April 21, 19

Docket No. 50-311

Mr. Steven E. Miltenberger
Vice President and Chief Nuclear
Officer
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. Miltenberger:

SUBJECT: EMERGENCY CORE COOLING SYSTEM SURVEILLANCE RELAXATIONS, SALEM NUCLEAR GENERATING STATION, UNIT 2 (TAC NO. M83267)

The Commission has issued the enclosed Amendment No. 118 to Facility Operating License No. DPR-75 for the Salem Nuclear Generating Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 24, 1992, and supplemented by letter dated February 2, 1993. Because of the requested implementation date for Salem, Unit 1 (restart from the refueling outage scheduled to begin on October 2, 1993), a separate amendment will be issued immediately before the scheduled refueling outage for Unit 1.

This amendment revises the emergency core cooling system (ECCS) surveillance test acceptance criteria, for ECCS flows, pump performance, and updates the bases section.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice. You are requested to notify the NRC, in writing, when the amendment has been implemented at Salem 2.

Sincerely,
/S/ A. Pelletier for
James C. Stone, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 118 to
License No. DPR-75
2. Safety Evaluation

cc w/enclosures:
See next page

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Docket File	MO'Brien(2)	CGrimes, 11E-21	JWhite, RGN-I
NRC & Local PDRs	JStone	SBrewer	APelletier
PDI-2 Reading	OGC	ACRS(10)	
SVarga	DHagan, 3206	OPA	
JCalvo	GHill(4), P1-22	OC/LFMB	
CMiller	Wanda Jones, P-370	EWenzinger, RGN-I	

OFC	:PDI-2/LA	:PDI-2/PM	:OGC	:PDI-2/D	:	:
NAME	:MO'Brien	:JStone	:	:CMiller	:	:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 21, 1993

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Sincerely,

Allison Pelletier for

James C. Stone, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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   License No. DPR-75
2. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 118
License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated April 24, 1992, and supplemented by letter dated February 2, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 118 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to restart from the seventh refueling outage, scheduled to end on June 2, 1993, or restoration of the design flow capabilities of the main steam safety valves, whichever is the latest.

FOR THE NUCLEAR REGULATORY COMMISSION

Charles L. Miller

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 21, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 118

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 5-6	3/4 5-6
3/4 5-6a	3/4 5-6a
B 3/4 5-2	B 3/4 5-2
B 3/4 5-3	B 3/4 5-3
B 3/4 6-2	B 3/4 6-2

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one safety injection pump or one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all safety injection pumps except the allowed OPERABLE safety injection pump to be inoperable below 312°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single POPS relief valve.

The surveillance requirements, which are provided to ensure the OPERABILITY of each component, ensure that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The safety analyses make the assumptions with respect to: 1) both the maximum and minimum total system resistance, and 2) both the maximum and minimum branch injection line resistance. These resistances, in conjunction with the ranges of potential pump performance, are used to calculate the maximum and minimum ECCS flow assumed in the safety analyses.

The maximum and minimum flow surveillance requirements in conjunction with the maximum and minimum pump performance curves ensures that the assumptions of total system resistance and the distribution of that system resistance among the various paths are met.

The maximum total pump flow surveillance requirements ensure the pump runout limits of 560 gpm for the centrifugal charging pumps and 675 gpm for the safety injection pumps are not exceeded.

The surveillance requirement for the maximum difference between the maximum and minimum individual injection line flows ensure that the minimum individual injection line resistance assumed for the spilling line following a LOCA is met.

SURVEILLANCE REQUIREMENTS (Continued)

f. By verifying that each of the following pumps develops the indicated Total Dynamic Head (TDH) when tested at the test flow point pursuant to Specification 4.0.5:

1. Centrifugal Charging pump ≥ 2338 psi TDH

2. Safety Injection pump ≥ 1369 psi TDH

3. Residual Heat Removal pump ≥ 165 psi TDH

g. By verifying the correct position of each of the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

HPSI System Valve Number	LPSI System Valve Number
21 SJ 16	21 SJ 138
22 SJ 16	22 SJ 138
23 SJ 16	23 SJ 138
24 SJ 16	24 SJ 138
	21 SJ 143
	22 SJ 143
	23 SJ 143
	24 SJ 143

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1. For Safety Injection pumps, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is ≥ 453 gpm, and

b) The total flow rate through all four injection lines is ≤ 647 gpm, and

c) The difference between any pair of injection line flow rates is ≤ 12.0 gpm, and

d) The total pump flow rate is ≤ 675 gpm.

