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 1	0 5 0 0 0 2 7 2	1 OF 1 0

TITLE (4)
Failure to Implement Portions of the Inservice Testing Program

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 1	3 1	8 6	8 6	— 0 0 2	— 0 0	0 2	2 8	8 6	Salem - Unit 2	0 5 0 0 0 3 1 1
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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)	1 0 0	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 - 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	

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)

During a routine surveillance of the Inservice Testing (IST) program by the Station Quality Assurance organization, it was discovered that current testing procedures did not reflect all testing specified in the program. A comprehensive review of the programs for both Units 1 and 2 identified valves in various systems for which testing was either incomplete, or not being conducted on the frequency prescribed by Section XI of the ASME Boiler and Pressure Code. All deviations between the IST programs and the current station testing procedures were identified and analyzed. Each discrepancy was reviewed on an individual basis to determine the potential effects on system operability. No instances were identified where the lack of complete testing resulted in a challenge to the continued safe operation of the Station. The "root" cause of this event was attributed to inadequate management attention to the IST program requirements. Testing procedures will be issued or revised, as necessary, to ensure compliance with the IST programs. Administrative and/or departmental procedures will be revised to better define the responsibilities and requirements of the IST program. Personnel involved with the IST program will receive training to ensure a complete understanding of their responsibilities and of the overall program requirements.

IE22 1/1

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

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

IDENTIFICATION OF OCCURRENCE:

Failure to Implement Portions of The Inservice Testing (IST) Program

Event Date: 01/31/86

Report Date: 02/28/86

This report was initiated by Incident Report No. **86-034**

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 - Rx Power 100 % - Unit Load 1120 MWe

BACKGROUND:

Technical Specification Surveillance Requirement 4.0.5 states:

> Inservice inspection of ASME Code Class 1, 2, and 3 components and inservce testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10CFR 50, Section 50.55a(g)(6)(i).

On February 3, 1979, PSE&G submitted the proposed Unit 1 Inservice Testing (IST) program to the Commission. This was followed by a revision to the program on March 10, 1980, and another on April 29, 1981. Between August 31 and September 3, 1982, the Commission conducted a routine inspection at Salem Generating Station to review the status of the IST program submittals. During the inspection, it was brought to PSE&G's attention that the current NRC position was that applicable components in the Fire Protection System [KP] and the Emergency Diesel Generator Auxiliary Systems [EK] be included in the IST program. In addition, even though 1RH1 and 1RH2, which are the Residual Heat Removal System [BP] common suction valves from the Reactor Coolant System [AB] Hot Legs, are not listed in Table 4.4-4 of Unit 1 Technical Specification 3.4.6.3, they too should be considered Reactor Coolant System pressure isolation valves requiring periodic testing per Surveillance Requirement 4.0.5. These findings were formally transmitted to PSE&G in Inspection Report No. 50-272/82-23 (dated October 5, 1982). The inspection report stated that the Commission was preparing a Safety Evaluation Report (SER) approving the IST program, and that a formal response to these findings would be expected sometime after the issuance of the SER.

BACKGROUND: (cont'd)

1RH1 and 1RH2 were subsequently included in Surveillance Procedure SP(O) 4.4.6.3 (Emergency Core Cooling - ECCS Subsystems), Revision 4, dated January 6, 1983, and the valves were first tested, with satisfactory results, on February 10, 1983. PSE&G then received the SER, dated April 12, 1983, approving the IST program as submitted, with the following conditions; 1) the RHR valves were to be included by July 12, 1983, and 2) the applicable components of the Fire Protection and Emergency Diesel Generator Auxiliary Systems be included by April 12, 1984. As previously stated, 1RH1 and 1RH2 had already been incorporated into the program and initially tested. Verification of this was transmitted to the Commission in a letter dated July 11, 1983. The Fire Protection and Emergency Diesel Generator Auxiliary Systems were reviewed and applicable components requiring testing were identified and included in the program. The Commission was notified of this in a letter dated April 12, 1984.

