TECHNICAL EVALUATION REPORT

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

DUKE POWER COMPANY
OCONEE NUCLEAR STATION UNITS 1, 2, AND 3

NRC DOCKET NO. 50-269, 50-270, 50-287 FRC PROJECT C5506

NRC TAC NO. 46847, 46848, 46849 FRC ASSIGNMENT 5

NRC CONTRACT NO. NRC-03-81-130 FRC TASK 134

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September 28, 1982

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Date: 9-28-82	Date: 9/28/82	Date: 9-28-82

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FOREWORD

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

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

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This Technical Evaluation Report (TER) documents an independent review of Duke Power Company's (DPC) 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 Oconee Nuclear Station Units 1, 2, and 3. This evaluation was performed with the following objectives:

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

Duke Power Company responded to IE Bulletin 80-04 in a letter to the NRC dated May 7, 1980 [3] and provided additional information for this review in a letter dated July 23, 1982 [4]. The information in References 3 and 4 has been evaluated along with pertinent information from the Oconee Nuclear Station Final Safety Analysis Report (FSAR) [5] to determine the adequacy of the Licensee's compliance with IE Bulletin 80-04.

2. ACCEPTANCE CRITERIA

The following criteria against which the Licensee's MSLB response was evaluated were provided by the NRC [6]:

1. PWR licensees' responses to IE Bulletin 80-04 shall include the following information related to their analysis of containment pressure and core reactivity response to a MSLB within or outside containment:

   a. A discussion of the continuation of flow to the affected steam generator, including the impact of runout flow from the AFW system and the impact of other energy sources, such as continuation of feedwater or condensate flow. AFW system runout flow should be determined from the manufacturer's pump curves at no backpressure, unless the system contains reliable anti-runout provisions or a more representative backpressure has been conservatively calculated. If a licensee assumes credit for anti-runout provisions, then justification and/or documentation used to determine that the provisions are reliable should be provided. Examples of devices for which provisions are reliable are anti-runout devices that use active components (e.g., automatically throttled valves) which meet the requirements of IEEE Std 279-1971 [7] and passive devices (e.g., flow orifices or cavitating venturis).

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

   c. A discussion of the ability to detect and isolate the damaged steam generator from continued feedwater addition during the MSLB accident. Operator action to isolate AFW flow to the affected steam generator within the first 30 minutes of the start of the MSLB should be justified. If 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 mintues 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 [8] (i.e., increase in core reactivity) can occur by the licensee's analysis, the licensee shall provide the following additional information:

   a. The proposed corrective actions to prevent containment overpressure or the violation of fuel design limits and the schedule for their completion.

   b. The interim actions that will be taken until the proposed corrective action is completed, if the unit is operating.

3. The acceptable input assumptions used in the licensee's analysis of the core reactivity changes during a MSLB are given in Section 15.1.5 of the Standard Review Plan [9]. The following specific assumptions should be used unless the analysis shows that a different assumption is more limiting:

   Assumption II.3.b.: Analysis should be performed to determine the most conservative assumption with respect to a loss of electrical power. A reactivity analysis should be conducted for a normal power situation as well as a loss of offsite power scenario, unless the licensee has previously conducted a sensitivity analysis which demonstrates that a particular assumption is more conservative.

   Assumption II.3.d.: The most restrictive single active failure in the safety injection system which has the effect of delaying the delivery of high concentration boric acid solution to the reactor coolant system, or any other single active failure affecting the plant response, should be considered.

   Assumption II.3.g.: The initial core flow should be chosen such that the post-MSLB shutdown margin is minimized (i.e., maximum initial core flow).

The acceptable computer codes for the licensee's analysis of core reactivity changes are, by nuclear steam supply system (NSSS) vendor, the following: CESEC (Combustion Engineering), LOFTRAN (Westinghouse), and TRAP (Babcock & Wilcox). Other computer codes may be used, provided that these codes have previously been reviewed and found to be acceptable by the NRC staff. If a computer code is used which has not been reviewed, the licensee must describe the method employed to verify the code results in sufficient detail to permit the code to be reviewed for acceptability.

4. If the AFW pumps can be damaged by extended operation at runout flow, the licensee's action to preclude damage should be reviewed for technical merit. Any active features should satisfy the requirements of IEEE Std 279-1971. Where no corrective action has been proposed, this should be indicated to the NRC for further action and resolution.

5. Modifications to 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" [10], and the regulatory positions in Regulatory Guide 1.97, Rev. 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" [11].

