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Beaver Valley Power Station, Unit No. 2
Docket No. 50-412 Licensee No. NPF-73
LER-96-006-00

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 96-006-00, 10 CFR 50.73(a)(2)(ii), "Potential Control and Protection System Interaction in Steam Generator Water Level Control."

T P Noonan
T. P. Noonan

LB/ds

Attachment

IE22

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NRC FORM 366 (4-95)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)

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

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TITLE

Potential Control and Protection System Interaction in Steam Generator Water Level Control

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 NAME	DOCKET NUMBER
									Beaver Valley Power Station Unit 1	05000334
10	24	96	96	006	00	11	22	96	N/A	

OPERATING MODE (9)	6		20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	0%		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
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in abstract
			20.405(a)(1)(iv)	X	50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		below and in Text
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (include Area Code)
T. P. Noonan, Vice President Nuclear Operations and Plant Manager	(412) 393-7622

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

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

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

On October 24, 1996, at approximately 1630 hours, with Beaver Valley Power Station (BVPS) Unit 2 shutdown in refueling MODE 6, it was determined that the plant had the potential for a control and protection system interaction in the Steam Generator Water Level Control System (SGWLCS) which could prevent the Reactor Protective System (RPS) from initiating a reactor trip on a low-low steam generator level. This was identified during an engineering review of Westinghouse Nuclear Safety Advisory Letter (NSAL) 96-004, "Control and Protection System Interaction," which addresses an IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations," concern with designs that share a common tap for steam generator level instruments and main steam flow instruments. BVPS Unit 2 steam generator level instruments 2FWS-LT476/486/496 share a common steam generator tap with steam flow instruments 2MSS-FT474/484/494 respectively. A postulated failure at the common tap, combined with a second postulated failure of a steam generator level instrument, as required by IEEE-279, could prevent the RPS from initiating a reactor trip and Auxiliary Feedwater (AFW) automatic start, if the steam generator water level decreased below the low-low level trip setpoint. Evaluation of NSAL-96-004 has determined that this condition is not applicable to BVPS Unit 1, due to a difference in design.

This event was reported on October 24, 1996, at 1825 hours, pursuant to the requirements of 10CFR50.72(b)(2)(i). This written report is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(ii)(B) as an event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis. The apparent cause of this event was a failure to adequately review and evaluate the steam flow and steam generator level instrument mechanical design configuration during the development and approval of the Technical Specification (TS) Amendment application that proposed the elimination of steam flow/feedwater flow mismatch coincident with low level trip at Unit 2.

Corrective actions for this event include: 1) a review of Unit 1&2 plant drawings was performed to identify any similar common tap concerns - none were identified, 2) an interim design modification to the SGWLCS is being implemented to ensure against the identified control and protection system interaction, with a final design to be completed and implemented during the next refueling outage, 4) appropriate plant procedures will be revised or developed, 5) a self-study guide to address the design change has been provided to Licensed Operators and Shift Technical Advisors, 6) an evaluation of the TS amendment process will be performed to identify enhancements to the coordination of design reviews between the Licensing and Engineering organizations.

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse - Pressurized Water Reactor (PWR)

Reactor Protective System (RPS) {JC}*

Feedwater/Steam Generator Water Level Control System (SGWLCS) {JB}*

Main Steam System {SB}*

Steam Generator 21A Level Transmitter 2FWS-LT476 {JB/LT}*

Steam Generator 21B Level Transmitter 2FWS-LT486 {JB/LT}*

Steam Generator 21C Level Transmitter 2FWS-LT496 {JB/LT}*

Steam Generator 21A Steam Flow Transmitter 2MSS-FT474 {SB/FT}*

Steam Generator 21B Steam Flow Transmitter 2MSS-FT484 {SB/FT}*

Steam Generator 21C Steam Flow Transmitter 2MSS-FT494 {SB/FT}*

* Energy Industry Identification System (EIIS) plant system and component codes are identified in the text as {EIIS:SS/CC}.

