U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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LICENSEE: Public Service Electric & Gas Company

FACILITY: Salem Generating Stations
Hancocks Bridge, New Jersey

DATES: December 5-19, 1994
February 13-24, 1995
March 14-15, 1995

INSPECTOR: Brian J. McDermott, Reactor Engineer — Date 3/24/95
Systems Section
Division of Reactor Safety

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SUMMARY: PSE&G worked to resolve nonconservatisms in the Pressurizer Overpressure Protection System (POPS) setpoint calculations for approximately two years (March 1993 through February 1995). In the process, PSE&G relied on an exemption from the requirements of 10 CFR 50.60 without NRC approval, failed to report a condition outside their plants' design-bases, and revised the POPS design-basis transient (described in the FSAR and Technical Specification Bases) without performing a safety evaluation pursuant to 10 CFR 50.59. During the inspection, engineering personnel stated that from the time the issue was identified in March 1993 they considered its safety significance to be low and that the plant was adequately protected. However, the POPS issue was not entered into an appropriate system for evaluating operability, safety significance, or reportability for over a year while options to assuage the problem were explored. After the issue was entered in an appropriate system, the design basis for POPS was changed and the evaluation required by 10 CFR 50.59 for identification of a possible unreviewed safety question, was not performed. Several apparent violations of NRC requirements identified during this inspection are being considered for enforcement action. The licensee's calculations for the revised POPS design-basis transient have been referred to the NRC Office Of Nuclear Reactor Regulation for review and pending their evaluation, this aspect of the issue will be unresolved.

DETAILS

1.0 INSPECTION SCOPE

This inspection evaluated the response of Public Service Electric and Gas (PSE&G) to information regarding the nonconservatisms identified in the technical specification setpoints for the pressurizer overpressure protection system (POPS) at both Salem units. The nonconservatisms in the original Westinghouse setpoint methodology were communicated to PSE&G in a letter from the vendor dated March 15, 1993. NRC Information Notice 93-58, "Nonconservatism in Low-Temperature Overpressure Protection for Pressurized-Water Reactors," dated July 26, 1993, reiterated the problems identified by Westinghouse.

2.0 FINDINGS

Background

The POPS uses two pressurizer power-operated relief valves (PORVs) to mitigate low temperature (<312°F) overpressure transients, keeping the peak pressure below the limits of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," for brittle fracture protection. The Appendix G limits are incorporated in technical specifications (TS) as pressure-temperature (P/T) curves specific to each unit's reactor vessel. The original design-basis mass addition transient for the POPS was based on the start of a safety injection pump (780 gpm) and its injection into a water solid reactor coolant system (RCS). POPS was designed to meet the single failure criterion, with either PORV having sufficient relief capacity to limit the peak pressure to less than the P/T curve limit.

An NRC safety evaluation report, dated February 21, 1980, associated with Amendment No. 24 to the Unit 1 TS, approved the Salem POPS setpoint of 375 pounds per square inch gage (psig), based on the calculated peak transient pressure of 446 psig and a 14 psi margin (at that time) below the Unit 1 Appendix G limit of 460 psig. Requirements for the Unit 2 POPS were incorporated into the unit's TS prior to initial startup and were approved based on the Unit 1 POPS safety evaluation.

The P/T limits for all reactor vessels decrease with successive operating cycles due to irradiation effects on the vessel materials. Therefore, margin between the peak transient pressure and the P/T limit will change as subsequent revisions of P/T curves are reviewed and approved by the NRC. The Salem Unit 1 P/T curves were revised in February 1990 in TS Amendment No. 108, which established a more restrictive limit of 450 psig at low temperatures. The Unit 2 P/T curves were approved (at the same time) in TS Amendment No. 86, which established a limit of 475 psig. These curves are valid for up to 15 effective full power years of operation.

