TECHNICAL EVALUATION REPORT

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

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON UNIT 2

NRC DOCKET NO. 50-261

FRC PROJECT C5506

NRC TAC NO. 46856

FRC ASSIGNMENT 3

NRC CONTRACT NO. NRC-03-81-130

FRC TASK 137

Prepared by

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Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: P. Hearn

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CONTENTS

FOREWORD

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

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

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This Technical Evaluation Report (TER) documents an independent review of Carolina Power and Light Company's (CP&L) 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 H. B. Robinson Steam Electric Plant Unit 2. This evaluation was performed with the following objectives:

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

CP&L responded to IE Bulletin 80-04 in a letter to the NRC dated May 9, 1980 [3] and provided additional information in a letter dated September 3, 1982 [4]. The information in References 3 and 4 has been evaluated along with pertinent information from the H. B. Robinson Unit 2 Updated 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 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 [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 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 is relied upon to isolate the MFW and AFW systems from the affected steam generator should satisfy the following criteria to be considered safety-grade:

   o Redundancy and power source requirements: The isolation valves should be designed to accommodate a single failure. A failure-modes-and-effects analysis should demonstrate that the system is capable of withstanding a single failure without loss of function. The single failure analysis should be conducted in accordance with the appropriate rules of application of ANS-51.7/N658-1976, "Single Failure Criteria for PWR Fluid Systems" [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

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 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 containment pressure response analysis, the Licensee stated [3]:

"The conservative assumptions used in the analysis for containment pressure following a main steam line break in containment have been reviewed as requested in IE Bulletin 80-04. The analysis documented in H. B. Robinson Unit 2 [original] FSAR, Page 14.25-10 included allowance for 100 seconds of auxiliary feedwater flow. We have extended the analysis to consider auxiliary feedwater flow for 10 minutes, as well as main feedwater flow for 10 seconds; a conservative estimate of the time for isolation of the system. The resultant containment pressure, including allowance for auxiliary feedwater flow to 10 minutes, is 34.4 psig compared with a design value of 42 psig...."

A main steam line break in containment will result in blowdown to containment with resultant increase in containment pressure and increase in cooling of the RCS. Increased cooling of the RCS would lead to low or falling pressurizer pressure and level. The operator would make the determination of a steam line break on the basis of abnormally low steam pressure in one or more steam generators; a continuously decreasing T_{avg} would also indicate a steam line break."

In regard to operator action, the Licensee stated [4]:

"The operator tasks required to identify the affected steam generator and isolate the AFW flow are taken from Emergency Instruction (EI)-1 Appendix B and are as follows:

1. Verify that steamline isolation has occurred. If not, manually initiate steamline isolation.

2. Verify the steam dump valves and atmospheric relief valves are closed to insure that the emergency has not resulted from an inadvertant opening of these valves.

3. If the reactor coolant pressure drops below 1300 psig, trip all reactor coolant pumps after safety injection pump operation is verified.

4. Determine if one steam generator has blowdown by observation of steam pressure and isolate the auxiliary feedwater flow to that steam generator.

The plant operations staff has evaluated the time required to respond to this accident and has determined that a trained operator responds in 2-3 minutes. The simulator training staff has also evaluated the response time for this event and has determined that a typical operator trainee response time is 2-5 minutes."

In response to a request for information regarding AFW flow rate, the Licensee stated [4]:

"In the original analysis, the back pressure value was conservatively calculated for the AFW pumps in the runout condition. The back pressure value calculation considered elevation differences and line resistance between the AFW pump's outlet and the steam generator inlet. The steam generator pressure was assumed to be atmospheric, i.e., zero gage. The runout flow at this pressure for each motor driven AFW pump is 316 gpm, as calculated in the original analysis."

In regard to the ability of the AFW pumps to operate without sustaining damage during a MSLB, the Licensee stated [3]:

"No difficulties are anticipated with extended auxiliary feed pump operation at runout conditions. Cavitation is not expected at the anticipated flow rate."

3.1.2 Evaluation

The Licensee's submittals [3, 4] concerning the containment pressure response following a MSLB and applicable sections of the H. B. Robinson updated FSAR [5] were reviewed in order to evaluate whether the following portions of the acceptance criteria were met:

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

The H. B. Robinson Unit 2 is a Westinghouse-designed, 3-loop, 2300-MWt plant.

