TECHNICAL EVALUATION REPORT

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

MAINE YANKEE ATOMIC POWER COMPANY
MAINE YANKEE ATOMIC POWER STATION

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CONTENTS

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 review of the Maine Yankee Atomic Power Company's 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 Maine Yankee Atomic Power Station. This evaluation was performed with the following objectives:

- o to assess the conformance of Maine Yankee's main steam line break (MSLB) analyses with the requirements of IE Bulletin 80-04
- o to assess Maine Yankee'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 Maine Yankee plant responded to IE Bulletin 80-04 in a letter to the NRC dated May 5, 1980 [3]. The information in Reference 3 has been evaluated along with pertinent information from the Maine Yankee Final Safety Analysis Report (FSAR) [4] to determine the adequacy of the Licensee's response to IE Bulletin 80-04.

2. ACCEPTANCE CRITERIA

The following criteria against which the Licensee's MSLB response was evaluated were provided by the NRC [5]:

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 [6] 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 [7]. 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).

*Other values for minimum DNBR may be acceptable if justified for certain fuel designs and DNBR correlations.

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 and 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 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" [8], 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" [9].

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:

   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" [10].

- o Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [11].
- o 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" [12].
- o 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 technical evaluatios report (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

The Licensee made the following statements regarding the impact of runout flow on the containment pressure response analysis [3]:

"The impact of runout flow from the auxiliary feedwater system on containment pressure was provided in Reference 3 [13], where it was determined to be bounded by the Reference 4 [14] analysis. Reference 4 [14] considered continuation of feedwater flow (8788 GPM 31% of total full power flow rate) to the damaged steam generator at flow rates in excess of both the main feedwater system and the runout flow of the auxiliary feedwater pumps. Although auxiliary feedwater was not directly considered, the sensitivity analyses performed as part of this study indicate that the key assumption with respect to containment pressure is how fast the intact steam generators are isolated by the action of the non-return and excess flow check valves and is rather insensitive to continuous auxiliary feedwater addition. In the worst case the peak containment pressure is 45 psig and occurs 112 seconds into the event.

The Reference 4 [14] analysis assumed continuation of main feedwater flow to the affected steam generator through the open main feedwater regulating bypass valve along with leakage past the closed main feedwater regulating valve (MFWRV). The continuation of feedwater and condensate flow to the affected steam generator through an open main feedwater regulating valve (i.e., failure of the MFWRV to close on turbine trip) has not been previously addressed. Continued addition of feedwater through a full open MFWRV has the potential for containment pressures above the containment design value of 55 PSIG. For a major break this would occur in approximately four minutes. The plant emergency procedures are being revised to direct the operator to trip main feedwater pumps and close the main feedwater MOVs if the MFWRV to the affected steam generator fails to close."

As part of the Maine Yankee plant's response to Item 3, the Licensee proposed the following corrective action in Reference 3:

"Failure of the MFWRV and/or MFWRV bypass valve to close or respond to their post turbine trip positions results in the potential for overpressurization of the containment and/or return to criticality following a main steam line break. As a result, the following design change has been initiated as a corrective action with implementation scheduled around June 1, 1981.

The design change would provide a safety grade closure signal from the low steam generator pressure excess flow check valve (EFCV) closure signal to both the MFWRV and the MFWRV bypass valves of the affected steam generator. This would isolate all main feedwater flow to the break. Redundancy would be provided by a safety-grade signal to trip all pumps in the main feedwater system (MFW, condensate, and heater drain pumps) on receipt of a coincident SIAS [safety injection actuation signal] and any low steam generator pressure EFCV closure signal. In conjunction with these changes, the auxiliary feedwater system would be modified to provide a safety grade closure signal from the low steam generator pressure EFCV closure signal to the associated auxiliary feedwater flow control valve in order to direct flow to the intact steam generators. These changes would prevent containment overpressurization and return to criticality for any steam line break transient."

Regarding the ability of the AFW pump to remain operable after extended operation at runout flow, the Licensee stated [3]:

"In the event of a steam-line rupture upstream of the excess flow check valve (EFCV), it is assumed that the operator isolates flow to the rupture stream generator within 10 minutes. As a result, the auxiliary feed pumps (AFW) may experience cavitation due to pump runout for a period 5 minutes. Maine Yankee has evaluated the effects to the auxiliary feed pump operating under these runout conditions and concludes that there will be no consequential loss of safety function capability.

At Maine Yankee, if the auxiliary feed pump is operating at runout conditions while discharging to a depressurized steam generator, moderate cavitation is expected to occur at the eye of the impeller and along the trailing edge of the impeller vane. Without pre-heat, there is no possibility of forming and sustaining large voids in the suction pipe and losing pump suction as a result. Since the pump is cooled by the water pumped, there is no threat of overheating. As the cavitational voids increase in size, the problem becomes self-correcting, because there is a rapid drop in pump efficiency, or flow, which in itself eliminates the voids. The result can be surging flow condition but not a loss of flow so long as the water pumped is cold. The effects of surging and collapsing voids are not expectd to cause damage since these forces are significantly less than the design capabilities of a boiler feed pump which, according to the manufacturer, are experienced at shut-off conditions.

