Docket File

Docket No. 50-272

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A. Schwencer

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

February 21, 1980

Docket No. 50-272

Mr. F. P. Librizzi, General Manager
Electric Production
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Public Service Electric and Gas Company
80 Park Place, Room 7221
Newark, New Jersey 07101

Dear Mr. Librizzi:

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-70 for the Salem Nuclear Generating Station, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your request dated June 29, 1978, as supplemented by letter dated September 27, 1979.

The amendment incorporates standard radiological safety technical specifications for operation and surveillance of a low temperature pressurizer overpressure protection system.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 24 to DPR-70
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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PUBLIC SERVICE ELECTRIC AND GAS COMPANY
PHILADELPHIA ELECTRIC COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. DPR-70

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

   A. The application for amendment by Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated June 29, 1978 as supplemented by letter dated September 27, 1979, 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;

   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-70 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 24 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 21, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 24

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

| Remove Pages | Insert Pages |
|---|---|
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| x | X |
| 3/4 4-3 | 3/4 4-3 |
| 3/4 4-30 | 3/4 4-30 |
| 3/4 4-31 | 3/4 4-31, 4-32 and 4-33 |
| 3/4 5-6 | 3/4 5-6 and 5-6a |
| B3/4 4-1 | B3/4 4-1 and 1a |
| B3/4 4-12 | B3/4 4-12 |
| B3/4 5-1 | B3/4 5-1 |

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ACTION (Continued)

Below P-7#:

a. With $K_{eff} \geq 1.0$, operation may proceed provided at least two reactor coolant loops and associated pumps are in operation.

b. With $K_{eff} < 1.0$, operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant or residual heat removal pump.*

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

## SURVEILLANCE REQUIREMENTS

4.4.1.1 With one reactor coolant loop and associated pump not in operation, at least once per 31 days determine that:

a. The applicable reactor trip system and/or ESF actuation system instrumentation channels specified in the ACTION statements above have been placed in their tripped conditions, and

b. If the P-8 interlock setpoint has been reset for 3 loop operation, its setpoint is $\leq$ 76% of RATED THERMAL POWER.

*All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

#A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than 312°F unless 1) the pressurizer water volume is less than 1600 cubic feet (equivalent to approximately 92% of level) or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

## SAFETY VALVES - SHUTDOWN

### LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG $\pm$ 1%.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

### SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

## REACTOR COOLANT SYSTEM

## PRESSURIZER

### LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

### SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two Pressurizer Overpressure Protection System relief valves (POPs) with a lift setting of less than or equal to 375 psig, or

b. A reactor coolant system vent of greater than or equal to 3.14 square inches.

APPLICABILITY: When the temperature of one or more the RCS cold legs is less than or equal to 312°F, except when the reactor vessel head is removed.

ACTION:

a. With one POPs inoperable, either restore the inoperable POPs to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both POPs have been restored to OPERABLE status.

b. With both POPs inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 8 hours; maintain the RCS in a vented conditin until both POPs have been restored to OPERABLE status.

c. In the event either the POPs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the POPs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each POPS shall be demonstrated OPERABLE by:

SURVEILLANCE REQUIREMENTS (Continued)

a. Performance of a CHANNEL FUNCTIONAL TEST on the POPS actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the POPS is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel at least once per 18 months.

c. Verifying the POPS isolation valve is open at least once per 72 hours when the POPS is being used for overpressure protection.

d. Testing in accordance with the inservice test requirements for ASME Category C valves pursuant to Specification 4.0.5.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vents(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

4.4.10.1.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads - The steam generator channel heads shall be ultrasonic inspected during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material. The stainless steel clad surfaces of the steam generator channel heads shall also be 100% visually inspected during the above outages and a television camera shall be used to make a videotape recording of the condition of this cladding. Each videotape shall be compared with those obtained during the previous outages to determine that the cladding does not show any abnormal degradation.

SURVEILLANCE REQUIREMENTS (Continued)

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 580 psig.

   2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

## ECCS SUBSYSTEMS - $T_{avg}$ < 350°F

### LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem# comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System $T_{avg}$ less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

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#A maximum of one safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 312°F.

## EMERGENCY CORE COOLING SYSTEMS

## ECCS SUBSYSTEMS - $T_{avg}$ < 350°F

### SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All safety injection pumps, except the OPERABLE pump allowed above, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 312°F by verifying that the motor circuit breakers have been removed from their electrical power supply circuits.

## 3/4.4 REACTOR COOLANT SYSTEM

### BASES

### 3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to < 36 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above 1.30. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (36 percent of RATED THERMAL POWER).

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a RHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more RCS cold legs less than or equal to 312°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting from the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

## 3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

## 3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

## 3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

## TABLE B 3/4.4-1 (Continued)

## REACTOR VESSEL TOUGHNESS

| COMPONENT | COMP CODE | MATERIAL TYPE | CU % | P % | NDTT F | 50 FT-LB/35 MIL TEMP F | | RTNDT F | MIN. UPPER SHELF FT-LB | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | LONG | TRANS | | LONG | TRANS |
| LOWER SHL. | 10DW | A533B1 | 19 | 011 | -40 | 45 | 77* | 17 | 138 | 90** |
| LOWER SHL. | 10DX | A533B1 | 19 | 012 | -70 | 58 | 89* | 29 | 124 | 81** |
| LOWER SHL. | 10DY | A533B1 | 19 | 010 | -40 | 46 | 93* | 33 | 124 | 81** |
| BOT.HD.SEG | 12DZ | A533B1 | 10 | 009 | 10 | 28 | 71* | 11 | 117 | 76** |
| BOT.HD.SEG | 12EA | A533B1 | 11 | 010 | -50 | 40 | 76* | 16 | 131 | 85** |
| BOT.HD.SEG | 12EB | A533B1 | 12 | 008 | 10 | 27 | 64* | 10 | 118 | 76** |
| BOT.HD.DOM | 13EC | A533B1 | 15 | 010 | -20 | 37 | 84* | 24 | 104 | 68** |
| WELD | 14ED | WELD | 16 | 019 | 0/$ | NA | -38 | 0 | NA | 97 |
| HAZ CORE | 15ED | HAZ | NA | NA | NA | -28 | NA | NA | 107 | NA |

