Public Service
Electric and Gas
Company

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SEP 19 1994
NLR-N94161
LCR 94-06

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

Gentlemen:

RECORD OF TELEPHONE CONVERSATION
REACTOR COOLANT SYSTEM FLOW RATE REQUEST FOR AMENDMENT
SALEM GENERATING STATION UNIT NO. 2
DOCKET NO. 50-311

The purpose of this letter is to document an August 4, 1994 telephone conversation between Public Service Electric and Gas Company (PSE&G) and NRC staff to discuss PSE&G's proposed 1% reduction in the minimum allowable Reactor Coolant System (RCS) flow. The proposed change to the Salem Generating Station (SGS) Unit No. 2 Technical Specifications was made via PSE&G letter dated February 3, 1994 (NLR-N94016). Attachment 1 summarizes the conversation, which centered around NRC questions relative to the current status of the RCS flow calculations and the evaluations of the impact of the proposed flow limit reduction on the SGS safety analyses. The information provided in this letter does not affect the No Significant Hazards Consideration presented in the February 3, 1994 amendment request.

Please contact us if there are any questions regarding this transmittal.

Sincerely,

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SEP 19 1994

Attachment
Affidavit
C Mr. T. T. Martin, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. J. C. Stone, Licensing Project Manager - Salem
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. C. Marschall (S09)
USNRC Senior Resident Inspector

Mr. K. Tosch, Manager, IV
NJ Department of Environmental Protection
Division of Environmental Quality
Bureau of Nuclear Engineering
CN 415
Trenton, NJ 08625

REF: NLR-N94161

STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

J. J. Hagan, being duly sworn according to law deposes and says:

I am 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 the Salem Generating Station, Unit Nos. 1 and 2, are true to the best of my knowledge, information and belief.

JJHagan

Subscribed and Sworn to before me
this 19th day of September 1994

Kimberly Jo Brown
Notary Public of New Jersey

My Commission expires on KIMBERLY JO BROWN NOTARY PUBLIC OF NEW JERSEY My Commission Expires April 21, 1998

ATTACHMENT 1

SUMMARY OF 8/4/94 TELECON - AMENDMENT REQUEST FOR SALEM UNIT 2 RCS FLOW REDUCTION

PARTICIPANTS:

NRC

J. C. Stone, NRR/Project Directorate I-2
J. I. Zimmerman, NRR/Project Directorate I-2
H. Balujkian, NRR/Reactor Systems Branch
W. O. Long, NRR/Containment Systems and Severe Accident Branch

PSE&G

K. R. Pike, Salem Reactor Engineering
D. A. Rothrock, Nuclear Fuels
W. J. McTigue, Nuclear Licensing

SUMMARY

The purpose of the telephone conversation was to address NRC questions resulting from their review of PSE&G's 2/3/94 amendment request for a 1% reduction in the minimum RCS flow allowed by the Salem Unit 2 Technical Specifications. The first set of NRC questions are related to RCS flow measurement:

1. The [February 3, 1994 amendment request] stated that PSE&G has found that the RCS flow calculation has a non-conservatism that may reduce the flow calculated by approximately 1,000 gpm. Has this been factored in for the latest total RCS flow calculation for Salem Unit 2?

2. What is the value for the latest total RCS flow calculation for Salem Unit 2? Does this include the 1,000 gpm non-conservatism?

In response to the above questions, PSE&G explained that the latest RCS flow calculation, performed on May 16, 1994, was 366,054 gpm. This value includes the approximate 1,000 gpm nonconservatism mentioned in the amendment request. The calculated flow value is approximately 2.5% above the current Technical Specification minimum value. It was calculated based on the latest set of feedwater flow measurement data. The decrease in calculated RCS flow at Salem Unit 2 over the past two refueling cycles is attributed to actual feedwater flow being greater than indicated flow, as discussed in Licensee Event Report (LER) 311/94-002, Supplement 4, dated August 12, 1994.

[Note that the LER Supplement was submitted after the phone conversation, and provides the current status of the feedwater flow investigations.] NRC noted that the margin to the current Technical Specification limit was small, and that the proposed change would increase the margin.

NRC then asked whether hot leg streaming was considered a factor in the reduction in calculated RCS flow. [Streaming in the hot leg may result in measured hot leg temperatures being slightly higher than actual temperatures, which would increase the measured differential temperature across the core and result in reduced calculated RCS flow.] PSE&G responded that hot leg streaming had been considered, and that it was not the cause of the downward trend in calculated RCS flow at Salem Unit 2. The streaming phenomenon is associated with plants with extremely low leakage core designs (low power fuel assemblies or filler rods at the core periphery). Salem does not presently use extremely low leakage core designs, compared to other Westinghouse plants which may be more susceptible to hot leg streaming. The degree of leakage is controlled by the core design process, and has not decreased significantly at SGS Unit 2 since Cycle 3. This suggests that the downward trend in calculated RCS flow over the past several refueling cycles at SGS Unit 2 is independent of any hot leg streaming effects. Additionally, the differential temperatures among the three temperature elements per hot leg is relatively low, further suggesting that hot leg streaming is not the cause of the downward trend in calculated RCS flow.