The following systems provide the necessary protection against a steam pipe rupture:

- Safety injection system actuation on:

  a. two out of three low pressurizer pressure signals

  b. two out of three differential pressure signals between any steam line and steam line header

  c. high steam line flow in two out of three steam lines (one out of two per line) in coincidence with either low reactor coolant system average temperature (two out of three loops) or low steam line pressure (two out of three lines)

  d. two out of three high containment pressure signals.

- The overpower reactor trips (nuclear flux and differential temperature) and the reactor trip occurring upon actuation of the safety injection system.

- Redundant isolation of the MFW lines. A safety injection signal will close all MFW control valves, trip the main MFW pumps, and close the MFW block valves (safety grade). In addition, normal control action will close the MFW control valves.

- Trip of the fast-acting safety-grade steam line isolation valves (designed to close in less than 5 seconds with no flow) on:

  a. high steam flow in two out of three steam lines in coincidence with either low reactor coolant system average temperature or low steam line pressure

  b. two (two out of three) high-high containment pressure signals.

Each steam line has a fast-closing stop valve with downstream check valve. These six valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For breaks upstream of the stop valve in one line, closure of either the check valve in that line or the stop valves in the other lines will prevent blowdown of the other steam generators. For all breaks, this arrangement precludes blowdown of more than one steam generator inside the containment.

The AFW system for the plant includes a single turbine-driven pump (600 gpm) and two motor-driven pumps (300 gpm each), each of which can supply all three steam generators.

The AFW flow from one motor-driven pump supplying a steam generator will ensure that the heat removal capacity will exceed the minimum level required for decay heat removal after a MSLB.

The following signals are used for automatic initiation of the AFW system:

Motor-driven pumps

- o Low-Low steam generator level (two out of three channels on any steam generator)
- o Trip of both main feedwater pumps
- o Loss of all ac power
- o Safety injection

Turbine-driven pump

- o Low-Low steam generator level (two out of three channels on two steam generators)
- o Undervoltage on 4kV buses 1 and 4 (one out of two channels per bus).

The above systems are designed to meet safety-grade and IEEE Std 279-1971 requirements.

The environmental qualification of safety-related electrical and mechanical components is being reviewed separately by the NRC and is not within the scope of this review.

The Licensee's analysis determined that for the worst-case MSLB, which included runout AFW flow for 10 minutes, the resultant containment pressure attained was 34.4 psig, compared to the design pressure of 42 psig.

Sufficient indications and alarms are available to the operator to determine that a MSLB has occurred; once this determination has been made, the operator has minimal actions to perform in order to isolate AFW flow to the ruptured steam generator. It is conservative to assume that the operator will complete all required actions within the 10 minutes assumed in the analysis.

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 [9] and Regulatory Guide 1.97 [10].

Since cavitaton of the AFW pumps is not expected to occur at runout conditions, no damage would be expected.

3.1.3 Conclusion

The Licensee's responses [3 and 4] and the H. B. Robinson updated FSAR [5] adequately address the concerns of Item 1 of IE Bulletin 80-04. The containment pressure response analysis and the design of the mitigating systems satisfy the NRC's acceptance criteria. Regarding Item 1, it is concluded that there is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition. In addition, since the AFW pumps will not experience cavitation at runout flow conditions, the pumps will be able to carry out their intended function without damage.

3.2 REVIEW OF REACTIVITY INCREASE ANALYSIS

The requirement from IE Bulletin 80-04, Item 2, is as follows:

"Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return-to-power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return-to-power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient."

3.2.1 Summary of Licensee Statements and Conclusions

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

"The worst case steam line break is assumed to occur at hot zero power condition outside containment with offsite power available. At this time, the steam generator secondary side water inventory is at a maximum, prolonging the duration and increasing the magnitude of the primary loop cooldown. With negative moderator temperature coefficient, this causes reactivity insertion into the core. For conservatism, the most reactive control rod is assumed to be stuck out of the core when evaluating the shutdown capability.

