MAR 28 1975

Docket No.: 50-346

Voss A. Moore, Jr., Assistant Director for LWR's, Group 2, RL

FSAR SECOND ROUND REVIEW

Plant Name: Davis-Besse 1
Licensing Stage: OL
Docket No.: 50-346
Responsible Branch and Project Manager: LWR 2-3, L. Engle
Requested Completion Date: February 3, 1975
Techniaal Review Branch Involved: Reactor Systems Branch
Description of Revimw: Final Positions
Review Status: Second Round Incomplete

A review of the Davis-Besse FSAR is being carried out to assess the acceptability of station design. The NSSS is a Babcock and Wilcox two-loop PWR, with two reactor coolant pumps per loop. It is to be noted that this Davis-Besse package consists of an unusually long list of questions, with a proportionately longer cover letter to explain the reasons. My concern for the Davis-Besse schedule was expressed as early as May 1973 (Reference 1) during our first-round review by pointing to several major areas not addressed in the FSAR. References 4 and 5 (see Enclosure 1) provided preliminary notices of the inadequacy of many of the responses to our first-round of questions last year. A major part of the enclosed inquiries are the result of unsatisfactory responses to our first-round questions.

References 1, 2, and 3 (see Enclosure 1) indicated that our first-round was INCOMPLETE due primarily to the absence of LOCA analyses across the break spectrum. Since an acceptable ECCS performance analysis still does not exist for Davis-Besse, our review remains incomplete through the second round. The areas which will require further discussion are specified in Enclosure 2 and should be addressed by Toledo Edison when they submit their ECCS analysis. We are providing Enclosure 2 at this time in an attempt to expedite the review by obtaining a responsive submittal. Of special interest to us is a clear rationale of how the design differences between Davis-Besse and the Oconee Class reactors influence the course of the LOCA.

Enclosure 3 provides the positions and additional information needed in areas other than ECCS analysis. Our review covered the FSAR through Revision 12. As indicated above, many of these questions resulted from first-round responses which were insufficient to allow an adequate review. For example Chapter 4.0, Question B in Enclosure 3 notes that the requested update of FSAR Table 1-3 was not provided. As another example, Chapter 15.0, Question D notes several inconsistencies in the provided feedwater line break discussion.

Memo 4

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We noted previously (Reference 3) that the FSAR did not evaluate the effects of fuel densification on normal operation, transients, and accidents. Revision 3 to the FSAR (question 4.4.2) committed to a mid-1974 topical report. Revision 6 slipped the mid-1974 date to December 1974. Revision 10 slipped the December commitment to April 25, 1975. The slip in dates for the Davis-Besse fuel densification report could place the evaluation of this concern outside the present review period.

Several areas of concern are currently under generic review by the Reactor Systems Branch. These areas are open items and must be considered by Toledo Edison for application to Davis-Besse at the conclusion of our studies.

- Loose Parts Monitoring (FSAR Subsection 5.2.5.3)
- Reactor Coolant Pump Overspeed (FSAR Appendix 5A)
- Fuel Rod Bowing (FSAR Subsection 4.4.3.7)
- ATWS (question 15.2.10)

Steady state and transient operation for Davis-Besse have been analyzed in Chapters 4.0 and 15.0 using the W-3 correlation. We are aware of the current transition of such analyses by Babcock and Wilcox to the B&W-2 correlation in other plants. Since the Davis-Besse FSAR was submitted under the W-3 correlation analyses, the B&W-2 correlation was not considered for application to Davis-Besse. Recent concerns on the appropriateness of the B&W-2 DNB correlation were discussed at a meeting with Babcock and Wilcox on January 27, 1975. The information presented at the meeting is currently under evaluation.

We note that revisions have been made to many figures since the FSAR was originally submitted. It is obviously quite time-consuming to search a P&ID in an effort to locate an unknown change and difficult to be sure that all changes are found. It is also frustrating to find a change without understanding why the change has been made. An example of such a revision is FSAR Figure 6-17, "Davis-Besse Nuclear Power Station Decay Heat Removal System and ECCS." Much time was spent examining the latest revision to this Figure (Revision 9) before two previously closed isolation valves were discovered now to be open. Not understanding the reasons for such a change, certain problems could exist with these open valves as are specified in Chapter 6.0 of the attached question list. Along these lines,

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it is our position that an itemization of all future revisions to FSAR figures (especially Engineered Safety Features) must accompany each new diagram with a clear rationale for the changes.

