TECHNICAL EVALUATION REPORT

# PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR POWER PLANT UNIT 3

NRC DOCKET NO. 50-286

NRC TAC NO. 46840

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 5

FRC TASK 131


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May 25, 1983


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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 an independent review of the Power Authority of the State of New York's (PASNY) 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 Indian Point Nuclear Power Plant Unit 3. This evaluation was performed with the following objectives:

- to assess the conformance of PASNY's main steam line break (MSLB) analyses with the requirements of IE Bulletin 80-04
- to assess PASNY'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 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 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,

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

The Power Authority of the State of New York responded to IE Bulletin 80-04 in a letter to the NRC dated May 8, 1980 [3] and provided additional information in a letter dated March 31, 1983 [4]. The information in References 3 and 4 has been evaluated along with pertinent information from the Indian Point Nuclear Power Plant Unit 3 Updated Final Safety Analysis Report (FSAR) [5] to determine the adequacy of the Licensee's compliance with IE Bulletin 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 the 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 Licensee 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 in 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 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."

### 3.1.1 Summary of Licensee Statements and Conclusions

In regard to the review of the containment pressure response analysis, the Licensee stated [4]:

"An evaluation of the transient response for the NSSS and the containment was performed for the single active failure of the main feedwater system feedwater control valve (FCV) during a main stream line break (MSLB) as a function of initial reactor power level, break size and main feedwater addition. The containment pressure response was detemined by the CONTEMPT code using the blowdown data calculated by the improved RETRAN-02, MOD-2 code for the NSSS transient response.

The methodology used in the evaluation of the MSLB with continued feedwater addition, is described as follows:

1. For different plant power levels, the main feedwater flow characteristics were determined for steam generator pressure dependence.

2. Various break types and sizes were selected.

3. A matrix was used to determine the three worst cases that produced the highest steam blowdown inventory, which were then analyzed for the containment pressure response.

The single active failure assumed is that the FCV fails in an 'as is' position. This assumption represents the worst possible single active failure that leads to a maximum feedwater flow into the steam generator.

The IP3 plant specific feedwater flow characteristics were developed for various initial power levels. The feedwater flow dependence on the steam generator pressure was determined for the 'as is' failure of the FCV where the feedwater flow would increase as the generator pressure decreases.

The three worst blowdown cases that were determined by the RETRAN results were evaluated for their containment response. These cases were:

1. The three square foot split break at hot zero power with the large initial generator water inventory, with auxiliary feedwater flow (main feedwater is not used at HZP conditions).

2. The three square foot split break at 100% power initial conditions, where the main feedwater flow reached the boiler feedpump runout conditions within the first minute of the transient.

3. A full break upstream of the flow restrictor at 70% power initial conditions, where the maximum feedwater flow continued throughout the entire transient without runout conditions being reached. This maximum blowdown case is also the same maximum feedwater flow case, as stated above.

Of the three worst blowdown cases analyzed, the maximum containment pressure was calculated to be 44.6 psia. This value is well below the containment design pressure of 62.0 psia. Accordingly, the integrity of the containment is not violated during this postulated accident. It was found that the peak pressure occurred within a hundred seconds of the beginning of the transient. Manual operator action was not relied upon in the analysis."

In response to a question about the ability of the AFW pumps to remain operable during a MSLB, the Licensee stated [3]:

"Each motor driven auxiliary feed pump is provided with a discharge pressure sustaining control system to prevent the pump from 'running out' on its curve. Runout flow conditions on the auxiliary feedwater pumps are also precluded by procedural requirements to maintain the auxiliary feedwater flow regulating valves in a throttled position. Should failure of the runout protection system result in the inoperability of the motor driven AFW pump feeding the damaged steam generator, the other motor driven AFW pump feeding intact steam generators and the steam driven AFW pump will remain operable and be available for maintaining the plant in a safe shutdown condition following the transient."

### 3.1.2 Evaluation

The Licensee's submittals [3, 4] concerning the containment pressure response following a MSLB and applicable sections of the Indian Point Unit 3 FSAR [5] 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 steam and feedwater isolation system
- Criterion 6 - Decay heat removal capacity
- Criterion 7 - Safety-grade requirements for MFW and AFW isolation valves.

Indian Point Unit 3 is a Westinghouse-designed, 4-loop, 3025-MWt plant.

The following systems provide the necessary protection against a steam pipe rupture:

- Safety injection system actuation on:

  a. two out of three (2/3) pressurizer low pressure signals

  b. 2/3 high differential pressure signals between any two steam generators

  c. high steam line flow in two out of four main steam lines (1/2 per line) in coincidence with either low reactor coolant system average temperature (2/4) or low steam line pressure (2/4)

  d. 2/3 high containment pressure signals

  e. 2/3 high-high containment pressure signals (two sets of 2/3).

- The overpower reactor trips (nuclear flux and differential temperature) and the reactor trip occurring upon actuation of the safety injection system.

- Redundant isolation of the MFW lines. Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the MFW valves, any safety injection signal will rapidly close all MFW control valves, close the MFW pump discharge valves (60-120 seconds), and trip the main MFW pumps upon closure of the discharge valves.