In conclusion, if the MY [Maine Yankee] AFW pumps operate at runout; we expect noisy operation, a fall off of pump performance but no damage to the pump."

3.1.2 Evaluation

The Licensee's submittal concerning containment pressure response analysis and applicable references 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

- 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.

A review of the Reference 14 analysis determined that the Licensee's original MSLB containment pressure response analysis considered a main feedwater flow of 8778 gpm. This worst case produced a containment pressure of 45 psig at 112 seconds into the event. Containment design pressure is 55 psig.

A later analysis [13] was performed to determine the effect of automatically initiated auxiliary feedwater. This analysis modeled the MFW flow more explicitly and determined that a MFW flow of 2320 gpm was delivered to the affected steam generator through the MFW regulating valve and MFW regulating valve bypass valve. The analysis also determined that AFW runout flow of 1775 gpm would be directed to the affected steam generator for a total feedwater flow of 4095 gpm.

As discussed in Reference 16, the Licensee installed several modifications in the MFW and AFW systems in order to mitigate the consequences of a MSLB.

The following sequence of events occurs in the event of a MSLB:

- o reactor trip when steam generator pressure reaches 478 psia
- o excess flow check valve closure signal generated when steam generator pressure reaches 393 psia
- o excess flow check valves, MFW regulating valves, and MFW regulating valve bypass valves shut on excess flow check valve signal
- o SIAS generated at 1622 psia pressurizer pressure
- o MFW, condensate, and heater drain pumps trip on coincident excess flow check valve signal and SIAS
- o excess flow check valve signal causes AFW flow control valve to isolate affected steam generator
- o coincident excess flow check valve signal and SIAS override AFW pump start for 5 minutes.

Redundant solenoids on the MFW regulating valves, MFW regulating valve bypass valves, and AFW flow control valves help ensure operation when required.

When the unaffected steam generator repressurizes, the associated AFW flow control valve will open. The AFW pumps will start after a 5-minute delay and provide sufficient AFW flow to the unaffected steam generator to ensure that system heat removal capacity exceeds the minimum level required for decay heat removal after a MSLB.

The AFW automatic initiation system and the MFW isolation system are designed to safety-grade and IEEE Std 279-1971 requirements. 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 design of the MFW isolation system is such that, in the event of a single failure of one of the components, the MFW system is adequately isolated to prevent delivery of main feedwater to the steam generators.

The addition of the closure of the MFW regulating valve and the MFW regulating valve bypass valve on a excess flow check valve signal, along with the trip of the MFW pumps on a coincident SIAS, has significantly reduced the severity of a MSLB accident by reducing the feedwater flow (from the original analysis estimate of 8788 gpm to the current value of 1775 gpm). Therefore, the Reference 14 analysis bounds the current plant design and no potential for containment overpressurization exists.

In regard to operation of the AFW pumps at runout flow, it can be concluded that no damage would be incurred by the pumps since the forces experienced during operation at runout flow are significantly less than the design capabilities of the pumps.

3.1.3 Conclusions and Recommendations

The Licensee's response and supporting references 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, there is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition. In addition, since the forces experienced by the AFW pumps when subject to runout flow are less than the design capabilities of the pumps, the pumps will be able to carry out their intended function without incurring 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:

"The revised steam line break analysis submitted in Reference 5 [15] included the effects of automatic initiation of auxiliary feedwater and continuation of main feedwater flow through the main feedwater regulating valve bypass valve. No return to power is predicted to occur for either condition. The Reference 5 analysis did not include the continuation of main feedwater flow to the affected steam generator through an open MFW regulating valve or MFW regulating valve bypass valve should either fail to respond to its post-turbine trip position. Continued feedwater addition in either mode would result in a return to criticality due to the excessive reactor cooldown and negative moderator temperature coefficient at end of cycle.

As previously described in the response to Item 1, emergency procedures are being revised to direct the operator to trip all pumps in the main feedwater system should the MFW regulating valve and/or MFW regulating valve bypass valve fail to close or respond to their post-trip positions."

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:

- 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.

A review of the Reference 15 analysis determined that the worst-case accident assumptions were: hot zero power (1 MWT), double-ended MSLB, 400 gpm per steam generator post-trip MFW regulating valve bypass valve flow, and a 5-minute delay of AFW actuation. The AFW pumps then provided runout flow (1775 gpm) to the ruptured steam generator. To ensure that the safety injection pumps flood the core with sufficient boron to prevent a restart, a 5-minute delay was incorporated into the automatic actuation circuitry for the AFW system.

This worst-case analysis determined that no return-to-power occurred, the minimum subcritical margin attained was 0.042%, and the DNBR remained greater than 1.30. However, as noted by the Licensee, failure of the MFW regulating valve and MFW regulating valve bypass valve to respond to a post-turbine trip signal would allow continued feedwater addition and a return-to-power.

