NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Davis-Besse Unit 1

DOCKET NUMBER (2): 0 5 0 0 0 3 4 6

PAGE (3): 1 OF 0 3

TITLE (4): Exceeding Reactor Power Limit Established Due to Low Measured RCS Flow

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 4	2 0	8 5	8 5	0 0 8	0 1	0 7	1 9	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 8

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: Erdal Caba

TELEPHONE NUMBER: AREA CODE 4 1 9 | 2 4 9 - 5 0 0 0

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	S J	F T	B 0 4 5	Y					

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)

While slowly increasing power to 96 percent on April 20, 1985, the operators suspected a problem with the computer heat balance calculation which read 95.2 percent. Generated megawatts indicated 903 MWe which corresponds to a reactor power of approximately 98 percent. Power was confirmed by a manual heat balance calculation to be 98.3 percent.

The unit was limited to 96 percent power due to Reactor Coolant System measured flow being approximately 2 percent lower than the minimum flow requirement in Table 3.2-1 of Technical Specifications. Therefore, for approximately 12 hours, the unit inadvertently did not comply with the action statement of Technical Specification 3.2.5. Reactor power was decreased to 95.4 percent as determined by a manual heat balance calculation. On April 30, 1985, the Reactor Coolant System Flow Test, ST 5042.03, was performed and showed the actual flow to be 406,533 gpm which is above the Technical Specification minimum flow requirement. Therefore, only indicated flow had been low.

1 Subsequent evaluation showed that during this same period of time that only one of the NIs was lower than the actual heat balance power by approximately 0.2 percent more than the allowable 2 percent. Because this was not known at that time, the inoperable channel was not placed in the tripped condition within one hour as required by Technical Specification 3.3.1.1.

The cause of the erroneous reading was found to be a failed feedwater flow transmitter which is part of the computer secondary heat balance calculation. Modification was made to plant procedures to provide a more detailed look at secondary heat balance calculations and its inputs.

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

Description of Occurrence: Following a maintenance outage which ended on April 13, 1985, reactor power was increased to 85 percent power. Reactor power was then slowly being increased to 96 percent power on April 20, 1985. The unit was limited in power due to the Reactor Coolant System, RCS, (AB), measured flow reading approximately 2 percent lower than the minimum required flow in Table 3.2-1 of Technical Specifications of 396,880 gpm.

At 0220 hours, the operators suspected a problem with the computer heat balance calculation which read 95.2 percent power. Generated megawatts read 903 MWe, which corresponds to approximately 98 percent power. A manual heat balance calculation confirmed reactor power to be 98.3 percent. The action statement of Technical Specification 3.2.5, which limits thermal power at least 2 percent below rated thermal power for each 1 percent the RCS flow is outside its limit was not being met.

Reactor power was decreased to 95.4 percent as determined by a manual secondary heat balance calculation. Subsequent evaluation showed that only one of the NIs was lower than the actual heat balance power by approximatley .2% more than the allowable 2 percent. Because this was not known at the time, the inoperable channel was not placed in the tripped condition within one hour per Technical Specification 3.3.1.1. Once reactor power was decreased, the inoperable NI channels were once again operable.

This event is being reported per 10CFR50.73(a)(2)(i)(B) as operation in excess of the action statement limit.

Designation of Apparent Cause of Occurrence: The cause of the occurrence was found to be a failure of feedwater flow (SJ) transmitter FTSP2B2. The root cause was that no procedures existed that periodically checked the heat balance inputs. Each feedwater flow loop has two transmitters which are averaged and fed into the computer secondary heat balance calculation. These transmitters fail to a value of 4996.8 KGPM, which corresponds to a heat balance calculation of 87 percent power. The more reactor power was increased over 87 percent power, the greater the error was in the heat balance calculation. It was later found that the transmitter had failed on March 24, 1985, when the unit was in a maintenance outage. The thermal power calculation is not affected until greater than 50 percent power when a secondary heat balance calculation is used.

Analysis of Occurrence: During the 1984 Refueling Outage, Burnable Poison Rod Assemblies, BPRA, (AC), were installed into the core as the result of a change to an 18 month cycle. Since this changed the flow distribution through the core, the minimum required RCS flow required for DNB considerations was reanalyzed. It was found that the minimum could be lowered to 389,644 gpm, which is less than what was being measured at the time of this problem. A Technical Specification change request had already been submitted to the NRC for approval prior to this problem.

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

ST 5042.03, RCS Flow Test, performed on April 30, 1985, showed the actual RCS flow to be 406,533 gpm which is greater than the present minimum required flow of 396,880 gpm specified in Table 3.2-1 of Technical Specifications. The flow determined from ST 5042.03 is an accurate flow determined by a manual heat balance calculation. Therefore, at the time of this event, even though measured flow was lower than the Technical Specification limit of 396,880 gpm, the actual RCS flow was much greater. The difference could be due to the new Rosemount flow transmitters installed during the 1984 Refueling Outage.

The remaining three NI channels for RPS hi flux trip were operable.

Corrective Action: Feedwater flow transmitter FTSP2B2 will be repaired or replaced with a new transmitter under Maintenance Work Order 1-85-135000. The computer heat balance calculation has been modified to use only the available feedwater flow transmitter in Loop 1.

ST 5030.01, RPS Heat Balance, performed daily to check nuclear instrumentation indicated power against the heat balance power was modified to provide an additional check of the heat balance calculations by comparing reactor power to a feedwater flow and generated megawatts curve. PT 5131.02, Verification of Computer Calculations, will perform the heat balance calculations weekly to individually verify the computer computations. It was also modified to include a comparison of the inputs with other available instrumentation to assure that computer points are valid. A Generic Guidance Memorandum was written to emphasize the critical nature of the instrumentation and specify equipment associated with the performance of the secondary heat balance calculations.

Failure Data: This was the first occurrence where the station operated at a power level higher than an action statement limit.

Report No: NP-33-85-12 DVR No(s): 85-060

TOLEDO EDISON

July 19, 1985

Log No. K85-1114
File: RR 2 (NP-33-85-12)

Docket No. 50-346
License No. NPF-3

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

Gentlemen:

Enclosed is Revision 1 to Licensee Event Report 85-008. The revisions to the report are indicated by a "1" in the left margin of each page. Please replace your previous copies of this report with the attached revision.

Yours truly,

S. M. Quennoz / DWB

Stephen M. Quennoz
Plant Manager
Davis-Besse Nuclear Power Station

SMQ/ljk

Enclosure

cc: Mr. James G. Keppler,
Regional Administrator,
USNRC Region III

Mr. Walt Rogers
DB-1 NRC Resident Inspector

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