SAFETY EVALUATION REPORT

OCONEE UNIT 1

DOCKET NO. 50-269

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1.0 INTRODUCTION

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License (Reference 2) implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR 50.46. The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendment as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, Duke Power Company (the licensee) has submitted an ECCS re-evaluation and related Technical Specifications. The re-evaluation and Technical Specifications were submitted in References 1 and 4 using the B&W ECCS evaluation model as described in Reference 7 and discussed in Section 2.0 of this Safety Evaluation Report. Also discussed in Section 2.0 are the results of a staff review of the plant-specific areas of single failures, long-term boron concentration, potential submerged equipment, partial loop operation, ECCS valve interlocks, and the containment pressure calculation. Section 3.0 provides the results of the staff review of the proposed Oconee 1 Technical Specifications, and Sections 4.0 and 5.0 present staff conclusions and references, respectively.

2.0 ECCS RE-EVALUATION

The background of the staff review of the B&W ECCS evaluation model and its application to Oconee is described in the staff SER for this facility dated December 27, 1974, issued in connection with the Order for Modification of License. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 (Reference 5) and the Supplement to the Status Report of November 1974 (Reference 6) which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier version of the B&W model. Together, the December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Oconee 1 ECCS evaluation which is covered by this safety evaluation report properly conforms to the accepted model. The licensee's July 9, 1975 submittal (Reference 1) contains documentation by reference to B&W Topical Reports of the revised ECCS model (with the modifications described in our December 27, 1974 SER) and a generic break spectrum appropriate to Oconee 1 (Reference 7 and 8, respectively). In addition, Duke Power Company included in this July 9th submittal a separate analysis of the worst break for

Oconee Unit 1, using the following plant-specific parameters:

a. Power level = 1.02 x 2568 Mwt. The generic analyses in BAW-10103 used 1.02 x 2772 Mwt.

b. Initial average fuel temperature assumed reflects the reload core (T = 3030°F for 18 kw/ft with 580°F sink temperature). The generic analyses used T = 3050°F.

c. Different pin dimensions were employed to reflect fuel changes.

d. Core flood tank line resistance was changed to reflect the as-built value for Oconee Unit 1 (6.5 versus 7.75 in generic analyses).

e. System enthalpies and steam generator heat loads were changed to reflect the lower power level of 2568 Mwt.

f. Initial pin pressures and oxide layer thicknesses were changed to reflect the different fuel in Oconee 1.

The generic analysis in BAW-10103 identified the worst break size as the 8.55 ft^2 double-ended cold leg break at the pump discharge with a C_D = 1.0. The below table summarizes the results of the LOCA limit analyses which determine the allowable linear heat rate limits as a function of elevation in the core for Oconee Unit 1:

Elevation (ft)	LOCA Limit (kw/ft)	Peak Cladding Temperature (°F)		Max. Local Oxidation (%)	Time of Rupture (sec)
		Ruptured Node	Unruptured Node		
Oconee 1					
2	16.0	1882	1930	3.40	10.90
4	17.5	1975	1978	3.17	12.39
6	18.0	2066	2146	5.46	15.55
8	17.0	1743	2110	5.19	15.01
10*	16.0	1642	1931	2.93	39.20

*See discussion in text.

The maximum core-wide metal-water reaction for Oconee 1 was calculated to be 0.557 percent, a value which is below the allowable limit of 1 percent.

As shown in the tabulation, the calculated values for the 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-10103 has also shown that the core geometry remains amenable to cooling and that long-term core cooling can be established.

The staff noted during its review of BAW-10103 that the LOCA limit calculation at the 10-foot elevation in the core showed reflood rates below 1 inch/second at 251 seconds into the accident (Section 7.2.5). Appendix K to 10 CFR 50.46 requires that when reflood rates are less than 1 inch/second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer. As indicated by the staff in References 5 and 6, a steam cooling model for reflood rates less than 1 inch/second was not submitted by B&W for staff review. The steam cooling model submitted by B&W in BAW-10103 is therefore considered to be a proposed model change requiring further staff review and ACRS consideration. Accordingly, B&W was informed that until the proposed steam cooling model is reviewed, the heat transfer calculation at the 10-foot elevation during the period of steam cooling specified in BAW-10103 must be further justified. In lieu of using their proposed steam cooling model, B&W has submitted the results of calculations at the 10-foot elevation using adiabatic heatup during the steam cooling period, where this period is defined by B&W as the time when the reflood rate first goes below 1 inch/second to the time that REFLOOD predicts the 10-foot elevation is covered by solid water. The new calculated peak cladding temperature, local metal-water reaction and core-wide metal-water reaction at the 10-foot elevation are 1946°F, 3.02%, and .647%, respectively. These values remain below the allowable limits of 10 CFR 50.46 and are acceptable to the staff. Until a steam cooling model has been accepted by the staff, these values will serve as the LOCA results for Oconee 1 at the 10-foot elevation.