NRC asked about the present status of the feedwater flow measurement efforts, to which PSE&G responded that the flow calibration constants for the in-plant measurement equipment is cross calibrated against Leading Edge Flow Measurement (LEFM) equipment. Feedwater flow measurements and other parameters relevant to RCS flow (e.g., T_{avg}) are being trended.

NRC asked what the discovered nonconservatism in the flow calculation was. PSE&G responded that the Reactor Coolant Pump (RCP) heat input was not subtracted from the heat balance equation, which is necessary because of the pumps' location relative to the T_{hot} and T_{cold} measurement locations (T_{cold} is measured upstream of the RCP's). While the power calorimetric calculation included the pump heat, the flow calculation did not. Correction of this error resulted in a reduction of approximately 0.3% in calculated RCS flow.

At this point, questions 3 and 4, relative to the evaluation of the plant safety analyses presented in the amendment request, were discussed:

3) For the accident analyses for Blowdown Reactor Vessel and Loop Forces(UFSAR Section 3.9.1.5) you state that the impact

on operating temperature is small enough to be considered negligible relative to the calculation of forces from a postulated RCS pipe break. Is the "negligible" impact based on a calculation or from a judgement?

NRC explained that because the amendment request does not explicitly state that a calculation was done, a more definitive discussion of the impact on the blowdown analyses is required.

PSE&G responded that the 1% reduction in flow causes a change in cold leg temperature of up to 0.4°F. The fluid density increase associated with such a small change in temperature was judged to have negligible effect on the calculation of forces from a postulated RCS pipe break.

When the amendment request was submitted, Salem's application of Leak-Before-Break (LBB) methodology was under NRC review, but had not been approved. LBB has since been approved for Salem (ref: NRC letter to PSE&G dated May 24, 1994), thereby eliminating the dynamic effects of double-ended primary loop pipe ruptures from the licensing basis. Although application of LBB clearly offsets any penalty associated with the proposed reduction in flow relative to blowdown forces, the judgement of no significant impact was applied to the limiting (double-ended) ruptures at the time of the amendment request.

PSE&G went on to note that the discussion of the containment subcompartment analyses in the amendment request also mentions LBB. As noted in the amendment request, the limitation of the vessel inlet and outlet pipe break to 75 square inches for subcompartment analyses, which more than offsets any affects of increased density associated with the 0.4°F temperature reduction, does not rely on LBB. Instead, the break sizes were determined using elastic analyses which determine the maximum possible break area based on pipe deflections.

4) For the accident analyses for Loss of External Electrical Load and/or Turbine Trip (UFSAR Section 15.2.13) you state that the evaluation concluded that the pressures would not be significantly affected by the proposed flow reduction. Is the "significantly" term based on a calculation or from a judgement?

The term "significant" is also used for the following accidents:

Loss of Offsite Power (UFSAR Section 15.2.9)
Feedwater System Pipe Break (UFSAR Section 15.4.3)
Reactor Coolant Pump Shaft Seizure (*Locked Rotor*) (UFSAR Section 15.4.5)

Spurious Operation of Safety Injection at Power (UFSAR Section 15.2.14)
Accidental Depressurization of the RCS (UFSAR Section 15.2.12)

PSE&G addressed each of the above analyses as follows:

- *Loss of External Electrical Load and/or Turbine Trip* - The conclusion that the pressures would not be significantly affected by the proposed flow reduction was based on sensitivity calculations. Another plant had been analyzed for a 4% flow reduction. Although the calculation was not Salem-specific, the plant that was modeled was similar to Salem and supports the conclusion.

- *Loss of Offsite Power* - The conclusion that natural circulation core cooling would not be significantly affected was based on judgement. A *Loss of Offsite Power* transient is bounded by the *Loss of Feedwater and Loss of Forced Reactor Coolant Flow* transients. Since these were not significantly impacted by the reduction in primary system flow, the Loss of Offsite Power transient was judged similarly.

- *Feedwater System Pipe Break* - The conclusion that the flow reduction would have no significant impact on the peak hot leg temperatures was based upon sensitivity calculations. These showed the results were not sensitive to reductions in RCS flow on the order of 1%.

- *Locked Rotor* - The conclusions that are made were based on both calculations and judgement. A calculation was done to verify the core thermal limits. Once verified, a judgement was made on the effect the reduction in flow would have on existing margins and the transient behavior.

- *Spurious Operation of Safety Injection at Power* - The conclusion that transient conditions of this event are not significantly altered by the proposed reduction in RCS flow was based on both calculation and judgement. A calculation was done to verify the core thermal limits.

- *Accidental Depressurization of the RCS* - The conclusion that transient conditions of this event are not significantly altered by the proposed reduction in flow was based on a calculation and judgement. A calculation was done to verify core thermal limits and hence, ensure that the reactor protection system prevented DNB in the core. Once verified, a judgement was made on the effect the reduction in flow would have on existing margins and the transient behavior.

During the conversation, NRC asked PSE&G to confirm the statement in the amendment request, that the Double-Ended Pump Suction (DEPS) LOCA was limiting for long term containment response. NRC noted that DEPS was not typically the limiting containment pressure case. PSE&G has since confirmed that the DEPS is the limiting case for Salem and according to Westinghouse, is limiting for some other plants.