PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

March 2, 1995

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

Attn: Document Control Desk

SALEM GENERATING STATION
LICENSE NO. DPR-70 and DPR-75
DOCKET NO. 50-272 and 50-311
UNIT NO. 1 and 2

LICENSEE EVENT REPORT NO. 272/95-001-00

This Licensee Event Report is being submitted pursuant to the requirements of Code of Federal Regulation 10CFR50.73(a)(2)(ii)(B). Issuance of this report is required within thirty (30) days of event discovery.

Sincerely,

T Hellmer

for J. C. Summers
General Manager -
Salem Operations

SORC Mtg. 95-026
FHW:vs

C Distribution
LER File

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NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Salem Generating Station - Unit 1	05000272	1 OF 7

TITLE (4) Technical Specification (TS) 3.0.3 Entry; Both Trains of the Solid State Protection System (SSPS) Being Inoperable

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
02	01	95	95	-- 001 --	00	03	02	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	(Specify in Abstract below and in Text, NRC Form 366A)
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 (Include Area Code)
Frederick Wiltsee, LER Coordinator	609 339-5163

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 15 single-spaced typewritten lines) (16)

On 2/1/95, Unit 1 and 2 entered TS 3.0.3 when both Solid State Protection System (SSPS) trains were declared inoperable after discovery that the AC power distribution within the SSPS is susceptible to a common mode failure. The NRC granted discretionary enforcement allowing 4 days to implement design changes to modify the power distribution arrangement within SSPS. During Unit 1 implementation of the design changes, a number of power supply related problems were encountered. On 2/3/95, Unit 1 initiated a shutdown in accordance with TS 3.3.3.1 for exceeding reactor trip bypass breaker closure time in support of SSPS design changes. Unit 2 entered TS 3.0.3 on 2/3/95, for both SSPS trains being inoperable, NRC rescinded discretionary enforcement at time of TS 3.0.3 entry. This occurred as a result of original design of the SSPS. The apparent cause of the SSPS power supply failures has been attributed to aged components and the lack of preventive maintenance. Design changes have been completed to rewire the SSPS power supply leads. New power supplies were installed and additional power supplies were cleaned, tested and returned to service. Evaluation is on going to determine appropriate action concerning preventive and predictive testing requirements for SSPS power supplies. Event discovery followed original identification of this issue by Diablo Canyon.

Plant and System Identification:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes appear in the text as {xx}

Identification of Occurrence:

Technical Specification(TS) 3.0.3 Entry; Both Trains of the Solid State Protection System (SSPS) Inoperable

Event Date: 2/1/95 and 2/3/95

Report Date: 3/2/95

This report was initiated by Incident Report No. 95-066, 95-073, and 95-075.

Conditions Prior to Occurrence:

Unit 1 Mode 1 Reactor Power 100 %

Unit Load 1151 MWe

Unit 2 Mode 2 Reactor Power 10E-08 amps

Unit Load 0 MWe

Description of Occurrence:

On February 1, 1995, Unit 1 and 2 entered TS 3.0.3 when both SSPS trains were declared inoperable after discovery that the AC power distribution within SSPS {JC} is susceptible to a common mode failure. SSPS input signals which originate in the turbine building (auto stop oil, stop valve limit switches, and reactor coolant pump breaker position signals) were susceptible to failure (short circuits), due to the consequential effects of design basis accidents, including earthquake, and environmental effects of pipe ruptures. The Nuclear Regulatory Commission (NRC) granted discretionary

Description of Occurrence: (cont'd)

enforcement allowing 4 days to implement design changes to modify the power distribution arrangement within SSPS and change fuse sizes.

This concern is due to the location of equipment, wiring, and junction boxes with respect to high energy lines and the non-seismic design of the turbine building which feeds the 15 and 48 volt power supplies and field contacts powered from the SSPS input bays.

Event discovery followed original identification of this issue by Diablo Canyon.

During Unit 1 implementation of the design changes, a number of power supply related problems were encountered. On February 3, 1995 at 1100 hours (hrs), a shutdown was initiated in accordance with TS 3.3.3.1 Action 1 due to exceeding time for having reactor trip bypass breaker closed in support of SSPS design changes. Unit 1 entered Mode 3 at 1700 hrs, entered TS 3.0.3 both SSPS trains being inoperable, and exited discretionary enforcement. On February 4, 1995 at 2230 hrs, the Unit was placed in Mode 5 for the completion of SSPS design changes.

Unit 2 entered TS 3.0.3 on February 3, 1995 at 1640 hrs, for both SSPS trains being inoperable, NRC Region I rescinded discretionary enforcement at time of TS 3.0.3 entry. At 1730 hrs (same day), a Unit shutdown from Mode 2 was initiated and Mode 3 was entered at 1820 hrs (same day). On February 4, 1995 at 0004 hrs, Mode 4 was entered and Mode 5 was entered on February 5, 1995, at 0431 hrs, for completion of SSPS design changes. The NRC was notified of the shutdown initiation, in accordance with Code of Federal Regulations 10CFR50.72(b)(1)(ii)(b).