SURVEILLANCE REQUIREMENTS (Continued)

2. For Centrifugal Charging pumps, with a single pump running:

   a) The sum of the injection line flow rates, excluding the highest flow rate, is ≥ 306 gpm, and

   b) The total flow rate through all four injection lines is ≤ 444 gpm, and

   c) The difference between any pair of injection line flow rates is ≤ 10.5 gpm, and

   d) The total pump flow rate is ≤ 560 gpm.

i. The automatic interlock function of the RHR System shall be verified within the seven (7) days prior to placing the RHR System in service for cooling of the Reactor Coolant System. This shall be done by verifying with a test signal corresponding to a reactor coolant pressure of 375 psig or greater, that the 2RH1 and 2RH2 valves cannot be opened.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as a part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA.

The limits on RWST minimum volume and boron concentrations ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, (2) the reactor will remain subcritical in the cold condition following a small LOCA assuming complete mixing of the RWST, RCS, and ECCS water volumes with all control rods inserted except the most reactive control assembly (ARI-1), and (3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area > 3.0 sq. ft.) assuming complete mixing of the RWST, RCS, and ECCS water and other sources of water that may eventually reside in the sump following a LOCA with all control rods assumed to be out (ARO). The limits on contained water volume and boron concentration also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig, and 2) the containment peak pressure does not exceed the design pressure of 47 psig during the limiting pipe break conditions. The pipe breaks considered are LOCA and steam line breaks.

The limit of 0.3 psig for initial positive containment pressure is consistent with the accident analyses initial conditions.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is ≤ 47 psig.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA or steam line break.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the design pressure. The visual inspections of the concrete and liner and the Type A leakage test are sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.118 TO FACILITY OPERATING LICENSE NO. DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

1.0 INTRODUCTION

By letter dated April 24, 1992, as supplemented by letter dated February 2, 1993, the Public Service Electric and Gas Company (PSE&G), Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) submitted a request for changes to the Salem Nuclear Generating Station, Unit No. 2, Technical Specifications (TSs). The requested changes apply to the Technical Specification (TS) Surveillance for the emergency core cooling system (ECCS). Specifically, the licensees proposed to change Surveillance Requirements 4.5.2.f and 4.5.2.h of TS 3/4.5.2, "ECCS Subsystems - $T_{avg} \geq 350°F$" and the associated Bases. The licensees are proposing changes to reduce the required minimum safety injection flows, increase the allowed maximum runout flows, and modify the acceptance criteria for ECCS pump performance. The licensees have requested these changes to add additional margin between the minimum and maximum pump flow requirements to facilitate testing of the ECCS subsystems. The February 2, 1993, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The licensees have proposed to express the pump discharge pressure in total dynamic head (TDH). This term is derived by subtracting the measured suction pressure from the pump discharge pressure and is a more accurate term than the discharge pressure. By making the proposed changes to the pressure requirements of TS surveillance 4.5.2.f, the licensees lower the minimum flow requirements and achieve greater operational flexibility.

The other portion of the proposal requests to change the minimum and maximum flow requirements in the flow balance test during shutdown. In the current surveillance requirement 4.5.2.h, a narrow band exists within which the flows must be adjusted. By increasing the acceptance criteria, the licensees intend to reduce maintenance expenditures and operational manipulations to achieve precise flows and also allow system resistance requirements and instrument inaccuracies to be directly applied to flow measurements.

Additional information, regarding the safety injection (SI) flow evaluation, was in a Westinghouse letter dated December 18, 1993, and provided by the licensee.

A change to the Bases, Sections B3/4.6.1.4 and B3/4.6.1.6, to substitute containment design pressure for the calculated peak containment pressure derived for the accident analysis was proposed. A change to Bases Section B3/5.6.1.5 to recognize that the initial containment air temperature was also applicable to the steamline break analysis was included.

