Public Service
Electric and Gas
Company

Steven E. Miltenberger
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U.S. Nuclear Regulatory Commission
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Gentlemen:

10 CFR 50.46(a)(1)(i) EXEMPTION REQUEST
TRANSMITTAL OF AFFIDAVIT
SALEM GENERATING STATION
DOCKET NO. 50-311

Public Service Electric and Gas Company's Request for Temporary Exemption from 10 CFR 50.46(a)(1)(i) for the Salem Generating Station, Unit 2 dated October 21, 1988 did not include an affidavit. The required affidavit is included in the attached resubmittal of the October 21, 1988 request.

Should you have any additional questions or comments, please do not hesitate to contact us.

Sincerely,

Steven E Miltenberger

Affidavit
Attachments (3)

c Mr. J. C. Stone
Licensing Project Manager

Mr. R. W. Borchardt
Senior Resident Inspector

Mr. W. T. Russell, Administrator
USNRC Region I

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REF: 10 CFR 50.46(a)(1)(i)
EXEMPTION REQUEST

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

Steven E. Miltenberger, being duly sworn according to law deposes and says:

I am Vice President and Chief Nuclear Officer of Public Service Electric and Gas Company, and as such, I find the matters set forth in our letter dated OCT 24 1988, concerning the Salem Generating Station, Unit No. 2, are true to the best of my knowledge, information and belief.

Steven E Miltenberger

Subscribed and Sworn to before me
this 24th day of October, 1988

Eileen M. Ochs
Notary Public of New Jersey

EILEEN M. OCHS
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires July 16, 1992

My Commission expires on ______________

ATTACHMENT 1

REQUEST FOR TEMPORARY EXEMPTION FROM CERTAIN ADMINISTRATIVE REQUIREMENTS OF 10 CFR 50.46(a)(1)(i)

In accordance with the requirements of 10 CFR 50.12(a), Public Service Electric and Gas Company (PSE&G) hereby requests a temporary exemption to certain administrative requirements of 10 CFR 50.46(a)(1)(i) for Salem Generating Station Unit 2.

I. SUMMARY OF THE CURRENT SITUATION

As a result of eddy current examinations performed during the Salem Unit 2 Fourth Refueling Outage, defective tubes were discovered in two steam generators which were indicative of the Westinghouse Series 51 Steam Generator Row 1, U-bend tangent cracking phenomenon. In addition to plugging the defective tubes, PSE&G decided to plug the Row 1 tubes in all four Salem Unit 2 steam generators as a precautionary measure. As a result of this decision, 2.7% of the Salem Unit 2 steam generator tubes have been plugged. The reduced flow as a result of this condition affects the peak cladding temperature and requires a Large Break LOCA ECCS reanalysis for Salem Unit 2.

During refueling operations a burnable poison rodlet assembly hold down nut, a locking weld pin and a hand held gamma measurement probe with cable connector were inadvertently dropped into the reactor cavity. Subsequent efforts to retrieve these items were unsuccessful. As a result, a decision was made to evaluate these objects as loose parts within the reactor coolant system. The potential for a partial flow blockage to or within fuel assemblies affects of the peak cladding temperature and requires a Large Break LOCA ECCS reanalysis for Salem Unit 2.

While these events do not present any safety concerns or operational considerations for the reasons detailed in Attachments 2 and 3, a formal reanalysis is required to confirm that Salem Unit 2 meets the applicable criteria of 10 CFR 50.46(b) based on the current plant configuration.

II. BASIS FOR THE EXEMPTION REQUEST

For plants licensed based on the 1978 Westinghouse Large Break LOCA model, Generic Letter 86-16 requires subsequent plant changes which affect the results of the model, to be reevaluated against an updated, approved model and submitted in accordance with 10 CFR 50.46(a)(1)(i). Since the reanalysis with the new

ECCS model cannot be completed for approximately 5 months (for the reasons discussed in Section III) and because Salem Unit 2 is scheduled to enter Mode 2 on October 27, 1988, PSE&G requests a one-time, temporary exemption from 10 CFR 50.46(a)(1)(i) based on the specific circumstances discussed in Section III and the technical arguments contained in the Westinghouse Safety Evaluations provided in Attachments 2 and 3.

