NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Salem Generating Station - Unit 2	0 5 0 0 0 3 1 1	1 OF 0 8

TITLE (4)
Reactor Trip From 66% With Resultant Safety Injection

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 5 0 0 0
0 7	2 5	8 4	8 4	0 1 8	0 2	1 1	2 7	8 5		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 6 6

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(b)
[ ] 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)	[X] 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)	Special Report
[ ] 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)
J. L. Rupp - Operations Licensing Engineer	6 0 9 3 3 9 - 4 3 0 9

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
X	A B	P S V	M 0 9 0	Y	B	A B	R V	C 7 1 0	Y
X	A B	R V	V 0 8 5	Y	B	C B	R V	C 7 1 0	Y

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On July 25, 1984, while performing the final steps of the Pressurizer Overpressure Protection System functional test, Reactor Coolant System pressure rapidly decreased upon opening PORV block valve 2PR6. The operator immediately attempted to close 2PR6; however, it failed to close in the required time, resulting in a reactor trip and safety injection. The Reactor Protection System and all Engineered Safety Feature and Emergency Core Cooling Systems functioned as designed during the transient. Following the safety injection, and subsequent closure of 2PR6, the plant recovered to normal operating parameters. The depressurization was caused by the inadvertent opening, and failure to reseat, of POPS relief valve 2PR47. Investigation of 2PR6 revealed a broken wire in the valve operator circuit. Testing revealed that the valve closure thrust was adequate, although at the minimum recommended value. In addition, it is suspected that the calculated "required" torque may not be adequate when an attempt is made to reverse the valve direction while the valve is in a mid-stroke position. These problems, along with others noted in the text of this LER, were satisfactorily corrected. This report fulfills the requirements of 10CRF 50.73(a)(2)(iv) and Technical Specification 3.5.2 (action b).

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NRC Form 366 (9-83)

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

IDENTIFICATION OF OCCURRENCE:

Reactor Trip From 66% With Resultant Safety Injection

Event Date: 07/25/84

Report Date: 11/27/85

This report was initiated by Incident Report No. **84-115**

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 - Rx Power 066 % - Unit Load 0700 MWe

DESCRIPTION OF OCCURRENCE:

On July 25, 1984, during routine power operation, conditions were being restored to normal in the final steps of the Pressurizer Overpressure Protection System (POPS) functional test. The POPS functional test is performed on two independent trains. It requires that the Power Operated Relief Valve (PORV), for the train being tested, be isolated from the Pressurizer. This is done by closing the associated PORV block valve. With the PORV block valve closed, the PORV can be stroke timed, as required, without affecting normal system operation.

The test was started and satisfactorily completed on Train B of the POPS. The system was then returned to normal (i.e., the PORV block valve was opened). The test on Train A of the POPS was also satisfactorily completed. When the PORV block valve (2PR6) was opened, Reactor Coolant System [AB] pressure began to rapidly decrease. The Reactor Operator immediately initiated a close signal to block valve 2PR6. However, when the valve failed to close in the required time (less than ten seconds), the operator immediately reverified that the PORV on Train A (2PR1) and the PORV and associated block valve on Train B (2PR2 and 2PR7 respectively) were closed. He then attempted to reduce the severity of the transient by manually starting a centrifugal charging pump [CB]. At the same time, the other operator began shedding load, in approximately one-hundred (100) MWe increments, to further reduce the effects of the transient. This action reduced the rate of pressure drop slightly; however, pressure continued to decrease. When pressure had decreased to 1865 psig, the logic for a reactor trip was met, and a reactor trip did occur. Pressure continued to drop to the safety injection initiation setpoint of 1765 psig, at which time, an automatic safety injection occurred. Following the safety injection and the subsequent closure of 2PR6, the plant recovered to normal operating parameters.

APPARENT CAUSE OF OCCURRENCE:

The depressurization transient was initiated by the inadvertent opening, and failure to reseat, of Pressurizer Overpressure Protection System (POPS) relief valve 2PR47. The transient was not able to be immediately terminated, due to the failure of 2PR6 to close in the required time frame, resulting in the reactor trip and safety injection. The failure of 2PR6 to close in the required time was attributed to either a broken wire in the valve operator, a minimum recommended torque switch setting, attempted reversal of the valve direction (while the valve was in a "mid-stroke" position), or a combination of all three. See the "Corrective Action" section of this LER for a more detailed description and resolution of these individual problems.

