# DRAFT ECCS SAFETY EVALUATION REPORT

## SUPPLEMENT

## DAVIS BESSE UNIT NO. 1

### 1.0 Introduction

In Section 6.3.3 of the FSAR, the applicant (Toledo Edison Company) incorporated by reference B&W topical reports BAW-10104 and BAW-10105 (References 1 & 2, respectively) into its application to operate Davis Besse Unit No. 1. Pursuant to the requirements of 10 CFR 50.46, B&W submitted these reports to demonstrate compliance with the ECCS Acceptance Criteria for its 177 fuel assembly plants with raised loops. The basis for acceptance of the principal portions of the B&W evaluation model were set forth in the staff's Status Report of October 1974 (Reference 3) and the Supplement to the Status Report of November 1974 (Reference 4). Together, the Status Report and its Supplement describe the B&W ECCS evaluation model and the basis for the staff's previous acceptance of the model. BAW-10104 describes the general features of the B&W ECCS evaluation model and reflects the modifications previously required by the staff (References 5 and 6). The original ECCS calculations applicable to Davis-Besse 1 were submitted in BAW-10105 (Reference 2) using the B&W evaluation model described in BAW-10104 (Reference 1). Later developments on the

8002120965

validity of these calculations determined the following:

1. The B&W method for calculating fuel cladding temperatures during the blowdown phase of a LOCA did not conform to Appendix K because it allowed for a return to nucleate boiling after critical heat flux conditions have been reached.

2. A steam cooling model was used in the Davis Besse 1 ECCS calculations which had not been reviewed by the staff.

3. Improper pin pressure assumptions were employed.

4. Incorrect values of certain loop resistances were used.

With regard to item 1 above, Reference 20 provides the staff evaluation of a revised nucleate boiling lockout logic proposed by B&W. The staff concludes that the revised logic is an appropriate change to be incorporated in the B&W Evaluation Model and that the overall effect on the change on peak clad temperature would be small ($\sim 6^{\circ}F$).

With regard to item 2 above, the staff has concluded that the steam cooling model used by B&W is acceptable.* Items 3 and 4 relate to input errors and are discussed in more detail in Section 2.0 of this Safety Evaluation Report Supplement.

Other model changes have taken place subsequent to the ECCS calculations in BAW-10105 (References 23 and 24). These changes have been accepted by the staff and their cumulative effect is not significant to the peak clad temperature.

---

*See comment in cover letter. The staff steam cooling model safety evaluation should be referenced here if published in time.

## 2.0 ECCS Analyses

The background of the staff's review of the revised B&W ECCS evaluation model and its application to Davis Besse Unit No. 1 is described in Reference 5. The applicant's FSAR contains documentation by reference to BAW-10105 of a generic break spectrum appropriate to Davis Besse Unit No. 1. It is the staff's understanding that the responses to questions submitted on BAW-10105 (References 18, 19) will be made a part of the topical report by B&W. A spectrum of break sizes, configurations, and locations were performed. These analyses identified the worst break as the 8.55 $ft^2$ double-ended break at the pump discharge. B&W responses to staff inquiries during its review of BAW-10105 determined that incorrect internal fuel pin pressures had been assumed in the ECCS calculations. B&W subsequently resubmitted analyses in Reference 18 with the corrected pin pressures. These revised analyses also included consideration of an additional flow resistance in the cold legs to account for HPI pumps injecting ECC water during reflood. The table below summarizes the results of the revised LOCA limit analyses which determine the allowable linear heat generation rate limits as a function of elevation in the core:

| Elevation (ft) | LHGR Limit (Kw/ft) | Peak Cladding Temperature (°F) | Maximum Local Oxidation (%) |
|---|---|---|---|
| 2 | 16.5 | 2133 | 4.01 |
| 4 | 17.2 | 2073 | 3.15 |
| 6 | 18.4 | 2166 | 5.25 |
| 8 | 17.5 | 2164 | 6.56 |
| 10 | 17.0 | 2194 | 7.17 |

Subsequent to this review, Toledo Edison Company informed the staff that an erroneous resistance value to the reactor vessel inlet nozzle had been used in the loss-of-coolant accident (LOCA) analysis. As a result, the applicant submitted a re-evaluation of the Davis Besse Unit No. 1 LOCA analysis based on the corrected inlet nozzle model and a revised system pressure distribution. (References 21 and 22) These results show that lower peak cladding temperatures would be obtained for the worst break analysis. The peak cladding temperature obtained for the reevaluation of the 6-foot LOCA limit analysis is 2133°F, a value 33°F lower than obtained in BAW-10105 (see tabulation on preceding page).