As indicated in a previous paragraph, Duke Power Company elected to provide a plant-specific calculation for Oconee Unit 1 utilizing selected as-built data. We have reviewed the input changes used (relative to BAW-10103) and believe them appropriate for Oconee Unit 1.

Our review of other plant-specific assumptions discussed in the following paragraphs regarding the Oconee 1 analyses addressed the areas of single failure criterion, long-term boron concentration, potential submerged equipment, partial loop operation, ECCS valve interlocks, 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 assumed all containment cooling systems operating to minimize containment pressure and has separately assumed the loss of one diesel to minimize ECCS cooling. We concluded in Reference 5 that the application of the single failure criterion was to be confirmed during subsequent plant reviews.

A review of Oconee 1 piping and instrumentation diagrams indicated that the spurious actuation of certain motor-operated valves could affect the appropriate single failure assumptions. A spurious actuation of core flooding tank (CFT) vent valves CF-5 or CF-6 would result in a decrease in CFT pressure. The rate at which this decrease occurs is controlled by a preset needle throttling valve downstream of the electrically-operated valve (CF-5 or CF-6). The predetermined position of the needle valve is provided by manually turning the local handwheel such an amount as to limit the rate at which a depressurization of the CFT could take place. A recent test at Oconee indicated that the tested valve setting allowed 17 minutes for the CFT pressure to decay from 625 psi to the low pressure alarm, 580 psi,when the electrically-operated valves were opened. Since it is clear that CFT pressure is important to mitigating the consequences of a LOCA, a Station Technical Specification must be adopted, either for the position of the manual throttling valves or for the motor-operated valves. Since presetting the throttling valve by turning the handwheel an amount equal to the aforementioned depressurization rate does not appear to be sufficiently accurate to serve as a safeguard against an uncontrolled CFT blowdown, we will require that the normally closed motor-operated valves CF-5 and CF-6 have their power disconnected and their associated breakers locked open.

A review was also conducted of the electrical schematics for ECCS motor-operated valves. It was determined that a single failure of valve interlocks could not affect the appropriate single failure assumptions.

To further minimize the potential for a water hammer due to the discharge of ECC water into a dry line, the staff requires that valves LP-21, LP-22, HP-24 and HP-25 be left in the open position during normal operation. This maintains the ECCS lines filled with a continual supply of water from the BWST due to the available static head built into the design. Such a configuration will also eliminate the need for one automatic safety action in the event of a LOCA; that is, the automatic opening of these valves to provide water to the ECCS and Building Spray pumps. In addition, Duke Power Company will be required to adopt a Technical Specification whereby a monthly venting of ECCS pump casings and high points in ECCS lines will be performed to ensure that no air pockets have formed. Such venting must also be performed prior to any ECCS flow tests.

2.2 Containment Pressure

The ECCS containment pressure calculations for Oconee Class plants were performed generically by B&W for reactors of this type as described in Reference 8. The NRC staff reviewed B&W's evaluation model and published the results of this review in References 5 and 6. 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. A containment pressure calculation specific to Oconee 1 was submitted in Reference 1.

Justification for the containment input data was submitted for Oconee 1 on October 10, 1975 (Reference 11). This justification allows comparison of the actual containment parameters for Oconee 1 with those assumed in References 1 and 8. Duke Power 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 minimum operation values for the spray water and service water temperatures were assumed. The containment pressure analysis was demonstrated to be conservative for Oconee Unit 1.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Oconee 1 is reasonably 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 buildups in the reactor vessel during the long-term cooling period after a LOCA. Duke Power Company has agreed to implement procedures for Unit 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 network similar to the concept described in BAW-10. . To employ a single failure . oof mode, Duke Power Company will make modifications to the existing DHR design during the next refueling outage. The proposal consists of the addition of two drain lines from the decay heat drop line to the sump. One line (installed upstream of the DHR isolation valves) will include two qualified motor-operated valves. The other line (installed downstream of the DHR isolation valves) will include one qualified motor-operated valve. The Licensee will be required to install a flow rate measuring device to confirm that a minimum of 40 gpm are available following a LOCA, and to facilitate system tests. With the addition of this flow device, this proposal is acceptable to the staff. A final review of the installed system will be conducted before startup.

