FENOC
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
Route 168
P.O. Box 4
Shippingport, PA 15077-0004

L. William Pearce
Site Vice President

724-682-5234
Fax: 724-643-8069

September 15, 2003
L-03-144

Beaver Valley Power Station, Unit No. 1
Docket No. 50-334 License No. DPR-66
LER 2003-005-00

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 2003-005-00, 10 CFR 50.73(a)(2)(i)(B), "Non-Conservative Reactor Coolant System Low Flow Reactor Trip Setpoint."

for 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)

JE22

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 4

4. TITLE
Non-Conservative Reactor Coolant System Low Flow Reactor Trip Setpoint

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 None	DOCKET NUMBER
07	18	2003	2003	– 005 –	00	09	15	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)		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)		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)	X	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 July 18, 2003, during a routine post-refueling outage review of the Reactor Coolant System (RCS) precision flow calorimetric data from a test procedure on RCS Total Flow Measurement, it was determined that one of the nine channels of RCS flow (3 channels per loop) was set non-conservatively with respect to Technical Specifications. Specifically, it was identified that channel F-1RC-424 on the B-RCS loop was set to 89.2 percent of full power RCS flow when the Technical Specification allowable value for loop flow is greater than or equal to 89.8 percent. The root cause of the non-conservative RCS flow setpoint is an inadequate process to verify RCS flow setpoints are within Technical Specification limits prior to reaching ten-percent power. Since, this condition existed longer than the allowed outage time of 6 hours permitted by Technical Specifications, it was a condition prohibited by plant Technical Specification 3.3.1.1; therefore this event is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B). Additionally, there were two occasions where maintenance removed an operable instrument from service placing the plant in a condition outside of the action statement. Therefore, Technical Specification 3.0.3 was unknowingly in effect and actions to commence plant shutdown were not initiated within one hour. Thus, these events were also conditions that are prohibited by plant Technical Specification and are reportable pursuant to 10 CFR 50.73(a)(2)(i)(B). Review of this event for effect on core damage frequency (CDF) for BVPS Unit 1 was performed and the cumulative safety significance of these events was determined to be small.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Beaver Valley Power Station Unit 1	05000334	2003 -	005 -	00	2 OF 4

PLANT AND SYSTEM IDENTIFICATION

Westinghouse-Pressurized Water Reactor (PWR)
Reactor Protection System (JC)
Reactor Coolant System (AB)

CONDITIONS PRIOR TO OCCURRENCE

Unit 1: Mode 1 at 100 % power

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

DESCRIPTION OF EVENT

On May 12, 2003, an Reactor Coolant System (RCS) flow measurement test was satisfactorily performed when the plant was at steady state full power conditions, following startup from its fifteenth refueling outage (1R15), to ensure that the Technical Specification requirement for RCS total flow was met. This test procedure also obtained loop voltage measurements for engineering review and subsequent normalization of the instruments to account for cycle specific flow differences which were expected to be minimal. On July 18, 2003, during this routine post-refueling outage review of the Reactor Coolant System (RCS) precision flow calorimetric data from a test procedure on RCS Total Flow Measurement, it was determined that one of the nine channels of RCS flow (3 channels per loop) was set non-conservatively with respect to Technical Specifications. The remaining eight channels continued to meet the Technical Specification requirement. Specifically, it was identified that channel F-1RC-424 on the B-RCS loop was set to 89.2 percent of full power RCS flow. Technical Specification 3.3.1, Reactor Trip System Instrumentation, Limiting Condition for Operation states that "At a minimum, the Reactor Trip System Instrumentation channels and interlocks of Table 3.3-1 shall be operable." The allowable value for indicated loop flow as identified by Functional Units 12, "Loss of Flow – Single Loop", and 13, "Loss of Flow – Two Loops", of Table 3.3.1 is greater than or equal to 89.8 percent. Therefore, the effective field setpoint for RCS Flow Channel F-1RC-424 was less than the allowable value required by Technical Specifications. There are three channels in each RCS loop and two trip signals in any loop will initiate a Reactor Trip. The total number of channels required by Technical Specification 3.3.1 is three per loop in each operating loop. With less than the total number of channels operable, Action 7 of Table 3.3-1 is required to be met. Action 7 states in part, "with the number of operable channels one less than the total number of channels, startup and/or power operation may proceed provided ... the inoperable channel is placed in the tripped condition within six hours and the minimum channels operable requirement is met..." The minimum channels operable is two per loop in each operating loop.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Beaver Valley Power Station Unit 1	05000334	2003	- 005 -	00	3 OF 4

REPORTABILITY

It has been determined that one RCS flow channel [F-1RC-424 on the B-RCS loop] contained a setpoint which would not meet plant Technical Specification requirements and hence, was not in an operable condition. This condition existed for approximately 82 days (April 28 to July 18) since plant startup from the fifteenth refueling outage (1R15). Therefore, this condition existed longer than the allowed outage time of 6 hours required by Action 7 in Table 3.3-1 of Technical Specifications 3.3.1.1 and was a condition prohibited by plant Technical Specification. This event is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B).