ANALYSIS OF OCCURRENCE:

The inadvertent opening of 2PR47, coupled with the failure of 2PR6 to close in the required time, resulted in an inadvertent depressurization of the Reactor Coolant System. The Reactor Protection System [JC] and all Engineered Safety Feature and Emergency Core Cooling Systems [JE] functioned as designed during the transient. Had 2PR6 remained open, the Safety Injection System [BQ] flow would have established and maintained an equalibrium pressure in the Reactor Coolant System. With the procedures in effect, the operator had sufficient direction to bring the plant to cold shutdown conditions. Section 15.2.12 of the Updated Final Safety Analysis Report (UFSAR) analyzes the accidental depressurization of the Reactor Coolant System. This section concludes that there is adequate core protection for the opening of a pressurizer safety valve, which is the limiting case. Since the opening of 2PR47 is considerably less severe than the opening of a safety valve (due to the significant difference in flow capacity), the conclusions reached in Section 15.2.12.4 of the UFSAR are valid for this occurrence as well. This was analytically substantiated by Westinghouse through their letter PSE-85-577 (dated May 13, 1985). Their analysis concluded that the event was, in fact, an inadvertent depressurization and not a Small Break Loss of Coolant Accident (SBLOCA), because during the transient the mass flow out of the PORV did not exceed the available makeup.

In addition, the steam generator tubes did not experience pressure reversal, and the primary side pressure decrease was of a magnitude and rate such that the transient is bounded by transients included in the steam generator cycle analysis. The integrity of the steam generator tube bundle was not adversely affected, and the need for an extended shutdown following the transient to inspect the tubes, in accordance with Technical Specification 4.4.6.3.c, was not required. During the Unit 2's subsequent refueling outage, 100% of No. 24 Steam Generator tubes were inspected. In addition, in excess of 3% of the tubes in No. 21, 22 and 23 Steam Generators were also inspected. The surveillance results supports the conclusions that the tube bundle was not affected.

ANALYSIS OF OCCURRENCE: (cont'd)

Therefore, this occurrence involved no undue risk to the health or safety of the public. However, due to the automatic actuation of the Reactor Protection System and the Engineered Safety Feature, this event was reported in LER 84-018-00 in accordance with the Code of Federal Regulations, 10CFR 50.73(a)(2)(iv), on August 24, 1984, within the required thirty (30) day period. On September 25, 1984, the original report was updated to fulfill the requirements of the ninety (90) day Special Report, which is required by Technical Specification 3.5.2 (Action b) upon actuation of the ECCS which injects water into the Reactor Coolant System. That update report (LER 84-018-01) incorrectly stated that the total accumulated actuation cycles of the ECCS to date was six (6). The correct number of accumulated actuation cycles, as of that date, was five (5).

CORRECTIVE ACTION:

The Unit was cooled down to Mode 5, and a thorough investigation of the incident commenced. The Station Operations Review Committee (SORC) met to reconstruct the sequence of events, and review the occurrence. In addition to the obvious questions of why 2PR47 inadvertently opened, and why 2PR6 took 4.5 minutes to close, SORC expressed the following concerns related to the occurrence, and requested that these concerns be addressed during the investigation.

(1) What caused 2RC43 (Reactor Head Vent Solenoid Valve) to pop open at the beginning of the transient?

Does this valve performance create the potential for future leakage paths?

(2) What was the cause of the elevated tailpipe temperature on 2PR5 (Pressurizer Code Safety Valve)?

(3) What was the cause of the leakage on the bellow seals on 2PR5?

(4) Is the relief valve (2CV241) on the Volume Control Tank sized to accommodate the recirculation flow from both charging pumps?

If so, why was the valve damaged from the high recirculation flow?

(5) The present design of 2PR6 allows the operator to reverse the direction of the valve travel. Is this an acceptable practice that will not damage the motor or limitorque gear train?

(6) If question 5 is not an acceptable practice, what other valves are of similar design to 2PR6?

CORRECTIVE ACTION: (cont'd)

(7) 2PR47 (POPS Relief Valve) had been previously seal welded to prevent leakage to the containment atmosphere. Could this work have adversely affected the valve, to cause it to pop open?

