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PSEG
Nuclear LLC

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

Gentlemen:

RESPONSE TO APRIL 3, 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 April 3, 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.

Attachment 2 provides revised Technical Specification (TS) pages and associated bases pages for inoperable Main Steam Safety Valves (MSSVs). The original marked-up pages contained in the November 10, 2000 submittal were based on the approval of LCR S99-13 (submitted September 26, 2000) prior to the approval of the increased power level. Based on discussion with the staff, the increased power level request for amendment will be approved prior to LCR S99-13 therefore the pages associated with inoperable MSSVs are being revised. TS tables 3.7-1 and 3.7-2 currently require the power range neutron flux high setpoints to be reduced due to the number of inoperable MSSVs. The reduced trip setpoints are based on a percentage of rated thermal power. The tables are being revised to maintain the trip setpoints at the same thermal power level for inoperable MSSVs as exists in the current TS.

Should you have any questions regarding this request, please contact Mr. Brian Thomas at (856)339-2022.

Sincerely,

G. Salamon
Manager - Nucelar Safety and Licensing

Attachments (2)

A001

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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 April 3, 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. In order to verify that General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," will continue to be met following power uprate, please provide the following information:

   a. Tables 2-1 and 2-2 of Attachment 1 to the reference transmittal provide the nuclear steam supply system (NSSS) design parameters that are used as the basis for the 1.4% power uprate for Salem Units 1 and 2. Additionally, please provide the corresponding parameters that are used in the current Salem design basis analyses.

PSEG Nuclear response to Q1a

Table 2-1 in the original Salem uprate submittal provided the NSSS design parameter cases generated and used as the basis for the 1.4-percent uprate for the Unit 1 Model F steam generators (SGs). Table 2-2 provided the NSSS design parameter cases that were generated and used as the basis for the Unit 2 Model 51 SGs. The attached Table 2-1a covers the current power level design values for Unit 1 with the Model F SGs and Table 2-2a covers the current design values for Unit 2 with the Model 51 SGs. All four tables are included here for completeness.

The 1.4-percent uprate resulted in changes to some of the calculated NSSS design parameters, compared to the parameters that form the current licensing basis. The changes included the following RCS temperatures:

- T_{hot} increased by 0.5°F
- T_{cold} decreased by 0.5°F

These small changes occurred since the T_{avg} was maintained at the current design values (566.0°F and 577.9°F) while increasing the core power by 48 MWt to 3459 MWt. The temperature changes reflect the additional heat-up from the uprated core.

In addition, the 1.4-percent uprate resulted in the following changes to the secondary-side parameters:

Steam Temperature decreased by 0.8°F
Steam Pressure decreased by 6 psi
Steam Flow increased by 1.4 percent

These small changes occurred based on a calculation of the steam generator and secondary-side performance resulting from the increased core power. A greater steam flow and reduced saturated temperature/pressure conditions are needed to obtain the increased power.

Note the power level was the only design input parameter which was modified.

Table 2-1 NSSS Design Parameters for Salem Unit 1 1.4-Percent Uprating (Model F SGs)

OWNER UTILITY: Public Service Electric & Gas
PLANT NAME: Salem
UNIT NUMBER: 1
BASIC COMPONENTS

Reactor Vessel, ID, in.	173	Isolation Valves	No
Core		Number of Loops	4
Number of Assemblies	193	Steam Generator	
Rod Array	17x17(1)	Model	F(2)
Rod OD, in.	0.374	Shell Design Pressure, psia	1200
Number of Grids	12	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/No
Number of Control Rods, FL	53	Pump Motor, hp	6000
		Frequency, Hz	60

	1.4 % Uprating			
THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power, %	101.4	101.4	101.4	101.4
MWt	3471	3471	3471	3471
10^6 BTU/hr	11,844	11,844	11,844	11,844
Reactor Power, MWt	3459	3459	3459	3459
10^6 BTU/hr	11,803	11,803	11,803	11,803
Thermal Design Flow, Loop gpm	82,500	82,500	82,500	82,500
Reactor 10^6 lb/hr	127.3	127.3	125.3	125.3
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.2	7.2	7.2	7.2
Reactor Coolant Temperature, °F				
Core Outlet	606.7	606.7	617.9	617.9
Vessel Outlet	601.8	601.8	613.1	613.1
Core Average	570.3	570.3	582.4	582.4
Vessel Average	566.0	566.0	577.9	577.9
Vessel/Core Inlet	530.2	530.2	542.7	542.7
Steam Generator Outlet	530.0	530.0	542.5	542.5
Steam Generator				
Steam Temperature, °F	515.0	512.7	527.8	525.5
Steam Pressure, psia	778	762	869	852
Steam Flow, 10^6 lb/hr total	15.05	15.04	15.10	15.09
Feed Temperature, °F	432.8	432.8	432.8	432.8
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging, %	0	10	0	10
Zero Load Temperature, °F	547	547	547	547

