TECHNICAL EVALUATION REPORT

PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION (B-69)

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

SALEM NUCLEAR GENERATING STATION UNITS 1 AND 2

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

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1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This Technical Evaluation Report (TER) documents an independent review of the Public Service Electric and Gas (PSE&G) compliance with the Nuclear Regulatory Commission's (NRC) IE Bulletin 80-04, "Analysis of a Pressurized Water Reactor Main Steam Line Break with Continued Feedwater Addition" [1], as it pertains to the Salem Nuclear Generating Station, Units 1 and 2. This evaluation was performed with the following objectives:

- o to assess the conformance of PSE&G's main steam line break (MSLB) analyses with the requirements of IE Bulletin 80-04
- o to assess PSE&G's proposed interim and long-range corrective action plans and schedules, if needed, as a result of the MSLB analyses.

1.2 GENERIC BACKGROUND

In the summer of 1979, a pressurized water reactor (PWR) licensee submitted a repo e NRC that identified a deficiency in the plant's original analysis of the containment pressurization resulting from a MSLB. A reanalysis of the containment pressure response following a MSLB was performed, and it was determined that, if the auxiliary feedwater (AFW) system continued to supply feedwater at runout conditions to the steam generator that had experienced the steam line break, containment design pressure would be exceeded in approximately 10 minutes. The long-term blowdown of the water supplied by the AFW system had not been considered in the earlier analysis.

On October 1, 1979, the foregoing information was provided to all holders of operating licenses and construction permits as IE Information Notice 79-24 [2]. Another facility performed an accident analysis review pursuant to receipt of the information in the notice and discovered that, with offsite electrical power available, the condensate pumps would feed the affected steam generator at an excessive rate. This excessive feed was not previously considered in the plant's analysis of a MSLB accident.

A third licensee informed the NRC of an error in the MSLB analysis for their plant. During a review of the MSLB analysis, for zero or low power at the end of core life, the licensee identified an incorrect postulation that the startup feedwater control valves would remain positioned "as is" during the transient. In reality, the startup feedwater control valves will ramp to 80% full open due to an override signal resulting from the low steam generator pressure reactor trip signal. Reanalysis of the events showed that opening of the startup valve and associated high feedwater addition to the affected steam generator would cause a rapid reactor cooldown and resultant reactor return-to-power response, a condition which is outside the plant design basis.

Because of these deficiencies identified in original MSLB accident analyses, the NRC issued IE Bulletin 80-04 on February 8, 1980. This bulletin required all PWRs with operating licenses and certain near-term PWR operating license applicants to perform the following:

"1. Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the aux [illegible] feedwater system and the impact of other energy sources, such [illegible]nuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

2. Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

    a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

    b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient.

3. If the potential for containment overpressure exists or the reactor-return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed."

1.3 PLANT-SPECIFIC BACKGROUND

PSE&G responded to IE Bulletin 80-04 in a letter to the NRC dated May 2, 1980 [3] and provided additional information in a letter to the NRC dated July 26, 1982 [4]. The information in References 3 and 4 has been evaluated along with pertinent information from the Salem Nuclear Generating Station Final Safety Anal..is Report (FSAR) [5] to determine the adequacy of the Licensee's compliance wi ..lletin 80-04.

2. ACCEPTANCE CRITERIA

The following criteria against which the Licensee's MSLB response was evaluated were provided by the NRC [6]:

1. PWR licensees' responses to IE Bulletin 80-04 shall include the following information related to their analysis of containment pressure and core reactivity response to a MSLB within or outside containment:

   a. A discussion of the continuation of flow to the affected steam generator, including the impact of runout flow from the AFW system and the impact of other energy sources, such as continuation of feedwater or condensate flow. AFW system runout flow should be determined from the manufacturer's pump curves at no backpressure, unless the system contains reliable anti-runout provisions or a more representative backpressure has been conservatively calculated. If a licensee assumes credit for anti-runout provisions, then justification and/or documentation used to determine that the provisions are reliable should be provided. Examples of devices for which provisions are reliable are anti-runout devices that use active components (e.g., automatically throttled valves) which meet the requirements of IEEE Std 279-1971 [7] and passive devices (e.g., flow orifices or cavitating venturis).