In summary the Safety Evaluations conclude that the requirements of 10 CFR 50.46(b), including the maximum peak cladding temperature (PCT) limit of 2200°F are satisfied and the safe operation of Salem Unit 2 is assured. The Safety Evaluations are based on sensitivity studies which indicate that the current maximum Large Break LOCA (LB LOCA) PCT value of 2130°F would increase by a maximum of 28°F as a result of the tube plugging modifications and a maximum of approximately 22°F as a result of the potential flow blockage effect of the loose parts. However, the actual maximum LB LOCA PCT, due to the combined effects discussed above, will not change due to the offsetting effect of reduced rod internal backfill pressure of currently used fuel assemblies previously evaluated but not yet incorporated into the existing analysis. The Safety Evaluations provide sufficient information to conclude that this exemption request does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated and does not involve a significant reduction in a margin of safety. Therefore, it can be concluded that this exemption does not involve a safety concern nor present an undue risk to the public health and safety and is consistent with the common defense and security.

III. SPECIFIC JUSTIFYING CIRCUMSTANCES

This request meets the criteria established by the NRC in 10 CFR 50.12(a)(2) in that special circumstances are present which warrant approval. Specifically, Paragraphs 50.12(a)(2)(ii), 50.12(a)(2)(iii) and 50.12(a)(2)(v) apply to the current situation. 10 CFR 50.12(a)(2)(ii) states:

> "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

PSE&G believes that the immediate submittal of the formally amended ECCS analysis is not necessary to achieve the underlying purpose of 10 CFR 50.46 provided that sufficiently detailed information is available to justify the safe operation of the station in accordance with the applicable regulatory requirements. Attachments 2 and 3 contain the Safety Evaluations which indicate that the sensitivity studies performed on the current ECCS analyses assure that the calculated PCT value is bounding and in compliance with 10 CFR 50.46(b). Additionally, PSE&G commits to complete the ECCS reanalysis, using the 1981 Westinghouse ECCS model with BASH, by March 31, 1989 and provide a final report to the NRC staff for review.

10 CFR 50.12(a)(2)(iii) states:

> "Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated."

Without approval of the temporary exemption request, PSE&G would be required to complete the ECCS reanalysis prior to return to power operation of Salem Unit 2. Completion of this reanalysis requires significant expenditures of engineering manpower by Westinghouse (i.e. developing revised computer codes, inputting amended parameters, running lengthy computer codes, verifying output results, and generating a final report and conclusion) and takes a minimum of 5 months to complete. In addition, during the time in which the analysis was being performed, Salem Unit 2 could not return to operational service even though physically ready which results in a severe financial penalty and the associated costs of replacement power. Finally, similar requests have been granted to Tennessee Valley Authority for Sequoyah Unit 1 and Pacific Gas and Electric Company for Diablo Canyon Unit 2 which, if not applied to Salem Unit 2 would result in a significant cost well in excess of that incurred by others.

10 CFR 50.12(a)(2)(v) states:

> "The exemption would provide only temporary relief from the applicable regulation and the licensee or applicant has made good faith efforts to comply with the regulation."

As indicated above, PSE&G is only requesting this exemption on a one-time, temporary basis and fully intends to complete the necessary reanalysis and provide a report to the NRC staff by March 31, 1989. In the meantime, PSE&G has completed Safety Evaluations which justify the safe operation of Salem Unit 2. The requirements of 10 CFR 50.46(b) have been satisfied as demonstrated in the Safety Evaluations and will be applied to the reanalysis as well.

IV. TECHNICAL CONSIDERATIONS

Attachment 2 contains a Safety Evaluation which addresses the technical concerns associated with plugging the steam generator tubes. Information is also provided in the form of the results of a sensitivity study which evaluates the effects of tube plugging on the current Large Break LOCA ECCS analysis and against the requirements of 10 CFR 50.46(b). From the detailed discussion provided, PSE&G has concluded that this exemption request does not represent a safety concern.

Attachment 3 contains a Safety Evaluation which addresses the technical concerns associated with the presence of several loose parts in the reactor coolant system. Information is also provided in the form of the results of a sensitivity study which evaluates the effects of loose parts on the current Large Break LOCA ECCS analysis and against the requirements of 10 CFR 50.46(b). From the detailed discussion provided, PSE&G has concluded that this exemption request does not represent a safety concern.

