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September 12, 1997
L-97-026

*Beaver Valley Power Station, Unit No. 1*
*Docket No. 50-334 License No. DPR-66*
*LER 97-016-01*

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 97-016-01, 10 CFR 50.73(a)(2)(i) and 10 CFR 50.73(a)(2)(ii), "Unit 1 Shutdown Required by T.S. 3.0.3 Due to Inoperable Steam Generator Low-Low Level Reactor Protection System Trip."

Ronald L LeGrand

R. L. LeGrand

LB/ds/bls

Attachment

IE22

9709220093 970912
PDR ADOCK 05000334
S PDR

DELIVERING QUALITY ENERGY

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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

**APPROVED BY OMB NO. 3150-0104**
EXPIRES 04/30/98
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

# LICENSEE EVENT REPORT (LER)

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

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TITLE
Unit 1 Shutdown Required by T. S. 3.0.3 Due to Inoperable Steam Generator Low-Low Level Reactor Protection System Trip

| 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 |
| 06 | 27 | 97 | 97 | 016 | 01 | 09 | 12 | 97 | | |

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| OPERATING MODE (9) | 1 | 20.402(b) | | 20.405(c) | | 73.73(a)(2)(iv) | 73.71(b) |
| POWER LEVEL (10) | 100 | 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) | X | 50.73(a)(2)(i) | | 50.73(a)(2)(viii)(A) | (Specify in abstract below and in Text NRC Form 366A) |
| | | 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 (include Area Code) |
|---|---|
| R. D. Hart, Senior Licensing Supervisor | (412) 393-5284 |

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 June 27, 1997 at 2007, 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 Protection System (RPS) from initiating a reactor trip on a low-low steam generator level. This was identified during an engineering review of the seismic qualifications of the main feedwater flow transmitters. This trip was required by IEEE 279-1971 as a diverse trip to the two-out-of-three low-low steam generator reactor trip because of the control and protection interaction of the steam generator level channels. This was reported in accordance with 10 CFR 50.72 (b)(1)(i)(A) and (b)(1)(ii)(B). BVPS Unit 1 was then taken to Mode 3. On June 28, an update to the original 10 CFR 50.72 report was provided to indicate that the steam generator low-low level trip was also inoperable.

On July 3, 1997, with the unit in Mode 3, Residual Heat Removal (RHR) sample isolation valve RH-214 was opened to allow sampling of the RHR system. When this valve was opened, no overpressure protection for penetration 97-1 existed and it was declared inoperable, resulting in a unit shutdown to Mode 5 as required by Technical Specification 3.6.1.1. This was reported in accordance with 50.72 (b)(1)(ii)(A).

The apparent cause of the initiating event was a misapplication of IEEE 279-1971 Section 4.7 Control and Protection System Interaction criteria during the original plant design.

Corrective actions for this event include: 1) a design modification to the SGWLCS was implemented to ensure against the identified control and protection system interaction, by installing a Median Signal Selector Switch in each of the three SGWLCS racks, 2) a design review of the Unit 1 and Unit 2 RPSs with respect to compliance with IEEE 279-1971 is being performed.

As potential exists for a control and protection system interaction for the SG level protection system, IEEE-279-1971 requires that a second failure must be postulated. If a second SG level instrument were to fail in an intermediate position, since there are three channels, the requisite RPS trip logic would not be satisfied and a reactor trip and auxiliary feedwater automatic start would not be generated as required when an actual SG low-low level condition occurred. The described condition does not meet IEEE-279 1971 criteria and is outside the Unit 1 design basis.

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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

Plant Protection System {JC/FT}*

*Energy Industry Identification System (EIIS) codes and component function identifier codes appear in the text as {SS/CCC}.

## IDENTIFICATION OF OCCURRENCE

Discovery Date: June 27, 1997

The feedwater flow transmitters have always been located in the non-seismically qualified turbine building. Although the location of the feedwater flow transmitters was known, a specific exception in regards to compliance with IEEE 279-1971 was not requested and documented.

## CONDITIONS PRIOR TO OCCURRENCE

Unit 1: Mode 1, 100% Reactor Power
Unit 2: Mode 1, 100% Reactor Power

There were no additional structures, systems or components that were inoperable that contributed to the event.