Setpoint Nonconservatism

On March 15, 1993, Westinghouse issued a Nuclear Safety Advisory Letter (NSAL-93-005B) informing PSE&G about the nonconservatisms in the setpoint methodology for POPS. The dynamic head, resulting from running reactor coolant pumps (RCPs) and the static head, due to elevation of sensors relative

to the reactor vessel midplane, were found not to have been considered in the original setpoint methodology. The static head error for Salem is relatively small, resulting in a 4.7 psi increase in the peak transient pressure. However, the dynamic head error is more significant. Each operating RCP will increase the difference between pressure at the reactor vessel midplane and that sensed by the POPS instrumentation by approximately 25 psi. Consequently, for a four-loop plant such as Salem, the sensed pressure (with all four RCPs running) could be as much as 100 psi less than the actual pressure at the reactor vessel midplane (the area of concern for P/T curves). These errors simply can be added to the original peak transient pressure since their effect is to offset (nonconservatively) the pressure at which POPS will actuate. NRC Information Notice (IN) 93-58, "Nonconservatism in Low-Temperature Overpressure Protection for Pressurized-Water Reactors," was issued on July 26, 1994. The IN noted that administrative restrictions, recommended by the Westinghouse NSAL, were intended to provide interim actions until either setpoints were verified to be accurate, or appropriately revised in TS.

In December 1993, after reevaluating (over a nine month period) the original POPS analysis to address the NSAL concerns, PSE&G determined that the corrected peak transient pressure would exceed the P/T limits of both units. Even with limiting the number of running RCPs to two, the corrected peak pressure would be 485 psig (applicable for either unit since the analyzed transient is the same). On December 30, 1993, the licensee dispositioned the issue by memorandum (MEC-93-917), administratively limiting the maximum number of RCPs in service to two when RCS temperature was below 200°F (limiting the dynamic error in the most restrictive area of the P/T curve), and increasing each unit's P/T limit by 10% using an unapproved American Society of Mechanical Engineers (ASME) Code Case N-514. The inspector noted that, at temperatures above 200°F and up to 312°F, the Appendix G P/T curves allow for much higher pressure limits.

The inspector considered that - at the point PSE&G became aware that the margins to TS P/T limits for Appendix G brittle fracture considerations were not only reduced but, in fact, lost (and the Appendix G limits could be potentially exceeded) - both Salem Units could be potentially operated in an unanalyzed condition (whenever below 312°F) which would be outside the plants' design bases. Therefore, the condition was reportable, and the licensee's failure to make such a report is an apparent violation of the reporting requirements of 10 CFR 50.72 and 73. **(EEI 50-272;311/94-32-01)** Further, the inspector noted that an exemption request for use of ASME Code Case N-514 had not been submitted by PSE&G until late December 1994. Use of the ASME code case would require preapproval by the NRC, either generically via regulatory guide or specifically for Salem by exemption from 10 CFR 50.60. The licensee's reliance on the then unapproved ASME Code Case N-514 for over one year without the required exemption is an apparent violation of 10 CFR 50.60. **(EEI 50-272;311/94-32-02)**

Less than one month after the issued had been dispositioned in Memorandum MEC-93-917, the licensee recognized that the ASME code case could not be used without prior NRC approval. The licensee then sought to credit the capacity of the residual heat removal (RHR) suction relief valve RH3 to augment the

analyzed POPS relief capacity. The spring-operated relief valve (RH3) has the same setpoint as POPS, but has a greater effective flow area and will actuate faster than a PORV once its setpoint is reached. A subsequent analysis by PSE&G confirmed the licensee's initial judgement that, with RH3 available, the peak pressure would remain below the Appendix G limit. The issue of crediting RH3 as part of POPS (without either a 50.59 safety evaluation or prior NRC approval by changing the POPS TS) was under consideration from mid-January through mid-April 1994. On April 19, 1994, a Discrepancy Evaluation Form (DEF 94-0060) was written to document the fact that relief valve RH3 was not credited the original POPS analysis for Salem or in the existing licensing and design basis (the NRC safety evaluation) for the system. The inspector noted that at this point, PSE&G had attempted to resolve the issue for over a year without entering the fundamental engineering question (the adequacy of the POPS setpoint) in either of the two existing PSE&G quality systems for resolution of Salem engineering discrepancies (the Incident Report System or DEF process). The inspector concluded that the licensee's failure to initiate corrective actions for this significant condition adverse to quality, is an apparent violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." **(EEI 50-272;311/94-32-03)**