DESCRIPTION OF OCCURRENCE:

During the performance of a routine surveillance of the IST program by the Station Quality Assurance organization in January 1986, it was discovered that 1RH1 and 1RH2 were no longer included in the surveillance testing program. Investigation revealed that the required testing had been deleted from SP(O) 4.4.6.3 prior to the next scheduled performance (September 8, 1984) by a temporary on-the-spot change to the procedure; apparently because the testing was not identified in the Technical Specifications. On March 21, 1985, following the two (2) year review cycle, SP(O) 4.4.6.3, Revision 5, was issued, at which time, 1RH1 and 1RH2 were permanently deleted from the procedure. This resulted because Technical Specification 3.4.6.3 did not require the valves to be tested, and because the reviewer was not familiar with the current IST program testing requirements.

Further investigation revealed that, although the amended IST program (which was submitted to the Commission on April 12, 1984) included testing requirements for specific components in the Emergency Diesel Generator Auxiliary Systems, implementing procedures had not yet been written. These findings were brought to the attention of Station Management, at which time, a comprehensive review of the IST programs for both Units 1 and 2 was ordered. This review identifed valves in various systems for which testing was either incomplete, or not being conducted on the frequency prescribed by Section XI of the ASME Boiler and Pressure Code. Each identified discrepancy was transmitted to the Engineering and Plant Betterment Department for review which, in turn, was instructed to notify the General Manager-Salem Operations of any potential effects on the operability of the system. There were no instances identified where the lack of complete testing resulted in a challenge to the continued safe operation of the Station. The results of that review are detailed in the "Analysis of Occurrence" section of this report.

APPARENT CAUSE OF OCCURRENCE:

The "root" cause of this event was inadequate management attention to the IST program requirements, resulting in:

1. administrative and departmental procedures not adequately defining the responsibilities and requirements of the IST program as specified in Vice President-Nuclear Procedure PLP-10,

2. poor coordination and communications between the department responsible for identifying the testing requirements and the departments responsible for implementing the testing,

3. the lack of aggressiveness on the part of personnel involved with the IST program to either a) resolve problems known to exist within the program, or b) to make these problems known to the responsible management staff, and

4. the program lacking a means of "closing the loop" to ensure that the requirements identified within the scope of the IST program were actually being accomplished.

Additionally, inadequate training was evidenced by the fact that some personnel involved in the program did not fully understand the relationship between Technical Specification and IST requirements.

This event also revealed an apparent weakness in the Quality Assurance program, which resulted in not identifying this problem in a timely manner.

ANALYSIS OF OCCURRENCE:

As stated, a comprehensive review of the IST programs for both Units was performed. Each testing requirement was compared with existing procedures to ensure total compliance. Each discrepancy was then reviewed on an individual basis to determine the potential effects on system operability. No instances were identified where the lack of complete testing resulted in a challenge to the continued safe operation of the Station. The components which were identified as having the "potential" to affect system operability, and the safety significance of incomplete testing follows:

1. 11-13AF53 (Unit 1); 21-23AF53 (Unit 2); 1AF71, 1AF72 (Common to both Units) - NOT MANUALLY EXERCISED - These are check valves located in the alternate suction header for the Auxiliary Feedwater Pumps [BA]. This line is normally drained because it is routed through the electrical switchgear room. The only time the line is filled is in the event that the Auxiliary Feedwater Storage Tank [BA] becomes unavailable as a source of auxiliary feedwater. To date, use of this header has never been required. Additionally, Technical Specification 3.7.1.3 provides alternatives to using the alternate suction header.

ANALYSIS OF OCCURRENCE: (cont'd)

1. (cont'd)

   The operators were instructed to consider this header inoperable until the required testing (manually exercising) of these valves could be performed. These valves have since been inspected, manually exercised and verified operational.

2. 11AF40, 12AF40 (Unit 1); 21AF40, 22AF40 (Unit 2) - NO FLOW RESPONSE TEST - These valves are air operated, fail open valves on the recirc lines for the motor driven Auxiliary Feedwater Pumps. These valves are required to open on low flow conditions (for the protection of the pumps) and to close as flow increases (to ensure full flow to the steam generators). Although not formally documented, it has been shown, with a reasonable degree of confidence, that the valves have been functioning properly. The monthly surveillance test of the pumps in accordance with Technical Specification 3.7.1.2 utilizes the recirc line to provide the flow path, thus requiring the valves to be open. When the pumps have been used to feed the steam generators, no problems have been encountered in achieving full flow as required. This would not have been possible if the valves were not closed.