/$ ESTIMATED (0 F OR 30FT-LB TEMP, WHICHEVER IS LESS)
* ESTIMATED (77FT-LB/54 MIL TEMP FOR LONGITUDINAL DATA)
** ESTIMATED (65 PER CENT OF LONGITUDINAL SHELF)

The OPERABILITY of two POPSs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more the RCS cold legs are less than or equal to 312°F. Either POPS has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a safety injection pump and its injection into a water solid RCS.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

BASES

### 3/4.5.1 ACCUMULATORS

The OPERABILITY of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

### 3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

The limitation for a maximum of one safety injection pump to be OPERABLE and the Surveillance Requirement to verify all safety injection pumps except the allowed OPERABLE 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.

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 Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained.

## 3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensure that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 21000 ppm boron.

## 3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as 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 concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATED TO AMENDMENT NO. 24 TO FACILITY OPERATING LICENSE NO. DPR-70

## PUBLIC SERVICE ELECTRIC AND GAS COMPANY, PHILADELPHIA ELECTRIC COMPANY, DELMARVA POWER AND LIGHT COMPANY, AND ATLANTIC CITY ELECTRIC COMPANY

## SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

## DOCKET NO. 50-272

### Introduction

By letter dated October 26, 1977 (Reference 5) Public Service Electric and Gas Company (PSEG) submitted to the NRC a plant specific analysis in support of the proposed reactor vessel overpressure mitigating system (OMS) for Salem Nuclear Plant Unit 1. This information supplements other documentation previously submitted by PSEG (References 2-4, 7).

The NRC staff has completed its review of all information submitted by PSEG in support of the proposed overpressure mitigating system and has found that the system provides adequate protection from overpressure transients. A detailed safety evaluation follows. PSEG, by Reference 8, has proposed for Salem Unit No. 1 the applicable sections of the Standard Technical Specifications developed for the Salem Unit No. 2 OMS and approved by the staff.

### Discussion

Over the last few years, pressure transient incidents have occurred in pressurized water reactors. As used in this report, "pressure transient" is an event during which the Technical Specification temperature/pressure limits of the reactor vessel are exceeded. All of these incidents occurred at relatively low temperature (less than 200°F) where the reactor vessel material toughness (resistance to brittle failure) is reduced.

The "Technical Report on Reactor Vessel Pressure Transients" in NUREG-0138 (Reference 6) summarizes the technical considerations relevant to this matter, discusses the safety concerns and existing safety margins of operating reactors, and describes the regulatory actions taken to resolve this issue by reducing the likelihood of future pressure transient incidents at operating reactors. A brief discussion is presented here.

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A. Vessel Characteristics

Reactor vessels are constructed of high quality steel made to rigid specifications, and fabricated and inspected in accordance with the time-proven rules of the ASME Boiler and Pressure Vessel Code. Steels used are particularly tough at reactor operating conditions. However, if subjected to high pressures at low temperatures, these steels are less tough and could possibly fail in a brittle manner. Accordingly, power reactors have always operated with restrictions on the pressure during cold startup and cold shutdown operations when the steel temperatures are relatively low.

At operating temperatures, the pressure allowed by Appendix G limits is in excess of the relief setpoint of currently installed pressurizer code safety valves. However, most operating PWRs did not have pressure relief devices to prevent pressure transients that would exceed Appendix G limits during cold conditions.

B. Regulatory Actions

By letter dated August 27, 1976 (Reference 1) the NRC requested that PSEG begin efforts to design and install plant systems to mitigate the consequences of pressure transients at low temperatures. It was also requested that operating procedures be examined and administrative changes be made to guard against initiating pressure transients. It was felt by the staff that proper administrative controls were required to assure safe operation for the period of time prior to installation of the proposed overpressure mitigating hardware.

PSEG responded (Reference 2) with preliminary information describing interim measures to prevent these pressure transients along with some discussion of proposed hardware. PSEG proposed to install hardware to provide a low pressure actuation setpoint on the existing pressurizer air operating relief valves.

PSEG participated as a member of a Westinghouse user's group which was formed to support the analysis effort required to verify the adequacy of the proposed system to prevent pressure transients. Using input data generated by the user's group, Westinghouse performed transient analyses (Reference 7) which are used as the basis for each plant specific analysis.

The NRC staff requested additional information concerning the proposed procedural and hardware changes. PSEG provided the required responses (References 3 and 4). Reference 5 transmitted the plant specific analysis for Salem Unit 1.

C. Design Criteria

Through this series of meetings and correspondence with PWR vendors and licensees, the NRC staff developed a set of criteria for an acceptable overpressure mitigating system. The basic criterion is that the mitigating system will prevent reactor vessel pressures in excess of these allowed by Appendix G. Specific criteria for system performance are:

(1) Operator Action: No credit can be taken for operator action for ten minutes after the operator is aware of a transient.

