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Beaver Valley Power Station, Unit No. 1 and No. 2
Docket No. 50-334 License No. DPR-66
Docket No. 50-412 License No. NPF-73
LER 2003-006-00

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

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

LER 2003-006-00, 10 CFR 50.73(a)(2)(ii)(B) and 50.73(a)(2)(v), "New Steam Generator Level Uncertainties Identified Which Exceed Available Setpoint Margins."

L. W. Pearce
L. William Pearce

Attachment

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
INPO Records Center (via electronic image)
Mr. L. E. Ryan (BRP/DEP)

IE22

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Beaver Valley Power Station Unit No. 1	05000334	1 OF 5

4. TITLE
New Steam Generator Level Uncertainties Identified Which Exceed Available Setpoint Margins

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Beaver Valley Unit 2	05000412
09	22	2003	2003	006	00	11	12	2003	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)	X	50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)	X	50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
L. R. Freeland, Manager Regulatory Affairs / Performance Improvement	(724) 682-5284

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On September 22, 2003, Westinghouse issued Nuclear Safety Advisory Letter 03-9, which addressed an error in the Westinghouse steam generator water level setpoint analysis due to previously incorrectly addressing a differential pressure which occurs across a mid-deck plate within the steam generator during steam flow. This differential pressure adversely affected the steam generator low-low level setpoint uncertainty for Beaver Valley Power Station (BVPS) Unit 1 and Unit 2. During operation prior to when the steam generator water level low-low setpoints at both BVPS Units were recently raised, there was inadequate margin available to offset this newly identified effect which needed to be addressed as a non-conservative bias in the uncertainty calculations. Therefore, the design basis feedwater line break analysis of record may not be valid since it relied upon steam generator level to initiate a reactor trip.

This event is reportable pursuant to 10 CFR 50.73(a)(2)(ii)(B) as an unanalyzed condition that significantly degrades plant safety since the impact of crediting another reactor trip function which may occur in place of steam generator low-low level during the specific postulated feedline break transient has not been calculated and is unknown. Similarly, this also represents a condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shut down the reactor and maintain it in a safe shutdown condition as described in each Unit's UFSAR and is reportable pursuant to 10 CFR 50.73(a)(2)(v)(A). The cause of this event was inadequate Westinghouse design analysis of steam generator mid-deck plate differential pressure. The safety significance of this event was low.

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PLANT AND SYSTEM IDENTIFICATION

Westinghouse-Pressurized Water Reactor {PWR}
Steam Generator Water Level Control System {JB}
Plant Protection System {JC}

CONDITIONS PRIOR TO OCCURRENCE

Unit 1: Mode 1 at 100 % power
Unit 2: Mode 2 at 100 % power

There were no systems, structures, or components that were inoperable at the start of the event that contributed to the event other than as described below.

BACKGROUND

Westinghouse issued Nuclear Safety Advisory Letter (NSAL) 02-3, Rev. 1, on April 8, 2002 which described a steam generator "mid-deck" differential pressure which is developed as a function of steam flow rate. This mid-deck differential pressure had not been previously considered in the existing instrument uncertainty calculations used in the reactor trip system setpoint methodology. The effect of the mid-deck differential pressure was addressed by the Beaver Valley Power Station (BVPS) Unit 1 and Unit 2 for several design basis accidents (DBAs). Analytical margin existed to address this new uncertainty for many DBAs. However, for the Feedwater Line Break (FLB) event, NSAL 02-3 concluded that "It has been determined that for a steam generator affected by a Feedwater Line Break that reverse flow through the feedring out of the steam generator nozzle and eventually out the break results in a reversal in sign of the mid-deck differential pressure effect and thus can be ignored for that event." Thus, Westinghouse concluded that the consideration for mid-deck differential pressure did not affect the setpoints used for the FLB safety analysis. Thus, it was believed that there was no impact for the FLB analyses described in the BVPS Unit 1 and Unit 2 Updated Final Safety Analysis Reports (UFSAR).