V. ENVIRONMENTAL ASSESSMENT

The regulatory requirements of 10 CFR 50.46(b) have been applied to the sensitivity study results contained in the Safety Evaluations. None of the specific limits are exceeded as a result of the requested exemption. Therefore, operation of the station with the exemption request in place does not place the station in an unacceptable or unanalyzed condition and does not affect radiological or nonradiological plant effluents. This conclusion is applicable to both the steam generator tube plugging and the loose parts monitoring issues.

Therefore, it can be concluded, that operation of Salem Unit 2 with the proposed exemption in place will not impact the environment nor result in a situation in which the environmental impact is increased beyond that already analyzed for Salem.

VI. DATE WHEN COMPLIANCE WILL BE ACHIEVED

PSE&G is currently in compliance with the requirements of 10 CFR 50.46(b) and has concluded that the enclosed Safety Evaluations adequately demonstrate interim compliance with the administrative portions of 10 CFR 50.46(a)(1)(i). However, in order to assure that the ECCS model for Salem Unit 2 is properly revised to reflect the current plant configuration, PSE&G commits to complete the subject reanalysis and provide a final report to the NRC by March 31, 1989.

VII. CONCLUSION

Based on the information discussed above and the Safety Evaluations provided in Attachments 2 and 3, PSE&G has concluded that:

1. The requirements of 10 CFR 50.46(b) regarding the results of ECCS analysis, including PCT, have been, are and will continue to be applied to Salem Unit 2. Accordingly, PSE&G concludes that operation of

the station with the exemption request in place will satisfy the intent of 10 CFR 50.46(a)(1)(i).

2. The Safety Evaluations were performed to justify delaying the submittal of the formal ECCS analysis while Salem Unit 2 returns to power operation and concludes that neither the plugging of steam generator tubes nor the presence of loose parts in the reactor coolant system represent a safety concern or create an environmental impact.

3. The requested exemption is authorized by law and will not present an undue risk to the public health and safety, and is consistent with the common defense and security.

4. Special circumstances exist which justify the approval of this exemption request including the financial costs and hardships which would be incurred if approval was not granted.

5. The proposed exemption does not represent a situation which is unique or isolated to Salem Unit 2 but has been encountered in the industry before and successfully approved and implemented by stations such as Sequoyah and Diablo Canyon.

PSE&G requests the NRC staff to review and approve this exemption request by October 27, 1988 in order to support entry into Mode 2 and ascension to power of Salem Unit 2.

ATTACHMENT 2

LARGE BREAK LOCA SAFETY EVALUATION FOR SALEM UNIT 2 FOR STEAM GENERATOR TUBE PLUGGING

I. BACKGROUND

During the Fourth Refueling Outage for Salem Unit 2, the eddy current examination of Row 1 tubes indicated 45 defective tubes on No. 24 Steam Generator and 46 defective tubes on No. 22 Steam Generator. The defects were indicative of the Westinghouse Series 51 Steam Generator U-bend tangent primary water stress corrosion cracking (PWSCC) problem in Row 1 tubes. Eddy current examination of the Row 2 tubes of Nos. 22 and 24 Steam Generators and the Row 1 and Row 2 tubes in Nos. 21 and 23 Steam Generators revealed no further evidence of "low" row U-bend tangent cracking.

Subsequent to these eddy current examination results, a decision was made by PSE&G to plug all Row 1 tubes in all four Salem Unit 2 steam generators. The plugging of all the Row 1 tubes results in 2.7% of the tubes being plugged. This safety evaluation has been prepared to indicate the acceptability of up to 3.5% tube plugging.

Since there is no change in thermal design flow used as a basis for the accident analysis, the results of the evaluation indicate the change to the NSSS design temperatures and pressures is a reduction in the steam pressure of 9 psi and steam temperature of 1.4°F.

II. LARGE BREAK LOCA (FSAR Section 15.4.1)

The current Large Break LOCA analysis for Salem Unit 2, performed using the Westinghouse 1978 Evaluation Model (EM) for 17x17 standard fuel resulted in a PCT of 2130°F for the limiting C_d=0.8 break. As shown in the Salem UFSAR Section 15.4.1.2, a subsequent "evaluation," done to account for NUREG-0630 burst and blockage effects, resulted in no F_Q reduction. An evaluation has been performed, based upon the 78EM, to consider the effects on the analysis of the increase in the allowable steam generator tube plugging to 3.5%.

