PSEG

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

Nuclear Business Unit

OCT 15 1997

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

Attn: Document Control Desk

MONTHLY OPERATING REPORT
SALEM UNIT NO. 1
DOCKET NO. 50-272

In compliance with Section 6.9.1.6, Reporting Requirements for the Salem Technical Specifications, the original monthly operating report for September, 1997, is attached.

Sincerely yours,

A. C. Bakken III
General Manager -
Salem Operations

RBK:tcp
Enclosures

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

DOCKET NO.: 50-272
UNIT: Salem 1
DATE: 10/10/97
COMPLETED BY: F. Todd
TELEPHONE: (609) 339-1316

MONTHLY OPERATING SUMMARY

MONTH SEPTEMBER 1997

The unit is in a refueling and a steam generator replacement outage and remained shutdown for the entire period. According to commitments from PSE&G and a subsequent confirmatory action letter from the NRC, the unit will remain shutdown pending completion of the following actions:

- Appropriately address long standing equipment reliability and operability issues.
- After the work is completed, conduct a restart readiness review to determine for ourselves the ability of the unit to operate in a safe, event free manner.
- After the restart review, meet with the NRC and communicate the results of that review.

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OPERATING DATA REPORT

OPERATING STATUS

1	Reporting Period SEPTEMBER 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	0.0	0.0	104380.5
6	Reactor reserve shutdown hours	0.0	0.0	0.0
7	Hours generator on line	0	0	100338.3
8	Unit reserve shutdown hours	0.0	0.0	0.0
9	Gross thermal energy generated (MWH)	0	0	31806229
10	Gross electrical energy generated (MWH)	0	0	105301000
11	Net electrical energy generated (MWH)	-3779	-26247	100148489
12	Unit Service Factor	0.0%	0.0%	51.9%
13	Unit Availability Factor	0.0%	0.0%	51.9%
14	Unit Capacity Factor (MDC)	0.0%	0.0%	46.8%
15	Unit Capacity Factor (DER)	0.0%	0.0%	46.4%
16	Unit Forced Outage Rate	100.0%	100.0%	38.1%

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

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

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OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

MONTH SEPTEMBER 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
3859	970901	F	720.0	F, C	4	Steam Generator Replacement

(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 SEPTEMBER 1997

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	0
2	0	18	0
3	0	19	0
4	0	20	0
5	0	21	0
6	0	22	0
7	0	23	0
8	0	24	0
9	0	25	0
10	0	26	0
11	0	27	0
12	0	28	0
13	0	29	0
14	0	30	0
15	0	31	0
16	0		

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COMPLETED BY: R. Knieriem
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SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS FOR THE SALEM UNIT 1 GENERATING STATION

The following items completed during **September 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**

1EA-1265, Pkg. 1, Feedwater Thermal Sleeve. This design change installs steam generator feedwater nozzle to piping transition pieces in the Salem Unit 1 replacement steam generators. The change resolves thermal fatigue concerns of steam generator feedwater inlet piping as reported by the NRC via I. E. Bulletin 79-13 and later by NRC Information Notice 93-20.

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.

1EA-1259, Pkg. 1, Steam Generator J-Nozzle Replacement This design change replaced the original carbon steel feed ring J-nozzles in the replacement Westinghouse F Steam Generators that were installed in Salem Unit 1 with J-nozzles made from inconel. The original J-nozzles in service at a number of plants have experienced wall thinning by erosion and corrosion. The original equipment manufacturer has issued technical bulletins on this problem and recommended the replacement.

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.

1EA-1248, Pkg. 1, Rebar Cut For Installation of Motion Restraint Structural and Runway Beams. This design change is a modification to the Containment Building that was necessary during the Salem Unit 1 Steam Generator replacement. The modification addresses the cutting of rebar in the base mat and in the annular slab that was necessary to accommodate the installation of temporary motion and restraint structures and runaway beams used to facilitate rigging and handling of the steam generators.

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.

1EA-1258, Pkg. 1, Addition of Steam Generator Secondary Side Inspection Openings. This design change modifies the Salem Unit 1 Model F replacement steam generators by adding four inspection openings in the shell and wrapper. The objective of the modification is to provide additional access openings for inspecting specific regions of the tube bundle and tube support plates and to facilitate future sludge lancing or chemical cleaning.