HYDRAULIC DESIGN PARAMETERS	
Mechanical Design Flow, gpm	99,600
Minimum Measured Flow, gpm total	337,920

FOOTNOTES:
(1) Parameters incorporate 17x17 RFA w/IFMs and the protective bottom grid.
(2) Unit 1 has Model F SGs.

Table 2-1a NSSS Design Parameters for Salem Unit 1
Current Power Level Design Values (Model F SGs)

OWNER UTILITY: Public Service Electric & Gas
PLANT NAME: Salem
UNIT NUMBER: 1
BASIC COMPONENTS

Reactor Vessel, ID, in.	173	Isolation Valves	No
Core		Number of Loops	4
Number of Assemblies	193	Steam Generator	
Rod Array	17x17(1)	Model	F(2)
Rod OD, in.	0.374	Shell Design Pressure, psia	1200
Number of Grids	12	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/No
Number of Control Rods, FL	53	Pump Motor, hp	6000
		Frequency, Hz	60

	— Current Power Level Design Values —			
THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power, %	100	100	100	100
MWt	3423	3423	3423	3423
10^6 BTU/hr	11,680	11,680	11,680	11,680
Reactor Power, MWt	3411	3411	3411	3411
10^6 BTU/hr	11,639	11,639	11,639	11,639
Thermal Design Flow, Loop gpm	82,500	82,500	82,500	82,500
Reactor 10^6 lb/hr	127.2	127.2	125.2	125.2
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.2	7.2	7.2	7.2
Reactor Coolant Temperature, °F				
Core Outlet	606.2	606.2	617.4	617.4
Vessel Outlet	601.3	601.3	612.6	612.6
Core Average	570.3	570.3	582.3	582.3
Vessel Average	566.0	566.0	577.9	577.9
Vessel/Core Inlet	530.7	530.7	543.2	543.2
Steam Generator Outlet	530.4	530.4	542.9	542.9
Steam Generator				
Steam Temperature, °F	515.6	513.3	528.5	526.2
Steam Pressure, psia	782	767	874	857
Steam Flow, 10^6 lb/hr total	14.84	14.84	14.90	14.88
Feed Temperature, °F	432.8	432.8	432.8	432.8
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging, %	0	10	0	10
Zero Load Temperature, °F	547	547	547	547

HYDRAULIC DESIGN PARAMETERS	
Mechanical Design Flow, gpm	99,600
Minimum Measured Flow, gpm total	337,920

FOOTNOTES:
(1) Parameters incorporate 17x17 RFA w/IFMs and the protective bottom grid.
(2) Unit 1 has Model F SGs.

Table 2-2	NSSS Design Parameters for Salem Units 2 1.4-Percent Uprating (Model 51 SGs)			
OWNER UTILITY: Public Service Electric & Gas				
PLANT NAME: Salem				
UNIT NUMBER: 1 and 2				
BASIC COMPONENTS				
Reactor Vessel, ID, in.	173	Isolation Valves		No
Core		Number of Loops		4
Number of Assemblies	193	Steam Generator		
Rod Array	17x17(1)	Model		51(2)
Rod OD, in.	0.374	Shell Design Pressure, psia		1100
Number of Grids	12	Reactor Coolant Pump		
Active Fuel Length, in.	144	Model/Weir		93A/No
Number of Control Rods, FL	53	Pump Motor, hp		6000
		Frequency, Hz		60

