Public Service
Electric and Gas
Company

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APR 28 1997

LR-N970273
LCR-S97-03

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

SUPPLEMENT TO REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
PRESSURIZER POWER OPERATED RELIEF VALVES
SALEM GENERATING STATION NOS. 1 AND 2
FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311

Gentlemen:

By letters dated January 31, March 14, and April 8, 1997 (LR-N97055, LR-N97144, and LR-N970221 respectively), Public Service Electric & Gas Company (PSE&G) submitted a request to amend Salem Generating Station (SGS) Units 1 and 2 Technical Specifications (TSs) 3.4.3 and 3.4.5 pertaining to the pressurizer power operated relief valves (PORVs). Attachment 1 to this letter contains supplemental information to the January 31, 1997 letter relative to available PORV stroke to mitigate an inadvertent safety injection, and to the March 14, 1997 letter regarding the PORV circuitry upgrade.

PSE&G has reviewed the previously submitted evaluation and has concluded that as a result of this supplement the conclusion of no significant hazards consideration remains valid.

Should you have any questions regarding this supplement, we will be pleased to discuss them with you.

Sincerely,

Louis F. Storz

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APR 28 1997

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REF: LR-970273
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STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

L. F. Storz, being duly sworn according to law deposes and says:

I am Senior Vice President - Nuclear Operations of Public Service Electric and Gas Company, and as such, I find the matters set forth in the above referenced letter, concerning Salem Generating Station, Units 1 and 2, are true to the best of my knowledge, information and belief.

Louis F Storz

Subscribed and Sworn to before me
this 28th day of April, 1997

Deloris D. Hadden
Notary Public of New Jersey

DELORIS D. HADDEN
Notary Public of New Jersey
My Commission Expires
03-29-2000

My Commission expires on

95-4933

ATTACHMENT 1

REF: LR-N970273

PORV CYCLING

In the January 31, 1997 letter, PSE&G stated (on page 7) that each PORV has the capability to fully stroke open and closed 305 times and could partially stroke open (up to 50% of full stroke) an additional 486 times, for a total of 791 strokes.

In March, 1997, as a result of NRC review and comments on the PORV circuitry, PSE&G requested a third party (Westinghouse) to review the Salem Unit 2 Evaluation of the Pressurizer PORVs for Inadvertent Safety Injection. The third party review, as well as additional work by PSE&G, identified that the evaluation had failed to account for the additional volume of tubing downstream of the PORV accumulator solenoid valve and the volume of the PORV diaphragm actuator. Accounting for the tube volume and the current PORV actuator excess active air volume of 5/8 inches (nominal), the PORV accumulator would have sufficient air for approximately 350 strokes (145 full + 205 partial).

As stated in the January 31, 1997 letter (in terms of valve Cv) the relieving capacity of a PORV at fifty percent stroke is essentially the same as the full stroke. The new calculated value (full and partial strokes) represents a reduction of strokes over what PSE&G stated in the January 31, 1997 letter. However, the ability of the PORVs to mitigate an Inadvertent SI event is not compromised by the results of the revised number of total strokes. PSE&G is considering field adjustment of the 5/8 inch nominal PORV excessive active air volume to increase the total number of available strokes, thus gaining additional design margin.

PORV CIRCUIT MODIFICATIONS

As stated in the March 14, 1997 letter, PSE&G has taken additional actions to eliminate single failure vulnerabilities in the PORV circuitry and upgrade the circuitry to qualify the PORVs as safety related. The modifications consisted of: 1) elimination of the non-safety related controller, PC455K, from the valve control circuitry and relocation of the PORV circuitry into the protection racks from the control racks, and 2) separation of the channels such that failure of any one channel or power supply will render, at most, one PORV inoperable. As a result of these changes the circuitry, the valves and associated controls will be safety related, redundant and independent.

