PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

APR 2 0 2001

LRN-01-0099

PSEG
Nuclear LLC

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

RESPONSE TO MARCH 16, 2001 REQUEST FOR ADDITIONAL INFORMATION IN REGARDS TO REQUEST FOR LICENSE AMENDMENT INCREASED LICENSED POWER LEVEL
SALEM GENERATING STATION, UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

On March 16, 2001, the NRC issued a request for additional information (RAI) to support the staff's review of the request for license amendment submitted by PSEG Nuclear LLC on November 10, 2000 requesting an increase in licensed power levels for Salem Generating Station Unit Nos. 1 and 2.

The response to the request for additional information is contained in Attachment 1. Should you have any questions regarding this request, please contact Mr. Brian Thomas at (856)339-2022.

Sincerely,

G. Salamon
Manager – Nuclear Safety and Licensing

Attachment

A001

C Mr. H. J. Miller, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. R. Fretz
Licensing Project Manager - Salem
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop O8B2
11555 Rockville Pike
Rockville, MD 20852

USNRC Senior Resident Inspector - Salem (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
P.O. Box 415
Trenton, NJ 08625

ATTACHMENT 1
SALEM GENERATING STATION
UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
INCREASED LICENSED POWER LEVEL

On March 16, 2001, the NRC issued a request for additional information (RAI) concerning PSEG Nuclear's request for amendment to increase the licensed power level for Salem Unit Nos. 1 and 2. This attachment provides the responses to the RAI questions.

NRC Question:

1. Section 4.1.5 in the November 10, 2000, Request for License Amendment states that the uprate will increase the decay heat that is transferred from the residual heat removal (RHR) system to the component cooling water system (CCWS) during accident or normal cooldown. It states that the uprate also increases the decay heat in the spent fuel pool (SFP) transferred by the SFP cooling system to CCWS. Will this additional heat load reflect itself as additional CCWS electrical demand on the safety-related power system?

PSEG Nuclear response to Q1

The increase in heat transferred to the component cooling water (CCW) system will not result in an additional demand on the safety related power systems that feed the CCW system. The increased heat loading does not change the design of the CCW system. Because the electrical supply system is designed to meet the design requirements of the CCW system, the increased heat loading will not increase the electrical demand on the safety-related power system.

NRC Question:

2. In Section 8.6 (500 kV Grid Stability), it stated that no stability issues were identified during a feasibility study performed in support of the proposed uprate. It is further stated that an impact study including stability analysis will be completed before implementation of the proposed change. Is PSEG requesting NRC approval of the uprate before the stability analysis is completed? Although we would expect that the change in 500 kV grid stability would be minimal for such a relatively small power increase, it is unclear how approval could be granted before the actual impact on grid stability is determined. Please explain.

PSEG Nuclear response to Q2

The review of the stability analysis by the PJM Interconnection LLC for the increase in power level has been completed. This review indicates that for one of the cases evaluated the operating parameters for PSEG Nuclear will require minor changes to the minimum MVAR limits due to the increase in power level. The Artificial Island Operating Guide (AIOG) which controls the MW and MVAR operating curves specified by the PJM Interconnection for Salem and Hope Creek will be revised to incorporate these changes as part of the implementation plan for the increased power level.

NRC Question:

3. How will the 1.4% power uprate affect the electrical transients associated with the loss of external load event? There is likely to be some additional generator overspeed as a result of the uprate. What will the effect be of the overspeed and associated overfrequency on the generator and the electrical loads connected to the Unit Auxiliary Power Transformers?

PSEG Nuclear response to Q3

During normal operation, Salem Units 1 and 2 are designed such that the non-safety related group busses are fed from the auxiliary power transformer when the generator is connected to the grid. Upon a trip of the generator breakers, loss of external load event, the group busses will fast transfer from the auxiliary power transformer to the station power transformer feed from the 500 KV grid. Therefore, any overfrequency of the generator as a result of the loss of external load will not impact the loads on the group busses. The main turbine overspeed mechanical trip is at 103% speed, with a backup electrical trip at 110% speed with both generator breakers open (loss of external load event).

