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	0 5 0 0 0 3 1 1	1 OF 0 8

TITLE (4)
Number 22 Steam Generator Safety Valves Inoperable

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 4	0 7	8 5	8 5	— 0 0 7 —	0 0	0 6	2 6	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3
POWER LEVEL (10): 0 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)	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)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 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)
J. L. Rupp - LER Coordinator	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	S 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 April 7, 1985, during Unit heatup operations following a refueling outage, four of the five main steam line code safety valves associated with No. 22 Steam Generator prematurely lifted and were gagged shut. Action was initiated to place the Unit in Hot Shutdown in accordance with LCO 3.0.3. The cause was determined to be incorrect lift settings, although the root cause could not be determined. Because the lift set testing requires steam generator pressure to be greater than 900 psig, cooling down would have precluded being able to correct the cause of the problem. Therefore, the cooldown was terminated, and the valves were reset and satisfactorily tested under the cognizance of the vendor. Since the Technical Specification basis was not compromised, and because the Commission was aware of the situation and concurred with PSE&G's actions at the time of the event, this event was initially classified as non-reportable. However, because the action requirements of LCO 3.0.3 were not complied with, the event was subsequently determined to be reportable in accordance with 10 CFR 50.73(a)(2)(i)(B), and the LER is being submitted greater than thirty days following the actual event date. A License Change Request is being submitted to revise the Technical Specification action requirements when more than three (3) safety valves are inoperable (due to setpoint differences) on any steam generator.

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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:

No. 22 Steam Generator Safety Valves Inoperable

Event Date: 04/07/85

Report Date: 06/26/85

This report was initiated by Incident Report No. **85-088**

CONDITIONS PRIOR TO OCCURRENCE:

Mode 3 - Rx Power 000 % - Unit Load 0000 MWe

DESCRIPTION OF OCCURRENCE:

On April 6, 1985, following a refueling outage, 22MS14 (No. 22 Steam Generator Safety Valve) started leaking through. At the time of the event, the Unit was in Mode 3 (Hot Standby), Reactor Coolant System [AB] heatup operations were in progress, with temperature and pressure at 495^{o}F and 2000 psig respectively, and No. 22 Steam Generator pressure was 650 psig. In addition, the Power Range Neutron Flux High trip setpoint was 17.5% in preparation for low power physics testing.

There are five (5) main steam line code safety valves associated with each of the four (4) steam generators. Technical Specification 3.7.1.1 requires all main steam line code safety valves associated with each steam generator to be operable with lift settings as specified below:

	Loop A	Loop B	Loop C	Loop D	Lift Setting (+- 1%)
a.	21MS11	22MS11	23MS11	24MS11	1125 psig
b.	21MS12	22MS12	23MS12	24MS12	1120 psig
c.	21MS13	22MS13	23MS13	24MS13	1110 psig
d.	21MS14	22MS14	23MS14	24MS14	1100 psig
e.	21MS15	22MS15	23MS15	24MS15	1070 psig

DESCRIPTION OF OCCURRENCE: (cont'd)

Technical Specification Action Statement 3.7.1.1.a states:

> With four (4) reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in Modes 1 (Power Operation), 2 (Startup) and 3 (Hot Standby) may proceed provided that, within four (4) hours, either the inoperable valve is restored to operable status or the Power Range Neutron Flux High trip setpoint is reduced as per below; otherwise, be in at least Hot Standby within the next six (6) hours and in Cold Shutdown (Mode 5) within the following thirty (30) hours.

MAXIMUM # OF INOPERABLE SAFETY VALVES ON ANY OPERATING S/G	MAXIMUM ALLOWABLE POWER RANGE HIGH FLUX SETPOINT
1	87
2	64
3	42