The reason for a reduction in peak cladding temperatures compared to those reported in BAW-10105 was due to improved reflooding rates in the core following a LOCA. The increased core reflooding rates are based on an improved system pressure distribution (i.e. the new reactor coolant system total pressure drop is less than the original assumed pressure drop). We have reviewed the proposed pressure drops, the derivation of the revised system pressure distribution and its impact on the LOCA limit analysis, and agree with the applicant that the proposed linear heat generation limits as a function of core elevation are in compliance with the criteria of 10 CFR 50.46. Also, we conclude that the reevaluation of the 6-foot LOCA limit is sufficient to determine the effect of the revised system pressure distribution on peak cladding temperature for the range of axial power distributions previously analyzed. As reported earlier, the peak cladding temperature following a LOCA was reduced when analyzed at the 6-foot elevation of the core. Similar effects would be expected at other elevations of the core. Although the reevaluated results are less severe than those reported in BAW-10105, the applicant will maintain the allowable linear heat generation rate limits for Davis Besse Unit No. 1 at the same values as previously identified in this report. Additionally, because of the changes described in references 20 through 24, the staff requires that Toledo Edison Company submit within 6 months additional analyses to further quantify existing margins. The additional analyses should, as a minimum, confirm previous evaluations with regard to worst break size, configuration, and allowable linear heat generation limits as a function of elevation in the core.

Also, we will require the applicant to provide operating reactor coolant system flow data for the Davis Besse Unit No. 1 which can be used to further verify the assumed total system pressure drop. The new pressure drops were based on standard calculation methods supported by operating plant pressure data and the results from scaled reactor vessel model flow tests. B&W has shown that although there are some design differences between Davis Besse Unit No. 1 and other B&W plants from which measured data were obtained, these differences have a negligible effect on total system pressure drop. We have reviewed the flow path resistances input to the REFLOOD ECCS evaluation code for the revised system pressure distribution, and have checked several flow paths resistance values. We find the methods to be appropriate for the derivation of loop resistances and accept the reported values as being appropriate for Davis Besse Unit No. 1.

Therefore, the staff concludes that the previous values quoted in the tabulation remain applicable to Davis-Besse 1.

The maximum core-wide metal-water reaction was calculated to be 0.66 percent, a value which is below the allowable limit of 1 percent. As shown in the tabulation, the calculated values for peak clad temperature and local metal-water reaction were below the allowable limits specified in 10 CFR 50.46 of 2200°F and 17 percent, respectively. BAW-10105 has also shown that the core geometry remains amenable to cooling and that long-term core cooling can be established.

Our review of other plant-specific assumptions discussed in the following paragraphs regarding the Davis-Besse 1 analyses addressed the areas of single failure criterion, long-term boron concentration, potential submerged equipment, partial loop operation, and the containment pressure calculation.

## 2.1 Single Failure Criterion

Appendix K to 10 CFR 50 of the Commission's regulations requires that the combination of ECCS subsystems to be assumed operative shall be those available after the most damaging single failure of ECCS equipment has occurred. Babcock and Wilcox has conservatively assumed all containment cooling systems operating to minimize containment pressure and has independently assumed the loss of one diesel to minimize ECCS cooling. We stated in Reference 3 that the application of the single failure criterion was to be confirmed during subsequent plant reviews.