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.

1EE-0333, Pkg. 5, Replace Service Water Pump Strainer Backwash Valve 11SW24. This design change replaces the existing Service Water Pump Strainer Backwash Diaphragm Valve (11SW24) with a ball valve which has a longer service life between preventive maintenance and is suited for frequent cycling of the backwash service. The existing fast actuating air operator is being replaced with a Bettis piston type actuator to slow the closure time of the valve to mitigate existing water hammer in this portion of the system.

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.

1EC-3679, Pkg. 1, Safety Injection Discharge Pressure Rerating. This design change increased the setpoint for the Safety Injection Pump Discharge Relief valves from 1750 PSIG to 1910 PSIG to prevent leakage, accelerated seat pitting, and chattering that are caused by unseating of the relief valve during surveillance testing.

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.

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1EC-3472, Pkg. 1, #11 and #12 RHR Pump Recirculation Flow High Setpoint Change. This design change changes the RHR Pump Recirculation Flow high set point to eliminate inadvertent and unnecessary cycling of the RHR minimum flow valves thereby reducing the probability of valve failures.

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.

1EC-3620, Pkg. 1, Salem Unit 1 Letdown Isolation Bypass. This design change installs a Letdown Isolation Bypass circuit to provide operator control over Letdown Isolation and Letdown Orifice Isolation valves. This will allow operators to establish a controlled bleed path from the Reactor Coolant System during a low pressurizer level condition as required by Westinghouse Emergency Response Guidelines.

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.

1EC-3393, Pkg. 2, Room Cooler Drain Piping. This design change provides gravity flow condensate drain piping for the safety injection pump room cooler located in the Auxiliary Building to prevent water drippage on equipment.

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.

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1EC-3651, Pkg. 1, Modification to Station Air Compressor Controls. This design change improves compressor controls to eliminate cycling of the compressor and reduces air leakage by the addition of an orifice and flow transmitter to the 3rd stage compressor discharge piping and the addition of automatic drain valves to the inter and after cooler drain tanks.

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.

1EE-0027, Pkg. 1, Modification to Station Air Compressor Controls. This design change installs a turbine flowmeter and a digital flow totalizer on the #13 Charging pump packing seal tank overflow line to quantify leakage. This will allow the pump seal leakage to be reclassified as an identified leakage.

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.

Minor Modification S-96-032, Mason-Neilan Air Operator Retainer Clip

This design change installs a retaining clip on the Mason-Neilan Air Operator to prevent inadvertent handwheel engagement. This resolves a postulated failure of the handwheel shaft declutching mechanism on Containment Fan Coil Unit Service Water Air Operated Inlet and Outlet valves and the Turbine Lube Oil and Auxiliary Cooling Heat Exchangers Service Water Bypass valve.

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 September, 1997.

Procedures Summary of Safety Evaluations

Procedure NC.NA-AP.ZZ-0002(Q)/Revision 2, Nuclear Business Unit Organization

This change shifts responsibilities within the Salem Operations Department, revises the Salem Operations Organization succession of authority, and changes titles in both the Hope Creek and Salem Operations Departments.

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

UFSAR Change Notices Summary of Safety Evaluations

UFSAR Change Notice 96-13, Post Accident Sampling System (PASS)

This change reconciles the UFSAR with the current configuration of the PASS to reflect that the containment air sample lines do not require heat tracing to prevent iodine plateout and that the PASS sampling lines and components meet or exceed the quality group and seismic classification requirements that are consistent with NRC guidance on PASS.

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 Notice 97-117, Nuclear Operations Department Organization Change

This change shifts responsibilities for the Nuclear Security and the Loss Prevention Organizations within the Nuclear Operations Department. The change shifts the reporting relationship for these organizations from the General Manager - Nuclear Maintenance to the General Manager - Hope Creek Operations for the Loss Prevention Organization; and the General Manager - Salem Operations for the Nuclear Security Organization.

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.

Deficiency Reports   Summary of Safety Evaluations

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

Other   Summary of Safety Evaluation

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