- Trip of the fast-acting steam line isolation valves (designed to close in less than 5 seconds with no flow) on:

  a. high steam flow in any two steam lines in coincidence with either low reactor coolant system average temperature (two out of four) or low steam line pressure (two out of four)

  b. high containment pressure signals (2/3 plus 2/3 high-high containment pressure).

Each steam line has a fast-closing stop valve and a check valve. These eight valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For a break upstream of the stop valve in one line, closure of either the check valve in that line or the stop valves in the other lines will prevent blowdown of the other steam generator.

The AFW system includes one turbine-driven pump (800 gpm) and two motor-driven pumps (400 gpm each). The AFW pumps are automatically started on receipt of any of the following signals:

Steam-driven AFW pump

- Low-low water level in any two of the four steam generators
- loss of outside power concurrent with a unit trip

Motor-driven AFW pumps

- low-low water level in any steam generator
- automatic trip of either main feed pump as indicated by loss of auto stop oil pressure in the turbine control system
- safety injection signal
- loss of outside power concurrent with a unit trip.

The motor-driven AFW pumps are protected against operating at runout conditions by a discharge pressure sustaining control. In addition, the AFW flow-regulating valves are throttled to prevent the pumps from experiencing runout flow.

Compliance of the above systems to safety-grade and IEEE Std 279-1971 [7] requirements 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 review did not determine if the instrumentation that the operator relies upon to follow the accident and isolate the affected steam generator conforms with the criteria in ANS/ANSI-4.5-1980 [10] and Regulatory Guide 1.97 [11].

The Licensee's analysis determined that the worst-case MSLB was that of a doubled-ended break upstream of the flow restrictor at 70% power. The worst-case single active failure was determined to be the failure of the MFW flow control valve (FCV) to shut, which allows full main feedwater flow to the steam generator until the MFW pump discharge valve shuts (120 seconds). This

results in a peak containment pressure of 44.6 psia at approximately 96 seconds, which is well below the containment design pressure of 62.0 psia. No operator action was assumed.

### 3.1.3 Conclusion

The Licensee's responses [3, 4] and the Indian Point Unit 3 FSAR [5] adequately address the concerns of Item 1 of IE Bulletin 80-04. The containment pressure response analysis and the design of the mitigating systems satisfy 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. In addition, since the AFW pumps are protected from experiencing runout flow conditions, the pumps will be able to carry out their intended function without damage.

## 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 [4]:

"An evaluation of the transient response for the NSSS and the containment was performed for the single active failure of the main feedwater system feedwater control valve (FCV) during a main stream line break (MSLB) as a function of initial reactor power level, break size and main feedwater addition. The containment pressure response was detemined by the CONTEMPT code using the blowdown data calculated by the improved RETRAN-02, MOD-2 code for the NSSS transient response.

The methodology used in the evaluation of the MSLB with continued feedwater addition, is described as follows:

1. For different plant power levels, the main feedwater flow characteristics were determined for steam generator pressure dependence.

2. Various break types and sizes were selected.

3. A matrix was used to determine the three worst cases that produced the highest steam blowdown inventory, which were then analyzed for the containment pressure response.

The single active failure assumed is that the FCV fails in an 'as is' position. This assumption represents the worst possible single active failure that leads to a maximum feedwater flow into the steam generator.

The IP3 plant specific feedwater flow characteristics were developed for various initial power levels. The feedwater flow dependence on the steam generator pressure was determined for the 'as is' failure of the FCV where the feedwater flow would increase as the generator pressure decreases.

The worst feedwater addition case was calculated to be the full break upstream of the main steam line flow restrictor at initial operating conditions at 70% rated power. For this worse case, the core remained subcritical throughout the transient because the negative Doppler reactivity feedback and the scram shutdown reactivity margin compensate for the positive reactivity feedback due to the moderator density increase caused by the overcooling transient."

### 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 Reference 4 analysis of the reactivity increase resulting from a MSLB was 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.

In the worst-case MSLB (70% initial power, a double-ended rupture upstream of the steam generator flow restrictor, and failure of the MFW FCV), after the scram resulting from the MSLB occurs, the core remains subcritical for the remainder of the transient and no return-to-power occurs.

### 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 there is no violation of 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 did not address corrective actions.

### 3.3.2 Evaluation and Conclusion

The Licensee's analysis determined that neither a containment overpressurization nor a reactor return-to-power with violation of the specified acceptable fuel design limits would occur from a MSLB. Therefore, it was concluded that no further action regarding IE Bulletin 80-04 is required of PASNY for Indian Point Unit 3.

## 4. CONCLUSIONS

With respect to Indian Point Unit 3, conclusions regarding the Power Authority of the State of New York's response to IE Bulletin 80-04 are as follows:

- 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.
- All potential water sources were identified, no reactor return-to-power is predicted, and there is no violation of the specified acceptable fuel design limits. Therefore, the Reference 4 MSLB reactivity increase analysis remains valid.
- No further action regarding IE Bulletin 80-04 is required.

## 5. REFERENCES


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