2.4 Submerged Valves

The applicant has conducted a review of equipment arrangement to determine if any valve motors inside the containment will become submerged following a LOCA. Based on this review, no valves were identified which would be flooded and which would affect short-term or long-term ECCS functions or containment isolation.

2.5 Partial Loop Analyses

To allow 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). Duke Power 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 (18.0 kw/ft) from the 4-pump analysis discussed in Section 2.0. The worst 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 1766°F. A staff review of all input assumptions and conclusions resulted in a set of inquiries which were answered in References 4 and 10. The results of a new analysis was submitted to reflect a more appropriate value of initial pin pressure. The original partial loop analysis in Reference 9 used an initial pin pressure of 1600 psi. As was demonstrated in the time-in-life sensitivity study in Reference 8, the worst pin pressure for this analysis should have been 760 psi. The maximum cladding temperature for the re-analysis is 1784°F, a value which is within the criterion of 10 CFR 50.46. Therefore, this analysis may be used to support Duke Power Company's proposed operation with one idle reactor coolant pump.

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

-le loop operation (i.e., operation with two idle pumps in one notifying the Commission. Each proposal for a scheduled single loop test will be considered on a case-by-case basis.

3.0 TECHNICAL SPECIFICATIONS

We have reviewed the Technical Specifications proposed by Duke Power Company to assure that operation of Oconee 1 is within the limits imposed by the Final Acceptance Criteria. These criteria permit an increase in the allowable heat generation rate from 15 to 16 kw/ft at the 10-foot elevation compared to the Interim Acceptance Criteria currently in effect. For Unit 1, the LOCA-related heat generation limits occur in the reload fuel (Batch 4). The limits for this fuel are not changed from those now in effect for the bottom half of the core.

Although the proposed rod insertion limits for Unit 1 are not changed from those which are currently in effect, the burnup at which Group 7 withdrawal begins was modified from 250 ± 10 EFPD to 245 ± 10 EFPD. This has made necessary a reduction in the allowable positive axial imbalance from 14% to 10% at full power. We find the revised Technical Specification (See Reference 4) to be acceptable.

4.0 CONCLUSIONS

The staff has completed its review of the Oconee 1 ECCS performance re-analyses and has concluded:

a. The proposed Technical Specificationsare based on a LOCA analysis performed in accordance with Appendix K to 10 CFR 50.

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

c. The single failure criterion will be satisfied provided that the modifications specified in subsection 2.1 of this Safety Evaluation Report are implemented.

d. The proposed procedures for long-term cooling after a LOCA are acceptab' to the staff. The implementation of these procedures d· ing the next refueling outage is required to provide assurance that the ECCS can be operated in a manner which would prevent excessive boric acid concentration from occurring.

e. The proposed mode of reactor operation with one idle reactor coolant pump is supported by a LOCA analysis performed in accordance with Appendix K to 10 CFR 50. 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.

5.0 REFERENCES

1. Letter from William O. Parker, Jr. to Mr. Angelo Giambusso dated July 9, 1975.

2. "Order for Modification of License, " Letter from Robert A. Purple to Mr. Austin C. Thies dated December 27, 1974.

3. Letter from William O. Parker, Jr. to Mr. Benard C. Rusche dated November 10, 1975.

4. Letter from William O. Parker, Jr. to Mr. Benard C. Rusche dated October 31, 1975.

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

6. "Supplement 1 to the Status Report by 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.

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

8. R. C. Jones, et al., "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," BAW-10103, Babcock and Wilcox, June 1975.

9. Letter from William O. Parker, Jr. to Mr. Angelo Giambusso dated August 1, 1975.

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

11. Letter from William O. Parker, Jr. to Mr. Benard C. Rusche dated October 10, 1975.