TECHNICAL EVALUATION REPORT

PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

BALTIMORE GAS AND ELECTRIC COMPANY

CALVERT CLIFFS NUCLEAR POWER PLANT UNITS 1 AND 2

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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 an independent review of Baltimore Gas and Electric Company's (BG&E) 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 Calvert Cliffs Nuclear Power Plant Units 1 and 2. This evaluation was performed with the following objectives:

- o to assess the conformance of BG&E's main steam line break (MSLB) analyses with the requirements of IE Bulletin 80-04
- o to assess BG&E'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

Baltimore Gas and Electric Company responded to IE Bulletin 80-04 in letters to the NRC dated February 12, 1980 [3] and May 21, 1980 [4] and provided additional information for this review in letters dated April 13, 1983 [5], May 17, 1983 [6], and June 8, 1983 [7]. The information in References 3-7 has been evaluated along with pertinent information from the Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report (FSAR) [8] 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 [9]:

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 [10] and passive devices (e.g., flow orifices or cavitating venturis).

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

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

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

2. If containment overpressure or a worsening of the reactor return-to-power with a violation of the specified acceptable fuel design limits described in Section 4.2 of the Standard Review Plan [11] (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 [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 (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 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 equipment that isolates the main feedwater (MFW) and AFW systems from the affected steam generator should be specified. The modifications of equipment 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].

- o Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [16].
- 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" [17].
- 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 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 <u>Summary of Licensee Statements and Conclusions</u>

In regard to the review of the containment pressure response analysis, the Licensee stated [6]:

"In response to the concerns raised by I&E Bulletin 80-04, BG&E recently performed several main steam line break (MSLB) analyses to evaluate the effects of a failed-open main feedwater regulating valve (MFRV) on peak containment pressure and core reactivity response. The purpose of these analyses was to identify the potential for exceeding containment design pressure or experiencing a return-to-power event, and to provide data that could be used to support any corrective actions that might be deemed necessary.

The results of these engineering-oriented analyses indicated that with the current feedwater system design, the consequences of a MSLB would be significantly worsened by the assumption of a failed-open MFRV.

On April 29, 1983 the Off-Site Safety Review Committee (OSSRC) reviewed these analytical results. The OSSRC determined that although a failure of the MFRV was not considered in the as-licensed design basis for Calvert Cliffs, an appropriate treatment of this non-safety grade component would have been to disallow any credit for its function. On the basis of this determination, the OSSRC concluded that the main steam line break analysis contained in the FSAR was erroneous and that the issue constituted an unreviwed safety question.

On April 30, 1983 a licensee event report was initiated pursuant to paragraph 6.9.1.8.h of the Technical Specifications to inform the NRC of our conclusions. Efforts were immediately begun to quantify the impact of a MFRV failure on the core reactivity and the containment pressure responses to a MSLB for the existing plant configuration. This information was required to support any decision with regard to the safety of continued operations.

An additional MSLB analysis was performed using FSAR methodology to determine the maximum peak containment pressure that would result from runout main feedwater flow to the affected steam generator. This case assumes that a full-size MSLB (guillotine rupture) occurs during full power operations. Other assumptions used in this analysis include:

a. The reactor coolant pumps are not manually tripped upon SIAS as required by the operating procedures;

b. Only half of the containment cooling and containment spray system capacity is available; and

c. The steam release contains 20% moisture carryover.

This analysis yielded a peak containment pressure of approximately 80 psig and peak temperature of 306°F.

A significant amount of conservatism is inherent in the analyses discussed above. The principal contributors of this conservatism are summarized below:

a. The reactor coolant pumps are assumed to continue running throughout the event. A more realistic assumption would be to assume that the pumps are manually tripped on SIAS in accordance with the operating procedures. Continued operation of the pumps results in higher heat transfer in the steam generators and consequently results in a higher peak containment pressure than if the pumps were tripped.

b. The analyses only assume credit for half the containment cooling capacity (coolers and sprays).

c. A delay time of 60 seconds is assumed for the delivery of water to the containment spray header. A more realistic assumption would be a delay time of 30 seconds. Earlier spray delivery would reduce peak containment pressure.

d. The main feedwater isolation valve is assumed to close in 80 seconds upon receipt of a steam generator isolation signal. Experience with surveillance testing of this valve indicates that it will close in 60 seconds, thus reducing the total amount of feedwater introduced to the affected steam generator.

e. Feedwater flow is assumed to continue at runout conditions during the period when the MFIV was closing. A more realistic treatment of feedwater flow would be to include the throttling effect of this valve as it closes. Consideration of this effect would decrease the total flow to the affected steam generator and would yield a lower peak containment pressure.

