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March 19, 1993
ND3MNO:3433

Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, Licensee No. DPR-66
LER 93-004-00

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

Gentlemen:

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

LER 93-004-00, 10 CFR 50.73.a.2.v.C, "Potential Post Small Break Loss of Coolant Accident Radiological Release."

L. R. Freeland
General Manager
Nuclear Operations

DJS/sl

Attachment

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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)

Potential Post Small Break Loss of Coolant Accident Radiological Release

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	1 8	9 3	9 3	0 0 4	0 0	0 3	1 9	9 3	Beaver Valley Unit 2	0 5 0 0 0 4 1 2
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 9 0	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	X 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)	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
L. R. Freeland, 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	
B	C B	X X X X	X X X X	N							

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

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

On February 18, 1993, Westinghouse Nuclear Safety Advisory Letter (DLW-92-701) identified a scenario in which the Volume Control Tank (VCT) outlet check valve (CH-18) could become a potential leak path outside of containment during a 2 through 8 inch small break loss of coolant accident (SBLOCA). The VCT outlet isolation valves upstream of CH-18 were considered to be the flowpath boundary valves and, the function of CH-18 was considered only to be isolation from another closed piping system.

Inspections of the complementary valve (2CHS-18) at Unit 2 had been completed during the last refueling outage in October 1992, and no deficiencies were noted. Although this inspection met the intent of the recommended actions in the Westinghouse letter, upon further analysis, it has been determined that a postulated failure of 2CHS-18 meets the criteria for a potential single failure which may prevent the control of a radiological release outside containment.

Emergency Operating Procedures for both Units have been revised to diagnose the problem based on increasing VCT level and instructs operators to isolate the flowpath.

DESCRIPTION OF EVENT

On February 18, 1993 Beaver Valley Power Station (BVPS) Unit 1 was determined to be in a condition which could have prevented the fulfillment of a system to control the release of radioactive material. Westinghouse Nuclear Safety Advisory letter (DLW-92-701), dated December 29, 1992, applicable to numerous plants, identified a scenario in which the VCT outlet check valve (CH-18) could become a potential leak path outside of containment during a 2 through 8 inch small break loss of coolant accident (SBLOCA). Neither the station, the industry through operating experience, nor Westinghouse in its original design, had previously identified the safety function of preventing post-SBLOCA leakage via this flowpath. During post-SBLOCA recirculation, the VCT outlet isolation valves would be closed to isolate the VCT from recirculation backflow. However, piping from the Seal Water Heat Exchanger (SWHX), while upstream of CH-18, is not isolable by the VCT outlet isolation valves. Any backleakage past CH-18 could pressurize the SWHX and associated piping. At both Units 1 and 2, the SWHX relief set pressure is below the discharge pressure of the recirculation pumps which would cause the relief valve to lift allowing post-SBLOCA recirculation flow to the VCT and beyond. This amount of containment sump water had not been analyzed for discharge to the VCT and has been determined to cause 10 CFR 100 dose guidelines to be exceeded if complete core failure source term is postulated with 10% release of Iodine activity, evaporated to the atmosphere, as required by Standard Review Plan criteria.

Inspections of the complementary valve (2CHS-18) at Unit 2 had been completed during the last refueling outage in October 1992, and no deficiencies were noted. Although this inspection met the intent of the recommended actions in the Westinghouse letter, after further analysis, on March 1, 1993 it was determined that a postulated failure of 2CHS-18 meets the criteria for a potential single failure which may prevent the control of a radiological release outside containment with the equivalent scenario as stated above for Unit 1.

The Nuclear Regulatory Commission was notified on February 18, 1993 (Unit 1) and March 1, 1993 (Unit 2) of these events in accordance with 10 CFR 50.72.b.2.iii.C.

CAUSE OF THE EVENT

The combination of the VCT outlet check valve (CH-18) and the SWHX relief valve as a single failure possibility during a small break loss of coolant accident (SBLOCA) scenario had not previously been evaluated. Neither the station, the industry through operating experience, nor Westinghouse in its original design, had previously identified the safety function of preventing post-SBLOCA leakage via this scenario.

CORRECTIVE ACTIONS

An operability evaluation was immediately performed for each unit to address this degraded condition in accordance with Generic Letter 91-18, and continued plant(s) operation was determined to be acceptable.