   b. A determination of potential containment overpressure as a result of the impact of runout flow from the AFW system or the impact of other energy sources such as continuation of feedwater or condensate flow. Where a revised analysis is submitted or where reference is made to the existing FSAR analysis, the analysis must show that runout AFW flow was included and that design containment pressure was not exceeded.

   c. A discussion of the ability to detect and isolate the damaged steam generator from continued feedwater addition during the MSLB accident. Operator action to isolate AFW flow to the affected steam generator within the first 30 minutes of the start of the MSLB should be justified. If operator action is to be completed within the first 10 minutes, then the justification should address the indication available to the operator and the actions required. Where operator action is required to prevent exceeding a design value, i.e., containment design pressure or specified acceptable fuel design limits, then the discussion should include the calculated time when the design value would be exceeded if no operator action were assumed. Where operator actions are to be performed between 10 and 30 minutes after the start of the MSLB, the justification should address the indications available to the operator and the operator actions required, noting that for the first 30 minutes, all actions should be performed from the control room.

d. Where all water sources were not considered in the previous analysis, an indication should be provided of the core reactivity change which results from the inclusion of additional water sources. A submittal which does not determine the magnitude of reactivity change from an original analysis is not responsive to the requirements of IE Bulletin 80-04.

2. If containment overpressure or a worsening of the reactor return-to-power with a violation of the specified acceptable fuel design limits described in Section 4.2 of the Standard Review Plan [8] (i.e., increase the core reactivity) can occur by the Licensee's analysis, the Licensee shall provide the following additional information:

   a. The proposed corrective actions to prevent containment overpressure or the violation of fuel design limits and the schedule for their completion.

   b. The interim actions that will be taken until the proposed corrective action is completed, if the unit is operating.

3. The acceptable input assumptions used in the licensee's analysis of the core reactivity changes during a MSLB are given in Section 15.1.5 of the Standard Review Plan [9]. The following specific assumptions should be used unless the analysis shows that a different assumption is more limiting:

Assumption II.3.b.:	Analysis should be performed to determine the most conservative assumption with respect to a loss of electrical power. A reactivity analysis should be conducted for a normal power situation as well as a loss of offsite power scenario, unless the licensee has previously conducted a sensitivity analysis which demonstrates that a particular assumption is more conservative.
Assumption II.3.d.:	The most restrictive single active failure in the safety injection system which has the effect of delaying the delivery of high concentration boric acid solution to the reactor coolant system, or any other single active failure affecting the plant response, should be considered.

Assumption II.3.g.: The initial core flow should be chosen such that the post-MSLB shutdown margin is minimized (i.e., maximum initial core flow).

The acceptable computer codes for the licensee's analysis of core reactivity changes are, by nuclear steam supply system (NSSS) vendor, the following: CESEC (Combustion Engineering), LOFTRAN (Westinghouse), and TRAP (Babcock & Wilcox). Other computer codes may be used, provided that these codes have previously been reviewed and found to be acceptable by the NRC staff. If a computer code is used which has not been reviewed, the licensee must describe the method employed to verify the code results in sufficient detail to permit the code to be reviewed for acceptability.

4. If the AFW pumps can be damaged by extended operation at runout flow, the licensee's action to preclude damage should be reviewed for technical merit. Any active features should satisfy the requirements of IEEE Std 279-1971. Where no corrective action has been proposed, this should be indicated to the NRC for further action and resolution.

5. Modifications to electrical instrumentation and controls needed to detect and initiate isolation of the affected steam generator and feedwater sources in order to prevent containment overpressure and/or unacceptable core reactivity increases must satisfy safety-grade requirements. Instrumentation that the operator relies upon to follow the accident and to determine isolation of the affected steam generator and feedwater sources should conform to the criteria contained in ANS/ANSI-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors" [10], and the regulatory positions in Regulatory Guide 1.97, Rev. 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" [11].