	1.4% Uprating			
THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3	Case 4
NSSS Power, %	101.4	101.4	101.4	101.4
MWt	3471	3471	3471	3471
10^6 BTU/hr	11,844	11,844	11,844	11,844
Reactor Power, MWt	3459	3459	3459	3459
10^6 BTU/hr	11,803	11,803	11,803	11,803
Thermal Design Flow, Loop gpm	82,500	82,500	82,500	82,500
Reactor 10^6 lb/hr	127.3	127.3	125.3	125.3
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	7.2	7.2	7.2	7.2
Reactor Coolant Temperature, °F				
Core Outlet	606.7	606.7	617.9	617.9
Vessel Outlet	601.8	601.8	613.1	613.1
Core Average	570.3	570.3	582.4	582.4
Vessel Average	566.0	566.0	577.9	577.9
Vessel/Core Inlet	530.2	530.2	542.7	542.7
Steam Generator Outlet	530.0	530.0	542.5	542.5
Steam Generator				
Steam Temperature, °F	508.5	501.1	521.4	514.0
Steam Pressure, psia	735	687	822	771
Steam Flow, 10^6 lb/hr total	15.03	15.01	15.08	15.05
Feed Temperature, °F	432.8	432.8	432.8	432.8
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging, %	0	20	0	20
Zero Load Temperature, °F	547	547	547	547
HYDRAULIC DESIGN PARAMETERS				
Mechanical Design Flow, gpm	99,600			
Minimum Measured Flow, gpm total	337,920			

FOOTNOTES:

(1) Parameters incorporate 17x17 Robust Fuel Assembly (RFA) with Intermediate Flow Mixing Grids (IFMs) and the protective bottom grid.

(2) Unit 2 has Model 51 SGs.

Table 2-2a	NSSS Design Parameters for Salem Units 2 Current Power Level Design Values (Model 51 SGs)

OWNER UTILITY: Public Service Electric & Gas
PLANT NAME: Salem
UNIT NUMBER: 1 and 2
BASIC COMPONENTS

Reactor Vessel, ID, in.	173	Isolation Valves	No
Core		Number of Loops	4
Number of Assemblies	193	Steam Generator	
Rod Array	17x17(1)	Model	51(2)
Rod OD, in.	0.374	Shell Design Pressure, psia	1100
Number of Grids	12	Reactor Coolant Pump	
Active Fuel Length, in.	144	Model/Weir	93A/No
Number of Control Rods, FL	53	Pump Motor, hp	6000
		Frequency, Hz	60

	Current Power Level Design Values		
THERMAL DESIGN PARAMETERS	Case 1	Case 2	Case 3
NSSS Power, %	100	100	100
MWt	3423	3423	3423
10^6 BTU/hr	11,680	11,680	11,680
Reactor Power, MWt	3411	3411	3411
10^6 BTU/hr	11,639	11,639	11,639
Thermal Design Flow, Loop gpm	82,500	82,500	82,500
Reactor 10^6 lb/hr	125.2	125.2	125.2
Reactor Coolant Pressure, psia	2250	2250	2250
Core Bypass, %	7.2	7.2	7.2
Reactor Coolant Temperature, °F			
Core Outlet	617.4	617.4	606.2
Vessel Outlet	612.6	612.6	601.3
Core Average	582.3	582.3	570.3
Vessel Average	577.9	577.9	566.0
Vessel/Core Inlet	543.2	543.2	530.7
Steam Generator Outlet	542.9	542.9	530.4
Steam Generator			
Steam Temperature, °F	514.8	522.1	501.9
Steam Pressure, psia	777	828	693
Steam Flow, 10^6 lb/hr total	14.84	14.87	14.80
Feed Temperature, °F	432.8	432.8	432.8
Moisture, % max.	0.25	0.25	0.25
Tube Plugging, %	20	0	20
Zero Load Temperature, °F	547	547	547

HYDRAULIC DESIGN PARAMETERS	
Mechanical Design Flow, gpm	99,600
Minimum Measured Flow, gpm total	337,920

FOOTNOTES:
(1) Parameters incorporate 17x17 PERF+ fuel with Intermediate Flow Mixing Grids (IFMs) and the protective bottom grid.
(2) Unit 2 has Model 51 SGs.

NRC Question:

1b. In Sections 5.3.3, you evaluated the reactor internal components for the uprated power conditions including the baffle/barrel region components, core barrel, baffle plate, baffle/former bolts, and lower core plate. Provide a summary of analytical results including the maximum calculated stresses and cumulative usage factors (CUFs) for these components. Also provide the code and code edition used for evaluation of the reactor internal components. If different from the code of record, please justify and reconcile the differences.