(2) Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

(3) Testability: The system must be testable on a periodic basis consistent with the system's employment.

(4) Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category I and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

The NRC staff also instructed the licensee to provide an alarm which monitors the position of the pressurizer relief valve isolation valves, along with the low setpoint enabling switch, to assure that the overpressure mitigating system is properly aligned for shutdown conditions.

D. Design Basis Events

The incidents that have occurred to date have been the result of operator errors or equipment failures. Two varieties of pressure transients can be identified: a mass input type from charging pumps, safety injection pumps, safety injection accumulators; and a heat addition type which causes thermal expansion from sources such as steam generators or decay heat.

On Westinghouse design plants, the most common cause of the pressure transients has been improper isolation of the letdown path. Letdown during low pressure operations is via a flowpath through the RHR system. Isolation of RHR with a charging pump running can initiate a pressure transient. Although other pressure transients occur with lower frequency, those which result in the most rapid pressure increases were identified by the NRC staff for analysis. The most limiting mass input transient we identified is inadvertent injection by the largest safety injection pump. The most limiting thermal expansion transient is the start of a reactor coolant pump with a 50°F temperature difference between the water in the reactor vessel and the water in the steam generator.

Based on the historical record of pressure transients and the imposition of more effective administrative controls, the NRC staff believes that the limiting events identified above form an acceptable bases for analysis of the proposed overpressure mitigating system.

## System Description and Evaluation

PSEG's overpressure protection system (pressurizer overpressure protection system - POPS) is similar to the "Reference Mitigating System" developed by Westinghouse and the user's group as described in Reference 2. PSEG has proposed to modify the actuation circuitry of the existing air operated pressurizer relief valves to provide a low pressure setpoint at 375 psig during startup and shutdown conditions. When the reactor vessel is at low temperatures, with the low pressure setpoint selected, a pressure transient is terminated below the Appendix G limit by automatic opening of these relief valves. A manual switch is used to enable and disable the low setpoint of each relief valve. An enabling alarm which monitors reactor coolant system pressure, the position of the enabling switch and the upstream isolation valve is provided. The POPS with its low pressure setpoint is enabled at a RCS temperature of 312°F during plant cooldown and is disabled at the same temperature during plant heatup. The NRC staff finds PSEG's POPS to be an acceptable concept for an overpressure mitigating system. Discussion and evaluation of this system follows.

### A. Air Supply

The Salem plant has two power operated relief valves (PORVs). They are gate valves that are spring closed and air opened. Each of the two PORVs receives actuating air from one of three available sources: two fully redundant control air headers and a backup air accumulator. The accumulators are sized to provide sufficient actuating air for up to 100 cycles of PORV opening and closing (about ten minutes of operation) should the normal air supplies both become unavailable. In the event of a control air header rupture or other low pressure condition, sensors act to reposition valves to isolate the affected portion of the system and automatically align a backup air supply. Pressure alarms are installed in the control room to alert the plant operators to a low air pressure condition. The staff finds the PORV actuating air supply system for Salem Unit 1 acceptable.

B. Electrical Controls

A Technical Evaluation Report that was prepared for us by the Lawrence Livermore Laboratory as part of our technical assistance program is attached to this Safety Evaluation as Appendix A.

The POPS design information detailed in this section was derived from Reference 5. The design for Salem Unit 1 POPS is a two-train system which uses separate and independent pressure transmitters to open the two pressurizer PORVs (1PRI and 1PR2) in the event that RCS pressure exceeds 375 psig. This automatic action takes place provided the system has been manually enabled by placing two key-locked pushbuttons in the "on" position.

Each PORV is actuated by its own logic relay which is energized by a bistable device. The bistable device is energized when the RCS pressure exceeds 375 psig. The existing installed pressure sensors are used to develop the signal for valve actuation. These are the same sensors which provide the signal for automatic closure of the residual heat removal (RHR) suction paths at 600 psig. We find this design acceptable.

C. Testability

Testability will be provided. PSEG has stated that verification of operability is possible prior to RCS low temperature operation by use of the remotely operated isolation valve, enable/disable switch and normal electronics surveillance methodology. Testing requirements have been proposed for incorporation in the Technical Specifications as discussed elsewhere in this evaluation.

Appendix G Curve

The Appendix G curve submitted by PSEG for purposes of pressure transient analysis is based on 13 effective full power years irradiation. The zero degree heatup curve is allowed since most pressure transients occur during isothermal metal conditions. Margins of 60 psig and 10°F are included for possible instrument errors. The Appendix G limit at 11°F according to this curve is 460 psig. The NRC staff finds that use of this curve is accetpable as a basis for determining proper POPS performance.

Setpoint Analysis

The one loop version of the LOFTRAN (Reference WCAP 79-07) code was used to perform the mass input analysis. The four loop version was used for the heat input analysis. Both versions require some input modeling and initialization changes were required. LOFTRAN is currently under review by the NRC staff and is judged to be an acceptable code for treating problems of this type.

The results of this analysis are provided in terms of PORV setpoint overshoot. The predicted maximum transient pressure is simply the sum of the overshoot magnitude and the setpoint magnitude. The PORV setpoint is adjusted so that given the setpoint overshoot, the resultant pressure is still below that allowed by Appendix G limits.

PSEG presented the following Salem Unit 1 plant characteristics to determine the pressure reached for the design basis pressure transients:

| | |
|---|---|
| SI Pump Flowrate @ 500 psig | 108.4 lb/sec |
| RCS Volume | 12,800 $ft^3$ |
| PORV Opening Time, Setpoint | 2 sec, 375 psig |
| SG Heat Transfer Area | 51,500 $ft^2$ |

Westinghouse identified certain assumptions used in LOFTRAN that are conservative and tend to overpredict the peak RCS pressure in the design base transients. These are listed below along with some plant parameters Westinghouse has assumed in the generic analysis that the licensees has identified to be conservative relative to the actual Salem Unit 1 values.