After extensive research and testing, the following conclusions were reached and corrective actions were taken:

REACTOR HEAD VENT VALVE: 2RC43 was found to be operating satisfactorily. This is a Target Rock solenoid valve, connected to the reactor vessel head through a line with a 11/32 inch orifice. Solenoid type valves are known to "burp" (pop open and reseat during pressure transients). A recently conducted head vent test revealed that this valve does, in fact, "burp" during pressure transients. During the test, the orifice limited the flow to forty-five (45) GPM with the Reactor Coolant System at normal operating pressure. The "burping" duration is very short and the "burped" liquid stays contained in the Reactor Coolant System. Additionally; the system has been designed such that, even if the valve were to fail open, a SBLOCA would not occur. As such, no concern exists with the continued use of the Target Rock solenoid valves in this application.

PRESSURIZER CODE SAFETY VALVES: Because of the elevated tailpipe temperatures on 2PR5, the valve was removed and sent to Wyle Laboratory for testing. The valve exhibited seat leakage during testing; therefore, a replacement valve was installed. The bellows on 2PR5 was successfully tested with nitrogen gas, and found to be intact. What was thought to be "bellows" leakage was actually attributed to leak-off lines from the safety valves, which tie into the pressurizer relief line. Blanks were installed in these lines to prevent recurrence.

The problem with drifting setpoints and leakage associated with the Pressurizer Code Safety Valves had been previously recognized, investigated and addressed. Some of the problems have been attributed to testing the valves at different valve body temperatures; i.e., early tests (performed in 1982) were conducted at temperatures as high as 500°F, with subsequent tests being performed at 300-350°F and the most recent tests at temperatures below 300°F. In April, 1984, prior to this occurrence, Wyle Test Procedure 1009 was revised to ensure that subsequent tests of PR3, PR4 and PR5 valves were conducted with the valves at the same mean temperature.

VOLUME CONTROL TANK RELIEF VALVE: Investigation revealed that the VCT relief valve (2CV241) is, in fact, sized properly. 2CV241 was not damaged; however, it suffered an isolated case of seating O-ring failure. The O-ring was replaced. As a conservative measure, the VCT was visually inspected for signs of overpressurization. The inspection results were satisfactory.

CORRECTIVE ACTION: (cont'd)

PORV BLOCK VALVES: Motor Operated Valve Analysis and Testing Systems (MOVATS) tested 2PR6. The valve operated satisfactorily, with a closure thrust of 6700 pounds. Per Velan (the valve manufacturer), the required valve closure thrust is 4900 pounds, indicating that the valve should have closed. Records indicate that this valve was reworked in April, 1984, at which time, the wedge and Limitorque operator were replaced. A calculation of the valve thrust and torque values, with information supplied by Limitorque, indicated that the valve operator had been supplied with a light torque switch spring. The close torque switch setting was found to be one and one-half (1 1/2). Velan's minimum recommended setting is one and one-quarter (1 1/4). Although the valve tested satisfactorily, Velan recommended that this setting be two and one-half (2 1/2). This setting would represent approximately 8000 pounds of thrust; and even if a heavy torque spring was present in the operator, it would not be damaging to the valve. The close torque switch setting of 2PR6 was raised to the recommended value.

Further investigation of this occurrence involving the spring pack containing a different torque switch spring than the original revealed that the difference was not reflected in the part number. Therefore, there was no reason to suspect that a different torque switch setting would be required. Because of this, a list was compiled of all Limitorque operators (which were presently installed) that had been purchased as exact duplicate replacements. This list was sent to Limitorque for evaluation. It was determined that none of the operators on the list had had any spring pack size changes; therefore no further adjustments to the torque switch settings were required. To prevent another such occurence, Limitorque has agreed to officially notify PSE&G of any changes in replacement components that could affect operation. This notification is to include information about the modification, and the setting requirements necessary to ensure compatibility of the replacement operator. To further ensure compatibility, Maintenance Procedure M3L verifies that the factory set torque switch settings of the replacement are the same as the old operator. This verification will be performed before the replacement is installed.