With respect to additions of feedwater to the steam generator, main feedwater flow at hot zero power when the accident initiates is approximately 100-150 gpm/steam generator. Main feedwater isolates after approximately 10 seconds, so main feedwater flow additions to the steam generator inventory are insignificant. Upon safety injection actuation, auxiliary feedwater flow is initiated. It is estimated that this flow would be established at approximately t + 40 seconds. At t + 38 seconds, safety injection has reached the core, and the cooldown reactivity transient has peaked and core power is declining. The auxiliary feedwater flow will not be sufficient to reverse this trend.

In summary, the core cooldown transient is driven by the blowdown of the full-steam generator. Continued small flow additions represented by auxiliary flow capability are not significant contributions to the reactivity transient."

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 [5] analysis of the reactivity increase resulting from a MSLB and References 3 and 4 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 no load conditions, a double-ended rupture at the steam generator exit, with offsite power available, the core returns to power at 14 seconds, a maximum core power of 45% is predicted at 38 seconds. Shortly thereafter, 20,000 ppm boron solution reaches the core, rapidly shutting down the reactor. The calculated return-to-power did not result in a violation of the specified acceptable fuel design limits.

AFW flow is initiated at approximately 40 seconds, since the 20,000 ppm boron reaches the core before AFW reaches the steam generators, and since the negative reactivity inserted by the boron significantly exceeds the positive reactivity inserted by the cooldown caused by the addition of AFW, the core peak power will not be affected.

In addition, it can be assumed that the core transient is insensitive to runout AFW flow for the following reasons:

- 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
- 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
- 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 boron 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 insensitive to the contribution of AFW flow, and therefore the assumptions of the FSAR analysis remain valid.

3.2.3 Conclusion

The Licensee's response 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. Therefore, the FSAR analysis [5] 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 discussed above, no potential for containment overpressurization exists, and the return to power response is very insensitive to the addition of auxiliary feedwater. Therefore, no corrective action is required."

3.3.2 Evaluation and Conclusion

The Licensee's analysis determined that neither a containment overpressurization nor a reactor return-to-power with a violation of the specified acceptable fuel design limits would occur from a MSLB. Therefore, it was concluded that no further action regarding IE Bulletin 80-04 is required of CP&L for the H. B. Robinson Steam Electric Plant Unit 2.

4. CONCLUSIONS

With respect to the H. B. Robinson Unit 2, the conclusions regarding Carolina 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 MSLB with continued feedwater addition.
- o The AFW pumps will not experience cavitation at 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 FSAR MSLB reactivity increase analysis remains valid.
- o No further action regarding IE Bulletin 80-04 is required.

5. REFERENCES

1. "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition"
   NRC Office of Inspection and Enforcement, February 8, 1980
   IE Bulletin 80-04

2. "Overpressurization of the Containment of a PWR Plant after a Main Line Steam Break"
   NRC Office of Inspection and Enforcement, October 1, 1979
   IE Information Notice 79-24

3. L. W. Fury (CP&L)
   Letter to J. P. O'Reilly (NRC, Region II)
   Subject: IE Bulletin 80-04, Main Steam Line Break with Continued Feedwater
   May 9, 1980

4. Standard Review Plan, Section 4.2
   "Fuel System Design"
   NRC, July 1981
   NUREG-0800

5. H. B. Robinson Steam Electric Plant Unit 2
   Updated Final Safety Analysis Report
   Carolina Power and Light Company, 1982

6. Technical Evaluation Report
   "PWR Main Steam Line Break with Continued Feedwater Addition - Review of Acceptance Criteria"
   Franklin Research Center, November 17, 1981
   TER-C5506-119

7. "Criteria for Protection Systems for Nuclear Power Generating Stations"
   Institute of Electrical and Electronics Engineers, New York, NY, 1971
   IEEE Std 279-1971

8. Standard Review Plan, Section 15.1.5
   "Steam System Piping Failures Inside and Outside of Containment (PWR)"
   NRC, July 1981
   NUREG-0800

9. "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors"
   American Nuclear Society, Hinsdale, IL, December 1980
   ANS/ANSI-4.5-1980

10. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"
Rev. 2
NRC, December 1980
Regulatory Guide 1.97

11. "Single Failure Criteria for PWR Fluid Systems"
American Nuclear Society, Hinsdale, IL, June 1976
ANS-51.7/N658-1976

12. "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
Rev. 3
NRC, February 1976
Regulatory Guide 1.26

13. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"
Rev. 1
NRC, July 1981
NUREG-0588