DESCRIPTION OF EVENT

On September 22, 2003, Westinghouse issued NSAL 03-9, entitled "Steam Generator Water Level Uncertainties. Information was also supplied through the Westinghouse Owners Group via WCAP-16115-P about the new steam generator level uncertainties identified in NSAL 03-9 that could adversely affect the current steam generator level setpoints in the Reactor Protection System. NSAL 03-9 and WCAP-16115-P determined that the conclusion in the prior NSAL 02-3, Rev. 1 regarding the mid-deck differential pressure was incorrect for feedline break analyses. NSAL 03-9/WCAP 16115-P now recognized that "there may be some size of feedline break where there is no reverse flow out of the steam generator with the ruptured line attached, but also no feed flow to that steam generator" (in other words, the effects would be similar to the effects of a Loss of Normal Feedwater event for the affected steam generator). As previously described in NSAL 02-3, BVPS would have to

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DESCRIPTION OF EVENT (Continued)

add a new bias to their steam generator low-low level reactor trip setpoints to address the differential pressure across the mid-deck for the Loss of Normal Feedwater transient. This mid-deck differential pressure adversely affects the steam generator low-low level setpoint uncertainty calculations. Therefore, the BVPS Unit 1 and Unit 2 reactor trip system setpoint calculations will need to be revised to address this new mid-deck differential pressure effect during a feedline break transient.

Westinghouse estimated that the mid-deck differential pressure would conservatively require an additional five percent level in the steam generator low-low level setpoint calculation for feedline break inside containment to offset this new bias for both BVPS Units. There was insufficient margin remaining in the setpoint calculations for both BVPS Units to offset this new additional bias, because the safety analysis limit is already set at 0 percent for steam generator low-low level during feedline break. Thus, this newly identified required bias would invalidate the current feedline break analysis results in both BVPS Units' feedline break safety analyses calculations of record since steam generator low-low level was credited as the parameter which tripped the reactor and therefore ensured that the feedline break safety analyses acceptance criteria were met.

During the period of discovery prior to the issuance of NSAL 03-9, BVPS Unit 1 had increased its steam generator low-low level reactor trip system setpoints by five percent as a proactive measure to offset potential subsequent adverse consequences identified by the ongoing Westinghouse investigation. (BVPS Unit 2 was in a refueling shutdown at this time; its steam generator low-low level reactor trip system setpoint was also increased by five percent during the refueling shutdown prior to returning to power operation). The increased steam generator low-low level setpoints continued to meet the applicable Technical Specification requirements for this setpoint since the change was in a conservative direction. With the increased steam generator low-low level setpoints, there was sufficient margin to account for the newly required bias and continue to allow the feedline break analyses to remain valid.

REPORTABILITY

Pursuant to 10 CFR 50.73(a)(1), a licensee shall report an applicable event or condition if it occurred within three years of the date of discovery. BVPS Unit 1 and Unit 2 had operated at full power within the last three years without the newly identified required bias for mid-deck differential pressure addressed in each Unit's reactor protection system setpoint methodology and without increasing their steam generator low-low level setpoints. This condition is inconsistent with each Unit's feedline break analysis of record as described in each Units' UFSAR since a valid steam generator low-low level reactor trip may not have occurred during a specific postulated feedline break transient inside containment. Therefore, this event is reportable pursuant to 10 CFR 50.73(a)(2)(ii)(B) as an unanalyzed condition that significantly degrades plant safety. As discussed in the following Safety Implications Section, other reactor trip functions may occur in place of the credited steam generator low-low level trip function during this specific feedline break transient. However, the impact of crediting another reactor trip function in place of steam generator low-low level during the specific

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REPORTABILITY (Continued)

postulated feedline break transient has not been calculated, is unknown, and, therefore, was a significant degradation of plant safety.

Given this prior potential lack of a valid feedline break analysis, this also represents a condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to shut down the reactor and maintain it in a safe shutdown condition as described in each Unit's UFSAR and is reportable pursuant to 10 CFR 50.73(a)(2)(v)(A).