(1) One PORV was assumed to fail.

(2) The RCS was assumed to be rigid with respect to metal expansion.

(3) No credit was taken for the reduction in reactor coolant bulk modulus at RCS temperatures above 100°F (constant bulk modulus at all RCS temperatures).

(4) No credit was taken for the shrinkage effect caused by low temperature SI water added to higher temperature reactor coolant.

(5) The entire volume of water of the steam generator secondary was assumed available for heat transfer to the primary. In reality, the liquid immediately adjacent and above the tube bundle would be the primary source of energy in the transient.

(6) The overall steam generator heat transfer coefficient, $\mu$, was assumed to be the free convective heat transfer coefficient of the secondary $h_{sec}$. The forced convective heat transfer coefficient of the primary, $h_{pri}$ and the tube metal resistance has been ignored thus resulting in a conservative (high) coefficient.

(7) The RCP startup time assumed in the heat input analysis was 1.64 sec whereas the actual SI pump startup time is 5.0 sec.

The staff agrees that most of these assumptions are conservative. It is prudent is assume the failure of one PORV.

A. <u>Mass Input Case</u>

The inadvertent start of a safety injection pump with the plant in a cold shutdown condition was selected as the limiting mass input case.

Westinghouse provided PSEG with a series of curves based on the LOFTRAN analysis of a generic plant design which indicates PORV setpoint overshoot for this transient as a function of system volume, relief valve opening time and relief valve setpoint. These sensitivity analyses were then applied to the Salem Unit 1 plant parameters to obtain a conservative estimate of the PORV setpoint overshoot. The staff finds this method of analysis to be acceptable.

Using the Westinghouse methodology, the Salem Unit 1 PORV setpoint overshoot was determined to be 71 psi. With a relief valve setpoint of 375 psig, a final pressure of 446 psig is reached for the worst case mass input transient. Since the 13 EFPY Appendix G curve limit at temperatures above 100°F is 460 psig, we concluded that the system performance is acceptable with a 375 psig low pressure relief valve setpoint.

B. <u>Heat Input Case</u>

Inadvertent startup of an RCP with a reactor coolant to secondary coolant temperature differential across the steam generator of 50°F, and with the plant in a water solid condition, was selected as the limiting heat input case. For the heat input case, Westinghouse provided PSEG with a series of curves based on the LOFTRAN analysis of a generic plant design to determine the PORV setpoint overshoot as a function of RCS volume, steam generator UA and initial RCS temperature. For this transient, the reference PORV selected was assumed to have a total opening time of three seconds from the instant signal to open is received until the valve reached the full open position.

The calculated maximum RCS pressure for the heat input transient for a fixed $\Delta T$ of 50°F depends on the initial RCS temperature and is given here:

| Initial RCS Temperature | Maximum RCS Pressure |
|---|---|
| 100°F | 416.2 psig |
| 180°F | 442.2 psig |
| 250°F | 456.5 psig |

In none of these cases, is the Appendix G curve limit exceeded.

The NRC staff, therefore, finds that the analyses of both the limiting mass input case and the limiting heat input cases show maximum pressure transients below those allowed by Appendix G curve limit.

Implementation Schedule

The Salem Unit 1 POPS was installed and tested (pre-operational check-out) during the refueling outage ending in November 1977. The system is fully operational.

Operating Procedures

To supplement the hardware modifications and to limit the magnitude of postulated pressure transients to within the bounds of the analysis provided by PSEG a defense in depth approach is adopted using procedural and administrative controls.

A number of provisions for prevention of pressure transients are contained in the Salem Unit 1 operating procedures. The procedures for startup (and jogging) a reactor coolant pump require that a steam bubble is established in the pressurizer prior to pump start, or the SG/RCS $\Delta T$ be verified to be less than 50°F.

Also, shutdown procedures have been revised to include provisions for maintaining a steam bubble in the pressurizer during plant cooldowns. For conditions that do not require opening of the RCS a low pressure steam bubble will be maintained during the cold shutdown conditions. By following these procedures, PSEG does not anticipate that the RCS will be operated or maintained in a water solid condition except during RCS fill and vent procedures.

During plant cooldowns, the power to both safety injection pumps (SIPs) is removed by racking out the power supply breakers when the RCS temperature is below 350°F. Also, SI header isolation valves are shut and their power is removed. The SIPs are de-energized whenever the RCS temperature is below 312°F except when a special surveillance test is being conducted. During these procedures, only one SIP is energized; thus the POPS is able to keep RCS pressure below the Appendix G limit should an inadvertent mass addition from the single SIP occur during this procedure.

The staff finds that the procedural and administrative controls described are acceptable. However, the staff has determined that certain procedural and administrative controls should be included in the Technical Specifications. These are listed in the following section.

Technical Specifications

To assure operation of the overpressure mitigating system, the licensee has accepted for Salem Unit No. 1 the staff's format for Standard Technical Specifications and identical wording of the applicable sections of the Technical Specifications developed for Salem Unit No. 2 (Reference 8). These changes to the Technical Specifications are consistent with the intent of the statements listed below.

1. Both PORVs must be operable whenever the RCS temperature is less than the minimum pressurization temperature (312°F), except one PORV may be inoperable for seven days. If these conditions are not met, the reactor coolant system must be depressurized and vented to the atmosphere or to the pressurizer relief tank within eight hours.