Based on the results of engineering analyses performed in response to Bulletin 80-04 concerns, BG&E is proceeding with feedwater system modifications that will provide at least two barriers to the continued addition of feedwater to the affected steam generator after a MSLB. Two modifications are currently being pursued, either of which should result in an acceptable peak containment pressure and core reactivity response.

The first modification includes an automatic trip of the feedwater system (main feed pumps, heater drain pumps, and condensate booster pumps) on high containment pressure.

The second modification involves decreasing the closure time for the MFIV to about 15 seconds. This modification requires installation of a new valve actuator or possibly replacement of the entire valve.

Current analyses indicate that the calculated peak containment pressure is very sensitive to the amount of feedwater that is introduced to the affected steam generator within the first minute of the event. Under calculated runout conditions, complete isolation must occur in less than 30 seconds. Under loss-of-forced-flow conditions; i.e., feedwater system pumps tripped, two-phase expansion (flashing) of water in the feed system piping will result in some continued flow to the affected steam generator.

The feedwater expansion phenomenon was conservatively modeled and included in the analysis. The results of the analysis indicate that, without rapid MFIV closure, the feedwater system must be tripped promptly to limit the total flow delivered to the affected steam generator. Although an automatic feature will be required to ensure that this function is properly performed in the event of a MSLB, we have temporarily modified the plant operating procedures to require a manual trip of the feedwater system upon receipt of a steam generator isolation signal (SGIS)."

In Reference 6, the Licensee discussed a further analysis which accounted for some of the conservatisms and incorporated the MFW pump trip at 6 seconds. A discussion of this analysis follows:

"Methodology

The MSLB containment response was calculated using methods similar to the circa 1972 methodology of the licensing analysis reported in the Calvert Cliffs FSAR. This analytical methodology assumes a large (6.3 ft^2) guillotine MSLB, with the mass/energy release modeled as pure steam (less credit for 20% moisture carryover) throughout the blowdown.

The analysis is based upon full load initial conditions, whereas the 1972 FSAR analysis predicted no load as the worst case (due to the higher intial steam generator water inventory). The full load case is limiting because the effects of runout MFW flow due to a failed-open MFRV are most adverse from full load conditions. At no load conditions, initial main feedwater (MFW) flow is negligible.

The SGN III computer code was used to calculate the mass/energy release. A steam bubble rise factor of 100 was used to obtain a pure steam release, and the blowdown from the pure steam release was reduced by 20% to credit moisture carryover. The blowdown results were coupled to the CONTRANS computer code to calculate the containment response.

Assumptions

A simplistic first approximation of 105% of full MFW flowrate, for 6.0 seconds, was assumed. The flashing of MFW (1,771 ft^3 at 436°F) which would result from the depressurizing steam generator was included in a conservative fashion assuming isentropic expansion. In addition, 4000 lbm of steam was included to account for the steam line inventory between the ruptured steam generator and its isolation valve. A reactor trip on high containment pressure actuation (at 4.75 psig) was credited. Credit was taken for both containment spray trains and all four containment atmosphere coolers. It should be noted that the assumption of full containment sprays does not have a significant impact on peak containment pressure due to the assumed delay time of 60 seconds for spray actuation. The analyses that will be provided in our follow-up submittal scheduled for June 1 [7] will assume the worse single failure in the containment spray/cooler systems. It is expected that the earlier spray actuation dicussed in the transmittal letter will more than compensate for the effects of the worst single failure. It was assumed that the reactor coolant pumps remain running throughout the event, which is consistent with the assumption that off-site power remains available. A loss of off-site power would trip the MFW train and the reactor coolant pumps, and would therefore minimize both the mass/energy added by MFW and the primary to secondary energy transfer. Such an assumption would have resulted in a less adverse containment response.

Results

The calculated peak containment pressure and temperature was 53 psig and 278°F, respectively, at 60 seconds into the event."