Corrective Action Process Initiated

The April 1994 DEF addressed the immediate safety concern by assuring the availability of relief valve RH3 and considering the safety margins discussed in the ASME code case. At this time the licensee also initiated a procedure revision to limit the number of running RCPs in Mode 5 (below 200°F) to one pump, thus further minimizing the dynamic head error in the most restrictive region of the P/T curves. Since the licensee concluded that there was no immediate operability concern, they sought to find other reasons why the Westinghouse nonconservatism did not apply to Salem. The inspector noted that, even after the issue was entered into the PSE&G DEF process, the condition outside the design basis was still not reported to the NRC.

The inspector independently assessed the availability and capability of relief valve RH3 for supplementing the POPS. Valve RH3 is available for RCS pressure relief when the RHR system is aligned for shutdown cooling. Review of Salem integrated operating procedures IOP-2, "Cold Shutdown To Hot Standby," and IOP-6, "Hot Standby To Cold Shutdown" showed that RHR shutdown cooling will be in service when POPS is required to be operable (<312°F). One reason valve RH3 was not credited by the NRC in the original 1980 POPS analysis was that an automatic closure interlock would shut the RHR suction valve on high RCS pressure, isolating RH3 from the RCS. However, this interlock was removed from both Salem units in the late 1980's. This change was generically reviewed by the NRC under Westinghouse Topical Report WCAP-11736, "Residual Heat Removal System Autoclosure Interlock Removal Report," and was subsequently approved by the NRC in a safety evaluation, dated August 8, 1989. The inspector also reviewed the valve's relief capacity, actuation response time, and calibration schedule. The inspector concluded that valve RH3 would be available to supplement POPS based on the procedural

requirements and would substantially reduce the peak transient pressure based on its design. However, crediting valve RH3 as part of POPS would, in the inspector's estimation, require a change to the Salem Technical Specifications.

The issue was once again closed (by memorandum, dated 5/26/94), based on a procedural requirement to achieve a pressurizer "bubble" (saturated conditions with a steam space) before starting a RCP. Because of this requirement, it was reasoned that only a correction for static head was necessary (relatively a small effect); therefore, the original analysis was concluded by PSE&G to be still valid. The inspector noted that the procedural requirement to have a pressurizer "bubble" before starting any RCPs was in place, and had been previously reviewed in the 1980 NRC safety evaluation report for POPS. Although the DEF was closed by PSE&G via this memorandum, further analyses to support license changes for (crediting valve RH3 and using the ASME code case) were continued in anticipation of future, more restrictive revisions to the P/T curves. During this analysis, the licensee determined that the effects of a running RCP on the POPS analysis should also be considered. However, no formal 50.59 safety evaluation had as yet been performed.

Revised Design-Basis Transient

On September 27, 1994, Problem Report (PR) No. 940927126 was initiated after the licensee determined that they could not rely on the establishment of a pressurizer "bubble" to resolve the problem. Since the original POPS analysis would not provide acceptable results after the effects of running RCPs were considered, engineering personnel established what they considered a more "realistic" transient as the design basis event for POPS.