2. Operability of POPS requires that the low pressure setpoint be selected, the upstream isolation valves open and the backup air supply charged.

3. No more than one high head SI pump may be energized at RCS temperatures below 312°F.

4. A reactor coolant pump may be started (or jogged) only if there is a steam bubble in the pressurizer, or if the SG/RCS $\Delta T$ is less than 50°F.

5. The POPS must be tested on a periodic basis consistent with the need for its use.

Summary

The administrative controls and hardware changes proposed by Public Service Electric and Gas Company provide protection for Salem Unit 1 from pressure transients at low temperatures by reducing the probability of initiation of a transient and by limiting the pressure of such a transient to below the limits set by Appendix G. The NRC staff finds that the overpressure mitigating system proposed by PSEG meets the criteria established by the NRC and is acceptable as a long term solution to the problem of pressure transients. Any future revisions of Appendix G limits for Salem Unit 1 must be considered and the overpressure mitigating system setpoint adjusted accordingly with corresponding adjustments in the license.

## Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

## Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: February 21, 1980

References

1. NRC (Kniel) letter to Public Service Electric and Gas Company, (PSEG) dated August 27, 1976.

2. PSEG (Librizzi) letter to NRC (Kniel) dated October 25, 1976.

3. PSEG (Librizzi) letter to NRC (Lear) dated March 25, 1977.

4. PSEG (Librizzi) letter to NRC (Lear) dated May 3, 1977.

5. PSEG (Librizzi) letter to NRC (Lear) dated October 26, 1977.

6. "Staff Discussion of Fifteen Technical Issues listed in Attachment G, November 3, 1976 Memorandum from Director NRR to NRR Staff", NUREG-0138, November 1976.

7. "Pressure Mitigating System Transient Analysis Results" prepared by Westinghouse for the Westinghouse user's group on reactor coolant system overpressurization, dated July 1977.

8. PSEG (Librizzi) letter to NRC (Schwencer) dated September 27, 1979.

APPENDIX A

SELECTED ISSUES PROGRAM

TECHNICAL EVALUATION OF THE
ELECTRICAL, INSTRUMENTATION, AND CONTROL DESIGN ASPECTS
OF THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM
FOR THE SALEM NUCLEAR POWER PLANT, UNIT 1

by

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This report documents the technical evaluation of the electrical, instrumentation, and control design aspects of the low temperature overpressure protection system for the Salem nuclear power plant, Unit 1. Design basis criteria used to evaluate the acceptability of the system include operator action, system testability, single failure criterion, and seismic Category I and IEEE Std-279-1971 criteria. This report is supplied as part of the Selected Electrical, Instrumentation, and Control Systems Issues Support Program being conducted for the U. S. Nuclear Regulatory Commission by Lawrence Livermore Laboratory.

TECHNICAL EVALUATION OF THE
ELECTRICAL, INSTRUMENTATION, AND CONTROL DESIGN ASPECTS
OF THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM
FOR THE SALEM NUCLEAR POWER PLANT, UNIT 1

## 1. INTRODUCTION

By letter to the Public Service Electric and Gas Company (PSEG) dated August 27, 1976, the U. S. Nuclear Regulatory Commission (NRC) requested an evaluation of system designs to determine susceptibility to overpressurization events and an analysis of these possible events, and proposed interim and permanent modifications to the systems and procedures to reduce the likelihood and consequences of such events. By letter dated October 25, 1976 and subsequent letters (refer to the Appendix), the Public Service Electric and Gas Company submitted the additional information requested by the NRC staff, including the administrative operating procedures and the proposed low temperature overpressure protection mitigating system. The system hardware includes sensors, actuating mechanisms, alarms, and valves to prevent a reactor coolant system transient from exceeding the pressure and temperature limits of the Technical Specifications for Salem Unit 1 as required by the Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), Appendix G.

The purpose of this report is to evaluate the Licensee's equipment and procedures based on the information provided (refer to the Appendix), and to define how well they meet the criteria established by NRC as necessary to prevent unacceptable overpressurization events.

## 2. EVALUATION OF SALEM UNIT 1

### 2.1 INTRODUCTION

Review of the Salem Unit 1 low temperature overpressure protection system design by PSEG was begun in 1976 at NRC's request. The overall approach to eliminating overpressure events incorporates administrative, procedural, and hardware controls, with reliance upon the plant operator as the principal line of defense. Preventive administrative/procedural measures include:

(1) Procedural precautions.

(2) Deenergization (power removed) of nonessential and essential components which are not required to be operable during the cold shutdown mode of operation.

(3) Maintenance of a non-water-solid reactor coolant system condition whenever possible.

(4) Incorporation of a low pressure relief setpoint for the existing power-operated relief valve (PORV) control logic.

The design basis criteria that were applied in evaluating the acceptability of the electrical, instrumentation, and control aspects of the low temperature overpressure protection system (OPS) are as follows:

(1) Operator Action. No assumption of operator action is made until ten minutes after the operator is aware, through an action alarm, that a pressure transient is in progress.

(2) Single Failure Criterion. The OPS shall be designed to protect the reactor vessel given a single failure which is in addition to the failure that initiated the pressure transient.

(3) System Testability. The OPS must be testable on a periodic basis prior to dependence on the OPS to perform its function.

(4) <u>Seismic Category I and IEEE Std-279-1971 Criteria.</u> The OPS should satisfy both the seismic Category I and IEEE Std-279-1971 criteria. The basic objective is that the OPS should not be vulnerable to a failure mode that would both initiate a pressure transient and disable the low temperature overpressure mitigating system. Events such as loss of instrument air and loss of offsite power must be considered.

