TECHNICAL EVALUATION REPORT

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

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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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.

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1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This Technical Evaluation Report (TER) documents a review of the Florida Power Corporation's 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 Crystal River Unit 3. This evaluation was performed with the following objectives:

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

Florida Power responded to IE Bulletin 80-04 in a letter to the NRC dated May 8, 1980 [3]. In Reference 3, the Licensee stated that they were evaluating their steam line rupture matrix system and would submit the evaluation to the NRC. Information concerning this evaluation was forwarded in letters dated June 2, 1980 [4], September 3, 1981 [5], and November 17, 1981 [6]. Florida Power responded to a request for additional information concerning this review on June 11, 1982 [7]. The information in Reference 3 has been evaluated along with pertinent information contained in References 4, 5, 6, and 7 and the Final Safety Analysis Report (FSAR) [8] 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 [9]:

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

   a. A discussion of the continuation of flow to the affected steam generator, including the impact of runout flow from the AFW system and the impact of other energy sources, such as continuation of feedwater or condensate flow. AFW system runout flow should be determined from the manufacturer's pump curves at no backpressure, unless the system contains reliable anti-runout provisions or a more representative backpressure has been conservatively calculated. If a licensee assumes credit for anti-runout provisions, then justification and/or documentation used to determine that the provisions are reliable should be provided. Examples of devices for which provisions are reliable are anti-runout devices that use active components (e.g., automatically throttled valves) which meet the requirements of IEEE Std 279-1971 [10] and passive devices (e.g., flow orifices or cavitating venturis).

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

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

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

2. If containment overpressure or a worsening of the reactor return-to-power with a violation of the specified acceptable fuel design limits described in Section 4.2 of the Standard Review Plan [11] (i.e., increase in core reactivity) can occur by the licensee's analysis, the licensee shall provide the following additional information:

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

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

3. The acceptable input assumptions used in the licensee's analysis of the core reactivity changes during a MSLB are given in Section 15.1.5 of the Standard Review Plan [12]. The following specific assumptions should be used unless the analysis shows that a different assumption is more limiting:

Assumption II.3.b.:	Analysis should be performed to determine the most conservative assumption with respect to a loss of electrical power. A reactivity analysis should be conducted for a normal power situation as well as a loss of offsite power scenario, unless the licensee has previously conducted a sensitivity analysis which demonstrates that a particular assumption is more conservative.
Assumption II.3.d.:	The most restrictive single active failure in the safety injection system which has the effect of delaying the delivery of high concentration boric acid solution to the reactor coolant system, or any other single active failure affecting the plant response, should be considered.
Assumption II.3.g.:	The initial core flow should be chosen such that the post-MSLB shutdown margin is minimized (i.e., maximum initial core flow).

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

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

5. Modifications to electrical instrumentation and controls needed to detect and initiate isolation of the affected steam generator and feedwater sources in order to prevent containment overpressure and/or unacceptable core reactivity increases must satisfy safety-grade requirements. Instrumentation that the operator relies upon to follow the accident and to determine isolation of the affected steam generator and feedwater sources should conform to the criteria contained in ANS/ANSI-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors" [13], and the regulatory positions in Regulatory Guide 1.97, Rev. 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" [14].

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

7. The safety-grade requirements (redundancy, seismic and environmental qualifications, etc.) of the equipment that 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" [15].

- Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [16].

- Environmental qualification: The isolation valves should satisfy the requirements of NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" [17].

- 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

Regarding the containment pressure response analysis for the Crystal River Unit 3, the Licensee stated in Reference 3:

"The containment pressure response analysis relative to a main steam line break inside containment did not include the impact of runout from the Auxiliary Feedwater System nor the impact of other energy sources. The runout flow of Auxiliary Feedwater and Main Feedwater were not considered, as CR-3 is provided with a steam line rupture matrix with the ability to detect and isolate the damaged steam generator, following a main steam line rupture.

The main steam line rupture matrix detects a main steam line failure via main steam line header pressure. Subsequent to a steam line rupture, both steam generators begin to blow down at the same rate. The steam line rupture causes an increase in the heat transfer from the reactor coolant to the feedwater. This initiates a cooldown of the reactor coolant system, which increases the reactor power, due to the large negative moderator coefficient, such that the reactor trips on high flux in approximately 6.5 seconds. Reactor trip causes the turbine stop valves to close, isolating the unaffected steam generator on the steam side. Loss of main steam line pressure will actuate the matrix thus initiating closure of the main feedwater block valve, the feedwater low flow block, the feedwater pump suction valves and the auxiliary feedwater isolation valves. Following the isolation of the main steam lines the unaffected steam generator pressure recovers. Upon pressure recovery of the unaffected steam generator, the pressure switches activate the ICS to open the feedwater valve to permit feedwater flow to maintain the two-foot minimum level in the unaffected steam generator. On the affected steam generator the feedwater isolation valves have been closed on low steam line pressure (no recovery), neither primary system pressure recovery nor a return of the reactor to power will reopen these valves. Hence, continued feedwater through the affected steam generator is precluded."