PSEG Nuclear response to Q1b

Since the Salem reactor internals were designed prior to the introduction of Subsection NG of the ASME Boiler and Pressure Vessel Code Section III, a plant specific stress report is not required. However, the design of reactor internals is evaluated according to Westinghouse criteria which are similar to the criteria described in Subsection NG of the ASME code. (Hence the acceptance criteria are the same as used in the original design of the plant and its original licensing basis.)

Lower Core Plate

New CUF calculations were performed for the Power Uprate program for the Salem plant. The reason for the new CUF calculations was due to the increase in internal heat generation seen by the Lower Core Plate. The acceptance criteria used for this evaluation were in compliance with the Westinghouse design criteria.

Normal & Upset (Level A & B)

Stress Intensity = 47.6 ksi, allowable $3S_m$ = 49.2 ksi,

Faulted (Level D)

P_m, P_L and $P_m + P_b$ - Unaffected by the power uprating

Fatigue:

S_{alt} = 31.2 ksi, allowable number of cycles $6x10^5$.

Calculated additional Cumulative Usage Factor (CUF) due to the 1.4% uprating = $[(500)/(6x10^5)]$ = 0.0008 which is insignificant compared to the reported CUF of 0.76 calculated in accordance with the Westinghouse criteria and the Allowable CUF of 1.0.

Baffle-Barrel Region Components (Core Barrel, Baffle Plates, bolting and former plates)

No new CUF calculations were performed. The effect of heat generation rates seen by these components due to power uprate conditions remains bounded for the power uprate conditions at the Salem plant.

NRC Question:

1c. In reference to Section 5.6.1, you stated that an evaluation confirmed that the existing fatigue usage factors for the reactor coolant system (RCS) piping and nozzles remain bounding due to the conservative nature of the analysis (e.g., a conservative grouping of several more severe transients). Discuss your basis for the statement and conclusion. Provide a summary of the maximum calculated stresses and CUFs at the most critical locations for RCS piping, primary equipment supports and nozzles, RCS branch nozzles and pressurizer surge nozzles, allowable limits, the code of record and code edition used for the power uprate conditions for NSSS piping and supports. If different from the code of record, justify and reconcile the differences.

PSEG Nuclear response to Q1c

As a result of the 1.4% plant uprating for the Salem plant, the plant design operating parameters have changed slightly from the present licensed parameters. These include parameters, which are key for analysis of the NSSS design transients used for component fatigue analysis of the various NSSS components.

For the purposes of design transient evaluation, in 1989-1990, a 3800 MWt rerating program feasibility study was completed for Salem Units 1 and 2. The efforts associated with the 3800 MWt rerating program included development of the applicable design transients. The design transients associated with the 3800 MWt rerating program have continued to be the defining design transient set against which subsequent plant changes (i. e., the 1.4-percent uprating for the Salem Unit 2) have been evaluated, and have not required revisions. The affected parameters for both the 3800 MWt rerating program design transient set and for the 1.4% uprating are shown in Table 1-1.

In the 1989 rerating program, the existing design transients were reviewed for their continued applicability for the 3800 MWt uprating, and were revised where appropriate (i.e., where the existing design transients were no longer bounding for the uprating conditions). The 3800 MWt rerating program design transient parameter changes for Thot, Tcold, and Tsteam bound those that would result from the 1.4% uprating.

TABLE 1 -1

PARAMETER REVISIONS OF CONCERN TO DESIGN TRANSIENT ANALYSES

	1.4% Uprate		3800 MWT Rerating	
	Low Tavg	High Tavg	Low Tavg	High Tavg
NSSS Power, MWt	3471	3471	3800	3800
RCS flow (TDF), gpm/loop	82,500	82,500	85,000	85,000
Thot,F(reactor vessel outlet)	601.8	613.1	602.9	620.0
Tavg,F	566.0	577.9	569.4	587.4
Tcold, *F (S/G outlet)	530.0	542.5	526.8	545.9
Tsteam, *F (1)	501.1	514.0	499.5	519.3
No-load temperature, *F	547	547	547	547
Stm/FW flow,lb/hr total(1)	15.01×10^6	15.05×10^6	16.66×10^6	16.80×10^6
Feedwater temperature, *F	432.8	432.8	446	446

(1) Steam temperature and steam/feedwater flow values for limiting 20% plugging case for the 1.4% uprating

The design transients used as the basis for the analyses of the RCS piping and nozzles have not changed as a result of the 1.4-percent power uprate. No new analyses of the RCL nozzles and piping were performed for the 1.4-percent power uprate and therefore no new codes were used. The design basis LOCA hydraulic forcing functions bound the uprate conditions.