3. 11-14AF920, 11-14AF921 (Unit 1); 21-24AF920, 21-24AF921 (Unit 2) - FULL CLOSE VERIFICATION NOT PERFORMED - These are check valves in the discharge lines from each Auxiliary Feedwater Pump. Their function is to prevent cross-connecting the pump discharges, thus preventing the loss of auxiliary feedwater flow to all steam generators through a break in one (1) line. Although there has been no documented testing to verify that the valves were seated to prevent backleakage, subsequent radiographic examination has verifed full closure of all valves. Forward flow has been verified each time the pumps were used to feed the steam generators.

4. 1RH1, 1RH2 (Unit 1) - NO LEAK RATE TEST - These valves isolate the Residual Heat Removal System [BP] suction from the Reactor Coolant System [AB] and provide containment isolation. As previously stated, these valves were originally included in the IST program, and were previously tested with satisfactory results until erroneously deleted during a procedural revision. Further review of the Residual Heat Removal (RHR) System indicates that if the valves had been leaking the RHR System would have become pressurized and, at 375 psig, the suction relief valve (1RH3) would have lifted and relieved to the Pressurizer Relief Tank. This would have resulted in an increased RCS leak rate which would would have been detected and addressed by existing leak rate procedures. **Note**: Testing of 2RH1 and 2RH2 (Unit 2) is required by the Technical Specifications and has always been performed; therefore, this discrepancy was unique to Unit 1. 1RH1 and 1RH2 have subsequently been leak tested with satisfactory results.

ANALYSIS OF OCCURRENCE: (cont'd)

5. Diesel Auxiliaries - Valves in the diesel support systems (fuel oil [DC], lube oil [LA], starting air [LC] and turbo boost air [EK]) which were identified as requiring testing - NO DOCUMENTED TESTING - These valves were reviewed on an individual basis. It has been determined that the monthly diesel testing as prescribed by Technical Specification 3.8.1 and Regulatory Guide 1.108 has adequately demonstrated the operability of these support systems.

6. 1SJ3 (Unit 1); 2SJ3 (Unit 2) - FULL CLOSE VERIFICATION NOT PERFORMED; 1SJ31 (Unit 1) and 2SJ31 (Unit 2) - NOT REVERSE FLOW TESTED - These are check valves in the suction of the Charging and Safety Injection Pumps [BQ] from the Refueling Water Storage Tank (RWST). These valves are required to seat to prevent backflow of water to the RWST during recirculation phases in the event of an accident. There are motor operated isolation valves in series with each of these valves which are closed, by procedure, during the sequence of transfer from the injection to recirculation phase during an accident. Therefore, in the event of an accident, if the valves had failed to seat, backflow to the RWST would have existed only until the motor operated isolation valves were closed. Subsequent radiography has verified that 1SJ3 and 2SJ3 are fully closed. 1SJ31 and 2SJ31 have been tested to verify closure upon reverse flow.

7. 1SJ107 (Unit 1); 2SJ107 (Unit 2) - NOT REVERSE FLOW TESTED - This is a check valve in the recirc line for the Boron Injection Tank (BIT) from the Boric Acid Storage Tank (BAST). This valve serves as a pressure boundary to prevent the high pressure injection flow through the BIT from being diverted to the BAST recirc line. However, there is also an air operated, fail closed valve (1SJ108) in this line which is actuated by a safety injection (SI) signal. The current surveillance procedures for testing the safeguards system has verified the proper operation of 1SJ108; therefore, it can be reasonably assumed that this valve would have closed as required upon receipt of an SI signal. 1SJ107 and 2SJ107 have subsequently been tested with satisfactory results.

8. 1CH30, 1CH150, 1CH117, 1CH151 (Unit 1); 2CH30, 2CH151 (Unit 2) - NO EXERCISE AND STROKE TIME TEST - These valves isolate portions of the Chilled Water System [KM] which are not required during an accident. These include the Penetration Area Cooling Units (PACU's) and the Chemistry Laboratory. The closure of these valves is initiated by the actuation of an SI signal. However, the operation of these valves, in response to this signal, is not verified by current surveillance procedures; therefore, it is assumed (for evaluation purposes) that the valves would not have closed upon receipt of an SI signal.

