TECHNICAL EVALUATION REPORT

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

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT UNIT 1

NRC DOCKET NO. 50-327

NRC TAC NO. 46857

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 5

FRC TASK 121

Prepared by

Franklin Research Center
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Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

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Lead NRC Engineer: P. Hearn

September 24, 1982

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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 a review of the Tennessee Valley Authority's (TVA) 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 Sequoyah Nuclear Plant Unit 1. This evaluation was performed with the following objectives:

- o to assess the conformance of TVA'S main steam line break (MSLB) analyses with the requirements of IE Bulletin 80-04
- o to assess TVA'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 flow 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 [illegible]tainment 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

TVA responded to IE Bulletin 80-04 in a letter to the NRC dated June 16, 1980 [3]. On May 11, 1982, TVA responded to a request for additional information required to complete this review [4]. The information in References 3 and 4 has been evaluated along with pertinent information from the Sequoyah and Watts-Bar Final Safety Analysis Reports (FSAR) [5, 6] to determine the adequacy of the Licensee's response to IE Bulletin 80-04.

2. ACCEPTANCE CRITERIA

The following criteria against which the Licensee's MSLB response was evaluated were provided by the NRC [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. Where 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 [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 and Wilcox). Other computer codes may be used, provided that these codes have previously been reviewed and found to be acceptable by the NRC staff. If a computer code is used which has not been reviewed, the licensee must describe the method employed to verify the code results in sufficient detail to permit the code to be reviewed for acceptability.

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

5. Modifications to 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 isolate 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

Under contract to the NRC, 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 item 3. Sections 3.1 through 3.3 of this report state the requirements of IE Bulletin 80-04 by subsection, summarize the Licensee's statements and conclusions regarding these requirements, and present a discussion of the Licensee's evaluation followed by conclusions and recommendations.

3.1 REVIEW OF CONTAINMENT PRESSURE RESPONSE ANALYSIS

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

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

3.1.1 Summary of Licensee Statements and Conclusions

The Licensee made the following statement concerning its response to Item 1 of IE Bulletin 80-04:

"The response to this item has been previously addressed by TVA's response to Sequoyah FSAR question Q6.56B."

In regard to a request for information concerning operator action after a MSLB, the Licensee stated [4]:

"The only action required of the operator to prevent exceeding containment design pressure following an MSLB is isolation of AFW to the affected steam generator. The time at which credit is taken for this action is 10 minutes.

As stated in the response to question 6.56B part 2(e), information is available to the operator immediately upon initiation of the accident. It is also stated that this information is given in EOI-2 and that operator action to terminate AFW flow to the affected steam generator will occur in approximately three minutes. Please note that the postulated accident involves completely blowing down one steam generator. The pressure in this steam generator will drop to about zero psig while the pressure in the other three will be over 200 psig. We believe the operator will be able to identify the faulted loop with relative ease and quickness given such information. Therefore, the assumption that operator action occurs at 10 minutes, which is consistent with the licensing basis of the plant, is justified and is in fact conservative.

Following an MSLB, both trains of RHR spray will be available to relieve containment pressure since no RHR flow through the reactor is needed. However, no credit is taken for RHR spray in the analysis and the operator is not required to use it."

3.1.2 Evaluation

The Licensee's submittal concerning containment pressure response analysis and applicable sections of the Sequoyah and Watts-Bar FSARs [4, 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

- Criterion 7 - Safety-grade requirements for MFW and AFW isolation valves.

The Sequoyah Nuclear Plant is a Westinghouse-designed, four-loop plant with an ice condenser containment.

Following a MSLB, the engineered safety features actuation system (ESFAS) initiates signals for reactor trip, safety injection, steam and feedwater isolation, emergency diesel generator startup, AFW system startup, and other safeguard systems required for accident mitigation.

The following systems are initiated by ESFAS:

- Safety injection system actuation on:

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

  b. high steam line flow coincident with either low steam line pressure in two lines or low-low T_{avg} signals

  c. high steam line differential pressure signals (2/3 per line)

  d. 2/3 high containment pressure signals.

