TECHNICAL EVALUATION REPORT

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

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE UNIT 1

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FOREWORD

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

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

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This Technical Evaluation Report (TER) documents the review of Arkansas Power and Light Company's (APL) 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 Arkansas Nuclear One Unit 1. This evaluation was performed with the following objectives:

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

Arkansas Power and Light Company responded to IE Bulletin 80-04 in letters to the NRC dated May 27, 1980 [3] and July 9, 1980 [4]. Additional information was provided in a letter dated July 30, 1982 [5]. The information in References 3, 4, and 5 has been evaluated along with pertinent information from the Arkansas Nuclear One Unit 1 Final Safety Analysis Report (FSAR) [6] 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 [7]:

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 [8] 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.

1. 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 [9] (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 [10]. 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" [11], 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" [12].

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" [13].

- Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [14].
- 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" [15].
- 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 [3]:

"As a result of a main steam line break inside the reactor building of ANO-1, the steam pressure in both Once Through Steam Generators (OTSG) would decrease quite rapidly. The rate of depressurization would of course depend upon the break size. For a steam line break accident, the reactor power would increase with the decreasing average reactor coolant temperature as a result of a negative moderator coefficient. The ICS will cause insertion of control rods in an attempt to limit the reactor power to 102 percent. If the break were large, the reactor power increase could not be limited sufficiently by the ICS and a reactor trip would occur due to high neutron flux and/or low reactor coolant pressure.

Following the reactor trip, the turbine will trip and the ICS will run back the feedwater flow. Due to the low OTSG pressure in the affected OTSG, the safety grade Steam Line Break Instrumentation and Control System (SLBIC) would actuate, isolating the affected OTSG by closing the respective feedwater isolation valve and both main steam block valves. A SLBIC signal also opens the steam supply to the turbine driven emergency feedwater pump. As the affected OTSG boils dry, the emergency feedwater actuation and control system will actuate the emergency feedwater system when it receives a OTSG level of less than 18 inches in either generator. This signal will actuate the motor driven emergency feedwater pump (the turbine driven pump has already been actuated by SLBIC) and align the emergency feedwater valves in both trains.

Upon realizing he has a steam line break accident, the operator, using Emergency Operating Procedure 1202.24, will determine the affected OTSG by observing the OTSG levels and pressures. Upon identifying the affected OTSG, the operator will close the affected OTSG's emergency feedwater system steam supply and feed valves, and open, if not presently open, the corresponding steam supply valve on the unaffected OTSG. The operator would then commence cooldown to cold shutdown utilizing the unaffected OTSG.

If for some unlikely reason the operator fails to isolate the emergency feedwater to the affected OTSG, it has been shown through analysis using the assumptions in Attachment A that the reactor building pressure would not reach the design pressure of 59 psig until approximately 3 hours and 45 minutes into the accident allowing more than sufficient time for the operators to take corrective action."

In regard to a request for additional information concerning the possibility and consequences of continued main feedwater addition to the affected steam generator after a MSLB, the Licensee stated [5]:

"The analysis provided you in our original response [3] did consider the effects of concern. Although not explicitly stated in that response, an assumption in the 'no operator action' case was the failure of SLBIC. SLBIC automatically isolates main feedwater thus its failure assumed the isolation did not take place. The assumption of 'no operator action' was made to preclude manual isolation of main feedwater by the operators. (it should be noted that not all B&W plants were designed with SLBIC system thus manual isolation of main feedwater was the only isolation means available. As such, B&W maintained the 'no operator action' assumption in all analysis to generically bound all plants.)"

In regard to the ability of the AFW pumps to remain operable during a MSLB, the Licensee stated [4]:

"Analyses were performed by the Architect Engineer, using plant specific data and input from the pumps manufacturer, to determine if the emergency feedwater pumps would remain operable after possible runout flow conditions following a main steam line break (MSLB). These analyses demonstrated, even assuming no operator action, that the emergency feedwater pumps will remain operable during and following a MSLB accident considering runout flow conditions."

3.1.2 Evaluation

The Licensee's submittals [3, 4, 5] concerning the containment pressure response following a MSLB and applicable sections of the Arkansas Nuclear One Unit 1 FSAR [6] were reviewed in order to evaluate whether the following portions of the acceptance criteria were met:

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

Arkansas Nuclear One Unit 1 is a Babcock and Wilcox-designed, two-loop, 2568-MWt plant.