ANALYSIS OF OCCURRENCE: (cont'd)

8. (cont'd)

The Chilled Water System was reviewed to determine the effect of these additional loads, both from a thermal load consideration as well as a total system flow consideration. The review indicated that because the PACU cooling requirements would have reduced significantly, due to the isolation of the Steam Generator Blowdown System [WI] and the reduction of heat load from the Control Room (the Control Room goes into a recirc mode and cooling of outside air is not required), the thermal load on the Chilled Water System would have been within the capacity of two (2) chillers. The total system flow requirements were reviewed and it was determined that without the isolation of the PACU's and Chemistry Laboratory, the additional flow imposed by the Emergency Air Compressor and the Emergency Control Area Air Conditioning System would not have resulted in flow in excess of the capacity of one (1) Chilled Water Pump. All of these valves have been subsequently tested with satisfactory results.

9. 11SW20, 13SW20, 1SW26 (Unit 1); 21SW20, 23SW20, 2SW26 (Unit 2) - NO STROKE TIME TEST - These Service Water System [BI] valves isolate the non-safeguards Turbine Generator Area from the safeguards portion of the system upon receipt of a Safeguards Equipment Control (SEC) signal. These are motor-operated valves with a stroke time of ten (10) seconds. This ensures that the valves are fully closed before the service water pumps start and provide flow for the Emergency Diesel Generators [EK] and Containment Fan Coil Units [BK]. A review of the loading sequence for the diesels indicates that the valves will receive power to the operators as soon as the diesel output breaker closes. The first service water pump does not start for nine (9) seconds, and the remaining two (2) pumps start at thirteen (13) seconds. The first group of Containment Fan Coil Units do not start until eighteen (18) seconds, while the remaining two (2) groups start at twenty-two (22) seconds. Therefore, a ten (10) second stroke time will assure that these valves are closed prior to the first service water pump coming up to speed; thus, limiting the flow from the pump to only the safeguards header. The closure of these valves in response to an SEC actuation is verifed during the performance of the safeguards actuaton tests performed each refueling outage. The 11 and 13SW20 valves have also been exercised during the performance of the monthly service water pump surveillance, although stroke time for the valves has not been recorded. However, during the performance of the safeguards testing, no problems have been encountered with the service water pumps, in terms of excessive flow or low nuclear service water header presssure, which would be indicative of the valves failing to close within the required time.

ANALYSIS OF OCCURRENCE: (cont'd)

9. (cont'd)

A review of the vendor supplied pump and motor data indicates that the pumps would not trip on overload due to excessive flow, but that system pressure would be low. Therefore, it can be inferred that the stroke time for these valves were, in fact, satisfactory. the SW20 valves for each Unit have been subsequently tested with satisfactory results; the stroke time of all four (4) valves has been verified to be less than ten (10) seconds.

10. 1SW308, 1SW311 (UNIT 1); 2SW308, 2SW311 (Unit 2) - NO EXERCISE OR STOKE TIME TEST - These are air operated, modulating, fail closed valves which operate in conjuction with the service water header relief valves (11 & 12SW49 and 11 & 12SW305) to provide overpressure relief protection for the service water headers by recirculating the service water pump discharge back to the river. In the event that the SW308 and SW311 valves opened and failed to close, the additional flow would have been within the capacity of the service water pumps, but would have resulted in a reduction of service water header pressure to approximately 100 psig. With this reduction of pressure, the service water headers would still have supplied adequate flow to the safeguards equipment. In the event that these valves had failed to open upon increasing service water header pressure, the associated SW49 and SW305 valves would have been capable of providing overpressure relief protection.

As stated, the above listed valves were those identified as having the "potential" to affect system operability. The remaining discrepancies fall into three (3) catagories:

1. **Check valves which were tested in only one direction.** These were reviewed on an individual basis to determine in which direction(s) testing was required. In all cases, except as noted previously, it was determined that the current testing has been satisfactory for the intended safeguards function of the check valve.

2. **Valves which were not tested on the frequency required by either Section XI of the ASME Code or as committed to in the IST program.** Many valves had been granted relief from being tested every three (3) months, as stipulated in article IWV-3410 due to operational considerations. Relief was granted to test these valves in cold shutdown and refueling. This was erroneously interpreted by the procedure writers to mean either cold shutdown or refueling, and the valves were included only in the refueling procedures. In other cases, the need to request relief from the three (3) month test requirement was overlooked, and these valves were included in either the cold shutdown or the refueling procedures; the decision being based on operating conditions required to accomplish the testing.

ANALYSIS OF OCCURRENCE: (cont'd)

2. (cont'd)

As previously stated, each discrepancy was individually reviewed and determined not to have a detrimental effect on system operability.