The applicant has concluded that no single active failure would more severely degrade ECCS than the previous assumptions stated above. A review of the Davis-Besse 1 piping and instrumentation diagrams and ECCS motor-operated valve electrical schematics were conducted by the staff. From these reviews the staff required valve changes in the LPI discharge lines, LPI-HPI crossover lines, and HPI mini-flow bypass lines. On the basis of the revised plant design, the staff concludes that a bounding single failure analysis has been performed for the Davis-Besse Unit No. 1 plant.

## 2.2 Containment Pressure

The ECCS containment pressure calculations for 177-FA raised loop plants were performed generically by B&W as described in Reference 2. The NRC staff reviewed B&W's evaluation model and published the results of this review in References 3 and 4. We concluded that B&W's containment pressure model was acceptable for ECCS evaluations. We required that justification of the plant-dependent input parameters used in the containment analyses be submitted for our review of each plant.

Justification for the containment input data was submitted for Davis-Besse 1 on September 5, 1975 (Reference 8). This justification allows comparison of the actual containment parameters for Davis-Besse 1 with those assumed in Reference 2. Toledo Edison Company has evaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat-removal systems with regard to the conservatism for the ECCS analysis. This evaluation was based on as-built design information. The containment heat removal systems were assumed to operate at their maximum capacities, and lowest expected values for the spray water and service water temperatures were assumed. The containment pressure analysis in BAW-10105 was demonstrated to be conservative for Davis-Besse 1.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Davis-Besse 1 is conservative and, therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR 50 of the Commission's regulations.

2.3 Long-Term Boron Concentration

The NRC staff has reviewed the proposed procedures and the systems designed for preventing excessive boric acid buildup in the reactor vessel during the long-term cooling period after a LOCA. Toledo Edison Company has agreed to implement procedures for Davis-Besse 1 which would allow adequate boron dilution during the long term and which will comply with the single failure criterion. These procedures will employ a hot leg drain and hot leg injection network similar to the concept described in BAW-10105. The hot leg drain mode will direct reactor coolant from the hot leg, down the decay heat line to the DHR pump suction. This coolant draining from the hot leg would then be mixed with the dilute water being pumped from the containment sump and would then be pumped back to the reactor vessel. Should a single active component failure not allow operation of the hot leg drain mode, the operator then has the alternative of selecting the hot leg injection mode to provide boron dilution. The procedure would be to use the relatively dilute water being pumped out of the containment sump during the long-term recirculation mode and route a minimum of 40 gpm of this sump water to the hot leg to provide dilution of the water in the upper plenum of the reactor vessel. The applicant will be required to demonstrate this minimum flow rate in each mode during preoperational testing. In addition, the applicant must install flow rate measuring devices to assure that a minimum of 40 gpm is continually available following a LOCA, and to facilitate system tests. With the addition of the flow devices and preoperational tests, this proposal is acceptable to the staff.

2.4 Submerged Valves

The applicant has conducted a review of equipment arrangement to determine if any components inside the containment will become submerged following a LOCA. Based on this review, decay heat suction valves DH-11 & DH-12 were identified as being located in an area that will be flooded. The applicant subsequently enclosed these valves in a water-tight compartment to ensure their operability during the long term after a LOCA. The staff will require that an acceptable leakage test of this enclosure be performed each refueling outage. Simple visual inspection would not be sufficient.

## 2.5 Partial Loop Analyses

To support an operating configuration with less than four reactor coolant pumps on the line (partial loop), the staff requires an analysis of the predicted consequences of a LOCA occurring during the proposed partial loop operating mode(s). Toledo Edison Company submitted an analysis for partial loop operation with one idle reactor coolant pump (three pumps operating) in Reference 9. Using a reduced power level of 77% of rated power, B&W performed this analysis assuming the worst case break ($8.55\ ft^2$ DE, $C_D = 1$) and maximum LHGR allowed by Technical Specifications for this mode of operation. Based on a sensitivity study referred to by the applicant in Reference 14, the break selected was located in the active leg of the partially idle loop. Placing the break at the discharge of the pump in an active cold leg of the partially idle loop (instead of at the discharge of the pump in an active cold leg of the fully active loop) yields the most degraded positive flow through the core during the first half of the blowdown and results in higher cladding temperatures. The maximum cladding temperature for the one-idle-pump mode of operation was 1675°F, a value which is within the criterion of 10 CFR 50.46. Therefore, this analysis may be used to support the applicant's proposed operation with one idle reactor coolant pump.