Upon electrical disconnection of 2PR6 for internal inspection, the limitorque operator was found to contain a broken wire. This seven strand wire carries control voltage to the valve for opening and closing functions. Oxidation of the strands indicated that two of the strands were broken for some time, with the other five strands indicating a more recent break. The bolts at the base of the valve were found to be slightly loose, which allowed a small amount of valve operator movement. Since the wire run was taut, it is suspected that vibratory action of the valve broke the wire. The break was enclosed in sleeving, and the wire apparently was making intermittent contact during valve vibration. The broken wire was repaired. 2PR7 was also inspected; however, no similar problems were noted.

CORRECTIVE ACTION: (cont'd)

2PR6 was disassembled and inspected. Inspection of the wedge rails revealed no signs of galling, and the valve internals were found to be in excellent condition. Upon investigation of 2PR6 design, which allows reversing of valve direction at any time, the Station was informed by Limitorque that this was an undesirable design, due to the possibility of shearing the keyway on the pinion gear. In addition, while performing calculations for required valve torque, it was discovered that the coefficient of friction value (which is used in the calculation) may not be valid during periods of direction reversal while the valve is in a "mid-stroke" position. What this means, is that the required torque, during valve reversal operations, may be greater than previously calculated. Since that possibility existed, and because Limitorque has expressed the opinion that this is an undesirable design, the circuits for all PORV block valves (in both Unit 1 and Unit 2) were modified to provide "seal-in" features to prevent direction reversal until the valves have completed their stroke (open or shut).

Additionally, because other safety-related systems contained motor operated valves which do not contain a "seal-in" feature in the motor operator control circuits, appropriate procedures were changed to caution the operators against reversing direction until the valves have completed their stroke. An engineering investigation was initiated, and Safety Evaluation S-C-X110-MSE-0325 subsequently reviewed all safety-related motor operated valves not containing this "seal-in" feature. The purpose of this safety evaluation was to evaluate the reliability of those valves by determining the thrust (or safety) margins available for unanticipated increases during valve operations resulting from, but not limited to, reverse stroking of the valve in mid cycles. As a result, the torque switch settings on the following valves are being increased to ensure the existence of adequate thrust margins; CC16 (Cooling Water to the RHR Heat Exchangers), SJ113 (Charging/Safety Injection and Safety Injection Suction Cross-Over), SJ30 (Safety Injection Pump Isolation from Refueling Water Storage Tank), and SJ134 (Cold Leg Injection). No other valves require any adjustments or modifications.

POPS RELIEF VALVES: Inspection of valve 2PR47 revealed that the valve was open. Particles from the valve magnet had lodged in the pilot stem, preventing the pilot valve from closing. 2PR47 is a solenoid valve, also known to "burp" during pressure transients. The valve apparently "burped" while testing 2PR1, and the magnetic particles wedged in the pilot stem and prevented the valve from reseating. Seal welding of the valve was not the cause of the failure. 2PR47 and 2PR48 served no purpose because the relief function of these valves had been previously replaced with modifications to the circuitry of the PORV valves (2PR1 and 2PR2). These modifications had already been accomplished due to previous problems with the POPS relief valves, and both 2PR47 and 2PR48 were scheduled for removal in the near future. Due to this occurrence, both valves were removed from the system. Unit 1 design does not contain POPS Relief Valves.

CORRECTIVE ACTION: (cont'd)

TRAINING: In addition, this reactor trip/safety injection event was evaluated by the Training Department. As a result of this evaluation, training on the incident was included in the Licensed Operator Requalification Program cycle for 1984-85. The training was incorporated into a "pressurizer vapor space break accident analysis" lecture and a simulator demonstration of the event. Training was also conducted on motor operated valve operations, including the recent design changes.

General Manager-
Salem Operations

JLR:tns

SORC Mtg 85-153

PSE&G

Public Service Electric and Gas Company P.O. Box E Hancocks Bridge, New Jersey 08038

Salem Generating Station

November 27, 1985

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-75
DOCKET NO. 50-311
UNIT NO. 2
LICENSEE EVENT REPORT 85-018-02
SUPPLEMENTAL REPORT

This update report is being submitted pursuant to the requirements of 10CFR 50.73(a)(2)(iv) and Technical Specification 3.5.2 (action b).

Sincerely yours,

J. M. Zupko, Jr.

J. M. Zupko, Jr.
General Manager-
Salem Operations

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