22MS14 was declared inoperable and Technical Specification Action Statement 3.7.1.1.a was entered at 0900 hours, April 6, 1985. With one safety valve inoperable and with the high flux trip setpoint at 17.5% the Unit was maintained in Hot Standby (temperature approximately 490°F), as authorized by the action requirements, while the problem was evaluated. Investigation revealed that the manual lifting device was binding the valve; i.e., the spindle nut was jammed against the forked lever. After backing the spindle nut away from the forked lever, 22MS14 seated. Technical Specification Action Statement 3.7.1.1.a was terminated at 1340 hours, April 6, 1985. Reactor Coolant System heatup operations continued, and at 2155 hours, with temperature and pressure at 520°F and 2235 psig respectively, and with steam generator pressure approximately 800 psig, the control room received indications of a safety valve prematurely lifting on No. 22 Steam Generator. At 2205 hours, visual observation confirmed that 22MS14 had lifted, and when the valve did not immediately reseat, an "Unusual Event" was declared. At 2305 hours, in accordance with the requirements of the Code of Federal Regulations 10CFR 50.72(a)(1)(i), the Commission was notified of the declaration of the "Unusual Event". Technical Specification Action Statement 3.7.1.1.a was again entered, 22MS14 was subsequently gagged shut and the "Unusual Event" was terminated at 0245 hours, April 7, 1985.

Later that day (April 7, 1985) 22MS12 and 22MS13 prematurely lifted and, at 1356 hours, both of these valves were also gagged shut. Technical Specification Action Statement 3.7.1.1.a was still in effect due to the inoperability of 22MS14.

DESCRIPTION OF OCCURRENCE: (cont'd)

At this time, there were a total of three (3) inoperable main steam line code safety valves (all associated with No. 22 Steam Generator). Since the Technical Specification requires the high flux trip setpoint to be equal to or less than 42%, and the actual setpoint remained at 17.5%, operation was in accordance with the action requirements. However, 22MS11 subsequently lifted, and it too was declared inoperable. Technical Specification 3.7.1.1 allows continued plant operation with a maximum of three (3) safety valves inoperable. Since the action requirements of this specification could no longer be met, Limiting Condition for Operation 3.0.3 was entered at 1948 hours, April 7, 1985.

Limiting Condition for Operation (LCO) 3.0.3 states:

> When a Limiting Condition for Operation is not met, except as provided in the associated action requirements, within one hour action shall be initiated to place the Unit in a mode in which the specification does not apply by placing it, as applicable, in:
>
> 1. At least Hot Standby within the next 6 hours,
> 2. at least Hot Shutdown within the following 6 hours, and
> 3. at least Cold Shutdown within the subsequent 24 hours.

The Unit was in Hot Standby at this time and, in accordance with LCO 3.0.3, action was immediately initiated to place the Unit in Hot Shutdown within the following six (6) hours. However, the reason for the valves being inoperable was incorrect lift setpoints. Therefore, compliance with the requirement to cooldown would have precluded being able to correct the cause of the problem, since setting of the safety valves requires steam generator pressure to be greater than 900 psig, which cannot be accomplished in Mode 4. In addition, the operability of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% rated thermal power coincident with an assumed loss of condenser heat sink; i.e., no steam bypass to the condenser, and inoperability of the atmospheric vents (MS10 Valves). Based on this review of Technical Specification 3.7.1.1 basis, because the Unit was in Mode 3 with the Main Steam Isolation Valves (MS167's) closed, knowing that the atmospheric vents were operational and the fact that cooling down would preclude correcting the problem, a management decision was made to terminate the cooldown, maintain the Unit in Mode 3 and correct the lift setpoint problem of the safety valves. LCO 3.0.3 was determined to be inappropriate in this situation, and was terminated at 2221 hours, April 7, 1985.

DESCRIPTION OF OCCURRENCE: (cont'd)

Based on the results of a review which revealed that the Technical Specification basis was not compromised, and because the Commission was aware of the situation and concurred with PSE&G's actions at the time of the event, this occurrence was initially classified as non-reportable. However, PSE&G subsequently determined that an LER explaining the circumstances surrounding this event is appropriate. This decision was based on 1) the desire to fully document this unique event, 2) PSE&G's intention to submit a License Change Request related to this Technical Specification, and 3) on discussions with the Commission (concerning the new reporting requirements) during a recent LER symposium. During those discussions, the Commission pointed out that they would expect an LER to be submitted any time the requirements of Limiting Condition for Operation 3.0.3 are invoked, even though the actions were not completed; i.e., mode changes were not actually accomplished. Following the LER symposium, the occurrence was re-evaluated, and on May 28, 1985, the event was reclassified as reportable. This LER is therefore being submitted within thirty (30) days following the determination of reportability rather than thirty (30) days following the actual event date.