- Reactor trip on overpower (neutron flux and differential temperature) and the reactor trip occurring in conjunction with the receipt of the safety injection signal

- Trip of the safety-grade fast acting steam line stop valves (designed to close in less than 5 seconds) on:

  a. high steam flow in two main steam lines in coincidence with either low-low T_{avg} or low steam line pressure in two lines

  b. high-high containment pressure

- Containment spray and air recirculation fans initiation on a high-high containment pressure (3.0 psig)

- Redundant isolation of the MFW lines and steam line isolation initiation on receipt of the safety injection signal (SIS). This signal closes all feedwater control and isolation valves (safety-grade), trips the MFW pumps, and trips the steam line stop valves. In addition, normal control action will close the MFW valves.

The AFW system, which consists of two 440-gpm motor-driven pumps and one 880-gpm turbine-driven pump is designed to be single-failure proof.' The following signals are used for automatic initiation of the AFW system:

- o low-low water level in any steam generator (motor-driven)
- o low-low water level in any two steam generators (turbine-driven pumps)
- o safety injection signal
- o loss of offsite power
- o loss of both MFW pumps.

The motor-driven pumps are equipped with pressure control valves which limit the runout flow to 450 gpm. The turbine-driven pump speed is controlled by a flow signal which limits flow to the steam generators to 880 gpm. Ten minutes after the MSLB, AFW to the affected steam generator is isolated by the operator who manually realigns the system for delivery of AFW flow to the unaffected steam generators.

A review of Section 7 of the Sequoyah FSAR disclosed that the ESFAS is designed to meet safety-grade and IEEE Std 279-1971 requirements. The environmental qualification of safety-related electrical and mechanical components is being reviewed seperately by the NRC and is not within the scope of this review.

A review of Section 6.2.1 of TVA's response to Sequoyah FSAR question Q6.56B disclosed that the Watts Bar FSAR Section 6.2.1 may be referenced for the Sequoyah Nuclear Plant. A review of the Watts Bar and Sequoyah FSARs determined that the MSLB analysis had considered the effects of continued feedwater addition to the affected steam generator. Failure of a main steam isolation valve, a diesel generator, a feedwater isolation valve, and AFW runout control were considered separately. The worst-case single failure is the failure of AFW runout control, at 102% of nominal full power, resulting in a flow of 2040 gpm to the faulted steam generator until isolation by the operator 10 minutes after the break. The peak upper and lower containment pressures of 8.8 psig and 9.4 psig, respectively, occur 10 seconds after the break and are less than the containment shell design pressure of 12 psig. The maximum differential pressure across the operating deck is 1.05 psid, which is less than the design pressure of 9.6 psig.

The operator has sufficient instrumentation to identify and isolate the ruptured steam generator. To isolate AFW from the rupture, the operator shuts the remote controlled AFW isolation valve. If this valve fails to close, the operator can trip the two feedwater pumps feeding the ruptured steam generator. (The flow from one AFW pump will provide sufficient decay heat removal capability.) All operator actions are performed from the contol room. It is conservative to assume that these actions will be completed within 10 minutes from the start of the MSLB.

The review did not determine whether the instrumentation upon which the operator relies 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].

3.1.3 Conclusion

The Licensee's responses [3, 4] adequately address the concerns of IE Bulletin 80-04. The containment pressure response analysis and the design of ESFAS satisfies the NRC's acceptance criteria. The AFW pumps are individually protected against the effects of runout flow. A single failure in the runout control system would affect only one AFW pump, leaving the two other pumps operable.

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

Regarding the reactivity increase resulting from a MSLB with continued feedwater addition, the Licensee stated [3]:

"The assumptions made for main and auxiliary feedwater flow as they apply to licensing basis steamline break transients have been reviewed. Several of the relevant assumptions used in all core transient analyses follow, and are further explained in the Sequoyah FSAR sections 6.2.1.3.11 and 15.5.4.

1. The reactor is assumed initially to be at hot shutdown conditions, at the minimum allowable shutdown margin.

2. For the Condition IV breaks, i.e. double-ended rupture of a main steam pipe, full main feedwater is assumed from the beginning of the transient at a very conservative cold temperature.