## 2.2 PSEG PRESSURIZER OVERPRESSURE PROTECTION SYSTEM DESIGN

The PSEG Pressurizer Overpressure Protection System (POPS) design information detailed in this section was derived from Reference 5 in the Appendix. The PSEG design for the Salem Unit 1 POPS is a two-train overpressurization mitigating system which uses separate and independent pressure transmitters to open the two pressurizer PORV's (1PR1 and 1PR2) in the event that reactor coolant system (RCS) pressure exceeds the preset value of 375 psig. This automatic action takes place provided the system has been manually enabled by placing two keylocked pushbuttons in the "on" position. The system will be enabled whenever the RCS is below 312°F.

Each PORV is actuated by its own logic relay which is energized by a bistable device. The bistable device is energized when the RCS pressure exceeds the setpoint. Existing installed pressure sensors are used to develop the signal for valve actuation. These are the same sensors which provide automatic closure of the residual heat removal (RHR) suction paths at 600 psig.

Operation of the POPS is governed by two administratively controlled, keylocked pushbuttons which perform three functions, as follows:

(1) When the RCS temperature is less than 312°F, the system is armed by depressing the "on" pushbutton for each POPS train.

(2) If the temperature should subsequently increase above 312°F, an actuation signal to open the motor-operated valves (MOV's) upstream of the PORV's is initiated as well as an alarm to indicate that the POPS is armed. In this mode of operation, the PORV will be opened automatically if the RCS pressure exceeds 375 psig.

(3) When the RCS temperature increases above 312°F, the "off" pushbutton for each POPS train is depressed. This action removes the opening permissive signal to the PORV, removes the opening signal from the associated MOV, and provides an alarm to indicate that the

system is disarmed if the temperature is subsequently decreased below 312°F. Upon actuation, the valves will open and will reset when system pressure decreases below 375 psig.

RCS pressure and temperature instrumentation are provided which permit the operator to monitor the above parameters. An alarm is provided on the main control console to inform the operator of a POPS initiation. Valve position indicator lights inform the operator when the valves have opened. In addition, a computer-generated alarm informs the operator of an impending pressure excursion beyond the Technical Specification limits.

The POPS is designed as a "protection grade" system in accordance with the applicable portions of IEEE Std-279-1971. The use of proven devices provides assurance that the system is compatible with other protection system equipment. The use of administrative controls to arm the POPS is considered acceptable due to the infrequency of low-pressure, low-temperature operation.

The effects of various failures have been considered in the POPS design. These failures include loss of control air and loss of station power. Due to the two-train design, failures within the POPS cannot cause a loss of protective function. Failures capable of causing an overpressurization event cannot cause failures within the POPS or prevent operation of the system.

An air accumulator is provided for each PORV in case of a loss of control air failure. The accumulators are sized to provide enough control air for up to 100 cycles of PORV valve opening and closing. The accumulators are designed to seismic Category I requirements, and are provided with an alarm which will alert the operator to a low air pressure condition. The accumulator design thus precludes a total loss of control air to the PORV's. A loss of station power failure will not affect the POPS since protection logic power is provided by inverters, and control power for the PORV's originates at the batteries.

In the event that one PORV opens on a false signal or upon transmitter failure at a time when protection is not required, a depressurization of the RCS would occur. Any such depressurization would be less severe than those analyzed in the FSAR, Section 14.1.2. The discharge through the PORV can be terminated by operator action, thus minimizing the effects of the transient.

## 2.3 EVALUATION OF SALEM UNIT 1 USING DESIGN BASIS CRITERIA

Salem Unit 1 was evaluated under the guidance of the four design basis criteria stated in Section 2.1 of this evaluation, and with specific attention given to various pertinent NRC staff positions resulting from these criteria. Sections 2.3.1 through 2.3.4 are concerned with the four design basis criteria.

### 2.3.1 Operator Action

In each design basis transient analyzed, no credit for operator action was assumed until 10 minutes after the initiation of the RCS overpressurization transient and after the operator is made aware of the overpressure transient. Operator awareness of the overpressure transient will be derived by the low temperature overpressure transient alarm.

PSEG states in Reference 5 that the POPS requires no operator action other than to enable the system prior to operation when the RCS temperature is less than 312°F. All other protective action is then performed automatically.

### 2.3.2 Single Failure Criterion

PSEG states in Reference 5 that the POPS is designed to protect the reactor vessel given a single failure in addition to the failure that initiated the overpressure transient. Redundant or diverse pressure protection channels are used to satisfy the single failure criterion. The POPS incorporates redundancy and separation of pressure transmitters, logic, and valves in a channelized system. Single failures within the POPS will not defeat the safety function. In addition, single failures which are capable of initiating a pressure transient cannot cause failures within the POPS which would render it unable to provide protection.

We conclude that the PSEG Salem Unit 1 POPS satisfies the NRC staff single failure criterion.

### 2.3.3 System Testability

The NRC staff position requires that the POPS control circuitry from pressure sensor to valve solenoid shall be tested prior to each heatup and cooldown. The PORV's should be tested during each refueling. Deviations from these criteria should be justified.