CAUSE OF EVENT

The cause is inadequate Westinghouse and industry review of the issues previously identified in NSAL 02-03 issued in 2002 which identified a concern with mid-deck differential pressure during steam flow in Westinghouse steam generators. The more recent evaluation which led to the issuance of NSAL 03-9 identified errors in the prior evaluation that were not bounded by previous assumptions.

SAFETY IMPLICATIONS

As stated in Westinghouse NSAL 03-9, a specific feedline break event can be affected. This event is represented as a Reactor Coolant System heatup transient. The primary criterion of interest for this ANS Condition IV event is that any fuel damage that may occur during the transient should be of a sufficiently limited extent such that the core will remain geometrically intact with no loss of core cooling capability. Based upon a more realistic Westinghouse assessment, the potential non-conservative effects identified in NSAL 03-9 would be compensated for by the actuation of other reactor trip functions and by reducing existing conservatisms in the analysis. A reactor trip would continue to be provided during a feedline break transient by either a delayed steam generator low-low level trip, or by one of several redundant trip functions such as over-temperature differential temperature, over-power differential temperature, high pressurizer pressure, or a safety injection signal. Similarly, auxiliary feedwater would continue to be initiated on either a delayed steam generator low-low level trip or on a safety injection signal. Since a reactor trip would still occur, the primary-to-secondary side heat load would be reduced via more realistic modeling, and the auxiliary feedwater system performance is unaffected by this issue, it is judged that the acceptance criterion for the event would be expected to continue to be met.

The possible increase in the Steam Generator Low-Low Level reactor trip setpoint process measurement uncertainties is considered to be of low safety significance, since a reactor trip signal would still be generated to mitigate the analyzed accidents, either by the steam generator low-low level trip or other diverse reactor trip signals. It is not expected that there would be a significant increase in CDF or LERF as a result of this increased uncertainty. Therefore, the safety significance of this event was low.

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CORRECTIVE ACTIONS

1. Administrative controls have been implemented to ensure that the steam generator low-low level setpoints will remain sufficiently increased at both BVPS Units above their minimum setpoint required by their Technical Specifications to address the newly identified steam generator level uncertainties pending Actions 2 and 3 below.

2. The NSAL information will be evaluated to determine the permanent changes needed to address the newly identified steam generator level uncertainties at both BVPS Units.

3. If the evaluation of the NSAL information determines that a license amendment request (LAR) is needed at either BVPS Unit, then a LAR will be initiated for the affected Unit(s) to revise the Technical Specification value for steam generator low-low level to be consistent with the reactor trip setpoint value assumed in the setpoint methodology.

Completion of the above and other corrective actions are being tracked through the corrective action program.

PREVIOUS SIMILAR EVENTS

A review of past Beaver Valley Power Station reportable events for the last five years found no similar BVPS Licensee Event Report involving nonconservative steam generator level setpoints or reactor trip setpoint uncertainties.

ATTACHMENT

Beaver Valley Power Station, Unit No. 1
License Event Report 2003-006-00

Commitment List

The following list identifies those actions committed to by FirstEnergy Nuclear Operating Company (FENOC) for Beaver Valley Power Station (BVPS) Unit No. 1 and 2 in this document. Any other actions discussed in the submittal represent intended or planned actions by Beaver Valley. These other actions are described only as information and are not regulatory commitments. Please notify Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement, at Beaver Valley on (724) 682-5284 of any questions regarding this document or associated regulatory commitments.

Commitment	Due Date
The NSAL information will be evaluated to determine the permanent changes needed to address the newly identified steam generator level uncertainties at both BVPS Units.	As tracked through the Corrective Action Program.
If the evaluation of the NSAL information determines that a license amendment request (LAR) is needed at either BVPS Unit, then a LAR will be initiated for the affected Unit(s) to revise the Technical Specification value for steam generator low-low level to be consistent with the reactor trip setpoint value assumed in the setpoint methodology.	As tracked through the Corrective Action Program.