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TECHNICAL EVALUATION REPORT

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

TOLEDO EDISON COMPANY

DAVIS-BESSE NUCLEAR POWER STATION UNIT 1

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FOREWORD

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

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1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This Technical Evaluation Report (TER) documents an independent review of the Toledo Edison Company'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 the Davis-Besse Nuclear Power Station Unit 1. This evaluation was performed with the following objectives:

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

Toledo Edison responded to IE Bulletin 80-04 in a letter to the NRC dated May 5, 1980 [3]. The information in Reference 3 and pertinent information from the Davis-Besse Final Safety Analysis Report (FSAR) [4] were evaluated 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 [5]:

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 [6] 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. The justification should address the indication available to the operator and the actions required, particularly those outside the control room. If operator action is required to prevent exceeding a design value, i.e., containment design pressure or departure from nucleate boiling ratio (DNBR), then the discussion should include the calculated time when the design value would be exceeded if no operator action were assumed.

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 the licensee's analysis shows that containment overpressure or a reactor-return-to-power with a DNBR less than 1.32 (1.30 for Tong correlation) [*] can occur, then the licensee shall provide the following additional information:

   a. The proposed corrective actions to preclude overpressure or reactor-return-to-power and a schedule for completion of those actions.

   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 [7]. 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).

*Other values for minimum DNBR may be acceptable if justified for certain fuel designs and DNBR correlations.

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

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

5. Modifications to the electrical instrumentation and controls needed to detect and initiate isolation of the affected steam generator and feedwater sources in order to prevent containment overpressure and/or unacceptable core reactivity increases must satisfy safety-grade requirements. Instrumentation that the operator relies upon to follow the accident and to determine isolation of the affected steam generator and feedwater sources should conform to the criteria contained in ANS/ANSI-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors" [8], 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" [9].

6. Auxiliary feedwater 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 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" [10].

- Seismic requirements: The isolation valves should be designed to Category I as recommended in Regulatory Guide 1.26 [11].
- 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" [12].
- Quality standards: The isolation valves should satisfy Group B quality standards as recommended in Regulatory Guide 1.26 or similar quality standards from the plant's licensing bases.

3. TECHNICAL EVALUATION

The scope of work included the following:

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

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

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

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

This report constitutes a TER in satisfaction of 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 the 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."

a. Summary of Licensee Statements and Conclusions

The Licensee made the following statements regarding the ability to detect and isolate a MSLB:

"Following a postulated double ended rupture of a main steam line inside the containment vessel, low pressure switches located in the main steam line outside containment will actuate the Steam and Feedwater Rupture Control System (SFRCS). The reactor trips on low reactor coolant system pressure. SFRCS will trip the turbine-generator, initiate closure of the main steam isolation valves in both main steam lines, initiate main feedwater isolation for both steam generators, and will start the auxiliary feedwater system.

"SFRCS will determine which is the affected steam generator. No auxiliary feedwater will be added to the affected steam generator, as SFRCS will align both auxiliary feedwater pumps to supply water only to the unaffected steam generator.

"Following main steam and main feedwater isolation of the steam generator, the unaffected steam generator will repressurize while the affected steam generator will continue to blow down and will not repressurize. The redundant SFRCS pressure switches on the main steam lines of each steam generator (set of 600 psig) are the means of detecting which is the affected generator. These pressure switches are interlocked with the valves in the auxiliary feedwater system to prevent the addition of auxiliary feedwater to the affected steam generator."

In regard to the review of the containment pressure response analysis for the Davis-Besse plant, the Licensee stated:

"During that period of time required to detect MSLB and close the main steam and feedwater valves, main feedwater will continue to flow to both the affected and the unaffected steam generators. This feedwater addition has been considered in the MSLB analysis and is described in the response to FSAR Question 15.4.8 and is tabulated in FSAR Table 15.4.4.2. We have reviewed the data used in the analysis and have found that the values used for feedwater addition following the MSLB are conservative.

"The values used in the analysis exceed the expected feedwater addition that would result from the most adverse response of the non-safety grade portion of the main feedwater system (control valves wide open and main feed pump turbine overspeed) and the most limiting single failure in the safety grade portions of the feedwater isolation system (failure of the main feedwater stop valves to close on the affected steam generator)."

The Licensee concluded:

"Since our review has determined that the existing FSAR analysis for containment overpressure following MSLB conservatively models continued feedwater addition, no further action is required."

Regarding the AFW pump's ability to remain operable after extended operation at runout flow, the Licensee stated:

"Because the auxiliary feedwater system only supplies water to the unaffected steam generator, there is no runout of the auxiliary feedwater pumps and their continued availability is not affected."

b. Evaluation

The Licensee's submittal concerning containment pressure response analysis and applicable sections of the Davis-Besse FSAR were reviewed in order to evaluate whether the following portions of the acceptance criteria were met:

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

A review of Section 7 of the Davis-Besse FSAR determined that the SFRCS system is designed as an engineered safety features (ESF) system to Seismic Category I and safety-grade requirements, and the initiating signals and circuits were designed to meet the criteria of IEEE Std 279-1971.