3. **Valves with remote position indication which were not tested every two (2) years as required by article IWV-3300.** All motor-operated valves are tested on a refueling frequency as part of the required Limitorque maintenance. All air operated valves are currently tested on a frequency of four and one-half (4 1/2) years as required by the limit switch maintenance. Review of these valves indicates that there are ample alternate methods of determining that the valve is traveling full stroke as indicated by the remote position indication. These include the leak testing which is performed each refueling for containment isolation valves, changes in stroke times recorded during exercising, and system parameters such as flow or temperature.

In conclusion, this problem was isolated to those valves specified within the IST program which require testing in accordance with Surveillance Requirement 4.0.5. All other testing requirements specified in the Technical Specifications has been performed. All deviations from the approved IST program were identified. Each valve has been reviewed on an individual basis and from a potential cumulative system impact. The review and the subsequent testing which has been performed to date indicate that the Units have not been operated in a degraded mode which would have posed a threat to the health or safety of the public. However, because all testing was not in accordance with the requirements of Technical Specification Surveillance Requirement 4.0.5, the event is reportable in accordance with the Code of Federal Regulations, 10CFR 50.73(a)(2)(i)(B).

CORRECTIVE ACTION:

The required testing was performed on all of the valves listed in sections 1 through 10 of the "Analysis of Occurrence" section of this report with the exeptions noted below. All test results were satisfactory with no discrepancies noted, which confirms the operability of these components during previous plant operation.

11 and 12AF40 and 1SW26 will be tested during the upcoming Unit 1 refueling outage presently scheduled for March 1986. 21 and 22AF40 will be tested the next time Unit 2 is in Mode 3 (Hot Standby). 2SW26 will be tested the next time Unit 2 is in Mode 5 (Cold Shutdown) or during the next scheduled refueling outage, whichever occurs first. The SW308 and SW311 valves are presently cleared and tagged for maintenance. These valves will be tested following completion of the maintenance activities.

CORRECTIVE ACTION: (cont'd)

All deviations between the Unit 1 and Unit 2 IST programs (submitted December 31, 1985 and January 28, 1986, respectively) and the current station testing procedures have been identified and analyzed on an individual basis. Those having the "potential" to affect system operability were discussed in detail in this report. For the remaining items:

1. testing procedures will be developed or existing procedures will be revised, as necessary, to ensure full compliance with the IST program requirements,

2. all 4.0.5 testing procedures will be reviewed and revised as necessary to ensure consistency, and

3. each 4.0.5 testing procedure will identify the specific testing requirement (i.e., Technical Specification, IST program, etc.) to preclude future deletions of testing requirements.

Additionally, a comprehensive review of compliance with the NRC SER (dated April 12, 1983) and of the IST program implementation was performed by Station Quality Assurance. This independent review revealed no deviations between the SER and current testing procedures other than those previously discussed in this report. Station Quality Assurance will provide surveillance over the IST program activities in accordance with the current Quality Assurance program. However, the Quality Assurance audit program will be modified during 1986 to include the ISI/IST programs as a separate functional element.

The IST program is being evaluated from an organizational standpoint. Appropriate administrative and/or departmental procedures will be revised to better define the responsibilities and requirements of the IST program as specified in Vice President-Nuclear Procedure PLP-10. Personnel involved with the IST program will receive training to ensure a complete understanding of their responsibilities and of the overall program requirements.

These corrective actions, addressing the procedural, programatic, training and Quality Assurance deficiencies noted in the "Apparent Cause of Occurrence" section of this report, should prevent similar occurrences. Amended IST programs for both Units will be submitted to the Commission by September 30, 1986, at which time, all newly developed procedures will be issued and all procedural revisions and training will be completed. The Quality Assurance organization will conduct another detailed review after the revised programs are submitted to verify compliance.

General Manager
Salem Operations

JLR:tns

SORC Mtg 86-013

PSEG

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

Salem Generating Station

February 28, 1986

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

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-70
DOCKET NO. 50-272
UNIT NO. 1
LICENSEE EVENT REPORT 86-002-00

This Licensee Event Report is being submitted pursuant to the requirements of 10CFR 50.73(a)(2)(i)(B). This report is required within thirty (30) days of discovery.

Sincerely yours,

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

JLR:ama

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