3. All auxiliary feedwater pumps are initially assumed to be operating in addition to the main feedwater. The flow is equivalent to the rated flow of all pumps at the steam generator design pressure.

4. Feedwater is assumed to continue at its initial flow rate until feedwater isolation is complete, approximately 10 seconds after the break occurs, while auxiliary feedwater is assumed to continue at its initial flow rate.

5. Main feedwater is completely terminated following feedwater isolation.

Based on the manner in which the analysis is performed for Westinghouse plants, the core transient results are very insensitive to auxiliary feedwater flow. The first minute of the transient is dominated entirely by the steam flow contribution to primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal-hydraulic transients in the core. The effect of auxiliary feedwater runout (or failure of runout protection where applicable) is minimal.

The auxiliary feedwater flow becomes a dominant factor in determining the duration and magnitude of the steam flow transient during later stages in the transient.

However, the limiting portion of the transient occurs during the first minute, both due to higher steam flows inherently present early in the transient and due to the introduction of boron to the core via the safety injection system.

In conclusion, the effects of runout auxiliary feedwater flows in the core transient for steam line break has been evaluated; and based on this evaluation, it has been determined that the assumptions presently made are appropriate for use as a licensing basis. The concerns outlined in the introduction to IE Bulletin 80-04 relative to, (1) limiting core conditions occurring during portions of the transient where auxiliary feedwater flow is a relevant contributor to plant cooldown; and (2) incomplete isolation of main feedwater flow, are not representative of the Westinghouse NSSS designs and associated Balance of Plant requirements."

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.

From the review of the FSAR analysis, it was determined that the analysis is conservative in its assumptions and that the assumptions are in accordance with those in acceptance criterion 3, with the exception of not assuming runout AFW flow.

As discussed in Section 3.1.2 of this report, the ESFAS isolates the main steam and main feedwater systems, starts the AFW system, and initiates other protective functions following a MSLB. The steam line break analysis considered four cases including (1) a break outside containment at no load conditions, (2) a break inside containment at the steam generator outlet and at no load, (3) same as case 1 but with concurrent loss of offsite power, and (4) same as case 2 but with concurrent loss of offsite power. In all four cases, there was a return to power with the peak occurring at less than 1 minute after the break; the peak reactivity was less than +0.005 in all four cases. In all four cases, the calculated return-to-power did not result in a violation of the specified acceptable fuel design limits.

The Licensee's conclusion that the core transient for the MSLB is insensitive to AFW flow is valid for the following reasons:

- o Early in the transient, the primary-to-secondary heat transfer rate (from the blowdown of the initial steam generator mass) is several orders of magnitude greater than that contributed by the additional AFW flow due to runout.
- o Later in the transient (when the majority of the initial mass has blown down), AFW flow becomes a dominant factor in determining the magnitude and duration of the transient.
- o The limiting core conditions will occur within the first minute due to the initial high cooldown rate contributing to the reactivity addition which is terminated by the introduction of 20,100 ppm boron solution into the core region.

Since the limiting core conditions occur before the AFW flow becomes a major contributing factor, it can be concluded that the core transient is insenitive to the contribution of AFW flow, and therefore the assumptions of the FSAR remain valid.

3.2.3 Conclusion

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

3.3 REVIEW OF CORRECTIVE ACTIONS

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

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

3.3.1 Summary of Licensee Statements and Conclusions

The Licensee stated:

"Based on the response to items 1 and 2, no corrective action is necessary."

3.3.2 Evaluation and Conclusion

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

4. CONCLUSIONS

Conclusions regarding Tennessee Valley Authority's response to IE Bulletin 80-04 relative to Sequoyah Nuclear Plant Unit 1 are as follows:

- o There is no potential for containment overpressurization resulting from a main steam line break (MSLB) with continued feedwater addition.
- o All potential water sources were identified. 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 reactivity increase analysis remains valid.
- o The auxiliary feedwater (AFW) pumps are individually protected against the effects of runout flow. A single failure of the runout control system will only affect one pump, leaving the other two pumps capable of continued operation.
- o No further action is required of the Licensee regarding IE Bulletin 80-04.

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