The results of the Reference 6 analysis indicated that an automatic MFW trip system would provide adequate protection against the potential effects of runout flow but that further attention needed to be focused on the areas of flashing in the MFW piping and delay times for containment spray actuation. Reference 7 addressed these concerns, as follows:

"Calculation of Main Feedwater Flow

Steam generator depressurization data from the latest available MSLB case was used as input to a detailed analysis of main feedwater flow during steam generator blowdown. The main feedwater transient analysis was performed using the RELAP-5 code. RELAP-5 explicitly models the effects of MFW flashing (which is terminated when isolation occurs at 80 seconds).

Since this anaysis yielded feedwater flowrates that were different than those that were used in the steam generator blowdown/depressurization case mentioned above, the steam generator blowdown and main feedwater transient analyses were reiterated to evaluate the impact on feedwater flowrates. The results of this iteration confirm the validity of the flowrate data (integral MFW mass input increased only 2%).

Steam Generator Blowdown Analysis

The SGN III computer code was used to calculate the mass/energy release. This NRC approved code is described in detail in Appendix 6B of CESSAR, 'Combustion Engineering Standard Safety Analyses Report,' Combustion Engineering, Inc., docketed December 19, 1973 (as amended). The SGN III code was, however, modified to provide a pure steam release, less credit for 20% moisture carryover from the 6.3 ft^2 break.

Containment Pressure and Temperature Analysis

The steam generator blowdown data discussed above yielded a peak containment pressure of 46.8 psig at 82 seconds with a peak temperature of 273°F. Except for the assumptions listed below, the containment pressure/temperature analysis used the same methods and assumptions in the 1972 MSLB analysis.

a. Containment Spray

Delay time in initiating containment spray was assumed to be 44 seconds from CSAS trip. This is the result of recent conservative calculations which account for: 1) completely dry spray header; 2) initial control valve actuator pressure of 60 psig; 3) actuator bleed valve flow coefficient; 4) actuator spring constant; 5) spray pump curve; 6) piping friction losses. One test case assuming a 60 second delay was run which resulted in containment pressure increase of only 0.08 psig, therefore, spray time delay has minimal effect (FSAR uses 35 seconds).

b. Off-Site Power

No loss of off-site power is assumed. This is consistent with the assumption of continued operation of the RCPs and feedwater pumps. As a result, no additional time delay was included in starting containment air coolers or spray pumps. (FSAR assumes loss of off-site power.)

c. Containment Heat Sinks

Revised heat sink data was taken from FSAR table 14.16-1. These include more metal and higher cooler duties. The FSAR MSLB used older heat sink data which was more conservative since vendor information and as-built information was not yet available. Table 14.16-1 represents accurate as-built conditions."

3.1.2 Evaluation

The Licensee's submittals [3-7] concerning the containment pressure response following a MSLB and applicable sections of the Calvert Cliffs Nuclear Power Plant Updated FSAR [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

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

Calvert Cliffs Nuclear Power Plant Units 1 and 2 are virtually identical, Combustion Engineering-designed, two-loop, 2700 MWt plants.

In the event of a MSLB, the following engineered safety features actuation systems (ESFAS) actuate to provide necessary protection:

- o Safety injection actuation signal (SIAS) occurs on:
  - a. two out of four (2/4) low pressurizer pressure signals (variable trip, minimum setpoint - 1750 psig)
  - b. 2/4 high containment pressure (4.75 psig).
- o Steam generator isolation signal (SGIS) actuates on 2/4 low steam generator pressure signals (570 psig). The SGIS initiates:
  - a. the closure of two fast-acting, safety-grade, main steam isolation valves (MSIVs) which are designed to shut in 6 seconds
  - b. the closure of two safety-grade main feedwater isolation valves (MFIVs) which are designed to shut in 80 seconds.
- o A reactor trip is generated by SIAS, high startup rate (2.6 decades/minute between 10^{-4} and 15% full power), low steam generator pressure, thermal margin/low pressure, or high power (107% maximum).
- o The reactor trip causes the main feedwater regulating valves (MFRVs) to ramp closed in 20 seconds and the feedwater bypass valves to open, permitting 5% MFW flow.
- o Containment spray actuation signal (CSIS) is generated upon receiving 2/4 high containment pressure signals (4.75 psig). The CSIS initiates the two containment spray trains with a total heat removal capacity of 240×10^6 Btu/hr and the containment cooling system which includes four air coolers, each with a heat removal capacity of 95×10^6 Btu/hr.