For short term corrective action, associated Emergency Operating procedures for both Units have been revised to diagnose the problem based on increasing VCT level and take action to isolate the flowpath. A time/motion study has been conducted and has shown that the radiation exposure to operator(s) for this action is within the normal 10CFR20 quarterly exposure limit of 3 rem and is less than GDC 19 requirements.

For the long term, BVPS is currently evaluating the rating of the Seal Water Heat Exchanger and associated piping to determine if a higher design pressure can be applied. This would allow the heat exchanger relief valves set pressure to be raised above the maximum discharge pressure of the recirculation pumps eliminating any possible flow back to the VCT and outside of containment. These design changes are scheduled to be completed during the next refueling outage at each unit. Unit 1 CH-18 will be included in the Preventative Maintenance (PM) Program for inspection (Unit 2 2CHS-18 is already in the PM Program). Unit 2 SWHX relief valve will be included in the Inservice Test (IST) Program (Unit 1 SWHX relief valve is already in the IST Program).

REPORTABILITY

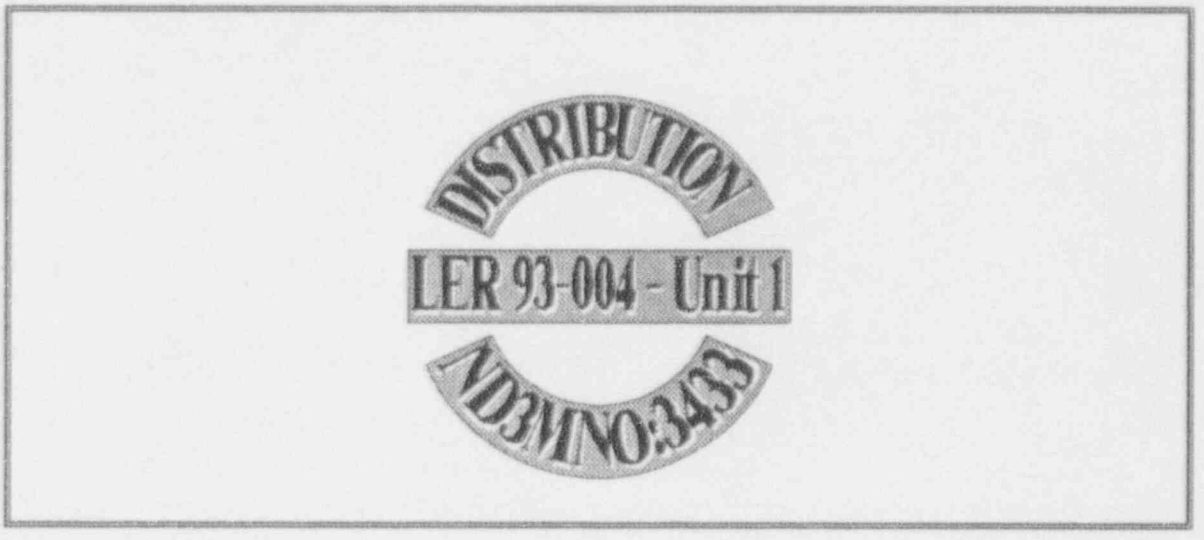
This event was reported to the Nuclear Regulatory Commission at 1435 hours on February 18, 1993 for Unit 1, and at 1304 hours on March 1, 1993 for Unit 2, in accordance with 10 CFR 50.72.b.2.iii.C, as a condition which could have prevented the fulfillment of a system to control the release of radioactive material.

SAFETY IMPLICATIONS

There are minimal safety implications because of this event. Per the FSAR, the maximum calculated peak clad temperature (PCT) for SBLOCA remains less than 2200 degrees F, such that complete core failure due to SBLOCA has been analyzed not to occur (less than 0.3% of total Zr/H2O reaction is expected to occur). A radiological evaluation has been performed and has shown that assuming 1% failed fuel, 10 CFR 100 limits are not exceeded. As an indicator of safety significance, the probabilistic risk assessment (PRA) of core damage due to SBLOCA for Unit 1 is 4.851E-7 and Unit 2 is 1.062E-6 per reactor year.

PREVIOUS OCCURRENCES

There were no previously reported similar events.



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