The above modification does not physically alter the pressurizer, and utilizes the existing four safety related pressurizer pressure sensing channels (PT-455 [CH-I], PT-456 [CH II], PT-457 [CH-III], and PT-474 [CH-IV]). These are the same safety related pressurizer pressure sensing channels that provide input for the Reactor Protection System and Engineered Safety Feature Actuation System. Two of the Class 1E pressure sensing channels, CH-I and CH-IV, share a single pressurizer pressure tap. This mechanical configuration, sharing a single pressurizer pressure tap, is not changed by this modification and was not described in the March 14, 1997 letter. However, this configuration is considered acceptable and in compliance with IEEE 279-1971 Section 4.2 "Single Failure Criterion."

The basis for acceptability of the existing mechanical configuration relative to the single failure criterion is provided below.

The Salem Generating Station (SGS)is committed to IEEE 279-1971 as stated in Section 7.1 of the UFSAR. IEEE 379-1972 "IEEE Trial-Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems" was issued to provide interim guidance on the application of the single failure criterion for nuclear power plants protection systems. Regulatory Guide (RG) 1.53 "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems," was issued to describe an acceptable method of complying with IEEE 379-1972 in order to assess compliance with IEEE 279-1971 Section 4.2 "Single Failure Criterion." These documents were issued approximately four and five years following the issuance of Salem Units 1 and 2 construction permits in September 1968, and were not part of the Salem original design. UFSAR Appendix 3A contains the Salem Generating Station position regarding RG 1.53. The PSE&G position is stated, in part, "...in general, meets the requirements of the regulatory guide. The Salem Station

Protection system is designed in accordance with the requirements of IEEE Standard 279-1971 which requires that any single failure within the Protection System shall not prevent proper protective action at the system level. The design of the Protection System included the application of the single failure criterion to the logic and actuators." The PSE&G RG 1.53 position contained in UFSAR Appendix 3A does not include the postulation of single failures in sensing lines.

The safety related instrument sensing lines are considered fluid systems in the SGS design in accordance with ANSI B31.1, and as such the portion of the instrument sensing line consisting of piping, including the root valve, is part of the Nuclear Class I, seismic Class I Reactor Coolant Pressure Boundary (RCPB). The instrument tubing downstream of the root valve is also part of the pressure boundary and is designated as Nuclear Class II, seismic Class I. These safety-related instrument sensing lines are protected from consequential damage resulting from postulated initiating events including high energy line breaks, jet impingement, missiles and pipe whip. Therefore, these pressure boundary safety related instrument sensing lines are considered passive fluid system components. As such, the failure of these sensing lines is either an initiating event (break), or not postulated to occur until the recirculation phase of the accident. Failures involving blockage (including valve disc separation from its stem) are also considered passive failures and are not postulated to occur until the recirculation phase of the accident.

Failures involving partial loss of pressure boundary integrity (limited leakage) of these sensing lines are highly unlikely because of their design, as described above. However, if a limited leakage failure of these safety related instrument sensing lines were to occur, it would be readily detectable allowing the operator to take appropriate actions. This limited leakage failure can be detected by 1) the performance of the technical specification required channel check by the operators, and 2) by the instrument bistables tripping, or trouble alarms associated with the depressurization of the sensing line. Additionally, relatively low leakage rates can be detected by the technical specification reactor coolant system leakage detection system. This system is comprised of the containment atmospheric particulate monitoring system, the containment sump level monitors, and either the containment fan cooler condensate flow rate or the containment atmosphere gaseous radioactivity monitoring system. The system is capable of detecting leakage rates as low as 1 gpm. This system was designed consistent with the recommendations of RG 1.45 "Reactor Coolant Pressure Boundary Leakage Detection."

Failures whose leakage rate is such that inventory can not be maintained by normal charging and result in a Safety Injection (SI) are not Condition II events. Loss of inventory that can not maintain an operational water level in the pressurizer are Condition III events as described in UFSAR Section 15.3. Emergency Operating Procedures are in place to guide the operators in mitigating the consequences of a small break LOCA event, including the reduction of safety injection flow if the system repressurizes due to the SI make up flow exceeding break flow. Condition III events do not require automatic PORV operation.

In conclusion, the SGS does not extend RG 1.53 and IEEE-379 to sensing lines, and the existing mechanical configuration is in compliance with the Salem licensing basis. PSE&G reviewed of the previously submitted evaluation concluded that as a result of this supplement the conclusion of no significant hazards consideration remains valid.