DESCRIPTION OF EVENT

On October 24, 1996, at approximately 1630 hours, with Beaver Valley Power Station (BVPS) Unit 2 shutdown in refueling MODE 6, it was determined that the plant had the potential for a control and protection system interaction in the Steam Generator Water Level Control System (SGWLCS) {EIIS:JB} which could prevent the Reactor Protective System (RPS) {EIIS:JC} from initiating a reactor trip on a low-low steam generator (SG) level. This was identified during an engineering review of Westinghouse Nuclear Safety Advisory Letter (NSAL) 96-004, "Control and Protection System Interaction," which addresses an IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations," concern with designs that share a common tap for SG level instruments and main steam system flow instruments. BVPS Unit 2 SG level instruments 2FWS-LT476/486/496 {EIIS:JB/LT} share a common tap with main steam flow instruments 2MSS-FT474/484/494 {EIIS:SB/FT} respectively. A postulated failure at the common tap, combined with a second postulated failure of a SG level instrument, as required by IEEE-279, could prevent the RPS from initiating a reactor trip and Auxiliary Feedwater (AFW) automatic start, if the SG water level decreased below the low-low level trip setpoint. Evaluation of NSAL-96-004 has determined that this condition is not applicable to BVPS Unit 1, due to a difference in design.

NSAL-96-004 is applicable to those plants with three narrow range SG water level channels that have removed the low feedwater flow trip function on the basis that a median signal selector (MSS) on the level control signal input precluded the need for back-up protection for a control/protection system interaction scenario. The Westinghouse design did not address the failure of the common tap, for which the low feedwater trip also provided redundant protection. In addition, the low feedwater flow trip provided the back-up protection necessary to satisfy IEEE-279 requirements. IEEE-279 requirements must be met pursuant to the requirements of 10CFR50.55a(h). In accordance with IEEE-279, Section 4.7.3, Single Random Failure: "Where a single random failure can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action, even when degraded by a second random failure."

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The NSAL is applicable to BVPS Unit 2, since the low feedwater flow trip function (steam flow/feedwater flow mismatch coincident with low steam generator level) was eliminated in favor of an installed MSS by a plant design change. Unit 1 retained the back-up trip and did not install the MSS, hence the issue described in the NSAL is not a concern for Unit 1.

CAUSE OF EVENT

The apparent cause of this event was a failure to adequately review and evaluate the steam flow and steam generator level instrument mechanical design configuration during the development and approval of the Technical Specification (TS) Amendment application that proposed the elimination of steam flow/feedwater flow mismatch coincident with low level trip at Unit 2.

During the development of Technical Specification Amendment No. 27 and the implementation of the subsequent design change, the focus and emphasis was upon addressing the instrumentation and control (I&C) aspects of IEEE-279. Review and evaluation of the mechanical design configuration of the SG steam flow and level instruments was inadequate, and the existence of the common SG tap and potential impact upon control/protection system interaction was not identified during the process.

ANALYSIS OF EVENT

A median signal selector was added to the SGWLCS before BVPS Unit 2 commenced initial operation in 1987 to address the IEEE-279 control/protection system interaction concerns. The primary objective of the MSS addition was to prevent a failed instrument channel from causing a disturbance in the controlled system which will initiate a plant transient. Formerly, the SGWLCS received only a single input channel of SG water level measurement. Failure of that channel could cause adverse control system action. With the MSS, all three level measurement channels became input to the control system and compared by the signal selector. The device selected the median signal for use by the control system, and control system action became based upon this signal. By rejecting the high and the low signals, the control system was prevented from acting on any single failed channel. Since no adverse control system action could now result from a single, failed instrument channel, a second random protection system failure (as would otherwise be required by IEEE-279) did not need to be considered. Signals resulting from a single failed high or low SG level measurement were rejected for control purposes, and therefore did not affect the system, thus eliminating the control and protection system interaction mechanism. This MSS design was approved by the NRC in Supplement 5 of NUREG-1057 as conforming to the requirements of paragraph 4.7 of IEEE-279 and hence, 10CFR50.55a(h). Westinghouse WCAP-11484, "Feedwater Control System Median Signal Selector," also confirmed the acceptability of the MSS.

Technical Specification Amendment No. 27 was issued February 20, 1990. The TS amendment request proposed deletion of the steam flow/feedwater flow mismatch coincident with low steam generator level trip. Trip deletion was based upon: 1) the fact that the BVPS Unit 2 UFSAR accident analysis does not credit this trip in mitigating the consequences of any analyzed accidents and 2) the MSS provides a sufficient safeguard against an adverse control/protection system interaction. The MSS was believed to effectively preclude the control/protection system interaction addressed for a decreasing SG water level and thus be substituted for the reactor trip on steam flow/feedwater flow mismatch coincident with low SG level to meet the control/protection system interaction criteria for IEEE-279.

