LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 | N J S G S 1 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

CON'T
01 | REPORT SOURCE L (6) | 0 5 0 0 0 2 7 2 (7) | 0 6 0 2 8 3 (8) | 1 2 1 3 8 3 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 In June, notification was received from Westinghouse Research Division of a potential
03 USQ. According to the notice, operation at full power following extended reduced power
04 operation could result in increased power peaking in the bottom of the core and possibly
05 exceeding limits for the Heat Flux Hot Channel Factor. Neither unit had operated in
06 the manner identified; and steps were immediately taken to preclude such operation.
07 Prompt notification was made to the USNRC in accordance with 6.9.1.8h.
08

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
09	R C (11)	B (12)	A (13)	F U E L X X (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 3	—	0 2 3	/	0 1	X	—	1

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
G (18)	Z (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	Y (24)	N (25)	W 1 2 0 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 Procedures were implemented that augment the Technical Specifications. The measured
11 target value is administratively controlled to ensure the plant is within the RSE
12 design tolerance under all operating conditions. We no longer have a safety question.
13 Westinghouse concurred with the action.
14

15 | FACILITY STATUS E (28) | % POWER 1 0 0 (29) | OTHER STATUS (30) N/A | METHOD OF DISCOVERY D (31) | DISCOVERY DESCRIPTION (32) Vendor Notification

16 | ACTIVITY RELEASED Z (33) | CONTENT OF RELEASE Z (34) | AMOUNT OF ACTIVITY (35) N/A | LOCATION OF RELEASE (36) N/A

PERSONNEL EXPOSURES
17 | NUMBER 0 0 0 (37) | TYPE Z (38) | DESCRIPTION (39) N/A

PERSONNEL INJURIES
18 | NUMBER 0 0 0 (40) | DESCRIPTION (41) N/A

LOSS OF OR DAMAGE TO FACILITY (43)
19 | TYPE Z (42) | DESCRIPTION N/A

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PDR ADOCK 05000272
S PDR

PUBLICITY
20 | ISSUED N (44) | DESCRIPTION (45) N/A | NRC USE ONLY

NAME OF PREPARER J. L. Rupp PHONE: (609) 339-4309

PSE&G

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

Salem Generating Station

December 13, 1983

Dr. Thomas E. Murley
Regional Administrator
USNRC
Region 1
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Dr. Murley:

LICENSE NO. DPR-70
DOCKET NO. 50-272
REPORTABLE OCCURRENCE 83-023/01X-1
SUPPLEMENTAL REPORT

Pursuant to the requirements of Salem Generating Station Unit No. 1 Technical Specifications, Section 6.9.1.8h, we are submitting supplemental Licensee Event Report for Reportable Occurrence 83-023/01X-1.

Sincerely yours,

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

JR:kll

CC: Distribution

1/1
IE22

Report Number: 83-023/01X-1

Report Date: 12-13-83

Occurrence Date: 06-02-83

Facility: Salem Generating Station Unit 1
Public Service Electric & Gas Company
Hancock's Bridge, New Jersey 08038

IDENTIFICATION OF OCCURRENCE:

Power Distribution Limits - Heat Flux Hot Channel Factor - Potentially Out-of-Specification.

CONDITIONS PRIOR TO OCCURRENCE:

Unit 1 - Mode 1 - Rx Power 100 % - Unit Load 1125 MWe.
Unit 2 - Mode 5 - Rx Power 0% - Unit Load 0%.

DESCRIPTION OF OCCURRENCE:

On June 2, 1983, during normal operation, notification was received from Westinghouse Research Division of a potential unreviewed safety question involving operation at full power following extended reduced power operation. According to the information received, continuous plant operation at less than 85% power for periods greater than 500 MWD/MTU, followed by a return to full power in the same cycle, could result in increased power peaking at the bottom of the core. This increase could potentially result in exceeding the Technical Specification limit for the Heat Flux Hot Channel Factor ($F_Q(Z)$). Both Salem Generating Station units were potentially affected, although at the time of notification neither unit had been operated in a pattern of concern, and Unit 2 was shutdown for refueling.

Prompt notification of the Resident NRC Inspector was performed on June 2, with written confirmation transmitted later that day. The Operations Department was informed of the potential problem and work immediately commenced on development of procedural changes to prohibit operation of the plant in a fashion leading to excessive values of $F_Q(Z)$. A followup report was submitted on June 15, 1983.

APPARENT CAUSE OF OCCURRENCE:

Initial investigation of the problem by the PSE&G Nuclear Fuels Group revealed that the reload safety evaluation (RSE) was based primarily on unrodded, full power depletions. In the RSE perturbation analysis of bounding operating conditions, peaking factors are compared to limits for an envelope around predicted axial flux difference (AFD) values. Subsequent plant operation is bounded by the RSE as long as the measured target flux condition is in good agreement with the predicted values.

Unrodded, low power operation, however, causes the axial flux to be significantly more positive than at full power. Continued operation at low power followed by a return to full power operation can

APPARENT CAUSE OF OCCURRENCE: (cont'd)

therefore result in a shift of the target flux position relative to that assumed in the RSE. If the shift is significant, the allowed plant operating AFD conditions may not be bounded by the reload analysis.

ANALYSIS OF OCCURRENCE:

The limits on heat flux and nuclear enthalpy hot channel factors ensure that the design limits on peak local power density and minimum DNBR are not exceeded and in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Based on the information that was received, existing Technical Specification requirements were insufficient to prevent exceeding the limit for $F_Q(Z)$. As noted, neither Salem core was operated in a manner outlined as being of concern, and measures were immediately taken to preclude such operation. Due to the presence of an oversight in the methods used in the transient or accident analyses which could possibly have resulted in operation in a manner less conservative than assumed in the analyses, the event was reportable in accordance with Technical Specification 6.9.1.8h.

CORRECTIVE ACTION:

As noted, operation of Salem Unit 1 was continued, and immediate measures were taken to insure that the Technical Specification bases were met. Procedures were drafted and incorporated in the Reactor Engineering Manual which now augment the Technical Specifications; the measured target value is administratively controlled to insure the plant is within the RSE design tolerance under all operating conditions. We therefore no longer have a safety question.

Subsequently, on August 12, 1983, we received a letter from the Westinghouse Nuclear Fuels Division recommending the same procedural changes that we had already incorporated. These measures, combined with existing Technical Specification requirements will insure safe operation of both units. Based on the information we have at the present time, a License Change Request is not necessary.

FAILURE DATA:

Not Applicable

Prepared By J. Rupp

General Manager -
Salem Operations

SORC Meeting No. 83-149