In the event of a MSLB, the SFRCS is designed to:

- o isolate the main steam and main feedwater system
- o start both AFW system pumps
- o align AFW flow to feed only the unaffected steam generator.

Steam generator level then is automatically maintained by the AFW system. The operator also has the option to take manual control of steam generator level or transfer control to the integrated control system (ICS), which will maintain a wider band of steam generator level.

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 qualfication of the instrumentation that the operator relies upon to follow accident and determine isolation of the affected steam generator was not determined.

Sufficient AFW flow is available to the unaffected steam generator to ensure that system heat removal capacity exceeds the minimum level required for decay heat removal after a MSLB.

Review of the steam line break analysis in the FSAR determined that the effects of continued feedwater addition had been adequately addressed. The SFRCS and AFW system are designed so that, even if a single failure to either system would occur, AFW flow would not reach the affected steam generator. The worst-case single failure is that of the MFW control valves (control-grade) remaining 100% open, allowing MFW pump runout to 135% full MFW flow to the affected steam generator for 17 seconds until the feedwater isolation valves (safety-grade) close. The FSAR analysis determined that the resultant release of mass and energy from the blowdown of one steam generator would increase the containment pressure to 21.4 psig, well below the design pressure of 36 psig.

The continued availability of the AFW pumps would not be affected since the pumps do not experience runout flow.

c. Conclusions and Recommendations

The Licensee's response and FSAR adequately address the concerns of Item 1 of IE Bulletin 80-04. The containment pressure response analysis and the design of the SFRCS 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 do not experience runout conditions, the pumps will be able to carry out 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."

a. Summary of Licensee Statements and Conclusions

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

"...the rupture of a main steam line between the steam generator and the main steam isolation valve (MSIV) represents the worst condition for accident analysis. Our review of feedwater addition described in the response to Item 1 of this Bulletin has also considered a break outside the containment and upstream of the MSIV. A break in this location will result in slightly less feedwater addition due to the proximity of the SFRCS pressure switch taps to the break and the consequent earlier SFRCS trip signal.

"Since our review has determined that existing FSAR analysis of reactivity increase following MSLB inside or outside containment considers all potential water sources, and conservatively models those sources, no further action is required."

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

Review of the FSAR analysis of the reactivity increase resulting from a MSLB determined that the analysis is conservative in its assumptions and that the assumptions are in accordance with those in Acceptance Criteria 3.

As discussed in Section 3.1.b of this report, the SFRCS isolates all potential water sources from the affected steam generator. In the worst case (double-ended MSLB between the steam generator and main steam isolation valve), no return to criticality occurs, the minimum subcritical margin during the transient is 0.69%$\Delta k/k$, and the minimum DNBR achieved during the transient is 1.42.

All potential water sources were considered in the FSAR analysis of the reactivity increase resulting from a MSLB and no further action is required.

c. 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, the SFRCS isolates all the potential water sources, no return-to-power occurs, and the DNBR remains greater than 1.30. Therefore, the FSAR analysis remains valid and no further action is required.

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."

a. Summary of Licensee Statements and Conclusions

The Licensee stated:

"Since our review has determined that the existing FSAR analysis for containment overpressure following MSLB conservatively models continued feedwater addition, no further action is required.

"Since our review has determined that the existing FSAR analysis of reactivity increase following MSLB inside or outside containment considers all potential water sources, and conservatively models those sources, no further action is required."

b. Evaluation

The Licensee's conclusions with respect to the corrective actions required as a result of the review were evaluated against Acceptance Criteria 2.

The existing FSAR analysis for containment overpressurization following a MSLB takes into account all potential water sources and no further action is required by the Licensee.

Since all potential water sources were identified in the FSAR analysis of the reactivity increase following a MSLB, no further action is required.

c. Conclusion and Recommendations

The Licensee's analysis determined that containment overpressurization or a worsening of a reactor return-to-power with a DNBR of less than 1.30 resulting from a MSLB would not occur. Therefore, it is concluded that no further action regarding IE Bulletin 80-04 is required of Toledo Edison for the Davis-Besse Nuclear Power Station Unit 1.

4. CONCLUSIONS

With respect to the Davis-Besse Nuclear Power Station Unit 1, conclusions regarding Toledo Edison'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 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 and no reactor-return-to-power occurs; therefore, the FSAR reactivity increase analysis remains valid.
- o No further action is required by the Licensee regarding IE Bulletin 80-04.

5. REFERENCES

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   "PWR Main Steam Line Break with Continued Feedwater Addition - Review of Acceptance Criteria"
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