	Loop 1	117.1	117.1	Nominal 117.5
	Loop 2	117.4	118.6	Nominal 117.5
	Loop 3	116.1	116.5	Nominal 117.5
	Loop 4	117.5	117.6	Nominal 117.5
	Δ flow (highest - to - lowest loop)	1.4	2.1	≤ 10.0
	Total combined flow (all loops)	468.1	469.8	≤ 470
	Minimum combined flow (lowest 3 branches)	350.6	351.2	≥ 345.8

Indiana Michigan
Power Company
Cook Nuclear Plant
P.O. Box 458
Bridgman, MI 49106
616 465 5901

INDIANA MICHIGAN POWER

May 27, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Operating License DPR-58
Docket No. 50-316

Document Control Manager:

In accordance with the criteria established by 10 CFR 50.73 entitled Licensee Event Reporting System, the following report is being submitted:

88-006-00

Sincerely,

W. G. Smith, Jr.

W. G. Smith, Jr.
Plant Manager

WGS:clw

Attachment

cc: D. H. Williams, Jr.
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M. P. Alexich
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