In the event of a MSLB, the following systems actuate to provide necessary protection:

- o The engineered safeguards actuation system (ESAS) initiates the high pressure and low pressure injection systems on receipt of the following:
  - a. two out of three (2/3) low reactor coolant pressure signals (1500 psig)
  - b. 2/3 high reactor building pressure signals (4 psig)
- o The reactor protection system (RPS) trips the reactor to protect aganst fuel damage on receipt of the following:
  - a. 2/4 overpower signals (105.5%)
  - b. 2/4 low reactor coolant system pressure signals (1800 psig)
  - c. 2/4 high reactor building pressure signals (4 psig)
- o Reactor building cooling system (4 units at 60×10^6 Btu/hr) is actuated on receipt of 2/3 high reactor building signals (4 psig)
- o Reactor building spray system (2 trains at 120×10^6 Btu/hr) is actuated on receipt of 2/3 high reactor building pressure signals (30 psig)
- o The steam line break instrumentation and control (SLBIC) is designed to isolate each steam generator by closing the main steam block valve and/or the feedwater isolation valve on each line upon receipt of 2/4 low steam generator pressure signals.

The emergency feedwater (EFW) system includes one motor-driven pump (672 gpm) and one turbine-driven pump (705 gpm) which are aligned so that either pump can supply both steam generators. The flow from either pump will ensure that the heat removal capacity exceeds the minimum level required for decay heat removal after a MSLB. The EFW system is automatically initiated on the following:

<u>Motor-driven Pump</u>

- o loss of both main feedwater pumps
- o loss of all four reactor coolant pumps
- o low steam generator level

<u>Turbine-driven Pumps</u>

- o loss of both main feedwater pumps

- o loss of all four reactor coolant pumps
- o low steam generator level
- o SLBIC signal

The SLBIC system is designed to meet safety-grade and IEEE Std 279-1971 [8] requirements; the ESAS and RPS are designed to safety-grade and IEEE Std 279-1968 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 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 [11] and Regulatory Guide 1.97 [12].

The Licensee's analysis assumed that the SLBIC fails to isolate main feedwater and that no operator action is taken to isolate main feedwater or emergency feedwater. The ICS is then assumed to control both main and emergency feedwater flow to maintain a minimum level in the steam generators. The Licensee's analysis determined that the containment design pressure of 59 psig would not be exceeded for 3 hours and 45 minutes. This is ample time for the operator to analyze the accident and take the appropriate actions to prevent exceeding the design pressure.

An analysis of the EFW pumps determined that the pumps would remain operable without operator action, when subject to runout flow conditions during a MSLB accident.

On October 15, 1980 [16] APL provided details of a safety-grade, automatically initiated, AFW system designed to feed only the unaffected steam generator in the event of a MSLB. The Licensee committed to install this system during the Arkansas Nuclear One Unit 1 fifth refueling outage currently scheduled for January 1983. The final design of this AFW initiation system was provided in Reference 17.

A review of References 16 and 17 determined that the proposed emergency feed initiation and control system (EFIC) is an instrumentation system designed to provide the following:

- o initiation of emergency feedwater (EFW)
- o control of EFW at appropriate setpoints (approximately 3, 20, and 31.5 feet)
- o level rate control when required to minimize overcooling
- o isolation of the main steam and main feedwater lines of a depressurized steam generator
- o the selection of the appropriate steam generator(s) under conditions of steamline break or main feedwater or emergency feedwater line break downstream of the last check valve
- o termination of main feedwater to a steam generator on approach to overfill conditions
- o termination of EFW to a steam generator on approach to overfill conditions
- o control of atmospheric dump valves to predetermined setpoint.

The EFIC logic issues a call for EFW auto-initiation when:

- o all four reactor coolant pumps are tripped
- o both main feedwater pumps are tripped
- o the level of either steam generator is low
- o either steam generator pressure is low
- o flux to MFW flow ratio trip is present.

Other functions of the EFIC logic are:

- o Issues a call for steam generator (SG) A main feedwater and main steamline isolation when SG A pressure is low
- o Issues a call for SG B main feedwater and main steamline isolation when SG B pressure is low
- o Signals approach to SG A overfill when SG A level exceeds a high level setpoint

- o Signals approach to SG B overfill when SG B level exceeds a high level setpoint

- o Provides for manually initiated individual shutdown bypassing of reactor coolant pumps, main feedwater pumps, and SG pressure initiation of EFW as a function of permissive conditions. The bypass(es) is automatically removed when the permissive condition terminates.

- o Provides for maintenance bypassing of an EFIC initiate logic.

