TECHNICAL EVALUATION REPORT

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1

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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.

Mr. F. W. Vosbury contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This technical evaluation report (TER) documents the Franklin Research Center's (FRC) review of the Southern California Edison Company's (SCE) response to 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 San Onofre Nuclear Generating Station Unit 1. This evaluation was performed with the following objectives:

- o to assess the conformance of SCE's main steam line break (MSLB) analyses with the requirements of IE Bulletin 80-04
- o to assess SCE'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 report to the 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 flow 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]. A second facility performed an accident analysis review after receiving 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 had not previously been considered in the plant's analysis of a MSLB accident.

Another licensee informed the NRC of an error in the MSLB analysis for a third plant. During a review of the MSLB analysis, for zero or low power at the end of core life, this 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 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.

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, Another licensee informed the NRC of an error in the MSLB analysis for a third plant. During a review of the MSLB analysis, for zero or low power at the end of core life, this 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.

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   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

SCE responded to IE Bulletin 80-04 in a letter to the NRC dated May 12, 1980 [3]. SCE's response advised the NRC that the initial scoping studies of the containment pressure and reactivity response to a MSLB would be available by May 16, 1980 and a final containment response analysis would be available by October 1, 1980. The NRC was presented with the preliminary results regarding the MSLB analysis in a meeting on May 13, 1980, and these results were formally submitted to the NRC by letter on June 10, 1980 [4]. On May 19, 1980 [5], SCE indicated that a review of the core reactivity respc e was required and that the results would be provided by July 1, 1980. In a letter dated July 16, 1980 [6], SCE advised the NRC that its response to Item 2 of IE Bulletin 80-04 would be provided by August 1, 1980. SCE forwarded its analysis of core reactivity on August 4, 1980 [7]. In October 1980, SCE submitted the "Reload Safety Evaluation, San Onofre Nuclear Generating Station Unit 1, Cycle 8" [8] which contained an analysis of the core reactivity response to a MSLB. On March 6, 1981 [9], SCE provided the NRC with the containment pressure response analysis for San Onofre Unit 1.

2. ACCEPTANCE CRITERIA

The following criteria against which the Licensee's MSLB response was evaluated were provided by the NRC [10]:

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 [11] 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. The justification should address the indication available to the operator and the actions required, particularly those outside the control room. If operator action is required to prevent exceeding a design value, i.e., containment design pressure or departure from nucleate boiling ratio (DNBR), then the discussion should include the calculated time when the design value would be exceeded if no operator action were assumed.

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 the licensee's analysis shows that containment overpressure or a reactor-return-to-power with a DNBR less than 1.32 (1.30 for Tong correlation) can occur, then the licensee shall provide the following additional information:

   a. The proposed corrective actions to preclude overpressure or reactor-return-to-power and a schedule for completion of those actions.

   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 [12]. 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 (CE), LOFTRAN (Westinghouse), and TRAP (B&W). 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. The 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" [13], 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" [14].

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 valves that isolate the main feedwater (MFW) and AFW systems from the affected steam generator should be specified. Isolation valves that are relied upon to isolate the MFW and AFW systems from the affected steam generator should satisfy the following criteria to be considered safety-grade:

   o 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" [15].

- Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [16].
- 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" [17].
- 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

Under contract to the NRC, the scope of work included the following:

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

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

   b. Evaluation of 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. Preparation of a TER for each plant based on the evaluations in Tasks 1 and 2 above.

This report constitutes a TER in satisfaction of Task 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."

a. Summary of Licensee Statements and Conclusions

In regard to the review of the containment pressure response analysis forwarded on March 6, 1981 [9], the Licensee stated the following:

"Preliminary studies to determine the effect of automatic initiation of the AFWS [AFW system] on the containment response to a MSLB inside containment were submitted by letter to the NRC dated June 10, 1980 [4] from K. P. Baskin to D. M. Crutchfield. The results of these analyses indicated potential for high containment pressures and temperatures from the conservatively postulated MSLB using current licensing basis assumptions. In the base case analyses auxiliary feedwater was assumed to be manually initiated at 10 minutes following a MSLB at a flow rate of 250 gpm. A sensitivity analysis of the full power case (1000 gpm for 90 seconds, 500 gpm until 10 minutes, 250 gpm thereafter) resulted in less than 1 psi increase in peak containment pressure.