6. AFW system status should be reviewed to ensure that system heat removal capacity does not decrease below the minimum required level as a result of isolation of the affected steam generator and also that recent changes have not been made in the system which adversely affect vital assumptions of the containment pressure and core reactivity response analyses.

7. The safety-grade requirements (redundancy, seismic and environmental qualifications, etc.) of the equipment that isolates the main feedwater (MFW) and AFW systems from the affected steam generator should be specified. The modifications of equipment that is relied upon to isolate the MFW and AFW systems from the affected steam generator should satisfy the following criteria to be considered safety-grade:

- Redundancy and power source requirements: The isolation valves should be designed to accommodate a single failure. A failure-modes-and-effects analysis should demonstrate that the system is capable of withstanding a single failure without loss of function. The single failure analysis should be conducted in accordance with the appropriate rules of application of ANS-51.7/N658-1976, "Single Failure Criteria for PWR Fluid Systems" [12].

- Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [13].

- Environmental qualification: The isolation valves should satisfy the requirements of NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" [14].

- Quality standards: The isolation valves should satisfy Group B quality standards as recommended in Regulatory Guide 1.26 or similar quality standards from the plant's licensing bases.

3. TECHNICAL EVALUATION

The scope of work included the following:

1. Review the Licensee's response to IE Bulletin 80-04 against the acceptance criteria.

2. a. Evaluate the Licensee's MSLB analyses for the potential of overpressurizing the containment and with respect to the core reactivity increase due to the effect of continued feedwater flow.

   b. Evaluate the Licensee's proposed corrective actions and schedule for implementation if the findings of Task 2a indicate that a potential exists for overpressurizing the containment or worsening the reactor return-to-power in the event of a MSLB accident.

3. Prepare a TER for each plant based on the evaluation of the information presented for Tasks 1 and 2 above.

This report constitutes a TER in satisfaction of item 3. Sections 3.1 through 3.3 of this report state the requirements of IE Bulletin 80-04 by subsection, summarize the Licensee's statements and conclusions regarding these requirements, and present a discussion of the Licensee's evaluation followed by conclusions and recommendations.

3.1 REVIEW OF CONTAINMENT PRESSURE RESPONSE ANALYSIS

The requirement from IE Bulletin 80-04, Item 1, is as follows:

"Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow."

3.1.1 Summary of Licensee Statements and Conclusions

In regard to the review of the containment pressure response analysis for Salem Units 1 and 2, the Licensee stated [3]:

"Response to this item is already covered in our response to the Question 5.106... This response to Question 5.106 is valid for both Units 1 and 2 of the Salem Generating Station."

The following is a summary of the Licensee response in Question 5.106:

"The Auxiliary Feedwater System is actuated shortly after the occurrence of a steam line break...the mass addition to the faulted steam generator from the Auxiliary Feedwater System was conservatively determined by using the following assumptions.

a. The entire Auxiliary Feedwater System was assumed to be actuated at the time of the break and instantaneously pumping at its maximum capacity.

b. The affected steam generator was assumed to be at atmospheric pressure.

c. The intact steam generators were assumed to be at the safety valve set pressure.

d. Flow to the affected steam generator was calculated from the Auxiliary Feedwater System head curves assumptions b and c above and the system line resistances. The effects of flow limiting devices were considered.

e. The flow to the faulted steam generator from the Auxiliary Feedwater System was assumed to exist from the time of rupture until realignment of the system was completed.

f. The failure of auxiliary feedwater runout control was considered separately as a single failure.

The auxiliary feedwater system has not been changed in any way that would affect conclusions of the original analysis.

a. The [MSLB] analysis...used the following auxiliary feedwater flow rates:

   1. With runout protection operational, a constant auxiliary feed flow of 1840 gpm to the faulted steam generator.

   2. Failure of runout control was simulated by assuming a constant auxiliary feedwater flow of 2040 gpm to the faulted steam generator.