2.0 EVALUATION

The licensees evaluated the effect of the proposed changes on the accident analysis in the Updated Final Safety Analysis Report (UFSAR). The staff's review of the applicable loss of coolant accident (LOCA), non-LOCA scenarios and the effect on pump performance are discussed below.

2.1 Non-LOCA Analyses

The non-LOCA accident analyses can be affected by the proposed change in one of two ways; either by the increase in the maximum flow limit or the decrease in the minimum flow. To verify that the proposed setpoints met the acceptance criteria, the licensees compared the proposed SI flows with existing SI flow curves from previous Salem reduction of SI flow analyses.

The licensees indicated that the only non-LOCA analysis impacted by the maximum safety injection performance is the spurious operation of the SI system with the reactor at full power. This event assumes the inadvertent actuation of the ECCS high pressure safety injection pumps during full power operation. During this event, all pumps are assumed to be available to deliver flow to the reactor coolant system (RCS), maximizing the pump performance. The results of the evaluation indicated that the proposed maximum SI flow rates remain lower than the previously evaluated flow rates and therefore, are bounded by existing analyses.

The non-LOCA analyses which were analyzed assuming minimum flow are: (1) steamline break (SLB) analysis to determine core response - i.e., the margin to departure from nucleate boiling (DNBR), (2) steamline break mass and energy release inside containment analysis to determine the containment pressure and temperature response, and (3) steamline break outside containment analysis for equipment qualification.

The minimum safety injection flows assumed in the previous evaluations are more conservative only for RCS pressures greater than 875 psia. Both the SLB Mass and Energy Inside Containment and Outside Containment analyses have minimum pressures that exceed 875 psia. The SLB Core Response analyses pressure does go below 875 psia, but causes only negligible changes in the calculated heat flux, pressure and core boron concentration from the current acceptable analyses.

The reduction in ECCS flow did not affect the DNBR margin in the analyses. Therefore, the licensee concluded that relaxation of the SI flow rates would have no impact on the previous analyses and the proposed TS changes are bounded by the existing analyses for all cases limited by minimum flow.

2.2 Steam Generator Tube Rupture (SGTR)

The maximum SI flow rates were assumed most limiting for the SGTR analysis because, maximum equilibrium flow rate maximizes the offsite radiological consequences. Therefore, in assessing the proposed flow rates, the licensees compared the revised maximum SI flow rates to the maximum SI flow rates used for the Salem SGTR analysis of record. The results indicated that in the applicable RCS pressure range, the revised maximum SI flows were less than the maximum SI flow rates used in the Salem SGTR analyses of record. Therefore, the licensees concluded that the offsite dose for a SGTR event would remain within the guidelines of 10 CFR Part 100. The staff finds the conclusion acceptable.

2.3 Small Break LOCA (SBLOCA) Analysis

Currently, the limiting break size for SBLOCA is 4 inches. Typically, a reduction in SI flow tends to reduce the limiting break size. The licensees performed an evaluation to determine if the 3-inch break size would become more limiting. The licensees concluded that the 4-inch break size remains most limiting by 11 °F.

The SBLOCA analysis of record, found in Section 15.3 of the Salem Unit 1 and 2 UFSAR, was performed using the 1975 WFLASH Westinghouse Small Break Evaluation Model. The peak cladding temperature (PCT) of 1465.3 °F, as reported in the USFAR, has had additional safety analysis and associated penalties of 262 °F added. The net resulting PCT, including penalties, is 1728 °F (1465.3 °F + 262.3 °F).

The licensees evaluated the new SI performance data with respect to the current SBLOCA analysis. The evaluation included the 4-inch break and the limiting single failure event, loss of one diesel generator with the loss of one SI train. The results indicated a degradation in the SI performance assumed in the original analysis and therefore the licensee assigned PCT penalties of 184 °F. The new PCT for the 4-inch break case is now 1912 °F (1728 + 184) °F. This value is below the regulatory limit of 2200 °F as stated in 10 CFR 50.46. The staff has reviewed the basis for the penalty and concludes that it conservatively bounds the effect of the flow reduction. Since the regulatory requirements of 10 CFR 50.46(b) remains satisfied, we find the evaluation acceptable.