Additional analyses have been performed to assess the effect of different initial power levels, break sizes, and automatic initiation of the AFWS to confirm that the limiting case has been analyzed and that the effect of automatic initiation of the AFWS is not significant."

The base case MSLB analyses assumed a double-ended steam line break and determined that containment pressure would peak at 53.0 psig in the 0% power case and at 47.6 psig in the full power case. The Licensee's additional analyses determined the following:

"The full power case with automatic AFW initiation is more limiting than the intermediate or zero power cases because SG [steam generator] depressurization is slower, due to higher primary stored energy and heat transfer capability, and hence, SG blowdown rates are higher for the full power case despite a smaller initial SG inventory.

The small break MSLB cases are less limiting than the corresponding double-ended MSLB's because the reduced break area limits the SG blowdown rate and extends the SG blowdown duration without significantly lengthening feedwater isolation time.

The full power case with automatic AFW initiation does not significantly increase peak containment pressure (50.0 to 50.5 psig) compared to the full power case with manual AFW initiation. While the addition of cold AFW contributes to SG depressurization, resulting in reduced SG blowdown rates, it also provides additional mass and energy release to containment.

The zero power case with automatic AFW initiation results in a calculated decrease in peak containment pressure (53.0 to 47.6 psig) as the additional mass and energy is released to containment at a rate which is less than the containment heat removal capability."

The Licensee concluded as follows:

"Thus automatic AFW initiation does not affect the MSLB results, i.e., significantly increase calculated containment pressures or temperatures.

In addition, the AFW flow rate of 500 gpm assumed for MSLB analysis is conservative considering the proposed AFWS design which incorporates a pump trip for runout protection of the motor-driven AFW pump to prevent excessive AFW flow and would bound any AFW flow actually obtained in the event of a large MSLB. In this case, the no load case analyzed previously assuming manual actuation of AFW at 10 minutes would become the limiting case. The acceptability of this case which bounds the results for all cases analyzed was documented in the June 10, 1980 submittal based on preliminary results and remains applicable."

Regarding the AFW pump's ability to remain operable after extended operation at runout flow, the Licensee stated the following:

"An automatic trip of the motor-driven auxiliary feedwater pump will be installed for pump runout protection on low discharge pressure.

The trip of the motor-driven auxiliary feedwater pump assures the availability of the pump for manual operation following depressurization of the steam generators."

b. <u>Evaluation</u>

The Licensee's submittal concerning containment pressure response analysis and applicable sections of the San Onofre Unit 1 Cycle 8 analysis [8] were reviewed in order to evaluate whether the following portions of the acceptance criteria were met:

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

The San Onofre Unit 1 steam system is unique in that the three individual main steam lines from the three steam generators feed into a common header which then branches into two lines which then exit the containment. There are no isolation valves inside the containment. In the event of a MSLB inside containment, all three steam generators would blow down.

In Reference 4, the Licensee conducted a preliminary analysis to determine the containment pressure response to a MSLB. This analysis determined that, with AFW manually initiated at 10 minutes, the limiting accident produced a peak containment pressure of 53 psig at 378 seconds. The Licensee then performed an additional analysis [9] with automatically initiated AFW to verify that the limiting case had been analyzed.

San Onofre Unit 1 is not equipped with a system to monitor the secondary side of the steam generators and isolate the steam side in the event of a MSLB. The safety injection system does not monitor any secondary parameters. High containment pressure (2 psig) initiates the safety injection actuation signal (SIAS). The SIAS trips the reactor, isolates the main feedwater (MFW) system, and aligns the MFW pumps for low pressure injection of borated water into the reactor coolant system. Since the SIAS causes the safety-grade MFW isolation valve, the MFW block valve, the MFW flow control valve, and the MFW flow control bypass valve in each line to shut within 8 seconds, a single active failure of any of these valves would not cause additional feedwater to enter the steam generators through the MFW system. The compliance of the SIAS circuitry and components with IEEE Std 279-1971 was not reviewed. 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 MSLB analysis conservatively assumes 125% (full power) or 5% (zero power) of nominal feedwater flow for 10 seconds after a MSLB occurs and before MFW isolation.

