PSEG

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

Nuclear Business Unit

DEC 12 1997

LR-N970795

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

Attn: Document Control Desk

MONTHLY OPERATING REPORT
SALEM UNIT NO. 2
DOCKET NO. 50-311

Gentlemen:

In compliance with Section 6.9.1.6, Reporting Requirements for the Salem Technical Specifications, the original Monthly Operating Report for November, 1997, is attached.

Sincerely,

A. C. Bakken III
General Manager -
Salem Operations

IE24

RBK/tcp
Enclosures

C Mr. H. J. Miller
Regional Administrator USNRC, Region 1
475 Allendale Road
King of Prussia, PA 19046

SALEM GENERATING STATION DOCKET NO.: 50-311
UNIT: Salem 2
DATE: 12/15/97
COMPLETED BY: R. Knieriem
TELEPHONE: (609) 339-1782

MONTHLY OPERATING SUMMARY - UNIT 2
NOVEMBER 1997

Salem Unit 2 began the month of November operating at full power. At 2216 November 20, load was reduced to 89.5% to perform a Steam Flow Differential Pressure Transmitter Test. The unit returned to 100% power at 0800 on November 21, and remained at full power for the remainder of the month.

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UNIT:	Salem 2
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TELEPHONE:	(609) 339-1316

OPERATING DATA REPORT

OPERATING STATUS

1	Reporting Period NOVEMBER 1997	Hours in Report Period	720
2	Currently Authorized Power Level (MWt)		3411
	Max Dependable Capacity (MWe-Net)		1106
	Design Electrical Rating (MWe-Net)		1115
3	Power level to which restricted (if any) (MWe Net)		None
4	Reason For Restriction (if any)		

		This Month	Yr To Date	Cumulative
5	No. of hours reactor was critical	720	2379	80463
6	Reactor reserve shutdown hours	0.0	0.0	0.0
7	Hours generator on line	720	2125	77355
8	Unit reserve shutdown hours	0.0	0.0	0.0
9	Gross thermal energy generated (MWH)	2448156	5913921	193694926
10	Gross electrical energy generated (MWH)	825430	1921763	80570361
11	Net electrical energy generated (MWH)	792965	1727434	76430068
12	Unit Service Factor	100.0%	26.5%	48.9%
13	Unit Availability Factor	100.0%	26.5%	48.9%
14	Unit Capacity Factor (MDC)	99.6%	19.5%	43.7%
15	Unit Capacity Factor (DER)	98.8%	19.3%	43.4%
16	Unit Forced Outage Rate	0.0%	73.5%	33.6%

17 Shutdowns scheduled over next 6 months (type, date, duration):

18 If shutdown at end of report period, estimated date of Startup:

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OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

MONTH NOVEMBER 1997

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENT

(1) Reason

A - Equipment Failure (Explain)
B - Maintenance or Test
C - Refueling
D - Regulatory Restriction
E - Operator Training/License Examination
F - Administrative
G - Operational Error (Explain)
H - Other

(2) Method

1 - Manual
2 - Manual Trip
3 - Automatic Trip/Scram
4 - Continuation
5 - Other (Explain)

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AVERAGE DAILY UNIT POWER LEVEL

MONTH NOVEMBER 1997

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	1099	17	1102
2	1104	18	1107
3	1103	19	1109
4	1104	20	1110
5	1103	21	1087
6	1103	22	1109
7	1100	23	1105
8	1100	24	1104
9	1101	25	1102
10	1095	26	1100
11	1098	27	1098
12	1102	28	1102
13	1104	29	1101
14	1103	30	1102
15	1104		
16	1099		

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SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS
FOR THE SALEM UNIT 2 GENERATING STATION

MONTH NOVEMBER 1997

The following items completed during **November 1997** have been evaluated to determine:

1. If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or
2. If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or
3. If the margin of safety as defined in the basis for any technical specification is reduced.

The 10CFR50.59 Safety Evaluations showed that these items did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. These items did not change the plant effluent releases and did not alter the existing environmental impact. The 10CFR50.59 Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

Design Changes Summary of Safety Evaluations

2EE-0147, Pkg. 1, Control Valve 2CV55 Replacement. This modification replaced the existing Centrifugal Charging pump Flow Control valve 2CV55 with a design that provides more reliable flow control during normal and depressurized Reactor Coolant system modes of operation.

This design change does not negatively impact any accident response. This design change does not increase the probability or consequences of either an accident or a malfunction of equipment important to safety. Therefore, this design change does not involve an Unreviewed Safety Question.

2EC-3329, Pkg. 1, Condenser Hotwell Level Control Modifications. This design change replaced the existing Hotwell Level instrumentation. It also modified Control Room indication to provide level indication for all six hotwells and provided trend recording for condensate overflow.

This design change does not negatively impact any accident response. This design change does not increase the probability or consequences of either an accident or a malfunction of equipment important to safety. Therefore, this design change does not involve an Unreviewed Safety Question.

2EC-3319, Pkg. 1, Feedwater Flow Nozzle Replacement. This design change involved the replacement of the Feedwater Flowmeter nozzles with ASME flow nozzles and added four new Chordal Type Leading Edge Ultrasonic flow meters.

This design change does not negatively impact any accident response. This design change does not increase the probability or consequences of either an accident or a malfunction of equipment important to safety. Therefore, this design change does not involve an Unreviewed Safety Question.

Temporary Modifications Summary of Safety Evaluations

There were no changes in this category implemented during November, 1997.

Procedures Summary of Safety Evaluations

NC.DE-AP.ZZ-0004(Q), Design Drawings. The proposed change involves nomenclature and responsibility changes related to the process for controlling engineering design drawings.

This UFSAR change does not negatively impact any accident response. This design change does not increase the probability or consequences of either an accident or a malfunction of equipment important to safety. Therefore, this design change does not involve an Unreviewed Safety Question.

UFSAR Change Notices Summary of Safety Evaluations

There were no changes in this category implemented during November, 1997.

Deficiency Reports Summary of Safety Evaluations

There were no changes in this category implemented during November, 1997.

Other Summary of Safety Evaluation

Safety Evaluation - 100% Power Operation With Degraded Advanced Digital Feedwater Control System (ADFCS) Median Signal Select (MSS) Function. This Safety Evaluation evaluated the proposal of "forcing" two of the three loop 2 steam flow channels to predetermined values for use in the non-safety related ADFCS during full power operation in response to the apparent failure of the 2FT523 and 2FA3472 channels.

This Safety Evaluation does not negatively impact any accident response. It does not increase the probability or consequences of either an accident or a malfunction of equipment important to safety. Therefore, this Technical Specification Bases change does not involve an Unreviewed Safety Question.