Based upon the large break LOCA sensitivity to steam generator tube plugging documented in WCAP 8986,("Perturbation Technique for Calculating ECCS Cooling Performance", February 1977), a penalty of 14°F would be estimated for a 4-loop plant with 3.5% tube plugging. However, studies with the 1981 Evaluation Model which incorporates the NUREG-0630 burst and blockage fuel rod models, have shown a sensitivity to tube plugging which was roughly double that cited in WCAP-8986. Thus, to account for potential non-conservatism associated with the fuel rod models in the 78EM of the established sensitivity, a conservative estimate of the penalty would be 28°F.

However, the analysis was also done using fuel parameters which are now overly conservative. Accounting for the lower rod internal backfill pressure of the 17x17 standard fuel currently in Salem Unit 2 results in a PCT benefit larger (>100^{o}F) than the combined penalty of 28^{o}F associated with tube plugging and approximately 22^{o}F penalty associated with the loose parts blockage presented in Attachment 3. Thus, the net effect would be no increase in peak clad temperature. Consequently, the evaluation in Section 15.4.1.2 to address NUREG-0630 "Clad Swelling and Rupture Models for LOCA Analysis" remains applicable.

Furthermore, this evaluation using the 78EM is known to be conservative with respect to more recent, approved evaluation models. Although the 1981 Evaluation Model has shown a higher sensitivity to tube plugging than the 78EM, the 1981 EM with BASH has demonstrated an even lower sensitivity to tube plugging. Indeed, the effect of using the improved analytical modeling techniques in either of these more recent models in total would result in a net benefit and a less limiting PCT for the plant.

Based on the discussion given above, the increase in the allowable steam generator tube plugging level to 3.5% does not result in an increase in the peak clad temperature for Salem Unit 2. Therefore, this change is acceptable and the resulting peak clad temperature does not change the current margin and remains within the regulatory limits.

ATTACHMENT 3

LARGE BREAK LOCA SAFETY EVALUATION FOR SALEM UNIT 2 FOR UNRECOVERED FOREIGN PARTS IN RCS

I. BACKGROUND

During the Salem Unit 2 Fourth Refueling Outage, it was visually determined that a burnable poison rodlet round hold down nut and weld pin were not in place, during reactor core fuel loading and repositioning in core map position C-10, in a reload core Region 2 location. The round hold down nut affixes the burnable poison rodlet top end plug to the burnable poison rodlet positioning upper plate which locates the burnable poison rodlets within the 17 x 17 fuel assembly. The top end plug has a threaded end shank, .216 inches in diameter, that is slotted to a depth to receive the positioning plate through hole and the round hold down nut with an internal .216 inches in diameter threaded hole. The hold down nut is slotted to the same size as the rodlet top end plug threaded shank to receive a .375 inches long pin that is tack welded in the shank and nut upon completing the rodlet - plate - nut assembly.

During core reloading in this same refueling outage, a gamma measurement hand probe, Eberline Model HP-290 was lost in the reactor vessel. When the manipulator crane trolley was moved, the probe with its cable connector was torn off and fell into the reactor cavity. At that time, only three fuel elements were loaded. These elements were off loaded to the spent fuel pit.

Neither of these aforementioned objects were found and a decision was made to evaluate these objects as loose parts within the reactor coolant system (RCS).

A potential effect of these loose parts is partial blockage of flow to the fuel assemblies or blockage within fuel assemblies. The worst case effect of this blockage on the Large Break LOCA analysis for Salem Unit 2 is discussed below.

II. LARGE BREAK LOCA (FSAR CHAPTER 15.4.1)

The current licensing basis Large Break LOCA analysis for Salem Unit 2 was performed using the 1978 Westinghouse Evaluation Model for 17x17 standard fuel and resulted in a Peak Cladding Temperature (PCT) of 2130^{o}F for the limiting C_d=0.8 break. A subsequent "3-page evaluation", done to account for NUREG-0630 burst and blockage effects, resulted in no reduction to the analyzed FQ of 2.32.

To determine the effect of these loose parts on the Large Break LOCA analysis an evaluation has been performed which considers locations of the loose parts within the RCS which could impact the Large Break LOCA PCT calculations. The core flow area assumed in the evaluation reflects the current Salem Unit 2 core with Westinghouse 17x17 standard fuel.