The above flow rates were held constant from time of break until realignment, which was assumed at ten minutes.

The auxiliary feedwater system is actuated shortly after the occurrence of a steam line break. In the analysis the auxiliary feedwater flow to the faulted steam generator was assumed to exist from the time of the rupture until realignment of the system was completed. The Auxiliary Feedwater System is manually realigned by the operator after 10 minutes. Therefore, the analysis assumes maximum auxiliary feedwater flow to a depressurized steam generator for full 10 minutes. The actions taken to terminate auxiliary feedwater to the faulted steam generator are discussed...below.

In the event a postulated main steam line break occurs, auxiliary feedwater to the affected steam generator must be terminated manually. Present design criteria allows ten minutes for the operator to recognize the postulated event and perform the necessary actions. However, the operator is expected to terminate auxiliary feedwater flow to the affected steam generator in much less time due to the amount of Class 1E indication provided to monitor plant conditions.

The information available to alert the operator of the need to isolate auxiliary feedwater to the affected steam generator is mounted on the control console in the control room. The pressure in each steam generator is monitored and displayed by two independent channels of instrumentation. Also, a bank of pen recorders indicates steam and feedwater flows for each steam generator; this allows the control room operator to readily view and compare the steam flow of one steam generator to the others.

The suction and discharge pressures of each auxiliary feedwater pump are indicated on the control console. The auxiliary feedwater flow indicators for each steam generator are mounted on the control console next to each other, allowing the operator to easily view and compare flows.

In addition to the above mentioned indications, high steam flow, low steam pressure, and steam-feed flow deviation conditions for each steam generator are alarmed on the main control console in the control room. Alarms for these conditions are also provided on the overhead annunciator.

Several failures can be postulated which would impair the performance of various steam break protection systems and therefore would change the net energy release from a ruptured line. Four different single failures were analyzed for each break condition. These were: 1) failure of a safeguards train; 2) failure of a main feed isolation valve; 3) failure of a main steam isolation valve; and 4) failure of auxiliary feedwater runout protection equipment.

The effect of these failures is to provide additional fluids which may be released to the containment via the break or reduce the heat removal capability of the containment safeguards systems.

Failure of the auxiliary feedwater isolation valve to close has not been considered. The maximum auxiliary feedwater flow that can be delivered to a faulted steam generator has been assumed in the anslysis for ten minutes with two cases being considered: 1) runout protection operational; 2) failure of runout protection. Only after ten minutes the operator takes action to isolate auxiliary feedwater to the broken steam generator. At that time if the remote controlled auxiliary feedwater isolation valve fails to close, the operator can trip the two auxiliary feedwater pumps feeding the broken steam generator until this valve or another in the line is manually closed."

In response to a request to provide information regarding operator response time, the Licensee stated [4]:

"Existing Westinghouse containment analyses use 10 minutes for isolation of a faulted steam generator and do not project containment peak pressure beyond such assumption. This ten minute isolation time criteria is the design basis for Westinghouse units, inclusive of Salem. In the event that the postulated main steam line break in the containment occurs, auxiliary feedwater to the affected steam generator is manually terminated by pushbutton operation in the control room. The control room operator, upon evaluating the alarms and indications symptomatic of this very recognizable accident, must depress the 'shut' pushbuttons for either one or two electrically operated control valves supplying the faulted steam generator. The number of valves to be closed depends upon the number of pumps feeding the affected steam generator at that time.

The instrumentation available to alert the operator of the need to isolate auxiliary feedwater to the affected steam generator is mounted on the control console in the control room. The pressure in each steam generator is monitored and displayed by several independent channels of instrumentation. Also, pen recorders indicate steam and feedwater flows for each steam generator; this allows the control room operator to readily view and compare the flows of one steam generator with the others.

The suction and discharge pressures of each auxiliary feedwater pump are indicated on the control console. The auxiliary feedwater flow indications for each steam generator are mounted on the control console next to each other, allowing the operator to easily view and compare flows.