APPARENT CAUSE OF OCCURRENCE:

The initial problem with 22MS14 leaking through was binding of the manual lifting device; the root cause being attributed to differential thermal expansion of the valve parts. The thermal growth experienced during Mode 3 heatup eliminated the clearance between the spindle nut and the forked lever. This caused the two (2) parts to act as a jacking device which lifted the spindle and disc insert, resulting in leakage. As previously mentioned, the spindle nut was backed away from the lever, reseating the valve. Both of these parts were subsequently removed from the valve in accordance with Design Change Request 2SC-1580, which also removed these parts from all twenty (20) safety valves. The lifting levers on safety valves have been deleted from Section III of the ASME Boiler and Pressure Code since 1977, since all utilities have an Inservice Inspection Program to test main steam isolation valves once per refueling outage. The environment in a nuclear power plant is also much cleaner than that of a fossil power plant, which precludes oxidation of the nozzle/disc area preventing sticking or galling; thus, the need for manual lifting levers is obviated.

Removal of the manual lifting devices did not affect the lift setpoints of the safety valves. The initial problem with 22MS14 manual lifting device causing the valve to leak through was unrelated to the subsequent events involving the premature lifting of this valve and the other three (3) safety valves associated with No. 22 Steam Generator. The cause of the premature lifting of the safety valves was incorrect lift set pressures.

APPARENT CAUSE OF OCCURRENCE: (cont'd)

All twenty (20) safety valves were lift set checked. Testing revealed the safety valves to require the following corrections (1 flat = approximately 8 to 10 psig):

21MS11 - none required	22MS11 - 20 flats
21MS12 - none required	22MS12 - 32 flats
21MS13 - none required	22MS13 - 25 flats
21MS14 - 1/2 flat	22MS14 - 32 flats
21MS15 - none required	22MS15 - 8 flats
23MS11 - 3 flats	24MS11 - none required
23MS12 - 1 flat	24MS12 - 1/4 flat
23MS13 - 2 flats	24MS13 - none required
23MS14 - 2 flats	24MS14 - 1/2 flat
23MS15 - none required	24MS15 - 1/4 flat

In all cases except one, the safety valves lifted conservatively (prior to the design lift set pressure; 21MS14 lifted approximately four (4) psig higher than the design set pressure of 1100 psig (+- 1%). As shown by the testing results, the safety valves associated with No. 22 Steam Generator required excessive adjustments to bring the lift set pressures within specifications. A detailed investigation was performed in order to establish a root cause for these incorrect lift setpoints. Several causes were postulated; however, investigations could not validate their credibility, and the root cause of the incorrect settings could not be determined.

The investigation revealed that all twenty (20) safety valves were previously lift set checked with satisfactory results on March 18, 1985. These settings were performed with main steam at 900 psig and the Reactor Coolant System at 531°F and 2235 psig. On March 19, 1985, the Unit was heated up to 547°F and 2235 psig (main steam at 547°F/1005 psig) in preparation for unit startup, and the safety valves did not lift at that time. On March 24, 1985, the Unit was cooled down to Mode 5 for unrelated maintenance activities associated with the reactor vessel head conoseals. The Unit remained in Cold Shutdown until March 31, 1985, at which time, Unit heatup operations were again commenced and the Unit was returned to Mode 4 operation. Mode 3 was achieved on April 6, 1985, at which time, the sequence of events (as delineated in the "Description of Occurrence" section) commmenced.

Because the safety valves did not lift during the period between March 19, and March 24, 1985, it appears that the valves were properly lift set tested on March 18, 1985. In addition, the same individuals who lift set tested the safety valves associated with No. 22 Steam Generator also lift set tested the safety valves for No. 24 Steam Generator.

APPARENT CAUSE OF OCCURRENCE: (cont'd)

As shown by the testing results following the event, No. 24 Steam Generator safety valve lift setpoints were satisfactory. Documentation also exists that the gages which were used during the testing were calibrated seventy-two (72) hours prior to their use, and these same gages were used during the lift set testing of both No. 22 and No. 24 Steam Generator safety valves.