6. AFW system status should be reviewed to ensure that system heat removal capacity does not decrease below the minimum required level as a result of isolation of the affected steam generator and also that recent changes have not been made in the system which adversely affect vital assumptions of the containment pressure and core reactivity response analyses.

7. The safety-grade requirements (redundancy, seismic and environmental qualifications, etc.) of the equipment that isolates the main feedwater (MFW) and AFW systems from the affected steam generator should be specified. The modifications of equipment that 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" [12].

- Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [13].
- Environmental qualification: The isolation valves should satisfy the requirements of NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" [14].
- Quality standards: The isolation valves should satisfy Group B quality standards as recommended in Regulatory Guide 1.26 or similar quality standards from the plant's licensing bases.

3. TECHNICAL EVALUATION

The scope of work included the following:

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

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

   b. Evaluate the Licensee's proposed corrective actions and schedule for implementation if the findings of Task 2a indicate that a potential exists for overpressurizing the containment or worsening the reactor return-to-power in the event of a MSLB accident.

3. Prepare a TER for each plant based on the evaluation of the information presented for Tasks 1 and 2 above.

This report constitutes a TER in satisfaction of Task 3. Sections 3.1 through 3.3 of this report state the requirements of IE Bulletin 80-04 by subsection, summarize the Licensee's statements and conclusions regarding these requirements, and present a discussion of the Licensee's evaluation followed by conclusions and recommendations.

3.1 REVIEW OF CONTAINMENT PRESSURE RESPONSE ANALYSIS

The requirement from IE Bulletin 80-04, Item 1, is as follows:

"Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow."

3.1.1 Summary of Licensee Statements and Conclusions

In regard to the review of the containment pressure response analysis, the Licensee stated [3]:

"The Oconee FSAR containment pressure response analysis for a postulated main steam line break inside containment considered two cases. The pertinent assumptions and system responses in each case are as follows:

Case 1

Following the main steam line break, feedwater was assumed to remain at 100% until the reactor trip occurred. Subsequently, the integrated control system (ICS) was assumed to close the main and startup control valves. The operator was assumed to take manual control of the feedwater to the affected OTSG [once-through steam generator] and ensure that the main and startup feedwater control valves remained closed. The unaffected OTSG was controlled at the minimum level (two feet) removing the decay heat generated in the core. The containment pressure response analysis calculated a 13 psi rise in the containment pressure, well below the design pressure of 59 psi."

Case 2

The second case was summarized in Reference 4 and is presented below:

"Assuming no operator action, feedwater will be delivered to the affected steam generator by the MFWS [main feedwater system] following a main steam line break to control steam generator level at the setpoint. More energy will be delivered to the Reactor Building with the MFWS rather than the EFWS [emergency feedwater system] in operation, due to the higher fluid enthalpy and higher flow capacity. As the Reactor Building pressure increases, the Reactor Building Spray System (RBSS) and the Reactor Building Cooling System (RBCS) will actuate and begin to remove energy from the building. These two systems are described in the Oconee FSAR. The feedwater delivered to the affected steam generator will continue to boil off and cool down the primary system. The increase in Reactor Building pressure causes the saturation temperature to increase, thereby decreasing the primary to secondary temperature difference across the steam generator tubes. This causes the heat transfer from the primary system to become limited to the heat being added to the primary system, which is the reactor decay heat. The Reactor Building pressure will continue to increase until the energy addition to the building is less than the energy removal by the RBSS and the RBCS.

The results of the FSAR analysis show that at 250 sec the Reactor Building has pressurized to 38 psig, and the heat transfer from the primary has become limited to the decay heat source. At 360 sec the energy removal capacity of the RBCS exceeds the decay heat source, and the pressurization of the building has peaked and begins decreasing. The peak Reactor Building pressure is significantly less than the design pressure of 59 psig. No operator action is assumed. This scenario bounds all credible steam line breaks within the Reactor Building."

In regard to the analyses, the Licensee stated [4]:

"The existing FSAR analysis summarized above did not explicitly address the impact of the runout flow from the auxiliary feedwater system. This is because continued feedwater addition to the affected steam generator by means of the main feedwater system is considered to be more limiting with respect to containment pressure response than the case involving auxiliary feedwater flow.