In addition to the above mentioned indications, high steam flow, low steam pressure, and steam/feed flow deviation conditions for each steam generator are alarmed on the main control console in the control room. Alarms for these conditions are also provided on the overhead annunciator.

Based on the number of class 1E indications provided to monitor plant conditions, the number of alarms provided to annunciate this accident, and the minimal operator actions required to terminate auxiliary feedwater to the faulted steam generator, operator action within 10 minutes is easily achievable and is justifiable as a design base."

3.1.2 Evaluation

The Licensee's submittals [3, 4] concerning the containment pressure response following a MSLB and applicable sections of the Salem FSAR [5] were reviewed in order to evaluate whether the following portions of the acceptance criteria were met:

- o Criterion 1.a - Continuation of flow to the affected steam generator
- o Criterion 1.b - Potential for containment overpressure
- o Criterion 1.c - Ability to detect and isolate the damaged steam generator
- o Criterion 4 - Potential for AFW pump damage
- o Criterion 5 - Design of steam and feedwater isolation system
- o Criterion 6 - Decay heat removal capacity
- o Criterion 7 - Safety-grade requirements for MFW and AFW isolation valves.

The Salem Units 1 and 2 are virtually identical Westinghouse-designed four-loop plants.

In the event of a MSLB, the following functions provide the necessary protection:

- o Safety injection system actuation from any of the following:
  - a. Two-out-of-three low pressurizer pressure signals
  - b. High differential pressure signals between steam lines
  - c. High steam line flow in two main steam lines (one-out-of-two per line) in coincidence with either low-low reactor coolant system average temperature or low steam line pressure in any two lines
  - d. Two-out-of-three high containment pressure.
- o The overpower reactor trips (neutron flux and differential temperature) and the reactor trip occurring in conjunction with receipt of the safety injection signal.
- o Redundant isolation of the main feedwater lines: normal control action closes the main feedwater regulating valves. A safety

injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater inlet stop valves (safety-grade).

- o Trip of the fast acting steam line stop valves (designed to close in less than 5 seconds) on:

  a. High steam flow in two main steam lines in coincidence with either low-low reactor coolant system average temperature or low steam line pressure in any two lines

  b. High-high containment pressure.

Fast-acting isolation valves are provided in each steam line that will fully close within 10 seconds of a large break in the steam line. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would blow down even if one of the isolation valves fails to close.

The two motor-driven AFW pumps automatically start on:

- o Loss of offsite power
- o Loss of MFW flow
- o Low-low level in one steam generator
- o Safety injection system actuation.

The turbine-driven AFW pump will start on:

- o Loss of offsite power
- o Low-low level in two steam generators
- o 4-kV bus undervoltage.

Finally, the AFW system is equipped with a runout protection system which regulates the discharge flow of the motor-driven pumps and controls the turbine-driven pump governor to limit flow so that the pumps will not sustain damage from operation during a MSLB.

All of the equipment required to mitigate the MSLB accident is safety-grade and complies with the requirements IEEE Std 279-1971.

The environmental qualification of safety-related electrical and mechanical components is being reviewed separately by the NRC and is not within the scope of this review.

The review did not determine if the instrumentation upon which the operator relies to follow the accident and isolate the affected steam generator conforms to the criteria in ANS/ANSI-4.5-1980 [10] and Regulatory Guide 1.97 [11].

A spectrum of blowdowns covering four power levels and three different break sizes were evaluated by the Licensee. The three break sizes considered at each power level (0, 30, 70, and 102% of nominal) were: a full double-ended rupture upstream of the steam line flow restrictor, a full double-ended rupture downstream of the steam line flow restrictor, and the largest split rupture that will not result in generation of a steam line isolation signal from the primary plant protection equipment. In the analysis of the third (split) break, reactor trip, feed line isolation, and steam line isolation are generated by high containment pressure signals. Additionally, all blowdowns used in the analyses were assumed to consist of dry steam.