Analysis of Occurrence:

A condition in which a fault in the circuitry for the turbine stop valve limit switches, autostop oil pressure switches or reactor coolant pump breaker position could possibly render the solid state protection system (SSPS) trains inoperable was identified at a plant of similar design. The review concluded that a single initiating event

Analysis of Occurrence: (cont'd)

(e.g., main steam line break or seismic event), could possibly render one or both trains of SSPS inoperable.

The postulated failure can compromise the SSPS power supplies, due to the location of the 15 AMP fuses within the input bays. This would result in a loss of power to the SSPS power supplies and subsequent loss of power to the SSPS logic and master relays (note that loss of the 48 volt power supply will cause a reactor trip).

Electrical terminal boxes contain two SSPS instrument channels for turbine stop valve position indication while another terminal box contains the auto stop oil input to SSPS. These channels are non-safety related (non 1E) inputs to SSPS and are not electrically isolated from the safety related (1E) portion of the SSPS.

If the steam jet from the faulted main steam line was to strike one of the electrical terminal boxes, or if a seismic event affected the terminal boxes, short circuits in the non 1E inputs to the SSPS could result. Since the non 1E channels are not electrically isolated from the SSPS, the short would cause the fuses for the associated 1E channels to open. The opening of the fuses for the 1E channels would result in the deenergizing of the power supplies for the logic circuitry of one or both trains of SSPS. Assuming the credible failure of a short to ground, the SSPS 15 and 48 VDC power supplies would deenergize from the short circuit.

At 0230 hrs on February 2, 1994, the NRC granted verbal enforcement discretion allowing 96 hours to restore operability of both SSPS trains. PSE&G committed to a formal written request on February 3, 1995. This allowed restoration of one train of SSPS to operable status and termination of TS 3.0.3. The requested duration of the enforcement discretion was from 0230 hrs on February 2, 1995 until 0230 hrs on February 6, 1995, or completion of modifications.

Throughout this period, both trains would remain functional. During the design modification, only one train at a time was rendered inoperable. During the modifications the redundant train was maintained operable, as well as the reactor trip function from SSPS. In support of this extension the following accident initiators were considered to be

Analysis of Occurrence: (cont'd)

applicable: 1) seismic (alone); 2) seismic event resulting in a loss of off-site power; 3) seismic resulting in a steam line break in the affected area; 4) fire; and 5) steam line break.

Other events such as turbine building crane operation, handling and/or dropping of heavy loads, missile generation, and tornado were considered as potential initiators, however, consequences were less severe or comparable to that of the seismic event.

The power supply issues identified during SSPS design change implementation were due to age related component failures (i.e. capacitors, transistors) and the lack of preventive maintenance. Power supplies were found with an excessive accumulation of dust. In addition, the following were identified: a wire was shorted to the rear mounted heat sink, a ground had propagated from the conductive circuit board stand-offs (the other power supplies utilize non-conductive standoffs).

Apparent Cause of Occurrence:

The cause of this event is "Design, Manufacturing, Construction/Installation", as classified in Appendix B of NUREG 1022. This occurred as a result of original design of the Solid State Protection System (SSPS).

The apparent cause of the SSPS power supplies failures has been attributed to aged components and the lack of preventive maintenance.

Prior Similar Occurrence:

Review of documentation did not show similar occurrences.

Safety Significance:

This event did not affect the health and safety of the public. It is reportable pursuant to

Safety Significance: (cont'd)

10CFR50.73(a)(2)(ii)(B). In addition, this report is intended to satisfy reporting requirements applicable to a potential 10CFR21 concern involving the SSPS.

A modified hot zero power steam line break (SLB) core response analysis (UFSAR Section 15.4.2) was performed in which all four steam generators blow down to the environment without operator action or automatic mitigation of any kind. The evaluation shows that the current UFSAR licensing basis analysis of SLB core response remains bounding and DNBR limits are not exceeded and apply to previous operating cycles.

Long term heatup analysis indicates that even though all four steam generators dry out, the actuation of a single motor-driven auxiliary feedwater (AFW) pump is sufficient to prevent reactor coolant system (RCS) over-pressurization, pressurize overfill, or hot leg boiling which are typical bounding criteria applied to long-term events such as loss of normal feedwater, loss of AC power to auxiliaries, and feedline rupture. The evaluation shows the long term consequences of the subsequent heatup of the reactor coolant system (RCS) was analyzed with acceptable results. In addition, the frequencies of these events are relatively low.

Reactor vessel integrity for the effects of pressurized thermal shock (PTS) were evaluated with acceptable results.

Corrective Action:

Design changes have been completed to eliminate the identified failure mechanism of SSPS power supplies.

New power supplies were installed in Unit 2 and one new power supply was installed in Unit 1. Three power supplies were cleaned and bench tested satisfactorily for return to service in Unit 1.

Evaluation is ongoing to determine appropriate action

Corrective Action: (cont'd)

concerning preventive and predictive testing requirements for SSPS power supplies.

Hellmer for

J. C. Summers
General Manager -
Salem Operations

FHW:vs
REF: SORC Mtg. 95-026