Low steam generator level in two out of three steam generators automatically initiates the AFW system. The motor-driven pump is expected to trip on low discharge pressure. The turbine-driven pump will lose its steam supply when the steam generators depressurize but, for purposes of the MSLB analysis, a supply pressure of 50 psig was assumed, corresponding to the approximate minimum containment pressure needed to deliver a conservative AFW flow rate of 250 gpm.

A review of the AFW pump circuity revealed that the low discharge pressure switch was not single-failure-proof. It must be assumed that the motor-driven pump will not trip and will deliver an additional 250 gpm to the steam generators. The Licensee's analysis does assume an initial flow rate of 500 gpm, and the Licensee states the following:

> "...the AFW flow rate of 500 gpm to the MSLB analysis is conservative considering the proposed AFWS design which incorporates a pump trip for runout protection of the motor-driven AFW pump to prevent excessive AFW flow and would bound any AFW flow actually obtained in the event of a large MSLB."

Because of the potential for a single failure of the low discharge pressure switch, the Licensee's assumption that the pump trip can be relied on to reduce the AFW flow rate is invalid. However, the 500-gpm flow rate would bound any AFW flow obtained during a MSLB because of the conservatism of the assumed flow rate of the turbine-driven AFW pump.

In the full-power, double-ended MSLB, the peak containment pressure, 50.5 psig, is reached in 145 seconds (the design pressure is 51.0 psig), after which time the containment heat removal rate is greater than the energy addition rate and the containment pressure starts to decrease. The Licensee's analysis assumes that after 10 minutes the operator will reduce AFW flow to 250 gpm to limit the cooldown rate.

The Licensee's assumption that the operator will be able to isolate auxiliary feedwater to the steam generators after 10 minutes may not be conservative for this analysis. In light of studies of operator responses to stressful situations (NUREG-1278 [18]), it cannot be expected that the operator will perform the proper corrective actions during the first 30 minutes of the accident and, therefore, it cannot be expected that the operator would be able to reduce the AFW flow to 250 gpm until approximately 30 minutes after the initiation of the accident. However, by the 10-minute point, the pressure would already have peaked and the containment heat removal rate would exceed the energy addition rate. Therefore, failure to reduce the AFW flow will reduce the rate of containment depressurization but will not affect the previously calculated containment pressure peak. The qualifications of the instrumentation that the operator relies upon to follow the accident and to determine isolation of the steam generators and feedwater sources was not reviewed.

A review of the Licensee's containment pressure response analysis determined that the analysis conservatively bounds the potential for continued feedwater addition and verified that containment overpressurization would not occur in the event of a MSLB.

The Licensee's statement that the trip of the motor-driven AFW pump would ensure the availability of the pump for operation following steam generator depressurization is not valid, since the runout protection circuit does not meet the requirements of IEEE Std 279-1971.

c. Conclusions and Recommendations

SCE's MSLB containment pressure response analysis adequately addresses the concerns of Item 1 of IE Bulletin 80-04 with regard to the potential for containment overpressurization resulting from continued feedwater addition. The analysis demonstrated that the peak containment pressure did not exceed the design pressure when continued feedwater addition was assumed.

The runout protection circuit for the motor-driven AFW pump does not meet the criteria of IEEE Std 279-1971 and therefore cannot be relied upon to protect the pump from damage due to operation at runout flow.

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."

a. Summary of Licensee Statements and Conclusions

In regard to the reactivity increase resulting from a MSLB with continued feedwater addition, the Licensee stated the following in Reference 7:

"In response to our request, Westinghouse reviewed the previous analysis of core response following a main steam line break for San Onofre Unit 1. The results of the review showed that no main or auxiliary feedwater had been been assumed in the previous analysis. Subsequently, Westinghouse performed a reanalysis of this event. The cases reanalyzed were a main steam line break (complete severance of a pipe) outside containment at no load conditions with offsite power available, and an accidental depressurization of the main steam system associated with the inadvertent opening of a single steam dump, relief, or safety valve with offsite power available. These cases conservatively assumed main feedwater flow addition until main feedwater isolation on the safety injection signal and auxiliary feedwater runout flow initiated coincident with the event. The results of the reanalysis confirmed that the main steam line break transient results for these cases are very insensitive to continued feedwater addition for San Onofre Unit 1. It is expected that the results for other no load cases previously analyzed and full load cases (previously shown to be less limiting) would also be insensitive to continued feedwater addition based on Westinghouse generic studies.