PSEG states in Reference 5 that testing provisions in the POPS circuitry allow for test opening of the PORV's prior to arming of the system below 312°F. The "test" pushbutton, when depressed, will operate the PORV provided that the associated upstream MOV is closed. Other portions of the POPS can be tested in a manner similar to other plant protection systems. The POPS design provides for testing of the analog circuitry any time the RHR suction valves from the RCS are closed. The PORV's (1PR1 and 1PR2) can be tested prior to entry into a water-solid condition by use of the POPS "functional test" pushbutton. The POPS is designed to function during low-temperature low-pressure operating conditions and, therefore, periodic testing of the system during power operation is not planned.

The safety evaluation report (SER), dated December 1977, by the NRC Reactor Safety Branch/Division of Operating Reactors (RSB/DOR) for the Salem Unit 1 OMS states that:

(1) Testability will be provided.

(2) PSEG has stated that verification of operability is possible prior to RCS low temperature operation by use of the remotely operated isolation valve, enable/disable switch, and normal electronics surveillance methodology.

(3) Testing requirements will be incorporated in the Technical Specifications as discussed in Section 4.2 of this evaluation.

We conclude that the PSEG Salem Unit 1 POPS satisfies the NRC staff system testability criteria. It is recommended that the NRC staff ensure that thorough surveillance of the POPS from sensor to valve solenoid and proper PORV testing are adequately described in the PSEG Salem Unit 1 Technical Specifications.

### 2.3.4 Seismic Design and IEEE Std-279-1971 Criteria

PSEG states in Reference 5 that the POPS design meets seismic Category I criteria for all equipment required to open the PORV's, and that the instrumentation and actuating circuitry meet the applicable requirements of the IEEE Std-279-1971 criteria.

We conclude that the PSEG Salem Unit 1 POPS satisfies the NRC seismic design and IEEE Std-279-1971 criteria.

## 2.4 ALARM SYSTEMS DESIGNS AND OPERATION

Specific details concerning alarm systems designs and operation for the POPS are described below.

### 2.4.1 High-Pressure Alarm

The NRC staff position requires that a high-pressure audio/visual alarm shall be used during low RCS temperature operations as an effective means to provide unambiguous information and alert the operator that a pressure transient is in progress.

PSEG states in Reference 5 that the high-pressure alarm system design is as follows:

(1) The high-pressure alarm annunciates on the main control board when the RCS temperature is less than 312°F and the RCS pressure is greater than 360 psig.

(2) The annunciator provides both visible and audible signals.

(3) Operator action is required to acknowledge the alarm.

(4) In addition, a computer-generated alarm informs the operator of an impending pressure excursion beyond the Technical Specification limits.

We conclude that this design satisfies the NRC staff position.

### 2.4.2 Isolation Valve Alarm

The NRC staff position requires that

(1) The upstream isolation valve shall be wired into the overpressure protection alarm so that the alarm will not clear unless the system is enabled and the isolation valve is open.

(2) The alarm shall be of the audio/visual type and provide unambiguous information to the operator.

PSEG states in Reference 5 that the isolation valve alarm system design is as follows:

(1) The upstream PORV isolation valves (1PR6 and 1PR7) are wired into the RCS POPS in such a way that hand-switch activation of the POPS will result in the opening of the isolation valves.

(2) An open-close indicator for each isolation valve is provided on the main control board.

We conclude that this design does fully satisfy IEEE Std-279-1971(4.20) and the NRC staff position.

### 2.4.3 Enable Alarm

The NRC staff position requires that

(1) An alarm shall be activated as part of the plant cooldown process to ensure that the PORV "low" setpoint is activated before the RCS temperature is equal to or less than 312°F.

(2) The alarm shall be of the audio/visual type and provide unambiguous information to the operator.

PSEG states in Reference 5 that the enable alarm system design is as follows:

(1) A PORV "low" reset alarm is activated when the RCS temperature is equal to or less than 312°F, and ensures that the PORV "low" setpoint is activated.

(2) Once the PORV's are reset to the "low" relief position, an annunciator window will remain lit to indicate the "low" PORV mode of operation. The annunciator will remain in this mode until the PORV's are reset to the "high" position.

(3) After the PORV's are set to the "low" position, the overpressure transient alarm becomes operational only at RCS temperature below 312°F. When the PORV's are reset to provide low temperature relief at 375 psig, plant cooldown can be resumed.

We conclude that this design satisfies the NRC staff position.

2.4.4 Disable Alarm

The NRC staff position requires that

(1) An alarm shall be activated as part of the plant heatup process to ensure that the PORV's are reset to the "high" setpoint when the RCS temperature is greater than 312°F.

(2) The alarm shall be of the audio/visual type and provide unambiguous information to the operataor.

PSEG states in Reference 5 that the disable alarm system design is as follows:

(1) During the plant heatup, normal operating procedures will maintain the RCS pressure below 375 psig until the RCS temperature exceeds 312°F. When the RCS temperature exceeds 312°F, normal operating procedures require that the PORV's are reset to the "high" setpoint.

(2) At the same time, the overpressure transient alarm will be deenergized when the RCS temperature exceeds 312°F. In order to ensure that the PORV's are reset to the "high" setpoint, an alarm will be activated

when the RCS pressure exceeds 375 psig. After the PORV's are reset to the "high" setpoint, normal heatup will continue accordingly.

We conclude that this design satisfies the NRC staff position.

2.4.5 PORV Open Alarm

The NRC staff position requires that an audio/visual alarm shall be activated to provide unambiguous information and alert the operator that a PORV is in the "open" position.

PSEG states in Reference 5 that the PORV open alarm system design is as follows:

> The pressurizer PORV's have open/shut indicators on the main control board.

We conclude that this design does fully satisfy IEEE Std-279-1971(4.20) and the NRC staff position.