In the event of a steam line break or feed line break, the EFIC system is designed to isolate the steam and feedwater lines and to provide emergency feedwater to the intact steam generator. The system is designed so that no single active failure will either prevent emergency feedwater from being supplied to the intact steam generator or allow emergency feedwater to be supplied to the broken steam generator.

To meet the requirements for steam line or feed line break protection, the following design was implemented:

- o Isolation - low steam pressure (below approximately 600 psig) in either SG will isolate the main steamlines and main feedwater line to the affected SG.

- o SG selection

  a. If both SGs are above 600 psig, EFW is supplied to both SGs.

  b. If one SG is below 600 psig, EFW is supplied to the other SG.

  c. If both SGs are below 600 psig, but the pressure difference between the two SGs exceeds a fixed setpoint (approximately 100 psig), EFW is supplied only to the SG with the higher pressure.

  d. If both SGs are below 600 psig and the pressure difference is less than the fixed setpoint, EFW is supplied to both SGs.

The EFIC system was designed to safety-grade and IEEE Std 279-1971 requirements.

3.1.3 Conclusion

The Licensee's responses [3, 4, 5] and the Arkansas Nuclear One Unit 1 FSAR [6] 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 acceptance criteria. The proposed EFIC system will provide safety-grade protection against a MSLB and eliminate the need for operator action to isolate emergency feedwater flow to the ruptured steam generator. The EFW pumps will remain operable when subject to runout flow conditions during 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 [3]:

"The steam line break accident has been analyzed in the Unit 1 FSAR in Section 14.2.2 considering no operator action. In this analysis, the affected OTSG is assumed to blow dry after the rupture at which time the minimum level control opens feedwater valves such that the OTSG maintains low-level. Assuming a minimum tripped rod worth with the maximum rod stuck out, the reactor will return to a maximum neutron power level of 2.6% at 44.5 seconds and return to subcriticality at 47.5 seconds. With the low level control valves maintaining a 30-inch minimum downcomer level in the affected OTSG, the average coolant temperature will remain below 475 degrees F until feedwater isolation on the affected OTSG is achieved."

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 Reference 3 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.

In the worst case MSLB, which assumes full power conditions, a double-ended rupture at the steam generator exit, and no operator action, a peak power of 106% occurs at 6 seconds, at which time a high flux reactor trip occurs, inserting the control rods. After the reactor trip, the core returns to criticality at 43.5 seconds, reaches a maximum neutron power of 2.6% at 44.5 seconds, and returns to subcriticality at 47.5 seconds. The predicted return-to-power does not result in a violation of the specified acceptable fuel design limits.

3.2.3 Conclusion

For the current plant design, the Licensee's responses [3, 4, 5] and FSAR adequately address the concerns of Item 2 of IE Bulletin 80-04. All potential sources of water were identified, and although a reactor return-to-power is predicted, there is no violation of the specified acceptable fuel design limits, and the FSAR analysis of the reactivity increase resulting from a MSLB remains valid.

.3 REVIEW OF CORRECTIVE ACTIONS

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

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

3.3.1 Summary of Licensee Statements and Conclusions

The Licensee stated [3]:

"Although the potential exists on ANO-1 for reactor building over pressurization, this event will not take place until 3 hours and 45 minutes into the steam line break accident. It is our position that there is more than sufficient time for the operator to isolate the affected OTSG and terminate the event. Thus no corrective action is proposed."

3.3.2 Evaluation and Conclusion

The Licensee's analysis determined that neither a containment overpressurization nor a reactor return-to-power with a resultant violation of the specified acceptable fuel design limits would occur from a MSLB. Therefore, it is concluded that no further action regarding IE Bulletin 80-04 is required of APL for Arkansas Nuclear One Unit 1.

4. CONCLUSIONS

With respect to Arkansas Nuclear One Unit 1, conclusions regarding Arkansas Power and Light 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 with continued feedwater addition.
- o The emergency feedwater pumps will remain operable when subject to effects of runout flow and therefore can be expected to carry out their intended function during the MSLB event.
- o All potential water sources were identified and, although a reactor return-to-power is predicted, there is no violation of the specified acceptable fuel design limits. Therefore, the Final Safety Analysis Report MSLB reactivity increase analysis remains valid.
- o No further action regarding IE Bulletin 80-04 is required.

5. REFERENCES

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11. "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors"
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16. D. C. Trimble (APL)
Letter to R. W. Reid (NRR)
Subject: Submittal of EFW Upgrade Proposal Design Information
October 15, 1980

17. D. C. Trimble (APL)
Letter to T. M. Novak (NRR ORB)
Subject: Final Submittal of EFW Upgrade Design Information
December 1, 1981