UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-75

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

DOCKET NO. 50-311

1.0 INTRODUCTION

By letter dated February 3, 1994, as supplemented September 19, 1994, and November 23, 1994, the Public Service Electric & Gas Company (PSE&G, the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit No. 2, Technical Specifications (TS) to reflect a reduction in Reactor Coolant System (RCS) flow.

2.0 EVALUATION

Salem Unit No. 2 has experienced a decrease in calculated RCS total flow over the past several refueling cycles. This has not been confirmed by the RCS elbow tap data, which tends to indicate flow has remained basically constant. Following the Unit 2 seventh refueling outage, RCS total flow was calculated to be slightly above the minimum required by Technical Specification 3.2.5. Recently, a review by PSE&G of the flow calculation procedure identified a non-conservatism that may reduce the calculated flow by approximately 1,000 gpm. PSE&G stated in the September 19, 1994, letter that the latest RCS flow calculation was 366,054 gpm, which includes the approximate 1,000 gpm non-conservatism. This value was stated as being approximately 2.5% above the current TS minimum value (357,200) which includes the flow measurement uncertainty.

The Loss of Flow Reactor Trip is set at 90% of design RCS flow per loop to prevent operation with significant flow reductions.

Because of the small margin between the calculated RCS total flow and the Technical Specifications, PSE&G has proposed to reduce the TS requirements for the Loss of Flow Reactor Trip and the RCS total flow rate by 1%. This required confirmation that adequate margin exists in the Loss of Coolant Accident (LOCA), Non-LOCA and Containment analyses to justify the proposed TS changes.

One of the requested changes is an editorial change to modify the departure from nucleate boiling (DNB) flow parameter in Table 3.2-1 from "Reactor Coolant System" to "Reactor Coolant System Total Flow Rate."

The proposed TS changes are given below:

A. Change Table 2.2-1 Reactor Trip System Instrumentation Trip Setpoints as follows:

   1. For Functional Unit 12, Loss of Flow, change the Note to read;

      "* Design flow is 86,430 gpm per loop."

B. Change Table 3.2-1 DNB Parameters as follows:

   1. Change the third parameter from "Reactor Coolant System" to read, "Reactor Coolant System Total Flow Rate."

   2. Change the limit for Reactor Coolant System Total Flow Rate to read: ≥353,700 gpm.

3.0 EVALUATION

To justify the proposed Technical Specification changes, PSE&G made evaluations to confirm that adequate margin exists in the LOCA, Non-LOCA and Containment analyses. The accidents which rely on adequate RCS flow were analyzed using an RCS flow reduction of 1% in the evaluations below.

3.1 LOCA ANALYSIS

The following Updated Final Saftety Analysis Report (UFSAR) LOCA analyses were evaluated for effects of a 1% reduction in RCS flow:

3.1.1 Large Break LOCA (UFSAR Section 15.4.1)
Small Break LOCA (UFSAR Section 15.3.1)
Steam Generator Tube Rupture (UFSAR Section 15.4.4)

The Large Break LOCA analyses assume an RCS flow of 345,600 gpm. The Small Break LOCA and Steam Generator Tube Rupture analyses assume a flow of 330,000 gpm. These values conservatively bound the proposed RCS minimum flow limit of 353,700 gpm. Therefore, the proposed change has no adverse impact on these analyses. The peak cladding temperature and hydrogen generation criteria of 10 CFR 50.46, and the offsite dose criteria of 10 CFR 100 continue to be met. Therefore we find the TS reduction in flow to be acceptable.

3.1.2 Long Term LOCA (Containment Integrity) Analyses (UFSAR Section 15.4.8.1)

The bounding break for containment response is a cold leg pump suction line break. The containment response analysis for this break would not be affected by the flow reduction. The proposed RCS flow reduction would result in increased hot leg break flow energy release. The licensee has determined that the associated peak containment pressure increase would be less than 0.15 psi. With such a slight increase, the pump suction line break remains bounding for containment response. The temperature increase and equipment environmental effects associated with a 0.15 psi pressure increase and no power level increase would be insignificant. It is therefore concluded that the proposed change is acceptable for containment response.