The auxiliary feedwater system currently in use at Oconee consists of a turbine driven pump and two motor driven pumps. Each steam generator can receive auxiliary feedwater flow from one motor-driven pump and the turbine driven pump. Operation of the auxiliary feedwater system at runout conditions would result in approximately 2050 gpm (700 gpm from the motor driven pump and 1350 gpm from the turbine driven pump) auxiliary feedwater addition into the affected steam generator. Since the flow capacity of the auxiliary feedwater system is less than that of the main feedwater pump (greater than 10,000 gpm), since the auxiliary feedwater temperature (90°F) is less than that of the main feedwater (460°F) and since the existing analysis considered the maximum possible cooldown of the primary system for a steam line break in the containment, it is conlcuded that the existing analysis of containment pressure response bounds the situation involving flow from the auxiliary feedwater system."

In regard to the ability of the main feedwater (MFW) and emergency feedwater (EFW) pumps to remain operable during a MSLB, the Licensee stated [4]:

"Excessive feedwater delivery (runout flow) from the MFWS is prevented by the high steam generator level trip of the MFW pumps and would be limited by the self-limiting heat transfer processes. Excessive feedwater delivery from the EFWS is prevented by the level control system and would have a minimal impact on the building pressure response due to the low enthalpy of the fluid and the low flow capacity of the EFWS in comparison to the MFWS."

3.1.2 Evaluation

The Licensee's submittals [3, 4] concerning the containment pressure response following a MSLB and applicable sections of the Oconee Nuclear Station FSAR [5] 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.

Oconee Nuclear Station Units 1, 2, and 3 are virtually identical, Babcock and Wilcox-designed, two-loop, 860 MWe plants.

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

- o Reactor trip on high flux (104.9%, two out of four channels) or on low reactor coolant system (RCS) pressure (1800 psig, two out of four channels)
- o The reactor trip signals:
  - a. turbine stop valves to trip
  - b. Integrated control system (ICS) to control steam generator level at the minimum level. (Control-grade)
  - c. ICS to close MFW control valves and startup control valves to each steam generator (control-grade)
- o High pressure injection (HPI) system is actuated upon:
  - a. two out of three (2/3) low reactor coolant system pressure signals (1500 psig)
  - b. 2/3 high reactor building pressure signals (4 psig)
- o Low pressure injection (LPI) system is actuated upon:
  - a. 2/3 low reactor coolant system pressure (500 psig)
  - b. 2/3 high reactor building pressure
- o Reactor building spray system (RBSS) is actuated on 2/3 very high reactor building pressure signals (10 psig)

- Three reactor building cooling units (RBCU) are actuated at 2/3 high reactor building pressure signals.

Each AFW system consists of two motor-driven pumps (450 gpm) and one turbine-driven pump (880 gpm). The motor-driven pumps are normally aligned to supply a single OTSG and the turbine-driven pump is aligned to supply both OTSGs. The flow from one motor-driven pump to the unaffected OTSG is sufficient to ensure that the system heat removal exceeds the minimum level required for decay heat removal after a MSLB.

The EFW systems of the three units may be cross-connected such that any unit may supply EFW to another unit.

The safety-grade steam generator level control system (SGLCS) provides automatic OTSG water level control while the EFW system is supplying feedwater to the steam generators. SGLCS is designed to automatically control and modulate EFW supply to the steam generators during all initiating conditions for the EFW system. Each OTSG has two independent level control systems each of which is capable of supplying a signal to the OTSG EFW level control valve. All automatic initiation logic and control functions are independent of the ICS.

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 worst-case MSLB is a double-ended rupture, at full rated power, with no operator action to isolate MFW. Water level in the affected OTSG is assumed to be maintained at the 2-ft minimum level by MFW. At 250 sec, the reactor building pressure reaches 38 psig; this amounts to a back pressure on the system which limits RCS temperature to a minimum of 248°, thus halting the sensible heat flow from the coolant system. Beyond this time, only decay heat is released. The mass and energy released from this time until the time when the energy removal rate exceeds the decay heat generation rate is less than that required to reach building design pressure (59 psig).

The heat removal rate of the RBCUs equals the decay heat rate at 6 min following the MSLB. Each of two RBSS trains provides an energy removal capacity of 120×10^6 Btu/hr and each of three RBCUs provides an energy removal capacity of 80×10^6 Btu/hr. The arrangement of the engineered safeguards power supplies ensure that at least one RBSS train and two RBCUs are available in the event of a fault on a bus, thus providing a minimum energy removal capacity of 280×10^6 Btu/hr.

The EFW pumps are protected from damage caused by operating at runout flow by the SGLCS which throttles the EFW flow to the OTSGs.