The penalties associated with the SBLOCA analysis are significant, i.e. greater than 50 °F as defined by 10 CFR 50.46, therefore the licensees are required to propose a reanalysis schedule. The licensees have committed, in a letter dated July 31, 1992, to reanalyze the LOCA analyses using the NRC approved *NOTRUMP* SBLOCA analysis code. *NOTRUMP* was approved by the NRC in a safety evaluation report dated May 23, 1985. Since the adjusted WFLASH SBLOCA analysis meets the requirements of 10 CFR 50.46(b), and the *NOTRUMP* code will yield lower temperatures, the staff finds the proposed reanalysis schedule acceptable.

2.4 Large Break LOCA (LBLOCA)

The minimum SI case of record, the double ended cold leg guillotine break, was analyzed using the Salem LBLOCA BASH Evaluation Model and is presented in Section 15.4 of the UFSAR. The PCT reported is 2091 °F with a discharge coefficient (C_D) of 0.4. There have been previous PCT penalties applied to the UFSAR PCT yielding a net PCT of 2112 °F.

The limiting single failure is the loss of one low head safety injection (LHSI) pump for the LBLOCA Evaluation Model. For the Salem analysis, credit was taken for operation of the other LHSI pump, but the other SI pumps on that train were not credited. By crediting the second centrifugal charging pump (CCP), the net SI performance exceeds that assumed in the Salem analysis of record. Consequently, there was no impact to the Salem BASH analysis for all minimum SI cases.

For the maximum SI case, the licensees determined that the net CCP and intermediate head safety injection pump (IHSIP) flow had increased slightly and the LHSI remained unchanged. The PCT for the maximum SI case increased slightly but it did not exceed the existing minimum SI case. Therefore, the licensees have concluded that the minimum SI case remains limiting.

The reduction in ECCS flows can cause SI short falls. The licensees evaluated these short falls in their blowdown hydraulic forces analysis, post-LOCA long-term cooling subcriticality calculation, and the analysis for hot leg switch-over to prevent potential boron precipitation. In all three cases, the licensees concluded that the SI performance does not significantly affect the analysis.

2.5 Increase Allowed Maximum Emergency Core Cooling System (ECCS) Pump Runout Flow Rates

The licensees have stated that the pump runout flow rate can be increased without adverse effects on electrical loading or pump integrity. Cavitation and motor horsepower capability are the two major concerns which must be addressed when increasing the pump runout operating conditions. Cavitation will occur if the net positive suction head (NPSH) required by the pumps is not satisfied by the available NPSH at the increased runout flow rates. Also, the pump motors must be capable for operating satisfactorily at the increased runout flow rates which could require increased horsepower. An evaluation of pump performance was conducted by the pump vendor, Dresser Pump Division (Pacific Pumps). Based on this evaluation the licensees concluded:

1) The minimum TDH of 2338 psi for the CCPs and 1369 psi for the IHSIPs is within their design basis and does not represent a challenge to their operability.

2) The system provides sufficient NPSH to support operation of the pumps at the increased runout flow rates. The licensees provided the following specific information: for the CCPs at 560 gpm, the calculated available NPSH is 38 feet which is amply greater than the required NPSH of 24 feet, for the IHSIPs at 675 gpm, the calculated available NPSH is 27 feet which is sufficiently greater than the required NPSH of 23 feet.

3) The increased runout flows would have no effect on the long-term mechanical and hydraulic performance of the pumps.

4) Motor horsepower requirements at the increased pump runout flows is within the capability of the pump motors. Since the Salem CCPs and IHSIPs have falling head characteristics that cause the pump brake horsepower curves to become flat at high flow rates (the curve was extrapolated from the original runout flow rate for the pumps), the motor horsepower required to operate the pumps at the proposed runout limits does not exceed the horsepower required to operate the pumps at the original runout limit. The increased runout flows would therefore not cause an increase in the electrical power required to operate the pump assemblies and would not negatively impact the emergency diesel generator by increasing loads beyond their applicable capabilities and ratings. Furthermore, horsepower requirements at the increased flows remain within the rated limits of the motors (including service factor).