## DESCRIPTION OF EVENT

On June 27, 1997, with Unit 1 in Mode 1 at 100 % power, it was determined that the Reactor Protection System (RPS) main feedwater flow transmitters FT-FW-476, -477, -486, -487, -496 and -497 {JC/FT} would be unable to perform their design safety function in the event of a design basis seismic event due to their placement in the non-seismic turbine building. The transmitters provide inputs into the steam flow / feed flow mismatch coincident with low steam generator level trip. This trip was provided as part of the original plant design as a diverse trip to the two-out-of-three low-low steam generator reactor trip. The logic for the steam generator low-low level trip does not satisfy IEEE 279-1971 Section 4.7 requirements due to control/protection interaction; i.e., the protection channel that also provides control input fails high causing closure of the main feed regulation valve and a subsequent failure of one of the remaining protection channels will not result in a reactor trip. To satisfy IEEE 279-1971 a diverse protection actuation must be available. As the feedwater flow transmitters installation was not seismically qualified, the steam flow / feed flow mismatch coincident with low steam generator level trip could not be credited as this diverse trip. This was reported in accordance with the requirements of 10 CFR 50.72 (b)(1)(ii)(B) and (b)(1)(i)(A).

Upon recommendation of the Nuclear Safety Review Board, the Plant Manager declared the feedwater flow transmitters inoperable due to their installation in the non-seismic turbine building. Since Technical Specifications 3.3.1.1, Table 3.3-1 item 15 only authorizes one feedwater flow transmitter to be inoperable, Technical Specifications 3.0.3 was entered and a cooldown to Mode 3 was commenced. On June 28, an update to the original 10 CFR 50.72 report was provided to indicate that the steam generator low-low level trip was also inoperable.

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

After Unit 1 had reached Mode 3, at 0343 on July 3, 1997, residual heat removal (RHR) sample isolation valve RH-214 was opened in order to allow sampling of the RHR system prior to placement into service. RH-214 had been administratively closed with a clearance per BVPS-1 Containment Penetrations Overpressure Protection Analysis/Review Report to provide overpressure protection for its associated penetration. When RH-214 was opened, no overpressure protection existed for penetration 97-1 and it was declared inoperable. Technical Specifications Limiting Condition for Operation 3.6.1.1 (Primary Containment Integrity) was entered, and the action required restoration of the penetration within one hour or the initiation of action to place the plant in Cold Shutdown within 30 hours. Because the valve was to remain open for greater than one hour to allow sampling, the unit was taken to Mode 5 and was reported pursuant to the requirements of 10 CFR 50.72 (b)(1)(ii)(A) at 0356.

This event is not applicable to Unit 2 because it has Steam Generator Level Median Signal Selectors installed.

## CAUSE

The apparent cause of the initiating event was a misapplication of IEEE 279-1971 Section 4.7 Control and Protection System Interaction criteria during the original plant design. The issue was not later recognized when Westinghouse letter DLW-90-509, 'Steam Flow Measurement at Low Power Levels', was issued in 1989. This recommended the addition of the median selector switch to eliminate the Control and Protection System interaction issue.

## CORRECTIVE ACTIONS

### COMPLETED:

1. A modification has been completed to install Steam Generator Level Median Signal Selectors. This modification ensures that the steam generator level protection system meets the requirements for Control and Protection System interaction of IEEE 279-1971, and thus removes the requirement to have a diverse backup trip.

2. An evaluation of the inputs and outputs of the Unit 1 RPS to ensure that none of the inputs or outputs from non- seismically qualified equipment or areas will unacceptably degrade the performance of the RPS has been completed.

3. The Unit 1 UFSAR Section 7.2 was revised regarding the reactor trips that were credited to occur in the accident analysis. The rationale used to determine how the failure of unqualified equipment which interfaces with the reactor trip system satisfies IEEE Standard 279-1971 was evaluated by 10 CFR 50.59 safety evaluations and determined not to represent an unreviewed safety question.

### FUTURE:

4. An evaluation of the inputs and outputs of the Unit 2 RPS to ensure that none of the inputs or outputs from non-seismically qualified equipment or areas will unacceptably degrade the performance of the RPS will be performed. Due to emergent activities, including issues associated with non-seismic fire suppression equipment affecting safety-related equipment (Reference LER 1-97-018-00) the original commitment date of August 31, 1997 could not be met. This corrective action will be completed by October 10, 1997.

5. This LER will be reviewed as part of the Engineering Support Personnel continuing training. This will be completed by March 31, 1998.

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

## SAFETY CONSEQUENCES

As potential exists for a control and protection system interaction for the SG level protection system, IEEE-279 1971 requires that a second failure must be postulated. If a second SG level instrument were to fail in an intermediate position, since there are three channels, the requisite RPS trip logic (low-low steam generator level, as credited for protective action in the design basis accident section, Chapter 14 of the Unit 1 UFSAR) would not be satisfied and a reactor trip and Auxiliary Feedwater automatic start would not be generated as required when an actual SG low-low level condition occurred. The described condition does not meet IEEE-279 1971 criteria and is outside the Unit 1 design basis. In the original design, the steam flow / feed flow mismatch trip was installed as a diverse backup trip in order to satisfy the IEEE 279-1971 requirements for control and protection interactions.

## PREVIOUS SIMILAR EVENTS

A review of LERs for the past two years identified no similar events.