As discussed in Section 3.1.2, the Licensee modified the AFW and MFW systems to prevent continued feedwater addition. These modifications ensure that the Reference 15 analysis is conservative in its assumptions.

3.2.3 Conclusion

The Licensee's response and Reference 15 analysis adequately address the concerns of Item 2 of IE Bulletin 80-04. All potential sources of water are identified, no return-to-power occurs, and the DNBR remains greater than 1.30. Therefore, the Reference 15 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."

3.3.1 Summary of Licensee Statements and Conclusions

In response to Item 3, the Licensee stated [3]:

"Failure of the MFWRV and/or MFWRV bypass valve to close or respond to their post turbine trip positions results in the potential for overpressurization of the containment and/or return to criticality following a main steam line break. As a result, the following design change has been initiated as a corrective action with implementation scheduled around June 1, 1981.

The design change would provide a safety grade closure signal from the low steam generator pressure excess flow check valve (EFCV) closure signal to both the MFWRV and the MFWRV bypass valves of the affected steam generator. This would isolate all main feedwater flow, to the break. Redundancy would be provided by a safety-grade signal to trip all pumps in the main feedwater system (MFW, condensate, and heater drain pumps) on receipt of a coincident SIAS and any low steam generator pressure excess flow check valve closure signal. In conjunction with these changes, the auxiliary feedwater system would be modified to provide a safety grade closure signal from the low steam generator pressure excess flow check valve closure signal to the associated auxiliary feedwater flow control valve in order to direct flow to the intact steam generators. These changes would prevent containment overpressurization and return to criticality for any steam line break transient.

As an interim measure, an additional closure signal from a safety-grade source, the low steam generator pressure excess flow check valve closure signal, has been provided to the E/P converters controlling the MFWRV and MFWRV bypass valves as a back-up to the turbine-trip override signal. This signal will close both the MFWRV and the MFWRV bypass valve associated with the affected steam generator following a steam line break.

Interim action that will be implemented as soon as practicable involves upgrading the emergency procedures to direct the operator to trip the main feedwater pumps and close the main feedwater MOVs in the event that the MFWRV or MFWRV bypass valves fail to close or respond to their post-trip positions. Operator action to isolate main and auxiliary feedwater to a broken steam generator and direct feedwater flow to the intact steam generators is already included in the Maine Yankee emergency procedures."

3.3.2 Evaluation

The Licensee's proposed corrective actions to provide excess flow check valve closure signal to close the MFW regulating valve and MFW regulating valve bypass valve and to provide a trip of the MFW, condensate, and heater drain pumps on coincident excess flow check valve signal and SIAS will provide single-failure-proof isolation of the MFW system from the steam generators.

The modification of the AFW flow control valve that provides it with a closure signal from the excess flow check valve signal in order to direct flow to the intact steam generator is vulnerable to a single failure. The potential for a single failure of the AFW flow control valve requires that AFW runout flow be considered in the MSLB analysis.

In Reference 16, the Licensee proposed the installation of a second AFW isolation valve to make the AFW system single-failure proof. The exact details of the proposed system, however, were not available for this review. The compliance of the AFW automatic initiation system with safety-grade requirements of NUREG-0737 is being reviewed separately by the NRC.

The Licensee's proposed interim actions are adequate.

3.3.3 Conclusion and Recommendations

The Licensee's proposed corrective and interim actions are adequate and ensure that the assumptions used in the containment pressure response and return-to-power analysis remain conservative. No further action by the Licensee is required in regard to IE Bulletin 80-04.

4. CONCLUSIONS

Conclusions regarding Maine Yankee Atomic Power Company's response to IE Bulletin 80-04 for Maine Yankee Power Station Unit 1 are as follows:

- o There is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition.
- o The forces experienced by the AFW pumps when subject to runout flow are less than the design capabilities of the pump; therefore, they will be able to carry out their intended function without incurring damage during a MSLB.
- o All potential water sources were identified, no return-to-power occurs, and the DNBR remains greater than 1.30; therefore, the Reference 15 reactivity increase analysis remains valid.
- o The Licensee's proposed corrective and interim actions are adequate and ensure that the assumptions used in the MSLB analysis remain conservative.
- o No further action by the Licensee is required regarding IE Bulletin 80-04.

5. REFERENCES

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9-Jan-80

14. Maine Yankee Atomic Power Company
Justification for 2630 MWT Operation of the Maine Yankee Atomic Power Station
July 1977
YAEC-1132

15. D. E. Vandenburgh (MY)
Letter to Office of Nuclear Reactor Regulation
Subject: Revised Steam Line Break Analysis
17-April-80

16. R. H. Groce (MY)
Letter to R. A. Clark (NRC, ORB)
Subject: Feedwater and Auxiliary Feedwater Modifications
28-April-81