5) The pumps will not cavitate and the motors will not overheat during extended operation under the identified conditions.

The NRC staff concurs that the increased pump runout flow rates, from 550 to 560 gpm for the CCPs and from 650 to 675 gpm for the IHSIPs, will not challenge the operability of the pumps. The associated changes to the Salem TS are acceptable.

2.6 <u>Bases Change For Calculated Peak Containment Pressure</u>

The licensees have requested to change BASES 3/4.6.1.4 <u>INTERNAL PRESSURE</u>, 3/4.6.1.5 <u>AIR TEMPERATURE</u>, and 3/4.6.1.6 <u>CONTAINMENT STRUCTURAL INTEGRITY</u> as follows:

> 3/4.6.1.4, the calculated peak containment pressure is deleted and a statement is added that for limiting pipe breaks, LOCA and steamline breaks, the containment peak pressure does not exceed the design pressure of 47 psig.

3/4.6.1.5, the Bases is revised to state that the initial temperature limit assumed for accident analysis is applicable to the steamline break analysis as well as the LOCA analysis.

3/4.6.1.6, the maximum pressure calculated for a LOCA is replaced with the design pressure of the containment.

By letter dated August 11, 1992, the State of New Jersey commented that "The Technical Specification bases for the containment should include·calculated peak pressure plus an allowable margin." By letter dated February 2, 1993, the licensees responded to the State of New Jersey's comment. The licensees have stated that the design pressure was the limiting parameter and as long as calculated peak pressure remained below the design pressure the structural integrity of the containment is assured.

Appendix J, Section II.I, of 10 CFR Part 50, defines P_a as the calculated peak containment internal pressure related to the design basis accident and specified in the TS or associated bases. L_a is defined in 10 CFR Part 50, Appendix J, Section II.K, as the maximum allowed leakage at pressure P_a. For Salem, the licensees have chosen to perform leak rate tests of the containment at the design pressure of 47 psig. The use of the design pressure in the bases for containment internal pressure and containment structural integrity is consistent with the TS for containment leak rate testing and structural integrity testing. Therefore, the staff finds this change to be acceptable.

The addition of the steamline break to the initial containment air temperature Bases recognizes that the results of the steamline break analysis is also dependent on the initial temperature. The staff finds this change to be acceptable.

3.0 <u>CONCLUSION</u>

The staff has reviewed the licensees' submittal proposing to change the ECCS surveillance requirements. Their submittal discussed the impact of reduced SI flows on LBLOCA, SBLOCA, non-LOCA and SGTR accidents. The staff has concluded that the licensees have demonstrated, by approved methods, that the reduction in ECCS pressure and flow for TS surveillance requirements 4.5.2.f and 4.5.2.h, respectively, are acceptable. The licensees have demonstrated that Salem Unit 2 will continue to meet the requirements of 10 CFR 50.46 and LBLOCA remains limiting following implementation of the proposed changes. In a letter dated July 31, 1992, the licensees committed to reanalyze the SBLOCA event using *NOTRUMP*. Therefore, the staff finds the licensees' proposed changes acceptable.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments on the No Significant Hazards determination but had a technical comment. (See Section 2.6 of the safety evaluation for resolution.)

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 40219). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: S. Brewer
A. Pelletier
J. Stone

Date: April 21, 1993

REFERENCES

1. Letter from S. LaBruna, Public Service Electric and Gas Company, to the U.S. Nuclear Regulatory Commission, "License Amendment Application ECCS Surveillance Relaxations Salem Generating Station," dated April 24, 1992.

2. Letter from J. Huckabee, Westinghouse, to J. A. Ranalli, Public Service Electric and Gas Company, "Clarification of Reduced Safety Injection Evaluations," dated December 18, 1992.

3. Letter from S. LaBruna, Public Service Electric and Gas Company, to the U.S. Nuclear Regulatory Commission, "Annual 10 CFR 50.46 Report," dated July 31, 1992.