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OCT 1 2 2000

PSEG Nuclear LLC

LRN-00-0380

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

Gentlemen:

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

In compliance with Section 6.9, Reporting Requirements for the Salem Unit 2 Technical Specifications, the operating statistics for **September 2000** are being forwarded. Also being forwarded, pursuant to the requirements of 10CFR50.59(b), is a summary of changes, tests, and experiments that were implemented in **September 2000.**

Sincerely,

D. F. Garchow
Acting Vice President - Operations

RBK
Attachments

C Distribution

IE24

INDEX

DOCKET NO.: 50-311
UNIT: Salem 2
DATE: 10/15/00
COMPLETED BY: R. Knieriem
TELEPHONE: (856) 339-1782

Reporting Period **September 2000**

OPERATING DATA REPORT

Design Electrical Rating (MWe-Net)	**1115**		
Maximum Dependable Capacity (MWe-Net)	**1106**		
	Month	**Year-to-date**	**Cumulative**
No. of hours reactor was critical	**720**	**6575**	**102680**
No. of hours generator was on line (service hours)	**720**	**6575**	**99357**
Unit reserve shutdown hours	**0**	**0**	**0**
Net Electrical Energy (MWH)	**760265**	**7091174**	**100013392**

UNIT SHUTDOWNS

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR (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/Scram
3 - Automatic Trip/Scram
4 - Continuation
5 - Other (Explain)

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Summary Of Monthly Operating Experience

- Salem Unit 2 began the month of September operating at full power.
- On September 18, Salem Unit 2 entered end-of-cycle power coastdown.
- Salem Unit 2 completed the month in coastdown at 90% power.

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

MONTH September 2000

The following items completed during **September 2000** 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

Service Water Column Separation Protection, 22 Nuclear Supply Header (2EC-3590, Pkg. 5)

This design change installed new 10" and 2" diameter piping and valves in the Service Water System to provide a tie-in to an accumulator vessel. This modification along with the other packages of this design change will eliminate the potential for water column separation and flashing in Service Water piping as a part of the resolution for concerns identified in NRC Generic Letter 96-06.

Review of this design change under 10CFR50.59 was required because the modification constitutes a change to the facility as described in the UFSAR. The

components installed by this change did not change the functionality of the Service Water System other than to address the concerns identified by NRC Generic Letter 96-06. Therefore, this design change would not increase the probability or consequences of an accident previously analyzed. Additionally, this change would not increase the probability or consequences of a malfunction of equipment important to safety. This change would not create any new accidents or malfunctions since no new failure modes were introduced. In addition the Technical Specification Bases were not affected and no changes to the Technical Specifications were required.

Modification 80008505, 4KV/125VDC Control Circuit Modification

This modification changed each vital 4KV circuit breaker 125VDC control power circuit breaker from a two pole molded case breaker to a three pole molded case breaker. This change was performed to address potential fire-induced electrical shorts and grounds that may cause spurious actuation of a 4KV breaker.

Review of this modification under 10CFR50.59 was required because replacement of the 4KV/125VDC breakers constituted a change to the facility as described in the UFSAR. This change will preclude the possibility of fire-induced shorts and grounds from spuriously operating any 4KV circuit breaker while bypassing local operator control. Therefore, this change would not increase the probability or consequences of an accident previously analyzed. Additionally, this change did not increase the probability or consequences of a malfunction of equipment important to safety. This change would not create any new accidents or malfunctions since no new failure modes were introduced. In addition the Technical Specification Bases were not affected and no changes to the Technical Specifications were required.

Temporary Modifications Summary of Safety Evaluations

There were no reportable changes in this category implemented during September 2000.

Procedures Summary of Safety Evaluations

Procedure SC.MD-CM.CAV-0006(Q), Rev 0, Temporary Blank Installation for Switchgear Penetration Area Ventilation System (SPAV) Exhaust Fans

This procedure was developed to support the installation of a blank flange to isolate a SPAV Exhaust Fan from the remainder of the system to facilitate maintenance to that fan. Installation of the blank flange will permit operation of the system while a fan is removed.

Review of this procedure under 10CFR50.59 was required because the installation of a blank flange to isolate a SPAV Exhaust Fan constitutes a change to the facility as described in the UFSAR, and a change to procedures described in the UFSAR. During the installation of the blank flange, temperatures of areas cooled by the SPAV system will be monitored so that corrective actions can be taken, if necessary, to prevent exceeding temperature limits in those areas. Therefore, this change would not increase the probability or consequences of an accident previously analyzed. Additionally, this change did not increase the probability or consequences of a malfunction of equipment important to safety. This change would not create any new accidents or malfunctions since no new failure modes were introduced. In addition the Technical Specification Bases were not affected and no changes to the Technical Specifications were required.