For each break condition, four different single failures were evaluated. These were (1) failure of a containment safeguards train, (2) failure of a main feed isolation valve, (3) failure of a main steam isolation valve, and (4) failure of the AFW runout protection equipment.

For both the limiting large break and small break analyses, failure of one of the containment spray trains proved to be the most limiting single failure. Both analyses used an AFW flow rate, assuming runout protection operational, of 1840 gpm to the affected steam generator until the flow was manually isolated 10 minutes into the accident. In the limiting large break case, a peak pressure of 39.1 psig occurred at 657 seconds after a 1.4-ft^2 break at 70% power. In the limiting small break case, a peak pressure of 42.8 psig occurred at 810 seconds after a 0.86-ft^2 split at 102% power. The peak pressure for both cases remained below the containment design pressure of 47 psig.

Sufficient indications and alarms are available to the operator to determine that a MSLB has occurred; once this determination has been made, the operator has to perform minimal actions to isolate AFW flow to the affected steam generators. It is conservative to assume that the operator will complete the required actions within the 10-minute time frame. Therefore, it is concluded that there is no potential for containment overpressurization.

3.1.3 Conclusion

The Licensee's responses [3, 4] and the Salem FSAR [5] adequately address the concerns of Item 1 of IE Bulletin 80-04. The containment pressure response analysis and the the design of the mitigating systems meet the NRC's acceptance criteria. Regarding Item 1, it is concluded that there is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition. The AFW pumps are protected from the effects of runout flow and therefore can be expected to carry out their intended function during the MSLB event.

3.2 REVIEW OF REACTIVITY INCREASE ANALYSIS

The requirement from IE Bulletin 80-04, Item 2, is as follows:

> "Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:
>
> a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,
>
> b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,
>
> c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient."

3.2.1 Summary of Licensee Statements and Conclusions

In regard to the reactivity increase resulting from a MSLB with continued feedwater addition, the Licensee stated [3]:

"We have reviewed the assumptions made for main and auxiliary feedwater flow as they apply to Salem Units 1 and 2 licensing basis steam line break transients. Several of the relevant assumptions used in all core transient analyses follow, and are further explained in the Salem Generating Station FSAR.

1. The reactor is assumed initially to be at hot shutdown conditions, at the minimum allowable shutdown margin.

2. For the Condition IV breaks, i.e., double-ended rupture of a main steam pipe, full main feedwater is assumed from the beginning of the transient at a very conservative cold temperature.

3. All auxiliary feedwater pumps are initially assumed to be operating, in addition to the main feedwater. The flow is equivalent to the rated flow of all pumps at the steam generator design pressure.

4. Feedwater is assumed to continue at its initial flow rate until feedwater isolation is complete, approximately 10 seconds after the break occurs, while auxiliary feedwater is assumed to continue at its initial flow rate.

5. Main feedwater flow is completely terminated following feedwater isolation.

Based on the manner in which the analysis is performed for Salem Units 1 and 2, the core transient results are very insensitive to auxiliary feedwater flow. The first minute of the transient is dominated entirely by the steam flow contribution to primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal-hydraulic transients in the core. The effect of auxiliary feedwater runout (or failure of runout protection where applicable) is minimal. Greater feedwater flows during the large steamline breaks serve to reduce secondary pressures, accelerating the automatic safeguards actions, i.e., steamline isolation, feedwater isolation and safety injection. The assumptions described above are therefore appropriate and conservative for the short-term aspect of the steamline break transient.

The auxiliary feedwater flow becomes a dominant factor in determining the duration and magnitude of the steam flow transient during later stages in the transient. However, the limiting portion of the transient occurs during the first minute, both due to higher steam flows inherently present early in the transient and due to the introduction of boron to the core via the safety injection system.