The Salem reactor coolant piping code of record is USAS B31.1, 1967. This piping code of record did not require performance of fatigue analyses. However, in response to Generic Letter 88-11, an ASME code structural evaluation of the pressurizer surge line and nozzles was performed. This evaluation was documented in WCAP-12914, Rev. 1, June 1992. In that evaluation the limiting fatigue usage factor occurs in the pressurizer surge line. The Salem Unit 2 limiting usage factor on either the Surge Line or RCL Surge nozzle is 0.6 and occurs in the pressurizer surge line. This value is considered still applicable for the 1.4-percent power uprate.

As stated in the November 10, 2000, Request for Amendment, the Salem Unit 1 RCS piping system was evaluated by reviewing the analyses performed for the Model F Steam Generator replacement, which was put into service in 1998. The results of this evaluation showed that the 1.4% power uprate was found to have negligible effect on the resultant loads. The coefficients of thermal expansion, allowable stresses, steam generator primary nozzle stresses still remain bounded by the current analyses. The summary of the analyses performed and the results obtained for the Model F SG's (stress vs. allowables and stress ratios) are documented in Salem UFSAR, Appendix 3B.

NRC Question:

2. In order to verify that the assumptions, analyses, and conclusions of Salem's programs associated with Generic Letter (GL) 89-10, GL 95-07, and GL 96-06 remain valid, please discuss the functionality of safety-related mechanical components (i.e., all safety-related valves and pumps, including air-operated valves (AOV) and power-operated relief valves) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in your GL 89-10 MOV program at Salem will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which functionality at the uprated power level could not be confirmed. Please discuss effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07, and on the evaluation of overpressurization of isolated piping segments for GL 96-06.

PSEG Nuclear response to Q2

Safety related system are designed and analyzed to mitigate the consequences of an accident or a transient, maintain containment integrity and provide long-term decay heat removal capability at 102% power level. Proposed 1.4% power uprate still remain bounded by the original design basis for the safety related SSC's. The Design parameters (pressure, temperature and flow) of the safety related systems are not impacted by the proposed power uprate. Therefore, the safety related components expected to perform as designed.

As discussed in Item 12.6 of the LCR submittal, safety related MOV's were reviewed for the uprate. There is no change in the limiting temperature, pressure, flow in any Emergency Safeguard System (ESF). The ESF systems design bound the proposed rerate since the calculations for these systems assumed (1) a 2% calorimetric error and (2) the ESF design rating. All Feedwater and Main Steam MOV calculations were based on the limiting condition (highest pressure differential), which occur at the no-load condition. Therefore, the proposed uprate does not impact the GL 89-10 program.

GL 96-06 addressed the overpressurization of isolated piping segments as a result of the environmental or internal heat sources. LOCA/MSLB analyses that affect piping segments inside containment or at containment penetrations have been performed at 102% power which bound the proposed 1.4% power uprate. Thus, the resultant environmental conditions remain bounding.

GL 95-07 addressed pressure locking (PL) and thermal binding (TB) of safety related power operated gate valves that are required to open to perform their intended safety function. At Salem stations the PL/TB concerns had been resolved either by drilling a hole in the disk or procedural changes in the operation of the valves to preclude susceptibility to this phenomena. For the proposed 1.4% power uprate the basis for the evaluations were re-reviewed. The results of the review confirmed that the process fluid temperatures and pressures and the accident environmental temperatures remain unchanged. Thus, the implemented resolutions for the susceptible valves are not affected by the proposed power uprate.

NRC Question:

3. In reference to Section 9, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. Provide a summary of the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Provide a summary of the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the code of record and code edition used for the power uprate conditions. If different from the code of record, justify and reconcile the differences.