- In addition, the Licensee has committed [6] to install a trip of the MFW train (MFW pumps, heater drain pumps, condensate booster pumps) on receipt of a high containment pressure signal (4.75 psig).

Each AFW system includes one turbine-driven pump and one motor-driven pump (700 gpm each). The auxiliary feedwater actuation system (AFAS) is designed such that in the event of a MSLB, the AFW system will align itself to feed only the intact steam generator. The AFW pumps would not be subject to runout conditions since the AFW system aligns itself to a pressurized steam generator. The flow from one pump to the unaffected once-through steam generator (OTSG) is sufficient to ensure that the system heat removal exceeds the minimum level required for decay heat removal after a MSLB.

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 above systems are designed to safety-grade and IEEE Std 279-1968 requirements. The compliance of these systems with IEEE Std 279-1971 requirements was not reviewed.

The review did not determine whether 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 and Regulatory Guide 1.97.

The Licensee's analysis determined that the worst case conditions for a MSLB inside containment were:

- 100% full power
- 6.3 ft^2 guillotine break at steam generator nozzle
- 44-second delay in containment spray
- no loss of offsite power (reactor coolant pumps and MFW pumps remain on)
- 20% moisture carryover
- full MFW flow for 6 seconds (trip on high containment pressure).

The analysis determined that a peak pressure of 46.3 psig occurred at 82 seconds (containment design pressure is 50 psig).

A prior analysis reported in Reference 6 determined that a peak pressure of 80 psig was reached. This was due to the failure of a MFRV to close, allowing full MFW flow until the MFIV shut at 80 seconds. This determination prompted the Licensee to commit to install a trip of the MFW pumps on high containment pressure and to consider a modification to the MFIVs so that they would close in 15 seconds.

3.1.3 Conclusion and Recommendations

The Licensee's responses [3-7] and the Calvert Cliffs Nuclear Power Plant FSAR [8] adequately address the concerns of Item 1 of IE Bulletin 80-04. The containment pressure response analysis and the design of the engineered safeguards 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. The AFW pumps are adequately protected against a runout flow condition and therefore will be able to carry out their intended function without incurring damage in the event of a MSLB.

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 [6]:

"In response to the concerns raised by I&E Bulletin 80-04 BG&E recently performed several main steam line break (MSLB) analyses to evaluate the effects of a failed-open main feedwater regulating valve (MFRV) on peak containment pressure and core reactivity response. The purpose of these analyses was to identify the potential for exceeding containment design pressure or experiencing a return-to-power event, and to provide data that could be used to support any corrective actions that might be deemed necessary.

The results of these engineering-oriented analyses indicated that with current feedwater system design, the consequences of a MSLB would be significantly worsened by the assumption of a failed-open MFRV.

On April 29, 1983 the Off-Site Safety Review Committee (OSSRC) reviewed these analytical results. The OSSRC determined that although a failure of the MFRV was not considered in the as-licensed design basis for Calvert Cliffs, an appropriate treatment of this non-safety grade component would have been to disallow any credit for its function. On the basis of this determination, the OSSRC concluded that the main steam line break analysis contained in the FSAR was erroneous and that this issue constituted an unreviewed safety question.

On April 30, 1983 a licensee event report was initiated pursuant to paragraph 6.9.1.8h of the Technical Specifications to inform the NRC of our conclusions. Efforts were immediately begun to quantify the impact of a MFRV failure on the core reactivity and containment pressure responses to a MSLB for the existing plant configuration. This information was required to support any decision with regard to the safety of continued operations.

An additional MSLB analysis was performed using FSAR methodology to determine the maximum peak containment pressure that would result from runout main feedwater flow to the affected steam generator. This case assumes that a full-size MSLB (guillotine rupture) occurs during full power operations. Other assumptions used in this analysis include:

a. The reactor coolant pumps are not manually tripped upon SIAS as required by the operating procedures;

b. Only half of the containment cooling and containment spray system capacity is available; and

c. The steam release contains 20% moisture carryover.