The Licensee further stated the following concerning proposed modifications to the steam line rupture matrix [3]:

"Since our February 26, 1980 transient at CR-3, Florida Power Corporation has been evaluating the steam line rupture matrix system at CR-3. As stated in our May 2, 1980 submittal to the NRC, Florida Power has requested GAI and B&W to evaluate removing the isolation of the emergency feedwater valves from the rupture matrix.... This evaluation of the containment pressure response and steam line rupture matrix will be submitted to the NRC for review upon completion. If the evaluation supports not isolating the EFW via the rupture matrix, a minor control change will remove valves FWV-161 and FWV-162 from the rupture matrix actuation logic."

The evaluation of the removal of the emergency feedwater isolation valves from the rupture matrix was forwarded in References 4, 5, and 6.

3.1.2 Evaluation

The Licensee's submittal concerning containment pressure response analysis [4, 6] and applicable sections of the Crystal River FSAR [8] 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.

The Crystal River Unit 3 is a Babcock and Wilcox-designed, 2560 MWt, two-loop plant.

Following a MSLB, the main steam line rupture matrix system (SLRMS) is designed to isolate main steam, main feedwater, and emergency feedwater when low steam header pressure is detected. Although SLRMS and EFW system was not designed in accordance with IEEE Std 279-1971, a single failure in these systems would not preclude the safe shutdown of the plant. Because of an incident in which emergency feedwater was isolated to both steam generators, the Licensee proposed to remove the emergency feedwater isolation valves from the SLRMS. To ensure that this change would not produce more severe consequences than previously analyzed for in the FSAR, the Licensee performed an evaluation of the MSLB analysis, assuming emergency feedwater was not isolated to ruptured steam generator. This analysis was performed as an interim measure prior to the installation of the emergency feedwater initiation and control (EFIC) system, which is a safety-grade system designed to detect a MSLB, isolate the main steam and main feedwater systems and direct emergency feedwater to the intact steam generator.

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

The review did not determine 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 and Regulatory Guide 1.97.

Review of the steam line break analysis in References 4 and 6 determined that the effects of continued feedwater addition had been adequately addressed. The analysis considered the impact of emergency feedwater flow of 880 gpm to the affected steam generator on the containment pressure response to a double-ended MSLB until isolation by the operator. The analysis determined that the containment design pressure of 55 psig would be exceeded in approximately one hour without operator action to isolate the affected steam generator. It is conservative to assume that operator action to isolate the affected steam generator would occur within one hour; therefore, it can be concluded that there is no potential for containment overpressurization.

Flow of emergency feedwater to the steam generators is limited by flow control valves which are preset at 22% open to pass a minimum of 500 gpm to each steam generator. This will protect the EFW pumps from operating at runout conditions.

3.1.3 Conclusion

The Licensee's response and steam line break analyses [4, 6] adequately address the concerns of Item 1 of IE Bulletin 80-04. Regarding Item 1, it is concluded that there is no potential for containment overpressurization resulting from a MSLB with continued feedwater addition. Further, the EFW pumps should not experience runout conditions following a MSLB; due to the preset flow control valves, the pumps will be able to perform their intended function without incurring 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

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

"The steam line break analysis in the FSAR has been reviewed and it has been determined that all potential water sources have been considered as required by the licensing basis assumptions. Therefore, no corrective action has been identified."

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:

- Criterion 1.c - Ability to detect and isolate the damaged steam generator
- Criterion 1.d - Changes in core reactivity increase
- Criterion 3 - Analysis assumptions.

The steam line break analysis of the reactivity increase resulting from a MSLB [5] was reviewed. As discussed in Section 3.1.2 of this report, the modified SLRMS isolates the main steam and main feedwater system, but permits emergency feedwater flow to both steam generators throughout the transient. The analysis, which considered a double-ended rupture of a main steam line, assumed an emergency feedwater flow of 880 gpm to the affected steam generator and boron delivery to the core from only one high pressure injection pump. Further, a 1% shutdown margin at hot zero power conditions and no loss of offsite power were assumed.

The results of the analysis showed that the core remains subcritical throughout the transient. A minimum subcritical margin of 0.10% occurs at 16 seconds into the transient followed by an increasing subcritical margin.

3.2.3 Conclusion

The Licensee's responses [3, 4, 7] and analysis [5] adequately address the concerns of Item 2 of IE Bulletin 80-04. All potential sources of water were identified, no return to power occurs, and there is no violation of the specified acceptable fuel design limits. Therefore, the Licensee's MSLB reactivity increase analysis [5] 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:

> "Per our response to Questions 1 and 2 above, Question 3 does not apply to Crystal River Unit 3."

3.3.2 Evaluation and Conclusion

The Licensee's analysis determined that neither a containment overpressurization nor a reactor return-to-power resulting from a MSLB would occur. Therefore, it was concluded that no further action regarding IE Bulletin 80-04 is required of Florida Power Corporation for Crystal River Unit 3.

4. CONCLUSIONS

With respect to the Crystal River Unit 3, conclusions regarding Florida Power Corporation'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 pumps will not experience runout conditions; therefore, they will be able to carry out their intended function without incurring damage during a MSLB.
- o All potential water sources were identified; no reactor return-to-power occurs, and there is no violation of the specified acceptable fuel design limits. Therefore, the Licensee's MSLB reactivity increase analysis remains valid.
- o No further action regarding IE Bulletin 80-04 is required.

5. REFERENCES

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   "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition"
   NRC Office of Inspection and Enforcement, February 8, 1980

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15. "Single Failure Criteria for PWR Fluid Systems"
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Rev. 3
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Regulatory Guide 1.26

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