3.1.3 Subcompartment Analyses (UFSAR Section 15.4.8.3)

The proposed reduction in RCS flow is estimated to result in a 0.4 degree F reduction in vessel inlet temperature which results in increased fluid density. This change in temperature would result in approximately a 0.1% increase in critical flow. The licensee was not able to make a sensitivity study since the licensing basis analysis model was not available. However, a review of the current subcompartment analysis results show that evaluations were found to be performed for breaks as large as 100 square inches at the reactor pressure inlet and outlet piping. Based on the evaluation of piping displacements resulting from LOCA, and gap sizes for pipe whip restraints, it was determined that the largest break consistent with the RCS piping configuration, is 75 square inches at the vessel inlet and outlet locations, and a single-ended break at all other RCS locations. This reduction in break size offsets any penalty associated with the reduced RCS flow. In addition, a Salem-specific leak-before-break submittal was made to NRC on July 6, 1993, justifying further relaxations in primary loop pipe break postulations. We therefore find this to be acceptable.

3.1.4 Blowdown Reactor Vessel and Loop Forces (UFSAR Section 3.9.1.5)

The proposed 1% reduction in RCS flow would have an effect on RCS temperature. Although the forces created by a postulated RCS pipe break are a function of temperature, the licensee has found 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. We therefore find this to be acceptable.

3.2 NON-LOCA ANALYSES

The following current Salem Non-LOCA analyses were evaluated for the effects of a 1% reduction in RCS flow.

3.2.1 Excessive Load Increase (UFSAR Section 15.2.11)

Four cases of a 10% step load increase from nominal full power conditions were analyzed, based on automatic versus manual rod control, and minimum versus maximum reactivity feedback parameters. The licensee's evaluation of this transient using the proposed reduction in RCS flow concluded that the Departure from Nucleate Boiling Ratio (DNBR) design basis is met. We therefore find this to be acceptable.

3.2.2 Excessive Heat Removal Due to Feedwater System Malfunction (FWM) (UFSAR Section 15.2.10)

The current UFSAR (Section 15.2.10) compares two cases for this transient. These two cases were analyzed; a full power case and a zero power case which was shown to be bounded by the Rod Withdrawal From Subcritical (RWFS) transient. These two transients are compared to each other, since both FWM and RWFS are positive reactivity insertions. It was determined that the departure from nucleate boiling ratio (DNBR) for the full power case would remain above the safety limit and that the zero power case remains bounded by RWFS. Also, it was determined that steam generator overfill would not be affected.

3.2.3 Accidental Depressurization of the Main Steam System (UFSAR Section 15.2.13)

Evaluation of this event shows that the DNBR analysis of record remains valid given the proposed reduction in RCS flow.

3.2.4 Major Secondary Side Pipe Rupture (UFSAR Section 15.4.2)

The evaluation of this event concluded that sufficient DNB margin exists to offset the reduced flow penalty.

3.2.5 Steam Line Break Mass/Energy Release (UFSAR Section 15.4.8.2)

It was found that the proposed reduction in RCS flow would not adversely affect the steamline break mass/energy releases.

3.2.6 Loss of External Electrical Load and/or Turbine Trip (UFSAR Section 15.2.13)

Four cases of a total loss of steam demand at full power, without a direct reactor trip, were analyzed, based on automatic rod control versus no rod control, and minimum versus maximum reactivity feedback parameters. Evaluation of this transient using the proposed reduction in RCS flow concluded that the DNBR design basis is met. An evaluation of the maximum primary and secondary system pressures following this transient was also performed. Based on sensitivity calculations, the evaluation concluded that the pressures would not be significantly affected by the proposed flow reduction.

3.2.7 Loss of Offsite Power (UFSAR Section 15.2.9)

The evaluation concluded that the DNBR remains above the safety limit. The licensing basis criteria for primary and secondary system pressure continues to be met, and pressurizer filling will not occur.

3.2.8 Loss of Normal Feedwater (UFSAR Section 15.2.8)

The evaluation concluded that sufficient margin exists relative to primary and secondary peak pressures and pressurizer filling, such that the conclusions stated in the UFSAR remain valid.