3.1.3 Conclusion and Recommendations

The Licensee's responses [3, 4] and the Oconee Nuclear Station FSAR [5] 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 EFW 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 [3]:

"The Oconee FSAR analysis of the reactivity increase resulting from a main steam line break considered four cases involving various potential modes of feedwater addition to the affected steam generator. In all cases, a minimum rod worth, based on the maximum worth rod considered stuck-out, and consistent with the minimum shutdown margin required by the Technical Specifications was utilized in the core reactivity calculation. The pertinent assumption and system responses for each of these cases are summarized below.

Case 1

In this case, the integrated control system (ICS) was assumed to initially close the main and startup feedwater control valves following the reactor trip and then the operator was assumed to maintain feedwater isolation of the affected steam generator. The minimum (two foot) level was maintained in the unaffected steam generator. Under these assumptions, the reactor was calculated to remain subcritical throughout the transient.

Case 2

In this case, also the ICS was assumed to close the main and startup feedwater control valves following the reactor trip; however, no credit was taken for operator action to maintain feedwater isolation of the affected steam generator. Consequently, feedwater flow by means of the main feedwater pump continued to the affected steam generator at a rate necessary to maintain a two foot level. The resulting cooldown of the primary system was calculated to cause a return to power of about 1% FP [full power] at approximately 170 seconds. The core then returned to subcritical conditions with the addition of highly borated water by the emergency core cooling system (HPI, CFT [core flood tanks], and LPI).

Case 3

The third steam line break analysis case assumed proper ICS action to initially close the main and startup feedwater control valves; however, no operator action to maintain feedwater isolation of the affected steam generator was assumed. The auxiliary feedwater pump was assumed to start on a low main feedwater pump discharge pressure signal. The ICS was assumed to maintain a minimum (two foot) level in both steam generators with a combination of main and auxiliary feedwater. The analysis predicted a return to 35 percent of rated power in approximately 65 seconds. Without the stuck rod and considering the nominal trip rod worth, the core was found to remain subcritical.

Case 4

The fourth main steam line break analysis case included the assumption of no ICS or operator action to change the feedwater control valve positions. The feedwater flow to the damaged steam generator was postulated to be 135% of the rated flow in one steam generator. It was assumed that the auxiliary feedwater system was not actuated. Under these conditions, the reactor was calculated to return to less than 8 percent of rated power approximately 166 seconds after the break before going subcritical again by injection of borated water by the ECCS [emergency core cooling system].

From the foregoing discussion, it is seen that the existing analysis of the steam line break accident considered several potential modes of feedwater addition to the affected steam generator from the main and auxiliary feedwater systems. Although the flow capacity of the auxiliary feedwater system has increased with the recent addition of the motor driven pump into each of the two secondary loops, the analysis for Case 3 above still represents the worst case core reactivity increase. Since the increases in the auxiliary feedwater flow capacity is very small (less than 5%) compared to the available total feedwater flow capacity and since the amount of feedwater flow into the steam generator is dictated by the steam generator level requirement."

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 to isolate MFW and the ICS is assumed to actuate the turbine-driven EFW pump. The reactor returns to a peak power of 35% at 65 sec and then returns to subcriticality. The calculated return-to-power does not result in a violation of the specified acceptable fuel design limits.

3.2.3 Conclusion

The Licensee's responses [3, 4] and FSAR [5] 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

The Licensee stated [3]:

"As demonstrated in the response to Item 1 above, the potential for containment overpressure is not introduced by postulated auxiliary feedwater pump operation at runout conditions. Furthermore, the existing emergency procedure includes operator guidance to prevent uncontrolled feedwater addition to the affected steam generator. The reactor return-to-power responses calculated in the FSAR still represent the limiting case for core reactivity increase. Therefore, no corrective actions are considered necessary at this time for Oconee Nuclear Station. It is pointed out that a probabilistic risk assessment study is being planned for Oconee. If the results of this study indicate the need for any corrective actions with respect to the steam line break accident, appropriate corrective actions will be considered at that time."

3.3.2 Evaluation and Conclusion

The Licensee's analysis determined that a containment overpressurization or a worsening of a reactor return-to-power with a resultant violation of specified acceptable fuel limits resulting from a MSLB would not occur. Therefore, it is concluded that no further action regarding IE Bulletin 80-04 is required of DPC for the Oconee Nuclear Station Units 1, 2, and 3.

4. CONCLUSIONS

With respect to the Oconee Nuclear Station Units 1, 2, and 3, conclusions regarding Duke Power 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 emergency feedwater (EFW) 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 No further action is required by the Licensee regarding IE Bulletin 80-04.

5. REFERENCES

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