PSEG Nuclear response to Q3

The BOP piping systems that are affected by the 1.4% power uprate were mainly the Turbine cycle systems. Systems, which had a slight change in their operating parameters, were: Main Steam, Bleed Steam, Condensate and Feedwater. To review the proposed power uprate's impact on these BOP systems a new heat balance was generated. The results of the heat balance were compared to the system piping and component design parameters. These systems were originally designed to operate at Turbine Valves Wide Open (VWO) heat balance parameters. Turbine VWO condition equates to about 104% reactor thermal power. Therefore, original piping and component design parameters (temperature, pressure and flow) remained bounding. Calculations of Record piping stress analyses were also reviewed. The input parameters (temperature and pressure) for the piping stress analyses, which used the original design values, remained also bounding. Thus, there was no new stress analysis runs. Since, the existing pipe stress analysis bounds the power uprate conditions, there are no changes to the loads on pipe supports, nozzles, penetrations, guides or anchorage for pipe supports. Therefore, there is no change to the code of record used for power uprate conditions.

Based on this, the design of the BOP systems bound the operating conditions expected as a result of the proposed 1.4% power uprate.

<u>NRC Question:</u>

4. Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate. Provide a summary of evaluation for power uprate effects on the high energy line break analysis, jet impingement, and pipe whip loads for the power uprate condition.

<u>PSEG Nuclear response to Q4</u>

Proposed power uprate does not affect the NSSS and cooling water systems since the flows in those systems are not changed. BOP system heat exchangers are mainly the Feedwater Heaters. They are designed to flow rates at Turbine Valves Wide Open conditions, which bound the flows expected as a result of the 1.4% power uprate. Therefore, flow induced vibration is not a concern as a result of the uprate.

The determination of which lines are subject to postulated High Energy Line Break (HEBA) is based on the line's temperature and pressure. Exclusion or inclusion is not based on flow rate in the line.

The rerate does not increase the design temperature and pressure in any line beyond original design conditions, and it does not increase the duration that lines are operated. Accordingly, the power uprate does not result in any changes to the high energy line break analysis, jet impingement or pipe whip loads.

Postulated break locations in these lines are based on (1) specified locations (terminal points) and (2) at high stress points. The rerate will not require any pipe stress reanalysis. Accordingly, the postulated pipe break locations will not change.

The mass and energy blowdown from an isolatable postulated break is based on the design parameters used for volume, temperature and pressure in the line. The mass energy used for the limiting RCS and MS line breaks assumed a core power of 3479 MWt (102% power) which bounds the proposed 1.4% power uprate.

Thus, since no new lines are added, no break locations added or changed, and no change to assumed blowdown from the postulated breaks, there is no impact on the HEBA analysis as a result of the proposed uprate.

SALEM GENERATING STATION
UNIT NOS. 1 AND 2
FACILITY OPERATING LICENSE DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
CHANGE TO FACILITY OPERATING LICENSES

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications pages for Facility Operating License No. DPR-70 are to replace the pages contained in Attachment 4 of LR-N00-0387, dated November 10, 2000. Please replace the pages in the November 10, 2000 submittal with the attached pages.

Technical Specification	Page
Table 3.7-1	3/4 7-2
Table 3.7-2	3/4 7-3
Bases 3/4.7.1.1	B 3/4 7-1

The following Technical Specifications pages for Facility Operating License No. DPR-75 are to replace the pages contained in Attachment 4 of LR-N00-0387, dated November 10, 2000. Please replace the pages in November 10, 2000 submittal with the attached pages.

Technical Specification	Page
Table 3.7-1	3/4 7-2
Bases 3/4.7.1.1	B 3/4 7-1

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	~~87~~ 85
2	~~64~~ 63
3	~~42~~ 41

TABLE 3.7-2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator*	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	~~60~~ 59
2	~~44~~ 43
3	~~29~~ 28

* At least two safety valves shall be OPERABLE on the non-operating steam generator.

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~16,655,268~~ 16.66 x 10^6 lbs/hr which is ~~115~~ 110.3 percent of the Maximum Calculated ~~total secondary~~ steam flow of ~~14,459,360~~ 15.10 x 10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (76)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	~~87~~ 85
2	~~64~~ 63
3	~~42~~ 41

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ~~16,655,268~~ 16.66×10^6 lbs/hr which is ~~115%~~ 110.4% of the ~~total secondary~~ Maximum calculated steam flow of ~~14,459,360~~ 15.08×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

For 3 loop operation

$$SP = \frac{(X) - (Y)(U)}{X} \times (76)$$

Where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line