The location which was determined to have the greatest potential to impact the Large Break LOCA transient was that which would block the reactor coolant flow through the active core fuel rod assemblies. The chemical and mechanical evaluation of these loose parts within the reactor vessel environment indicates that the plastic, rubber, aluminum and tin-lead alloy of the HP-290 probe will disperse into benign particles within the reactor coolant. The remaining metal components of the lost objects will remain intact; however, some deformation can be expected. An evaluation of the size of the remaining objects indicates that only the lock pin, the ground wire and the stainless steel central wire of the HP-290 probe have the potential to enter any of the fuel assembly subchannels. The remaining objects will not be able to pass through the fuel assembly bottom nozzles.

The current licensing basis analysis for Salem resulted in a peak clad temperature at the fuel rod burst elevation of 6 ft. In this case, the temperature excursion is driven by the heat release from the zirc-water reaction on the fuel rod cladding. The time of fuel rod burst is dependent upon the cooling provided by the positive and negative flow through the core during the blowdown portion of the transient. If the presence of the loose parts will cause the fuel rod to burst earlier than is currently predicted, an increase in PCT will result. To determine if the timing of fuel rod burst would be altered by the presence of the loose parts in the RCS, it was postulated that during the LOCA event, all the objects are lodged beneath the bottom nozzle except the lock pin, the ground wire and the HP-290 central wire which are lodged next to the hot rod in the hot assembly. In this situation flow through the subchannel adjacent to the hot rod is blocked at the core entrance and flow in the subchannel is impeded due to the presence of the lock pin and wire. The maximum effect of the flow impedance due to the presence of the lock pin and wire will occur if the objects are lodged at the burst elevation. In this case, the maximum flow redistribution resulting from the blockage will effect the local heat transfer at the burst elevation. The presence of the objects lodged in the fuel assembly bottom nozzle will have no effect on the timing of the fuel rod burst since full flow recovery will occur approximately 30 inches downstream of the blocked nozzle. Therefore, only the flow impedance in the subchannel has the potential to impact the PCT calculation. An evaluation of the maximum blockage within the subchannel at the burst elevation indicates that 36% of the subchannel or approximately 0.13% of the hot assembly would be blocked due to the presence of the lock pin and wire. A subchannel blockage of this magnitude was evaluated and found to potentially create a PCT increase of 22.21°F.

The licensing basis large break LOCA analysis for Salem was performed using fuel performance parameters which are now overly conservative. Accounting for the lower rod internal backfill pressure of the fuel currently in Salem Unit 2 results in a PCT benefit larger ($\geq 100^{o}F$) than the combined penalty of the approximately $22^{o}F$ penalty associated with the blockage and the $28^{o}F$ penalty due to the increase steam generator tube plugging presented in Attachment 2. Thus, the net effect would result in no increase in peak clad temperature.

Recent development of a Best-Estimate Large Break LOCA model and test performed to determine the effects of fuel assembly flow blockage have demonstrated that even large amounts of flow blockage (<90%) result in a PCT benefit. The benefit is related to breakup of the entrained water droplets which are present during a LOCA. However, current LOCA models developed in response to 10CFR50.46 and Appendix K to 10CFR50 do not have the sophistication to model non-equilibrium effects and the presence of entrained water droplets during blowdown. Thus, sensitivity studies based on Appendix K models, result in a calculated increase in PCT. Furthermore, the expected location of the loose parts during a LOCA would be at either the top or bottom of a grid depending upon the flow direction. The local power is lower and heat transfer is much higher in the region around grids than calculated by the Westinghouse Evaluation Models. Credit for these effects would offset the $22^{o}F$ penalty associated with the loose parts in the RCS.

Based on the discussion given above, the presence of the loose parts will not result in an increase in peak clad temperature for Salem Unit 2. Therefore, the continued operation of Salem Unit 2 with the loose parts in the RCS is acceptable. A calculated PCT of $2130^{o}F$ is low enough that there are no concerns for meeting the maximum local Zirconium water reaction limit of less than 17%, the core wide Zirconium water reaction of less than 1%, the ability to retain a coolable geometry as a result of thermal effects, or the ability to cool the reactor long-term.