In addition, because one of three channels on the B-RCS loop was (unknowingly) inoperable since 1R15, two of the three channels would have been inoperable whenever one of the other two channels on the B-RCS loop had been put into an inoperable status during routine maintenance. This occurred when the RCS flow test procedures were last run for F-1RC-425 and F-1RC-426, on June 9 and June 16, 2003 respectively, since F-1RC-424 was inadvertently inoperable at these times. During these two times, the plant was in a condition beyond Action 7 of Technical Specification 3.3.1.1 and therefore was in Technical Specification 3.0.3. Each time, action to commence plant shutdown was not initiated within one hour; thus, these events were also conditions that are prohibited by plant Technical Specification 3.0.3 and are reportable pursuant to 10 CFR 50.73(a)(2)(i)(B).

CAUSE OF EVENT

The root cause of the non-conservative RCS flow setpoint is an inadequate process existed to verify RCS flow setpoints are within Technical Specification limits within the required time frame. Specifically, Technical Specification 3.3.1.1 requires RCS flow setpoints to be greater than or equal to the allowable value above ten-percent reactor power (when the RCS flow instruments are required to be operable). As determined by investigation, the measured RCS flow at each transmitter is subject to change at each refueling outage. Hence, with no programmatic process in place to verify that RCS flow setpoints met the Technical Specification requirements prior to exceeding ten-percent power, a violation of the requirements occurred.

SAFETY IMPLICATIONS

Review of these events for effect on core damage frequency (CDF) for BVPS Unit 1 was performed. RCS flow instrument F-1RC-424 provides one out of three input signals to the RCS low flow reactor trip in the event of low flow from the B-RCS loop. The three input signals for RCS low flow in the A-RCS loop and the C-RCS loop remained operable. Since the other two instruments in the B-RCS loop would continue to provide 2/3-input signals, an RCS low flow reactor trip would still be actuated in the event of loss of primary flow in the B-RCS loop. Additionally, when F-1RC-425 and F-1RC-426 were tested on June 9 and 16, 2003, respectfully, the instruments were placed in the tripped position.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Beaver Valley Power Station Unit 1	05000334	2003	- 005 -	00	4 OF 4

During the time that the instruments were in a tripped condition, the Reactor Protection System would have performed its safety function if necessary. Both Beaver Valley Power Station Units' Probabilistic Risk Assessments (PRA) do not explicitly model these instruments. Therefore, the incorrect setting on one channel will have no direct impact to the PRA CDF or Large Early Release Frequency (LERF). The probability of having more than one instrument channel not working is considered low and the impact to the PRA would still be insignificant. Based on the information discussed above, the cumulative safety significance of these events was determined to be small.

CORRECTIVE ACTIONS

1. The setpoint on channel F-1RC-424 was adjusted to comply with Technical Specification requirements.
2. The 1R15 scaling calculation was revised to document RCS flow setpoints based on normalized RCS flow at 100 percent power.
3. An Engineering Change Package was developed to translate design basis setpoints from the scaling calculation into applicable plant procedures following 1R15.
4. All RCS flow setpoints were reviewed at Unit 1 and Unit 2 to ensure Technical Specifications requirements were met.
5. A methodology to accurately determine RCS flow setpoints at both units was developed in a format compatible for incorporation into applicable plant procedures.
6. The appropriate procedures at both units will be revised in accordance with the methodology to implement and validate the RCS flow setpoints prior to exceeding 10 percent power following refueling outages (when the RCS flow instrument is required to be operable).

Corrective action completion is being tracked through the Corrective Action Program.

PREVIOUS SIMILAR EVENTS

A review of past Beaver Valley Power Station Licensee Event Reports for the last three years found two similar events regarding Reactor Protection System Instrumentation at BVPS Unit 1 and Unit 2.

Unit 2 LER 2003-001, "Lag Time Constant for Steam Line Pressure Channel Used in the Reactor Protection System Found Out of Tolerance"

Unit 2 LER 2002-003, "Calibration Discrepancies in Delta Temperature Tau Time Constant Values Used in Reactor Protection System"

ATTACHMENT

Beaver Valley Power Station, Unit No. 1
License Event Report 2003-005-00

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 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 appropriate procedures at both units will be revised in accordance with the methodology to implement and validate the RCS flow setpoints prior to exceeding 10 percent power.	As tracked through the Corrective Action Program.