The analysis that was performed to bound the core reactivity response used a methodology similar to that described in Reference [18] with the exception that a 1300 gpm auxiliary feedwater flow was assumed to initiate at 180 seconds (and was not isolated), a six-second MSIV closure time was assumed, and a 60-second MFIV closure time was used. Two hot-full-power cases were examined, one assuming a single stuck-out CEA and one assuming that all CEAs scram and both safety injection trains operate.

For the case with the stuck CEA, negative reactivity credit was assumed during return-to-power due to the local heating of the inlet fluid in the hot channel which occurs near the stuck CEA. This credit is based on three dimensional coupled neutronic thermal-hydraulic calculations performed with the HERMITE/PORC Code. As a result of the continued excess auxiliary feedwater flow, this analysis resulted in a peak return-to-power of about 10% at 400 seconds and would show some fuel failures.

For the case where all CEAs scram, the resulting core reactivity is about 0.5%. There is no return-to-power, and no fuel failures are predicted."

The Licensee then performed a similar analysis [6] assuming that the MFW pump trip was installed. The following discussion presents the results of the analysis:

"The SLB event with LOAC power on turbine trip is presented here. This case maximizes the moderator reactivity insertion, therefore maximizing the potential for the post trip return to power and consequent lower DNBRs. This occurs because LOAC power caused the Reactor Coolant Pumps (RCPs) to coastdown. The decreasing coolant flow is assumed to result in no flow mixing at the core inlet plenum. Thus, cold edge temperatures were used to calculate the moderator reactivity insertion.

The results of the analysis show that the HFP SLB causes the secondary pressure to rapidly decrease until a reactor trip on low steam generator pressure is initiated at 1.8 seconds. The CEAs drop into the core at 3.2 seconds and terminate the power and heat flux increases.

The Steam Generator Isolation Analysis Setpoint is reached at 1.8 seconds. At 2.7 seconds, the MSIVs begin to close and are completely closed at 8.7 seconds. The blowdown from the intact steam generator is terminated at this time.

A LOAC power on turbine trip is assumed to occur at 3.2 seconds. At this time, RCPs start coasting down and the diesel generators start coming on line. At 13.2 seconds, the diesel generators reach full speed and shutdown sequencer is initiated to load emergency systems. At 21.8 seconds the safety injection actuation signal is on and diesel generators switch from shutdown sequencer to LOCA sequencer to load emergency systems. At 26.8 seconds HPSI pump is loaded on line and at 56.8 seconds the HPSI pump reaches full speed.

An AFW isolation signal based on steam generator differential pressure is initiated at 2.9 seconds. At 22.9 seconds, the AFW block valve associated with the steam generator with lowest pressure (i.e., ruptured steam generator) is completely closed.

At 7.2 seconds, an AFAS is assumed based on low steam generator level. The steam admission valve to the AFW pump is opened at 12.2 seconds and the steam driven AFW pumps reaches full speed and delivers AFW flow to both steam generators at 16.7 seconds. At 22.9 seconds, AFW to the affected steam generator is terminated due to closing of its AFW block valve. The motor driven AFW pump is loaded on line by diesel generators at 51.8 seconds and is assumed to reach full speed and deliver AFW flow to the intact steam generator instantaneously.

The continued blowdown from the ruptured steam generator causes the core reactivity to increase. The ruptured steam generator blows dry at 265.4 seconds, which terminates the cooldown of the RCS. A peak reactivity of .41% at 312.0 seconds is obtained. A return to power of 7.09% of 2700 MWt occurs. Mcbeth DNBR values calculated remain above the 1.30 limit and therefore, it is concluded that critical heat fluxes are not exceeded. The decrease in moderator reactivity as well as negative reactivity inserted due to boron injection via the HPSI pump terminate the approach to criticality and the core becomes more subcritical."

The Licensee concluded [6]:

"The results of the steam line break event analyzed above show that the critical heat fluxes are not exceeded. It is therefore concluded that the consequences of a steam line break event with the worst single failure of a main feedwater component as well as the most restrictive single active failure in the safety injection system are within the criteria set for the event."

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.

The FSAR analysis of the reactivity increase resulting from a MSLB and References 3 through 6 were reviewed. From that review, it was determined that the analysis is conservative in its assumptions and that the assumptions are in accordance with those in Acceptance Criterion 3.