Since an analysis of ECCS cooling performance with one idle reactor coolant pump in each loop has not been submitted, power operation in this configuration must be limited by Technical Specifications to 24 hours.

Single loop operation (i.e., operation with two idle pumps in one loop) is prohibited by current Technical Specifications without notifying the staff. Each proposal for a scheduled single loop test will be considered on a case-by-case basis.

## 3.0 Conclusions

The staff has completed its review of the Davis-Besse Unit No. 1 ECCS performance analyses and has concluded:

a. The ECCS minimum containment pressure calculations were performed in accordance with Appendix K to 10 CFR 50.

b. With the modifications described herein, the single failure criterion will be satisfied.

c. The proposed procedures for long-term cooling after a LOCA are acceptable to the staff. The implementation of these procedures before startup is required to provide assurance that the ECCS can be operated in a manner which would prevent excessive boric acid concentration from occurring.

d. The proposed mode of reactor operation with one idle reactor coolant pump is supported by a LOCA analysis. Operation with one idle pump in each loop is restricted to 24 hours. Requests for single loop operation will be reviewed on a case-by-case basis.

e. Additional analyses are required within six months to further quantify existing margins.

References


1. B. M. Dunn, et al., "B&W's ECCS Evaluation Model," BAW-10104, Babcock and Wilcox, May 1975.

2. W. L. Bloomfield, et al., "ECCS Evaluation of B&W's 177-FA Raised-Loop NSS," BAW-10105, Babcock and Wilcox, June 1975.

3. "Status Report by the Directorate of Licensing in the Matter of Babcock & Wilcox ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," dated October 1974.

4. "Supplement 1 to the Status Report to the Directorate of Licensing in the Matter of Babcock and Wilcox ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," dated November 13, 1974.

5. Letter from A. Schwencer to Mr. Kenneth E. Suhrke, dated January 8, 1976.

6. Letter from John F. Stolz to Mr. Kenneth E. Suhrke, dated January 16, 1976.

7. "Minimum Requirements for ECCS Break Spectrum Submittals," dated April 25, 1975.

8. Letter from Lowell E. Roe to Mr. A. Schwencer, dated September 5, 1975.

9. Letter from Lowell E. Roe to Mr. A. Schwencer, dated October 8, 1975.

10. Letter from Kenneth E. Suhrke to Mr. A. Schwencer, dated December 15, 1975.

11. Letter from Kenneth E. Suhrke to Mr. John F. Stolz, dated April 6, 1976.

12. Letter from Lowell E. Roe to Mr. A. Schwencer, dated July 9, 1975.

13. Letter from Lowell E. Roe to Mr. A. Schwencer, dated July 21, 1975.

14. Letter from Lowell E. Roe to Mr. Benard C. Rusche, dated May 17, 1976.
15. Letter from Lowell E. Roe to Mr. Benard C. Rusche, dated June 4, 1976.
16. Letter from Lowell E. Roe to Mr. Benard C. Rusche, dated November 8, 1976.
17. Letter from Lowell E. Roe to Mr. John F. Stolz, dated January 3, 1977.
18. Letter from Kenneth E. Suhrke to Mr. John F. Stolz, dated April 6, 1976.
19. Letter from Kenneth E. Suhrke to Mr. John F. Stolz, dated June 8, 1976.
20. Letter from Steven A. Varga to Mr. Kenneth E. Suhrke, dated February 18, 1977.
21. Letter from Lowell E. Roe to Mr. John F. Stolz, dated February 8, 1977.
22. Letter from Lowell E. Roe to Mr. John F. Stolz, dated April 1, 1977.
23. Letter from Kenneth E. Suhrke to Mr. D. B. Vassallo, dated August 20, 1976.
24. Letter from Kenneth E. Suhrke to Mr. Denwood F. Ross, Jr., dated June 7, 1976.