In conclusion, PSE&G and Westinghouse have evaluated the effect of runout auxiliary feedwater flows in the core transient for steamline break, and based on this evaluation, have determined that the assumptions presently made are appropriate for use as Salem licensing basis. The concerns outlined in the introduction to IE Bulletin 80-04 relative to 1) limiting core conditions occurring during portions of the transient where auxiliary feedwater flow is a relevant contributor to plant cooldown, and 2) incomplete isolation of main feedwater flow, are not representative of the Salem Generating Station, Units 1 and 2."

3.2.2 Evaluation

The Licensee's analysis of the core reactivity increase resulting from a MSLB with continued feedwater addition was reviewed in order to evaluate whether the following acceptance criteria were met:

- Criterion 1.c - Ability to detect and isolate the damaged steam generator
- Criterion 1.d - Changes in core reactivity increase
- Criterion 3 - Analysis assumptions.

The Licensee's response and the FSAR analysis of the reactivity increase resulting from a MSLB were reviewed. From that review, it was determined that the analysis is conservative in its assumptions and that the assumptions are in accordance with those in Acceptance Criterion 3, with the exception of considering runout AFW flow.

The worst-case analysis assumed complete severance of a pipe inside the containment at the outlet of the steam generator with the plant at no load conditions and offsite power available. This analysis determined that, although a return-to-power is predicted, there is no violation of the specified acceptable fuel design limits.

The Licensee's conclusion that the core transient for the MSLB is insensitive to runout AFW flow is valid for the following reasons:

- Early in the transient, the primary to secondary heat transfer rate (from the blowdown of the initial steam generator mass) is several orders of magnitude greater than that contributed by the additional AFW flow due to runout.
- Later in the transient (when the majority of the initial mass has blown down), AFW flow becomes a dominant factor in determining the magnitude and duration of the transient.
- The limiting core conditions will occur within the first minute due to the initial high cooldown rate contributing to the reactivity addition which is terminated by the introduction of boron into the core region.

Since the limiting core conditions occur before the AFW flow becomes a major contributing factor, it can be concluded that the core transient is insensitive to the contribution of AFW flow, and therefore the assumptions of the FSAR analysis remain valid.

3.2.3 Conclusion

The Licensee's response and FSAR adequately address the concerns of Item 2 of IE Bulletin 80-04. All potential sources of water were identified and, although a return-to-power is predicted, there is no violation of the specified acceptable fuel design limits. Therefore, the FSAR analysis of the reactivity increase resulting from a MSLB remains valid.

3.3 REVIEW OF CORRECTIVE ACTIONS

The requirement from IE Bulletin 80-04, Item 3, is as follows:

"If the potential for containment overpressure exists or the reactor-return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed."

3.3.1 Summary of Licensee Statements and Conclusions

The Licensee stated:

"Based on our response to items 1 and 2 above, potential for the containment overpressure does not exist and the potential for the reactor to return to power does not worsen with due considerations to the NRC

Bulletin 80-04. PSE&G has determined that no corrective actions are required at Salem Units 1 and 2 based on the NRC Bulletin 80-04."

3.3.2 Evaluation and Conclusion

The Licensee's analyses determined that neither a containment over-pressurization nor a reactor return-to-power with a violation of the specified acceptable fuel design limits would result from a MSLB. Therefore, it is concluded that no further action regarding IE Bulletin 80-04 is required of PSE&G for Salem Nuclear Generating Station Units 1 and 2.

4. CONCLUSIONS

Conclusions regarding Public Service Electric and Gas Company's response to IE Bulletin 80-04 with respect to Salem Nuclear Generating Station Units 1 and 2 are as follows:

- o There is no potential for containment overpressurization resulting from a main steam line break (MSLB) with continued feedwater addition.
- o The auxiliary feedwater pumps are protected from the effects of runout flow and therefore can be expected to carry out their intended function during the MSLB event.
- o All potential water sources were identified and, although a reactor return-to-power is predicted, there is no violation of the specified acceptable fuel design limits. Therefore, the Final Safety Analyis Report reactivity increase analysis remains valid.
- o No further action is required by the Licensee regarding IE Bulletin 80-04.

5. REFERENCES

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