The original transient was simply the start of a safety injection pump (the intermediate head pump delivers 780 gpm) and its injection into a water-solid RCS. The licensee's revised transient is mechanistic and relies upon procedural controls for limiting possible injection sources. The revised transient begins with the reactor in Mode 5 (<200°F), the positive displacement (PD) charging pump in service and one RCP running, whereafter an inadvertent safety injection (SI) signal would cause the centrifugal charging pump (high head SI at 560 gpm) to start, the PD charging pump to trip, and the isolation of letdown to the chemical and volume control system. Evaluation of this transient (mitigated by a single PORV having a 375 psig setpoint) using the GOTHIC computer code resulted in a predicted peak pressure of 438 psig, below the P/T limits of each unit. Therefore, by limiting the magnitude of the mass addition, the licensee was able to reduce the predicted peak transient pressure and justify the existing TS setpoint for POPS.

By the end of September 1994, the licensee believed they had reached a final resolution and closed the issue (for the third time in nine months) because the revised transient could be mitigated by the original POPS hardware with the existing 375 psig TS setpoint. Although the licensee had not changed the TS setpoint, they had changed its technical justification by revising the limiting transient upon which the setpoint is based. The relevant Technical Specifications for the Salem Unit 1 POPS are 3.4.9.3 and Bases 3/4.4.9.3; For Salem Unit 2 they are TS 3.4.10.3, and Bases 3/4.4.10.3. Further, the new

transient, which changed the design basis for POPS, also invalidated the NRC's SER upon which Amendment No. 24 - and both units' POPS TS - was based. The description of the limiting transient and the design bases for POPS in Salem FSAR Section 7.6.3.3 were, therefore, no longer correct and current.

The inspector noted that, since March 1993, several PSE&G corrective action programs were used but none was effective in resolving the Salem POPS issue. Another (more recent) opportunity to evaluate all the relevant considerations of this issue was missed: as of the conclusion of the exit meeting on December 19, 1994, no safety evaluation was performed to determine if the change in the POPS design-basis transient had created an "unreviewed safety question." 10 CFR 50.59 requires licensees to evaluate changes to the plant or its procedures (including methods and modes of operation), prior to those changes being effected, to assure no unreviewed safety question exists. The licensee's failure to perform this safety evaluation is, therefore, an apparent violation of 10 CFR 50.59. **(EEI 50-272;311/94-32-04)**

<u>"New" Transient Amended</u>

In November 1994, the licensee recognized that an error - recently identified in their configuration baseline document - would adversely effect their assumptions for the revised POPS mass addition transient. The configuration document had incorrectly assumed that the positive displacement (PD) charging pump trips off on a SI signal; however, if off-site power is available when the SI signal occurs, the pump continues to run and trip signals are blocked (until the SI signal is reset). After discovering this error, analysis for the limiting POPS transient was revised to include the mass addition of the PD charging pump and resulted in a calculated peak pressure of 474 psig.

PSE&G Incident Report (IR) 94-419, dated November 17, 1994, documented this latest discovery and concluded that the Unit 1 POPS no longer met its design basis single failure criterion because a single PORV could no longer mitigate the transient. PSE&G reported this to the NRC under 10 CFR 50.72 as an unanalyzed condition for Salem Unit 1. IR 94-419 provided justification for the continued operation of Unit 1 based on RHR relief valve RH3 being available to augment POPS. With the three valves (two PORVs and RH3) available below 312°F, sufficient relief capacity was reasoned (by the licensee) to be provided and the single failure criterion could be met. However, the licensee considered Unit 2 to be "not reportable" because with a single PORV the peak transient pressure was still 1.0 psi below its P/T curve limit.

The inspector concluded that, since the margins to safety for overpressure protection (viz, peak pressure versus Appendix G P/T limits) had either been significantly reduced or lost altogether (depending upon which transient and assumptions are adopted as limiting), the new "limiting" transient represented a potential unreviewed safety question.

RCS Vent Path

TS for both Salem units require cold overpressure protection be provided by either the redundant PORVs (the POPS system) or a reactor coolant system vent of greater than or equal to 3.14 square inches (in^2). Venting the RCS is an alternative to having the POPS operable and would be accomplished after depressurizing the RCS. The TS action statement for POPS requires that, in the event a PORV fails and cannot be restored within seven days, the reactor must be depressurized and vented through the 3.14 in^2 vent within the next eight hours.