The Licensee's analysis, performed using the CESEC computer code, determined that the worst case conditions for the core reactivity response following a MSLB were:

- o 100% full power
- o Runout MFW flow until the MFIVs close at 80 seconds
- o 1300 gpm of AFW
- o Failure of one high pressure safety injection (HPSI) pump
- o End of cycle moderator temperature coefficient
- o 6.3 ft^2 guillotine steam line break
- o Loss of ac power.

This analysis determined that, in the worst case scenario, a peak reactivity of 0.41% would occur at 312 seconds with a return-to-power of 7.09% at 319 seconds. The decrease in moderator reactivity and the addition of negative reactivity due to boron injection via the HPSI system terminates the return-to-power and rapidly causes the core to become subcritical. No fuel design limits are exceeded.

3.2.3 Conclusion

The Licensee's response [6] and FSAR [8] adequately address the concerns of Item 2 of IE Bulletin 80-04. All potential sources of water were identified in the FSAR analysis, and although a reactor return-to-power is predicted, the specified acceptable fuel design limits are not exceeded. 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

Based on the results of engineering analyses performed in response to IE Bulletin 80-04 concerns, BG&E investigated two feedwater system modifications that would provide barriers to the continued addition of feedwater to the affected steam generator after a MSLB.

The first modification adds an automatic trip of the feedwater system (main feed pumps, heater drain pumps, and condensate booster pumps) on high containment pressure. BG&E has indicated that they are proceeding with the installation of the automatic trip system with expected completion at both units by November 17, 1983 [6].

The second modification investigated was the decreasing of the closure time for the MFIV to about 15 seconds. This modification would provide an additional barrier to the MFW during the MSLB; however, it was not considered in the analysis.

3.3.2 Evaluation

The Licensee's analysis determined that the installation of the automatic trip of the feedwater system on high containment pressure is necessary to ensure that neither a containment overpressurization nor a reactor return-to-power with a resultant violation of specified acceptable fuel limits would occur from a MSLB.

The Licensee stated that the condensate booster pumps and heater drain pumps motor control centers are located in the auxiliary building switchgear room, which is a safety-related equipment area (seismic Category 1).

The Licensee's commitment [6] to use a safety-grade trip source and safety-grade components up to the final trip relays and to substitute safety-grade final trip relays to trip the condensate booster pumps and heater drain pumps on high containment pressure meets the requirements of Acceptance Criterion 7.

The Licensee stated that the main feedwater pumps are turbine-driven. The steam supply is regulated by a hydraulically controlled valve; the hydraulic supply is controlled from a panel adjacent to the main feedwater pumps in the turbine building. The Licensee stated [19] that the safety-grade trip signal originates in the cable spreading room and activates a safety-grade relay in the control room which controls a solenoid valve in the steam valve's hydraulic controller. In addition, the Licensee reported that the main feedwater pumps would not deliver their rated supply without the booster pumps. The installation of the safety-grade trip of the main feedwater pumps partially complies with Acceptance Criterion 7 because the steam valve and hydraulic control panel are located in a non-seismic Category 1 area. However, the combination of:

- o the safety-grade trip of the condensate booster pumps and heater drain pumps
- o the inability of the MFW pumps to develop rated flow without the booster pumps (because of loss of net positive suction head)
- o the MFW pump steam supply shuts on loss of power to the hydraulic control panel solenoid

provides a realiable system to terminate or significantly limit the MFW flow to the ruptured steam generator, and therefore meets the intent of Acceptance Criterion 7.

3.3.3 Conclusion

The corrective action proposed by BG&E for Calvert Cliffs Nuclear Power Plant Units 1 and 2 is acceptable and no further corrective action is required.

4. CONCLUSIONS

With respect to the Calvert Cliffs Nuclear Power Plant Units 1 and 2, conclusions regarding Baltimore Gas and Electric Company's response to IE Bulletin 80-04 are as follows:

- o There is no potential for containment overpressurization resulting from a main steam line break (MSLB) with continued feedwater addition.
- o The auxiliary feedwater (AFW) pumps are adequately protected against a runout flow condition and therefore will be able to carry out their intended function without incurring damage in the event of a MSLB.
- o All potential water sources were identified and, although a reactor return-to-power is predicted, the specified acceptable fuel design limits are not exceeded; therefore, the FSAR reactivity increase analysis remains valid.
- o The corrective action proposed by the Licensee is acceptable and no further action is required regarding IE Bulletin 80-04.

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