NRC Question:

(4) In Section 6.1 of the submittal, Steam Generator Tube Rupture (SGTR) Evaluation, PSEG stated that the current licensing basis SGTR analysis was performed at 104.5% reactor power. It also stated that the proposed 1.4% increase in power will result in a decrease of steam pressure, and hence, an increase in break flow. In order to evaluate the impact the 1.4% power uprate will have on the SGTR event evaluation, please clarify the following:

   a. License Amendments 190 (Unit No. 1) and 173 (Unit No. 2) indicate a reactor power of 105.5% was used for the SGTR analysis. Please verify at what power level your current licensing basis is for the SGTR event analysis.

b. Revision 15 of the Final Safety Analysis Report (FSAR) states that the conservative upper limit for reactor coolant transferred to the steam generator secondary side is 125,000 lbs. In subsequent revisions, the stated mass transfer is 137,250 lbs. These subsequent revisions also state that the operators will take 50 minutes to isolate the steam generators. The mass release of 137,250 lbs. is then equated to a 55 minute operator action time for isolation. Since the original analysis of record is for 125,000 lbs., is the operator action time still bound by the original analysis of record? If not, what are the differences between the calculations? How were the mass/energy releases determined? What were the changes made, if any, to the operator actions?

To assist in answering the questions associated with question number (4), please provide a table indicating the specific initial conditions (including reactor power), assumptions, operator actions, and results of the steam generator tube rupture analysis for FSAR Revisions 15, 16, and 18 and Section 6.1 of the proposed power uprate. Clarify the bases for the specific assumptions, initial conditions, and operator actions that changed between the revisions and discuss how they relate to the proposed power uprate.

PSEG Nuclear response to Q4a & b

The NRC SER for Salem Unit 1 Amendment 190 and Salem Unit 2 Amendment 173 states that the NRC assessed the Steam Generator Tube Rupture (SGTR) event at a power level of 3600 MWt.

In reviewing the dose analysis information submitted to the NRC in support of Amendments 190/173, the initial coolant activity including a pre-accident Iodine spike, the accident initiated spike Iodine appearance rate and the secondary side activity were actually based on a thermal power level of 3558 MW. The rupture flow from the reactor coolant system (RCS) to the main steam system (MSS) via the faulted generator was calculated using an assumed power level of 3600 MWt. Because the Iodine appearance rate and the secondary side activity were based upon a lower power level, the current licensing basis SGTR event as a whole supports a power level of 3558 MWt.

The following table provides a summary of the changes from Revision 15 to Revision 18 of the Salem UFSAR.

Parameter	Revision 15	Revision 16	Revision 18
Power Level	3558	3558	3558
Operator Action Time	30 minutes	55 minutes	30 minutes
Mass Transfer from faulted Steam Generator	125,000 lbs	137,250 lbs	137,250 lbs

Revision 16 of the Salem UFSAR incorporated the dose analysis changes associated with the control room design changes that were approved as Amendments 190 (Unit 1) and Amendments 173 (Unit 2). In the PSEG dose analysis for this change, this analysis used a primary to secondary mass transfer in the faulted steam generator of 137,500 lbs as calculated in a reanalysis of the system thermal and hydraulics at the higher power level of 3600 MWt. A 30-minute operator action assumption was used in the conservative Westinghouse hand-calculation method for the mass transferred. An explicit transient calculation was performed by Westinghouse (in contrast with the conservative Westinghouse hand-calculation method) that showed that the SGTR would actually require 55 minutes to transfer the entire 137,250 lbs (predicted by the conservative hand-calculation method) of mass to the secondary side of the steam generator. Thus, it was determined that the operators need only meet a 55 minute action time to isolate the faulted steam generator to support the mass releases derived using the conservative hand-calculation method for mass transfer.

In Revision 18 of the Salem UFSAR, section 15.4.4.4 was revised to clarify the changes made in Revision 16 in regards to operator action time and the mass transferred from the faulted steam generator. Section 15.4.4.4 states that:

"...The current licensed method used to calculate the mass released from the faulted steam generator...has been shown to be conservative with respect to mass released over an assumed 30-minute operator action time. The amount of mass released, as predicted by the current licensed method, from the faulted steam generator over the 30-minute assumed operator action time is mush larger than expected mass release if the transient was to be modeled explicitly. An explicit modeling method was used to evaluate the equivalent amount of operator action time that would be available that yields an equivalent mass release to that calculated by using a 30-minute operator action time with the current licensed method. This time was found to be 55 minutes. Since the operator is able to isolate the faulted steam generator within 50 minutes from event initiation, the amount of mass released is not expected to exceed that calculated using a 30-minute isolation time with the current licensed method. Therefore, the 30-minute assumption used in the current licensed analysis for the time to isolate the faulted steam generator is conservative since it results in a bounding mass release calculation."

The current licensed method for calculating the mass releases is consistent with the original hand calculation method used by Westinghouse in the original SGTR analysis for the Salem facility (i.e., no change in methods has been implemented since the original SGTR analysis). The only changes that have been implemented are input assumptions such as power level.

Since the analysis for STGR supports a power level of 3558 MWt, this analysis bounds the change in licensed power level of 3459 MWt.