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March 29, 1990
ND3MNO:2045

Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-64
LER 90-006-00

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

Gentlemen:

In accordance with Appendix A, Beaver Valley Technical Specifications, the following revised Licensee Event Report is submitted:

LER 90-006-00, 10 CFR 50.73.a.2.ii.B, "Plant Operation in Excess of Licensing Basis".

Very truly yours,

T P Noonan

T. P. Noonan
General Manager
Nuclear Operations

sl

Attachment

IE22
1/1

cc: Mr. William T. Russell
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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)
Beaver Valley Power Station, Unit 1	0 5 0 0 0 3 3 4	1 OF 0 4

TITLE (4)

Plant Operation in Excess of Licensing Basis

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 2	2 7	9 0	9 0	— 0 0 6 —	0 0	0 3	2 9	9 0	N/A	0 5 0 0 0
									N/A	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)	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)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	X 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	
Thomas P. Noonan, General Manager Nuclear Operations	4 1 2	6 4 3 - 1 2 5 8

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	I D	C P U	W 1 2 0	N					

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 2/27/90 at 0100 hours, Operations personnel performed a daily heat balance calculation (HBC) which showed a calculated reactor power of 2663.3 megawatts thermal (MWt), or 100.4%. Turbine load was immediately reduced until reactor power was less than 2652 MWt. An investigation was initiated, since no reactor power changes were performed within the last 24 hours. A computer program (Task 3D), that performs averaging calculations used in the HBC, was found to have aborted on 2/18/90. This program retained the last calculated value until it was updated on 2/26/90. All HBCs performed from 2/18/90 through 2/27/90 were invalid utilizing the computer values. The HBC parameters affected were steam generator feedwater flow, pressure and steam flow for all three loops. The cause for the aborted computer program was internal error in data acquisition. Operations personnel have been instructed to independently verify computer values against control room instrumentation. There were no safety implications. A review of thermal output, utilizing data from an independent computer system, showed plant operation at 100.4%. This is below the evaluation contained in the Updated Final Safety Analysis Report, Section 3.4.3.4.1, which discusses steady state operation at 101% reactor power and concludes that core safety limits are not exceeded.

DESCRIPTION

On 2/27/90 at 0100 hours, Operations personnel performed a daily heat balance calculation which showed a calculated reactor power of 2663.3 megawatts thermal (MWt), or 100.4%. Turbine load was immediately reduced until reactor power was less than 2652 MWt. An investigation was initiated to determine the cause of this discrepancy, since no reactor power changes were performed within the last 24 hours. This investigation determined that a computer program (Task 3D "Average and Integrate"), that performs averaging calculations used in the heat balance calculation, was found to have aborted on 2/18/90. This program retained the last calculated value until it was updated at 1302 hours on 2/26/90. All heat balance calculations performed from 2/18/90 through 2/27/90 were invalid utilizing the computer calculated average values. The heat balance calculation parameters affected were steam generator feedwater flow, pressure and steam flow for all three loops. Additional averaging parameters not used in the heat balance calculation were also affected. None of these computer parameters are used for Technical Specification surveillance verifications. The axial flux difference (AFD) alarm during this time period was also inoperable as this computer program also provides average values of axial flux difference.

CAUSE

The cause for this event, as determined by the computer engineer was a failure of the computer to synchronize file addresses, in Task 3D, between two files, prompting the computer to abort the task. This type of computer problem is not readily detectable by the operators because there are no error messages or computer prompts displayed as a result of an aborted program at the operators computer console. Reactor power was raised above 100% due to two adjustments of the excore nuclear instrumentation. These adjustments, performed on 2/19/90 and 2/24/90, reduced the excore nuclear instrumentation indication downward on each occasion (0.3% on 2/19/90 and 0.5% on 2/24/90). A review of plant thermal output for each of these occasions, utilizing an independent computer system, showed actual power to be less than 100%. Subsequently, when reactor power was increased on each occasion to the 100% indicated value, this resulted in a cumulative effect of increasing actual power. This effect was not identified until the performance of a heat balance calculation performed on 2/27/90, after the computer was rebooted on 2/26/90. A review of the independent computer system, showed actual power to be greater than 100% (100.4%) from 0145 hours on 2/26/90 to 0130 hours on 2/27/90.

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

CORRECTIVE ACTIONS

The following corrective actions have been taken as a result of this event:

1. All heat balance calculations performed during the time period from 2/18/90 through 2/27/90, were reverified using data from the Plant Variable Computer (PVC), which receives information independently from the plant computer. These verifications showed thermal output greater than 100% for the time period of 2/26/90 0145 hours to 2/27/90 0130 hours.

2. A review of Axial Flux Difference values, recorded by the operator using control board indicators, for the time period from 2/18/90 through 2/27/90 was performed. This review showed that axial flux difference was within the required band the entire time period.

3. Operations personnel have been provided with administrative guidance requiring a verification that plant computer heat balance parameters are responding to statistical variations by observing these values against control room indications.

4. A review of reactor coolant system gross activity samples, for the time period, was performed and no abnormalities were found.

5. Operations personnel have been instructed to verify heat balance calculations utilizing independent parameters whenever the excore nuclear instrumentation indication is adjusted in a downward (indication lowered) manner.

PREVIOUS OCCURRENCES

There have been no previously reported events of this type. A review of the computer engineers log showed aborted computer programs on three occasions since February 1987. Task 3D was found to have aborted on 5/1/89, however, there were no operational concerns as the plant had been operating at 90% since 4/17/89, as part of a fuel extension evolution in support of the upcoming Seventh Refueling Outage.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

REPORTABILITY

This event is being reported in accordance with 10 CFR 50.73.a.2.ii.B, as an event that is outside the Licensing basis.

SAFETY IMPLICATIONS

There were no safety implications as a result of this event. A review of reactor coolant system gross activity samples, for the time period, was performed and no abnormalities were found indicative of potential fuel damage. The Updated Final Safety Analysis Report, Section 3.4.3.4.1, discusses steady state operation at various power levels to assess the core thermal design to determine the maximum heat removal capability in all flow subchannels. This included testing at 101% reactor power. This testing showed that the core safety limits, as presented in the Technical Specifications, are not exceeded. Additionally, operation at 102%, above the design rating of 2652 MWt, has been assumed and evaluated for various operational transients in Sections 14.1.7, 14.1.8, 14.2.5, 14.2.6 and 14.3 of the Updated Final Safety Analysis Report. These evaluations concluded that there were no safety implications resulting for these events.