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 is minimal. The primary side pressure, on which the low pressurizer pressure safety injection signal is based, decays at a slightly faster rate with the addition of auxiliary feedwater. This accelerates the safety injection signal actuation

(< 0.5 seconds sooner) as well as allowing a slightly greater safety injection flowrate with the faster pressure decay. The overall results are, therefore, negligibly impacted with the addition of auxiliary feedwater flow.

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.

Hence, the conclusions documented in the previously submitted main steam line break core response analysis for San Onofre Unit 1 remain valid and applicable."

b. Evaluation

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

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

The Cycle 8 analysis [8] of the reactivity increase resulting from a MSLB was reviewed, and it was determined that the analysis is conservative in its assumptions and the assumptions are in accordance with those in Acceptance Criterion 3.

The Cycle 8 worst-case analysis assumed a doubled-ended MSLB with offsite power available and continued nominal MFW flow at full power plus an additional 10% flow to simulate AFW runout flow.

The peak core thermal power achieved during the transient was 43.1% of full power. The minimum DNBR was greater than 1.30.

The Cycle 8 analysis verifies the conclusion of the previous analyses [19] (which did not include the addition of MFW or AFW runout flow) that the DNBR remains greater than 1.30.

The Licensee's conclusion that AFW flow is not a dominant factor in determining the peak power/reactivity attained is valid because the limiting portion of the accident will occur within the first minute because the high steam flow from the break will cause a high cooldown rate which will add significant reactivity to the core. Since the initial AFW flow will be several orders of magnitude less than the initial steam flow, it will not contribute a significant amount of reactivity to the core. By the time the AFW flow becomes a significant contributing factor to the cooldown rate, the safety injection system will have flooded the core with a high concentration of boron and effectively shut down the core.

c. Conclusion

The Licensee's response and Cycle 8 analysis adequately address the concerns of Item 2 of IE Bulletin 80-04. All potential sources of water were identified and were included in the analysis. A return to power occurs, but the DNBR remains greater than 1.30. Therefore, the Cycle 8 analysis remains valid and no further action is required.

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."

a. Summary of Licensee Statements and Conclusions

The Licensee did not propose any corrective actions in regard to Item 3 of IE Bulletin 80-04.

b. Evaluation

No corrective action by SCE is required for Item 1 of IE Bulletin 80-04, relating to the potential for containment overpressure, or for Item 2, relating to the reactivity increase resulting from a MSLB.

Corrective action by the Licensee is required in regard to the potential for damage to the motor-driven AFW pump, because the runout protection circuit does not meet the requirements of IEEE Std 279-1971.

c. Conclusion and Recommendations

The Licensee's analysis determined that containment overpressurization or a worsening of a reactor return to power with a DNBR less than 1.30 would not result from a MSLB. Therefore, no further action by SCE is required in regard to the analyses performed in response to IE Bulletin 80-04.

The Licensee should provide the NRC with (1) corrective actions proposed to prevent potential damage to the motor-driven AFW pump due to operation at runout flow and (2) a schedule for completion of those actions. In addition, if San Onofre Unit 1 is in operation, the NRC should be provided with the Licensee's proposed interim actions to be taken until the proposed corrective actions are completed.

4. CONCLUSIONS

Conclusions regarding SCE's response to IE Bulletin 80-04 with respect to San Onofre Nuclear Generating Station Unit 1 are as follows:

- There is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition. Thus, there are no electrical requirements to detect or isolate auxiliary feedwater flow.
- All potential water sources were identified. Although a reactor return-to-power occurs, the DNBR remains greater than 1.30 throughout the transient. Therefore, the previous reactivity increase analysis remains valid.
- No further action by SCE is required regarding analyses performed in response to Items 1 and 2 of IE Bulletin 80-04.
- The AFW pump runout protection circuit does not meet the requirements of IEEE Std 279-1971. Failure of the circuit could result in pump damage from runout flow.
- SCE should provide (1) proposed corrective and interim actions required to prevent damage to the motor-driven AFW pump due to operation at runout flow and (2) a schedule for completion of those actions.

5. REFERENCES

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San Onofre Nuclear Generating Station Unit 1
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