UFSAR Change Notices Summary of Safety Evaluations

UFSAR Change Notice SCN00-034, Engineering Evaluation S-C-CVC-NSE-0911 - Use of PRC-01 Media in Chemical Volume Control System Mixed Bed Demineralizers

This evaluation considered the use of PRC-01 media in the Chemical Volume Control System mixed bed demineralizers. PRC-01 media was developed to remove extremely fine CO^{58} particles as well as the soluble species transported in the Reactor Coolant System, especially when the unit is shutdown for refueling.

Review of this change under 10CFR50.59 was required because the change constitutes a change to the facility as described in the UFSAR and would change procedures as described in the UFSAR. The proposed change will incorporate a more efficient resin for removal of fine particulate material from the letdown stream. The use of PRC-01 will not affect the functionality of the Chemical Volume Control System demineralizers. Therefore, this change would not increase the probability or consequences of an accident previously analyzed. Additionally, this change would not increase the probability or consequences of a malfunction of equipment important to safety. This change would not create any new accidents or malfunctions since no new failure modes were introduced. In addition the Technical Specification Bases were not affected and no changes to the Technical Specifications were required.

Other - Summary of Safety Evaluations

Salem Unit 2 Cycle 12 Reload Safety Evaluation in All Modes (DS.8-0014), Revision 1

This Safety Evaluation considered the Salem Unit 2, Cycle 12 reload specific evaluation of safety parameters required to confirm the validity of the existing safety analysis. Revision 1 to this Safety Evaluation incorporated an alternate approach to addressing

fractured holddown spring screws on the top nozzle of irradiated fuel assemblies using Inconel Alloy 600 screws. Specifically, the Inconel Alloy 600 holddown screws have a potential to fracture during their second cycle of operation, or may potentially be fractured prior to reinsertion in the upcoming cycle. The alternate approach was developed by the fuel vendor and involves reuse of the assemblies without repair provided certain criteria are met. Although considered acceptable, this approach involves consideration under 10CFR50.59 as a degraded or non-conforming condition as described in Generic Letter 91-18.

Review of this change under 10CFR50.59 was required because the change warrants consideration to accept a degraded or non-conforming condition as described in Generic Letter 91-18. The proposed change will incorporate an alternate approach to address fractured top nozzle holddown screws to reuse without repair provided certain criteria are met. Therefore, this change would not increase the probability or consequences of an accident previously analyzed. Additionally, this change would not increase the probability or consequences of a malfunction of equipment important to safety. This change would not create any new accidents or malfunctions since no new failure modes were introduced. In addition the Technical Specification Bases were not affected and no changes to the Technical Specifications were required.

Change to Commitment Associated With NRC Generic Letter 88-17 to Close the Containment Equipment Hatch Prior to Operation With Reduced Reactor Coolant Inventory (Mid-loop Operation)

As a part of its response to NRC Generic Letter 88-17, PSEG Nuclear LLC committed to procedurally require that the Containment Equipment Hatch would remain closed during mid-loop operation unless the core would remain covered for at least four hours if Residual Heat Removal System flow is lost. PSEG Nuclear LLC further committed that any deviation from that position would be justified by a Safety Evaluation. This safety evaluation considered an alternative to that commitment. The alternative involves the use of a temporary Outage Equipment Hatch that can be closed to provide containment closure prior the onset of core boiling following a loss of Residual Heat Removal System flow during mid-loop operation.

This commitment change was reviewed under 10CFR50.59 to satisfy the commitment to do so in PSEG Nuclear LLC's response to NRC Generic Letter 88-17. The temporary Outage Equipment Hatch is fully capable of being closed to provide containment closure prior to core boiling in the event of a loss of Residual Heat Removal System flow during mid-loop operation. Establishment of containment closure prior to core boiling satisfies the requirement of NRC Generic Letter 88-17. Therefore, this change would not increase the probability or consequences of an accident previously analyzed. Additionally, this change did not increase the probability or consequences of a malfunction of equipment important to safety. This change would

not create any new accidents or malfunctions since no new failure modes were introduced. In addition the Technical Specification Bases were not affected and no changes to the Technical Specifications were required.