A postulated (single) failure of the common tap would simultaneously create a maximum differential pressure signal on the affected SG level and main steam flow instruments. The SG level signal from the affected level transmitter would fail high (sensor electrical signal output would go to maximum value) and the main steam flow signal from the affected flow transmitter would fail low (sensor electrical signal output would go to minimum value). The median channel selector sums all three water level channels for the SG and subtracts the highest and the lowest channels. Thus a failed high level channel is automatically subtracted out of the feedwater level control circuit. The MSS would function to filter out the non-median SG level input from the two remaining level instrument channels, before input to the SGWLCS on that SG. In addition, one of the two remaining SG level instruments is assumed to fail as is, above the low-low level setpoint. The low steam flow signal would cause the main feedwater controller to modulate the Main Feedwater Regulating valve to a closed position. Actual SG level would decrease. A reactor trip and AFW start on a low-low SG level signal would not occur, since the 2 out of 3 trip logic would not be satisfied.

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CORRECTIVE ACTIONS

1. A review of Unit 1 and Unit 2 plant drawings for the steam generators and the nuclear steam supply system (NSSS) was performed October 28, 1996, to determine whether unidentified common impulse taps existed which could have a potential for control /protection system interaction. None were identified.

2. As an interim corrective action, a modification will be made to the SGWLCS prior to entering Mode 3 from the current outage. The modification will install a 2-position selector switch into each of the three SGWLCS process racks to directly input the desired steam flow channel (Channel IV) into the SGWLCS. The current Control Board steam flow instrument selector switches will be disabled. The modification will also provide a common Control Room annunciator, to alert the Operators whenever any of the undesired (Channel III - shared tap) steam flow signals are selected. The modification will allow use of the Channel III steam flow transmitter if the steam generator level transmitter (Channel III narrow range level) with which the steam flow transmitter shares a tap is declared INOPERABLE . This will primarily be the case when steam flow Channel IV is removed for maintenance or quarterly calibrations. Under these conditions, the level transmitter failure procedure will be implemented, and ACTION statement 7 of BVPS Unit 2 Technical Specification (TS) 3.3.1.1, Table 3.3-1 will be entered. The TS ACTION statement requires the Channel III SG level bistables to be tripped in six hours if Channel IV steam flow cannot be restored to OPERABLE. If Channel IV steam flow malfunctions during normal operations, procedures and TS allow manual control of the associated feedwater control system to stabilize the plant, selection of Channel III steam flow as the controlling channel, and restoration to automatic feedwater control.

3. Final design corrective action(s) will be identified and implemented by the completion of the next refueling outage (2R07).

4. The appropriate operating procedures will be revised/developed to address the impact of this modification upon the operation of the plant prior to entering Mode 3 from the current outage.

5. I&C procedures affected by this modification will be revised by June 30, 1997. I&C procedures affected by this modification have been placed on hold and will be appropriately revised before they are performed.

6. A study guide which addresses the SGWLCS modification and associated procedure changes was provided to Licensed Operators and Shift Technical Advisors for self-study on November 14, 1996 to be completed prior to entering Mode 3.

7. Licensing currently has measures in place to ensure that TS amendments are fully supported by engineering design reviews. An evaluation will be completed by February 28, 1997, which will review the overall TS amendment process to identify enhancements to the coordination of design reviews between the Licensing and Engineering organizations.

REPORTABILITY

This event was reported on October 24, 1996, at 1825 hours, pursuant to the requirements of 10CFR50.72(b)(2)(i). This written report is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(ii) (B) as an event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis.

SAFETY IMPLICATIONS

Westinghouse stated in NSAL 96-004 that a search of INPO records was conducted which identified no reported instances of tap or impulse line failure. The probability of a tap/impulse line failure coupled with another level channel of the same redundant set failing has been identified as extremely low. Based upon this information, there were no safety implications to the health and safety of the public as a result of this event.

SIMILAR EVENTS

A review of Licensee Event Reports for the past two years identified no similar events.