## 2.5 PRESSURE TRANSIENT REPORTING AND RECORDING REQUIREMENTS

The NRC staff position is that a pressure transient which causes the POPS to function, thereby indicating the occurrence of a serious pressure transient, is a 30-day reportable event. In addition, pressure-recording and temperature-recording instrumentation are required to provide a permanent record of the pressure transient. The response time of the pressure/temperature recorders shall be compatible with pressure transients that increase at a rate of approximately 100 psig per second.

PSEG states in Reference 2 that four 0°F-to-700°F temperature recorders are installed in the control room to verify compliance with the 10 CFR 50, Appendix G pressure-temperature limits during startup, shutdown, or periods of cold shutdown. The recorders monitor the hot-leg and cold-leg temperatures on each of the four loops. A pressure recorder and two pressure indicators are also installed in the control room to monitor the hot-leg pressure. These instruments are kept in service during all modes of operation.

We conclude that this implementation, if properly incorporated in the PSEG Salem Unit 1 Technical Specifications, satisfies the NRC staff position.

## 2.6 DISABLING OF ESSENTIAL COMPONENTS NOT REQUIRED DURING COLD SHUTDOWN

The NRC staff position requires the deenergizing of safety injection system (SIS) pumps and the closure of safety injection (SI) header/ discharge valves during cold shutdown operations.

PSEG states in References 3 and 5 that the disabling of essential components not required during cold shutdown is as follows:

(1) During plant cooldown, the power to both SI pumps is removed by racking out the power supply breakers when the RCS temperature is below 350°F. Also, SI header isolation valves are shut and their power is removed.

(2) The SI pumps are deenergized whenever the RCS temperature is below 312°F except when a special surveillance test is being conducted. During these procedures, only one SI pump is energized. This allows PCPS to maintain the RCS pressure below the 10 CFR 50, Appendix G limit in case an inadvertent mass addition from the single SI pump occur during this procedure.

We conclude that this implementation, if properly incorporated in the PSEG Salem Unit 1 Technical Specifications, satisfies the NRC staff position.

## 3. TECHNICAL SPECIFICATIONS

The Technical Specifications information detailed in this section was derived from the RSB/DOR SER entitled, "Safety Evaluation Report of the Overpressure Mitigating System for Salem Nuclear Plant Unit 1", dated December 1977.

To ensure operation of the POPS, the Licensee is to submit for NRC staff review its Technical Specifications for incorporation into the license for Salem Unit 1. The Licensee should ensure that the proposed Technical Specifications are compatible with other Licensee requirements and are consistent with the intent of the statements listed below:

(1) Both PORV's must be operable whenever the RCS temperature is less than the minimum pressurization temperature (312°F); however, one PORV may be inoperable for seven days and still meet the single failure criterion. If these conditions cannot be met, the primary system must be depressurized and vented to the atmosphere or to the pressurizer relief tank within eight hours.

(2) Operability of the POPS requires that the low-pressure setpoint will be selected, the upstream isolation valves opened, and the backup air supply charged.

(3) No more than one high-heat SI pump may be energized at RCS temperatures below 312°F.

(4) A reactor coolant pump may be started or jogged only if there is a steam bubble in the pressurizer, or if the SG/RCS $\Delta T$ is less than 50°F.

(5) The POPS must be tested on a periodic basis consistent with the need for its use.

(6) Failure of the POPS to operate when required is a reportable event.

## 4. CONCLUSIONS

The electrical, instrumentation, and control (EI&C) design aspects of the low temperature pressurizer overpressure protection system (POPS) for Salem Unit 1 were evaluated using those design criteria originally prescribed by the NRC staff and later expanded during subsequent discussions with the Licensee.

We recommend that the NRC staff find the following EI&C aspects of the PSEG Salem Unit 1 POPS design acceptable:

(1) Operator action

(2) Single failure criterion

(3) Seismic Category I and IEEE-279-1971

(4) High pressure alarm

(5) Enable alarm

(6) Disable alarm.

## REFERENCES


1. NRC (Kniel) letter to PSEG (Librizzi) dated August 27, 1976.
2. PSEG (Librizzi) letter to NRC (Kniel) dated October 25, 1976.
3. PSEG (Librizzi) letter to NRC (Lear) dated March 25, 1977.
4. PSEG (Librizzi) letter to NRC (Lear) dated May 3, 1977.
5. PSEG (Librizzi) letter to NRC (Lear) dated October 25, 1977.
6. "Staff Discussion of Fifteen Technical Issues Listed in Attachment G, November 3, 1976 Memorandum from Director NRR to NRR Staff," NUREG-0138, November 1976.
7. "Pressure Mitigating System Transient Analysis Results" prepared by Westinghouse for the Westinghouse User's Group on Reactor Coolant System Overpressurization, July 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-272

PUBLIC SERVICE ELECTRIC AND GAS COMPANY,
PHILADELPHIA ELECTRIC COMPANY,
DELMARVA POWER AND LIGHT COMPANY, AND
ATLANTIC CITY ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 24 to Facility Operating License No. DPR-70, issued to Public Service Electric and Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees), which revised Technical Specifications for operation of the Salem Nuclear Generating Station, Unit No. 1 (the facility) located in Salem County, New Jersey. The amendment is effective as of the date of issuance.

The amendment incorporates Standard Radiological Technical Specifications governing operation and surveillance of the low temperature pressurizer overpressure protection system.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated June 29, 1978 as supplemented by letter dated September 27, 1979, (2) Amendment No. 24 to License No. DPR-70, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Salem Free Public Library, 112 West Broadway, Salem, New Jersey. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 21 day of February, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors