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 8

TITLE (4)
Reactor Trip and Safety Injection Due to Reactor Coolant Pump 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 NAMES	DOCKET NUMBER(S)
0 6	0 7	8 8	8 8	— 0 0 7 —	0 0	0 7	0 5	8 8	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(a)
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 (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)
Mr. Thomas P. Noonan, Plant Manager	4 1 2 6 4 3 - 5 1 0 1

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
A	A B				B	L C	M S T R	J 0 7 5	Y
B	A B	V	B 3 4 4	Y					

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 6/7/88, at 2155 hours, with Unit I at 100 percent reactor power, an operator inadvertently tripped the 'C' Reactor Coolant Pump Breaker while performing an air compressor test. The loss of the RCP caused a Reactor Trip on low flow in a single loop. An excessive cooldown due to a combination of design features led to a low pressurizer pressure safety injection 29 seconds after the trip. These factors include modifications to the Main Feedwater Regulation Valves to increase their flow and stagnant flow in a 'C' loop instrument manifold. The operators followed the plant's emergency procedures to stabilize the plant in Hot Standby. An Unusual Event was declared at 2206 hours and terminated at 2400 hours. Several minor problems were noted during the trip. After equipment repair, the reactor was taken critical at 2200 hours on 6/8/88. Design changes to the main feedwater regulating valves and instrument manifolds will be made to prevent similar excessive cooldowns. The root cause of this event was a cognitive error on the part of a non-licensed operator who, through inattention to labeling, did not notice that he was tripping the wrong breaker. The operator was disciplined, and this event will be reviewed by all Operations personnel to emphasize the lessons learned.

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On 6/7/88, at approximately 2100 hours, Unit 1 was operating at 100 percent power. Preparations were underway to perform Station Operations Surveillance Test (OST) 1.34.4 "Station Air Compressor SA-C-1A Operational Test". A non-licensed Nuclear Operator (NO) was assigned to perform the test. This test requires the running air compressor which, in this case, was the 'B' compressor (SA-C-1B) to be tripped by opening its power supply breaker, which should cause the backup air compressor (SA-C-1A) to automatically start. The power supply for SA-C-1B is 480V Bus 2C; the actual supply breaker is designated ACB 2C5.

The Nuclear Operator informed the Reactor Operator of his intention to trip the breaker. The NO marked the correct air compressor (SA-C-1B) and breaker (ACB-2C5) on the appropriate portion of the surveillance test, and then proceeded to the Normal (Non-Emergency) Switchgear area with a copy of the surveillance test in hand. He proceeded directly to the 4160V 'C' Bus (4160V Bus 1C) instead of the 480V 'C' Bus (480V Bus 2C) which was the actual power supply of the compressor. The operator, concentrating on the breaker designation rather than the equipment designation mounted on the breaker cubicle door, then proceeded to locate breaker C5, which was actually ACB-1C5 and supplied Reactor Coolant Pump 1C.

At 2155 hours, the NO tripped breaker 1C5. This action immediately stopped the 'C' Reactor Coolant Pump (RC-P-1C) and caused a low flow condition (less than 92% measured flow) in a single loop. Such a condition immediately resulted in a reactor trip since reactor power was above the P-8 (one loop low flow trip interlock) setpoint of 31 percent power. The Reactor Operator noticed the trip of the RCP, the reactor trip breakers opening, and control rods falling into the core. Several alarms annunciated simultaneously to indicate the trip, although no actual first out (red and white flashing) annunciator was observed. Meanwhile, the NO in switchgear heard the breaker he had just tripped cycling, along with other breakers cycling, and the standby alarm announcing a plant trip. At that point he noticed the label on the breaker he had tripped and realized it was the 'C' Reactor Coolant Pump.

The available Control Room personnel immediately began following Beaver Valley Emergency Procedure (EOP) E-0 "Reactor Trip on Safety Injection" and verified the automatic action of the plant's safety features. At this point,29 seconds after the initial trip, a Safety Injection (SI) occurred. The SI signal was generated from low pressurizer pressure (below 1845 psig) as sensed by two of the three pressurizer pressure protection channels. The low pressure was induced as a result of an excessive plant cooldown caused primarily by an excessively long period of time with the condenser steam dumps open, as well as by modifications to the feedwater valves that increased feedwater flow rates. As a result of the Safety Injection, the operators continued following the instructions of Emergency Procedure E-0 for a Safety Injection and placed the plant in Hot Standby by 2210 hours, 6/7/88. The Safety Injection was terminated (charging flow reestablished and normal letdown) at that time.

A total of 2216 gallons of borated water were injected before the SI was terminated. In addition, all valves required to close upon a Containment Isolation Phase A (CIA) signal did so, although TV-SS-112A (Pressurizer Vapor Space Sampling Isolation Valve) exhibited dual indication at its benchboard position lights.

Several minor equipment problems were noted during the plant stabilization. Two Circulating Water Pumps and a Condensate Pump tripped during the onsite to offsite bus transfer because a DC current spike resulting from the breaker transfer itself actuated their overcurrent protection relays.

Another problem involved a failed air start motor solenoid on the No. 1 Emergency Diesel Generator (DG-EE-1). The generator did start properly; however, the air start motor pinion for the left bank of air start motors repeatedly kept attempting to engage. The DG was secured at 2217 hours and declared to be inoperable at 2400 hours 6/7/88.

An Unusual Event was declared due to an Emergency Core Cooling System actuation at 2206 hours 6/7/88, in accordance with the Beaver Valley Emergency Preparedness Plan (EPP) Procedure I-1, Tab 6. The event was terminated at 2400 hours 6/7/88, due to the establishment of a safe shutdown condition in the reactor and the continued availability of long-term core cooling, in accordance with Section 9 of the EPP. The Nuclear Regulatory Commission was notified at 2212 hours in accordance with the one-hour provision of 10CFR 50.72.a.1.i (Declaration of Emergency Class) and 10CFR 50.72b.1.iv (ECCS discharge). This LER is being submitted under 10CFR 50.73.a.2.iv.

Station Instrument and Control investigated the problems with the sample valve TV-SS-112A1 and the Diesel Generator air start solenoid. The valve limit switches were adjusted on the valve operator and the valve was returned to service at 0320 hours 6/8/88. The start solenoid was repaired,and all four air start motors (Ingersoll-Rand Model D89RH-46) replaced by 1640 hours on 6/8/88, at which time the diesel generator was declared to be operable. The reactor was taken critical at 2200 hours 6/8/88.

To date at Beaver Valley Unit 1 there have been twenty (20) operational and two (2) preoperation safety injections.

Since a Diesel Generator start and DG equipment problems were included in the description of this event, the reliability figures for the two Unit 1 Diesel Generators has been provided.

	Last 20 Starts	Last 100 Starts
DG 1-1	1.00	.99
DG 1-2	1.00	1.00

These figures include the latest start.

Cause of Event

The 'C' Reactor Coolant Pump, RC-P-1C was tripped due to a cognitive error, a lack of attention to detail, on the part of a non-licensed operator. Such a lack of attention to detail is exemplified by the NO's over reliance on the breaker designation to identify power sources rather than equipment mark numbers.

The Safety Injection was caused by an excessive plant cooldown attributable to several factors. Foremost among these factors is a previously identified component/design problem, and its inter-relationship with the precise nature of this plant trip. Reactor Coolant System (RCS) temperatures, both protection and control, are sensed by means of Resistance Temperature Detectors (RTDs) located on bypass flow manifolds for each loop. These manifolds contain ten Rockwell T58 valves with disc/stem separation problems. Five (5) of these valves are on the 'C' Bypass Manifold. These valves, identified as problems since 1984, were analyzed by the vendor and Westinghouse at that time. It was concluded that no problem existed with the valves under normal flow conditions in that they would behave as check valves and let flow past. However, under reverse flow conditions they would act to stop the flow. Since forward flow is the normal operating condition at BVPS, it was concluded the problems safety impact was minimal. The trip of the 'C' RCP on 6/7/88, coupled

with full forced flow in the other two loops, led to a reverse flow condition in the 'C' RCS Loop. Thus, stagnant flow conditions were achieved in the 'C' Bypass Manifold and the average temperature as sensed by the instruments in that manifold would not decrease as rapidly as in the other two loops.

The steam dumps receive a modulating signal from auctioneered high control average temperature (Tavg) which is compared to the programmed load Tavg. Such a large deviation from the no-load Tavg controlling setpoint of 547°F kept the dumps open until the 'A' and 'B' RCS loops reached the low-low Tavg (P-12 interlock setpoint 543°F) as sensed by two of three RCS protection temperature channels. The dumps closed at that point but the large volume of steam removed cooled down and dropped the pressure of the primary coolant enough to satisfy the Safety Injection Logic.

The secondary cause of the rapid cooldown relates to the behavior of the feedwater system. In order to reduce the pressure drop across the valve and thus reduce flow-induced oscillation, the Main Feedwater System was modified during the Sixth Refueling Outage by changing the trim on the Main Feedwater Regulating Valves (MFRVs) and reducing the Main Feedwater Pump impeller diameter. These changes had the net effect of increasing feedwater flow rate during any transient condition that would require the MFRVs to fully open. The extra available feedwater (151% nominal full power flow as opposed to the previous valve of 131%) helped cool the RCS more rapidly.

The only reactor trip experienced at Unit 1 that involved a similar loss of coolant flow occurred in 1979 (see LER 79-14). This trip occurred prior to the identified manifold valve problems and no safety injection occurred at that time.

Safety Analysis

The safety implications of this event were minimal because all appropriate Engineered Safety Feature (ESF) equipment, including the Reactor Protection System (RPS) and Safety Injection System (SIS) actuated in response to valid signals. Specifically, the RPS acted to insert the Control Rods into the core upon receipt of a Low Flow Reactor Trip signal, thus removing the active heat source. Furthermore, the Turbine Driven Auxiliary Feedwater Pump

(FW-P-2) started upon receiving a low-low level signal from two of the three level transmitters in the 'C' Steam Generator six seconds after the trip. The Motor Driven Auxiliary Feedwater Pump started upon receipt of a second low-low level signal in a second steam generator. All of these pumps provided additional flow (as the Main Feedwater Pumps were still running) to remove decay heat. These actuations were sufficient to keep the plant transient within the analyzed limits of Section 14.2.7.2 of the Beaver Valley Unit 1 Updated Final Safety Analysis Report (UFSAR) especially since that analysis assumed the more severe case of a locked coolant pump rotor.

The Main Feedwater Regulating Valves opened fully in response to decreasing Steam Generator level as designed. The increased full open flow of 151% full power flow is bounded by the design failed open flow rate of 170% full flow assumed in the Feedwater System Malfunction Analysis of Unit 1 UFSAR Section 14.1.9.2. The feedwater valves shut properly upon the receipt of a low Tavg (554°F) signal as sensed by two of the three RCS average temperature channels to terminate normal feed flow to the generators, while the condenser steam dumps shut at the Low-Low Tavg (P12) interlock. These two actions limited the cooldown transient, although, for reasons discussed above did not prevent a Safety Injection.

The Safety Injection was generated in response to a valid Low Pressurizer Pressure (1845 psig) signal, and all equipment again responded properly. A second charging (High Head Safety Injection Pump) started, along with a second River Water Pump, the two Diesel Generators, and the two Low-Head Safety Injection Pumps. All valves required for aligning the High Head Safety Injection Pumps to take suction from the Refueling Water Storage Tank and to discharge through the Boron Injection Tank moved to their proper position. All Containment Isolation Valves changed position correctly in response to the Phase A Signal. No actual breach of RCS piping took place and pressure quickly recovered from its lowest point of 1845 psig.

Therefore, the event, although more of a challenge to various safety systems than expected, did not create a transient more severe than provided for by the station design. No major equipment damage or release of radioactive materials to the public occurred.

Immediate Corrective Actions

Several pieces of equipment that functioned improperly during the event were repaired prior to reactor startup.

1. The dual indication problem with Pressurizer Vapor Space Sampling Isolation Valve was corrected by adjusting the limit switch. The valve was returned to service at 0320 hours on 6/8/88

2. The air start solenoid for the left bank of our start motors for the No. 1 Diesel Generator was replaced along with all four motors. The DG was returned to service at 1640 hours on 6/8/88.

3. The pump motors for the Cooling Tower Pumps (CT-P-1A and 1B) and the Condensate Pump (CN-P-1B) were bridged and meggered by Station Electrical Maintenance on 6/8/88. The condition of the motors was found to be satisfactory. Relay Technicians from Duquesne Light's Substations Department checked the overcurrent relays and found them to be working properly.

4. The Nuclear Operator involved was reprimanded.

Long Term Corrective Action

1. It was believed that a previous increase in the Main Feedwater Regulating Valve Stroke Time to eliminate a water hammer concern also contributed to the addition of more feedwater to the steam generators. As a consequence the Nuclear Engineering Department reevaluated the water hammer concern and concluded on 6/11/88 that the valves could safely be restored to their former stroke time of approximately five (5) seconds. The valves were modified and restroked on 6/11/88, the new average stroke time is 5.85 seconds.

2. The RTD Bypass Manifolds are to be completely removed as part of a temperature instrumentation modification scheduled for the Seventh Refueling Outage.

3. The surveillance test for the Station Air Compressors, involving as it does a deliberate trip of the running air compressor, is considered an unnecessary challenge to the plant. Although the compressors are not safety related equipment, their loss can impact various plant systems (for instance, station air holds open the Main Steam Isolation Trip Valves) and cause a severe transient. Therefore, the frequency of that test will be changed from monthly to every refueling outage. A review will be made of all other balance of plant surveillance tests to determine whether any similar concerns exist.

4. The event (LER 88-007) will be reviewed and discussed by all operations Shift personnel and will also be included as a topic for the 1988-1989 Operations Retraining Cycle.

Duquesne Light

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July 7, 1988
ND3SPM:0249

Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
LER 88-007-00

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 88-007-00, 10 CFR 50.73.a.2.iv.," Reactor Trip and Safety Injection Due to Reactor Coolant Pump Trip".

Very truly yours,

T. P. Noonan

T. P. Noonan
Plant Manager

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Attachment

IE22
1/1

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