3.2.9 Feedwater System Pipe Break (UFSAR Section 15.4.3)

The evaluation of this event was based on sensitivity studies by Westinghouse and showed that the proposed reduction in RCS flow would result in a small decrease in steam generator mass, and no significant impact on peak hot leg temperatures. Based on additional calculations this slight decrease in inventory would result in a slight reduction in margin to the hot leg saturation temperature, but the RCS would remain subcooled (RCS subcooling is used to conservatively demonstrate that core cooling is maintained throughout the transient). Therefore the current licensing basis analyses contain sufficient margin to accommodate the decay heat removal penalty associated with the proposed reduction in RCS flow.

3.2.10 Partial Loss of Forced RCS Flow (UFSAR Section 15.2.5)
Complete Loss of Forced RCS Flow (UFSAR Section 15.3.4)

Evaluation of these transients showed that the effects of the proposed reduction in RCS flow on DNB and RCS pressure can be accommodated by existing margin, such that the conclusions in the UFSAR remain valid.

3.2.11 Reactor Coolant Pump Shaft Seizure (Locked Rotor) (UFSAR 15.4.5)

The licensee treats this event as an American Nuclear Society (ANS) Condition IV, Limiting Faults event, which is the same category as Large Break LOCA. Therefore, this event is evaluated to ensure that core cooling capability is maintained. DNB is assumed to occur for this transient; peak clad temperature (PCT) is one of the criterion. Evaluation of this transient showed that the effects of the proposed reduction in RCS flow on peak fuel clad temperature and RCS pressure can be accommodated by existing margins, such that the conclusions in the UFSAR remain valid.

3.2.12 Rod Withdrawal from Subcritical Condition (UFSAR Section 15.2.1)
Rod Withdrawal at Power (UFSAR Section 15.2.2)
Rod Cluster Control Assembly Misalignment (UFSAR Sections 15.2.3 and 15.3.5)

Startup of an Inactive Loop (UFSAR Section 15.2.6)
Uncontrolled Boron Dilution (UFSAR Section 15.2.4)

Evaluation of these transients showed that the effects of the proposed reduction in RCS flow on DNB can be accommodated by existing margins, such that the conclusions in the UFSAR remain valid.

3.2.13 Rupture of a Control Rod Drive Mechanism Housing (UFSAR Section 15.4.7)

The licensee treats this event as an ANS Condition IV, Limiting Faults event, which is the same category as Large Break LOCA. Therefore, this event is evaluated to ensure that core cooling capability is maintained. DNB is assumed to occur for this transient; PCT is one of the criterion. To demonstrate that gross fuel damage would not occur, that the core would remain in a coolable geometry, and that the RCS would remain intact, the licensee applied the following criteria to this event:

1) The average fuel pellet enthalpy at the hot spot is less than 200 cal/gm (360 BTU/lbm).

2) Fuel melt at the hot spot is limited to less than the innermost 10% of the fuel pellet.

3) Peak RCS pressure is less than that which would cause stresses to exceed 120% of the design pressure.

The evaluation of this transient showed that the effects of the proposed reduction in RCS flow on peak fuel clad temperature and RCS pressure can be accommodated by existing margins, such that the above criteria would continue to be met, and the conclusions in the UFSAR remain valid.

3.2.14 Spurious Operation of Safety Injection at Power (UFSAR Section 15.2.14)

The evaluation found that since the transient conditions of this event are not significantly altered by the proposed reduction in RCS flow, the conclusions in the UFSAR remain valid.

3.2.15 Accidental Depressurization of the RCS (UFSAR 15.2.12)

The evaluation found that since the transient conditions of this event are not significantly altered by the proposed reduction in RCS flow, and the OT-delta-T setpoint is not changed, the conclusions in the UFSAR remain valid.

The above evaluations of the relevant Chapter 15 transients take into account the 1% decrease in RCS flow rate. These evaluations indicate that the 1% reduction in RCS flow is acceptable as the consequences of previously analyzed accidents have been found to remain within acceptable licensing basis limits. The proposed change in wording for the title in Table 3.2-1, DNB Parameters, from "Reactor Coolant System" to "Reactor Coolant System Total Flow Rate" is an editorial change for clarity and is therefore acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 14028). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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