The possibility of someone tampering with the valves was investigated. Although this is a possibility, it was determined to be highly unlikely for the following reasons. The safety valves are located in a controlled access area. All personnel entering this area are logged in by a security guard. The person tampering with the valves would have to be knowledgeable about safety valves, would have to remove the valve cap, cotter pin, forked lever and manual lifting lever. It normally takes two maintenance men with a wrench and a cheater bar to move the three inch adjusting nut one flat. To remove the associated parts from five (5) safety valves, move the adjusting nuts a total of 117 flats and reassemble all valves under the pressure of willful tampering, while lowering the safety valve setpoints below operating pressure, is hardly conceivable and highly unlikely.

ANALYSIS OF OCCURRENCE:

As defined in the Technical Specification basis for the main steam safety valve operability and in section 15.2.7.1 of the UFSAR, the capacity of the safety valves is based on being able to relieve the steam flow at the Engineered Safeguards design flow (~105% steam flow at rated power) from the steam generators without exceeding 110% of the steam system design pressure of 1085 psig, assuming a turbine trip and reactor trip from full power without the operation of the condenser steam dump system or the atmospheric relief valves (MS-10's).

However, during startup operations, the heat production which would cause a sudden increase of pressure in the steam generators is not present since the reactor is maintained in a subcritical condition. Therefore, if the atmospheric steam reliefs (which have a capacity to relieve 10% of full steam flow) are available, the safety valves are not required to be operable. In addition, the only sources of pressure increase within the steam generators are readily controllable by the operator. These sources include heat input from the Reactor Coolant System [AB] (where the only source of heat is from the operation of the reactor coolant pumps) and from a level increase in the steam generators. The heat input from running the reactor coolant pumps is a slow developing transient. Overpressurization due to a water solid condition in the steam generators can be a relatively rapid transient; however, there are ample indications and alarms available to the operator to warn of an approach to a water solid condition well in advance of the condition existing.

ANALYSIS OF OCCURRENCE: (cont'd)

As previously stated, the main steam safety valves lifted in the conservative direction (prior to the design lift set pressure), and the integrity of the Main Steam System [SB] was not in question. Additionally, testing revealed that the safety valves reseated as designed. Engineering Evaluation No. S-2-G210-MEE-051 concluded that plant safety, at no time, was compromised. Therefore, this occurrence involving the inoperability of the safety valves, and the decision to maintain the Unit in Hot Standby, involved no undue risk to the health or safety of the public. However, because the requirements of LCO 3.0.3 were not complied with, the event was determined to be reportable in accordance with the Code of Federal Regulations, 10CFR 50.73(a)(2)(i)(B).

CORRECTIVE ACTION:

All main steam line code safety valves were retested and adjusted (to +-1% tolerance) under the cognizance of the vendor. Repeated testing verified the lift settings to be stable, and the lift set errors could not be duplicated. Wire seals were then installed on the valve caps to ensure that the integrity of the lift settings were not compromised. As previously stated, the manual lifting devices were removed from all safety valves in accordance with an approved design change.

In addition, the Maintenance Department lift set test procedure (M20A-1) was reviewed for completeness and accuracy. Although the vendor confirmed that temperature effects during testing would not cause the magnitude of changes which were encountered with No. 22 Steam Generator Safety Valves lift set pressures, the following revisions are being implemented. The procedure is being revised to include the temperature of the Main Steam System and the time at which each lift set test was performed. The revision also requires at least four (4) hours warm-up of the Main Steam System before starting lift set testing, and a minimum of five (5) minutes to elapse between successive lifts on each valve to allow the valve temperatures to stabilize.

Technical Specification 3.7.1.1 was reviewed for "Mode" applicability. Based on this review, a License Change Request is being submitted to revise the action requirements for this specification when more than three (3) safety valves are inoperable (due to setpoint differences) on any steam generator.

General Manager-
Salem Operations

JLR:tns

SORC Mtg 85-092

PSE&G

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

Salem Generating Station

June 26, 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 85-007-00

This Licensee Event Report is being submitted, pursuant to the requirements of 10CFR 50.73(a)(2)(i)(B), within thirty days of reportability determination.

Sincerely yours,

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

JLR:tcs

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