The inspector could find no specific justification for the TS-required vent area of 3.14 in^2. However, the inspector concluded that the vent area required in TS should be adequate based on: (1) the flow from an unrestricted opening of 3.14 in^2 would encounter less resistance than that through a single PORV; (2) a single PORV must be shown to provide sufficient relief capability even with its delay for actuation; and (3) the vent area is passive protection and, therefore, does not need to be redundant. The inspector noted that while no formal analyses were available to support the 3.14 in^2 area or compare it to actual PORV capacity, the full-open port area of a single PORV is approximately 2.2 in^2. The "equivalent throat area" (a term used by Westinghouse in WCAP-11640, March 1988) of a full-open PORV would be adjusted for hydraulic resistance, and factors affecting this correlation were provided in a December 8, 1992, memorandum to PSE&G from the valve vendor, Copes Vulcan. The vendor's memorandum depicts the estimated flow coefficient as a function of valve lift or opening during its 1.5-second stroke. The Salem PORVs are 2-inch diameter Model D-100 "plug-in-cage" valves, with flow coefficients on the order of 50. This flow coefficient can be used, along with previously compiled EPRI test data for these type valves, to calculate a so-called equivalent area-corresponding to a smoothly convergent nonflashing (sonic flow) nozzle - that licensee thermal hydraulic engineers estimated to be 1.21 in^2.

The RCS vent area of 3.14 in^2, by itself, has no hydraulic meaning unless a geometry can be assumed so that resistance and loss factors can be calculated. Nonetheless, a single PORV gagged-open is clearly enveloped in the RCS vent configuration by a simple two-inch diameter flanged opening when corrected for flow losses; this effective area is on the order of 2-2.5 in^2. The licensee, in fact, utilizes several options to establish the RCS vent including removal of a steam generator primary-side manway, removal of one or more code safety valves, or gagging-open the PORV's as an alternative to POPS valves set to automatically relieve at 375 psig.

Code Case Approval

The inspector reviewed the licensee's documentation and interviewed personnel involved with the POPS issue during the 20 months between the NSAL issuance in March 1993 and PSE&G's 50.72 notification in November 1994. The inspector concluded that there was an adequate assurance of safety, based on the additional relief capacity of valve RH3 and the margin that can be gained with use of ASME Code Case N-514. Based on the inspector's discussions (prior to February 1995) with representatives from the NRC Office of Nuclear Reactor

Regulation (NRR), ASME Code Case N-514 represented a technically acceptable position, although a plant specific exemption would be required.
On December 16, 1994, a conference call between PSE&G and NRC representatives was held to discuss the licensee's more immediate actions to resolve certain aspects of the POPS issue. During this call, the licensee committed to limit the number of RCPs in service (per existing procedures) when RCS temperature is below 200°F, and to maintain procedural controls preventing an intermediate head safety injection pump from injecting into the RCS. These commitments were formally submitted in a letter to the NRC from PSE&G issued later that same day.

On December 22, 1994, PSE&G submitted an application for NRC approval of ASME Code Case N-514. Included in the submittal were the calculations supporting the new design-basis transient for POPS. Without the code case, PSE&G credited valve RH3 on Unit 1 to meet the design basis single failure criterion for POPS. However, for Unit 2, the licensee did not credit valve RH3 because the peak pressure, based on a single PORV, was predicted to be 1.0 psi below the unit's P/T limit. By letter dated February 13, 1995, the NRC issued an exemption from the requirements of 10 CFR 50.60 for Salem Units 1 and 2. This exemption permits using the safety margins recommended in ASME Code Case N-514 in lieu of the safety margins required by Appendix G to 10 CFR 50. Therefore, each unit's P/T curve limits for POPS were increased by 10%; the Unit 1 and 2 limits became 495 and 522 psig, respectively.

3.0 CONCLUSIONS

The inspector considered several aspects of PSE&G's actions to resolve the POPS issue over the past 20 months as inadequate or inappropriate:

- The POPS issue was initially dispositioned to show that the requirements of 10 CFR 50.60 were met, invoking an ASME code case that had not received prior NRC approval.

- When inclusion of the setpoint nonconservatism put the Salem Units outside the POPS design basis, reports to the NRC were not made pursuant to 10 CFR 50.72 and 73.

- No safety evaluation pursuant to 10 CFR 50.59 was performed prior to revising the POPS design-basis transient (described in the Salem FSAR) in September 1994. As of the December 19, 1994, exit meeting, a 50.59 evaluation had not been completed. Several times during the licensee's attempts to resolve the POPS questions, the margins to safety for POPS (defined in the February 1980 NRC safety evaluation) were found to be reduced, but not appropriately evaluated.

- It took almost two years (March 1993 to February 1995) for PSE&G to take appropriate actions to address the NSAL nonconservatism. The lowest priority possible was assigned to this issue within the Operational Experience Feedback program in March 1993, and the issue was not entered into an appropriate PSE&G quality program for resolving engineering discrepancies for over a year.

The adequacy of the new design basis for POPS is currently under review by the NRC's Office of Nuclear Reactor Regulation. Pending NRC assessment of the licensee's proposed limiting design-basis transient for POPS, this issue is unresolved. **(URI 50-272;311/94-32-05)**

The licensee's: (1) reliance on ASME Code Case N-514 without NRC approval, (2) failure to report a condition outside the Salem design basis, and (3) failure to perform an adequate safety evaluation of the revised POPS design-basis transient are all apparent violations of NRC requirements. Further, the process used to address the issue (memorandum superseding memorandum) was considered to be fragmented and not appropriate for potentially safety-significant issues. The inspector further concluded that the corrective action processes that were engaged (a year late) did not appropriately resolve a condition outside the plant's design basis.

4.0 MANAGEMENT MEETINGS

Licensee representatives were informed of the scope and purpose of this inspection at an entrance meeting conducted on December 5, 1994. Findings were periodically discussed with the licensee throughout the course of this inspection. A telephone conference call was conducted between NRC and PSE&G representatives on December 16, 1994, to discuss the licensee's plans to resolve several aspects of the POPS issue. During this call, PSE&G committed to taking several actions and subsequently documented these commitments in a letter to the NRC issued later that day.

The inspector met with the principals listed below to summarize preliminary findings on December 19, 1994. The licensee acknowledged the preliminary findings and conclusions, with no exceptions taken. Further, the bases for the preliminary conclusions did not involve proprietary information, nor was any such information expected to be included as part of the written inspection report.

Public Service Electric & Gas Company

L. Catalfomo	Operations Manager
J. Morrison	Salem Technical Department Manager
M. Morroni	Controls Maintenance Manager
J. Ranalli	Nuclear Mechanical Engineering Manager
D. Smith	Principal Engineer, Nuclear Licensing
J. Summers	General Manager, Salem Operations Manager

NRC

C. Marschall	Senior Resident Inspector, Salem
D. Moy	Reactor Engineer, DRS
J. White	Chief, RPS2A, DRP

A final summary was provided to PSE&G representatives by telephone on March 23, 1995, discussing the results of the NRC inspection and evaluation that took place after the December 1994 site visit. Another telephone conversation was held on March 29, 1995, to discuss the issue of PORV flow characteristic's (refer to page 6, "RCS Vent Path") between E. Kelly of the NRC and the following PSE&G representatives:

C. Lambert	Manager, Nuclear Engineering Design
Vijay Chandra	Technical Consultant, Thermal Hydraulics
Gita Narasimhan	Mechanical Engineer
Mahesh Danak	Mechanical Engineer