Original Signed By

Victor Stello, Jr., Assistant Director
for Reactor Safety
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Enclosure:
References

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<u>ENCLOSURE 1</u>

REFERENCES

1. Letter from Victor Stello, Jr., to Richard C. DeYoung, Jr., dated May 24, 1973.

2. Letter from Victor Stello, Jr. to Richard C. DeYoung, Jr., dated July 3, 1973.

3. Letter from Victor Stello, Jr., to Richard C. DeYoung, Jr., dated December 5, 1973.

4. Letter from Thomas M. Novak to I. A. Peltier, dated February 15, 1974.

5. Letter from Thomas M. Novak to I. A. Peltier, dated July 25, 1974.

ENCLOSURE 2

ECCS PERFORMANCE

In a letter to Mr. A. Schwencer dated August 19, 1974, and in a recent revision to the Davis-Besse FSAR, Toledo Edison indicated that the completion of the LOCA analysis and ECCS design by Babcock and Wilcox is expected to be completed in December of 1974. With regard to the LOCA analysis, 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," and Supplement 1 indicate that the applicability of the generic model to Davis-Besse has yet to be verified. Justify the appropriateness of each section in Appendix K to the Davis-Besse design, and include the following additional information.

1. Describe the major design differences between Davis-Besse and the Oconee Class reactors.

2. Provide detailed comparison tables between Davis-Besse and the generic analysis of key parameters (such as number of vent valves, vent valve "K", loop resistance factors, ECCS design comparison, core inlet fluid temperature, core flow, reactor pressure, and other relevant parameters) employed in the ECCS analysis, especially those parameters of greatest effect on peak clad temperature and metal-water reaction.

3. Confirm that the volumetric average fuel temperature at the maximum power location assumed in the Davis-Besse analysis is equal to or greater than that calculated in the approved version of TAFY.

4. With regard to question 4.2.7 on vent valve design, the statement is made that the number and size of the vent valves are shown to be acceptable by ECCS analyses. This is understood; however, also of concern are any variations in the manner in which these venting areas are applied in the ECCS analyses. Also, it is noted that valve venting areas available on Davis-Besse are less than on any other Babcock and Wilcox reactor, including other plants with raised loops. Verify that the methods of calculating and applying such venting areas have not changed for Davis-Besse.

5. Discuss the probability of unacceptably high concentrations of boric acid in the core region during long term cooling due to continuous evaporation. Describe any analyses you have performed to indicate that precipitation will not occur. The description should clearly identify the criteria, method of analyses, and equipment required.

6. List and describe any other computer codes, such as SAVER, which are used in the Davis-Besse analysis, but are not referenced in the generic model description.

7. Examine, and provide new analyses for Davis-Besse where appropriate, each sensitivity study used as a basis for the ECCS model in BAW-10091. Completely justify any study which is not re-analyzed for Davis-Besse. Compare each study to those design aspects of Davis-Besse and the Oconee Class reactor of greatest influence on the outcome. Should the result of any sensitivity study on Davis-Besse be in opposition to those studies in BAW-10091, provide a thorough quantitative explanation of the phenomenological differences produced and the reasons for these differences.

8. Provide, or reference in Chapter 16.0, the linear heat generation rate limit as a function of axial elevation.

9. With regard to the response to question 6.3.12 (small breaks), BAW-10075 is currently under review by the Regulatory staff. The conclusions of this review will be applicable to Davis-Besse. Also, compare the results, using 10 CFR 50, Appendix K, of a 0.5 ft^2 break with the large break analytical model and the small break analytical model. Also, include a plot of peak clad temperature versus break size using the small break model up to, and including, the 0.5 ft^2 break.

10. With regard to question 6.3.3 (CFT line break), the response is insufficient to allow an adequate evaluation. Resubmit the core flooding tank (CFT) line break considering the requirements of 10 CFR 50, Appendix K. Explain why the peak clad temperature for the 0.3 ft^2 break (referenced in BAW-10075) was 1090° F, but for the 0.44 ft^2 break (CFT line) was only 932° F (FSAR page 6-81d). For the CFT line break, FSAR page 6-81c states that the reactor trips at a primary system pressure of 2050 psig. Isn't the low pressure trip at 1900 psig? Also, if the analytical techniques and assumptions were similar to BAW-10064 (as stated on FSAR page 6-81b), why did the higher power plant (Davis-Besse) produce the lower peak clad temperature?

11. With regard to question 6.3.4, describe the design of the CFT relief valve and, if this valve can be remotely actuated, analyze an inadvertant opening of this valve occurring during a loss-of-coolant accident (in lieu of a diesel failure).

12. Describe the consequences of a loss-of-coolant accident during a startup or shutdown (whichever is the worst-case) while the CFT tanks are routinely isolated, and the coolant system pressure is assumed to be at the maximum expected during the time of CFT isolation.

13. For a break in a reactor coolant line, it appears that a single failure of an HPI train could result in all available HPI water flowing through the break. When the break is small enough such that HPI flow is needed to cool the core (reactor coolant pressure too high for LPI flow), it is of concern whether sufficient ECCS flow exists. Examine a variety of small and intermediate break sizes to assure that a sufficient flow split occurs in the remaining HPI train to cool the core. Credit for operator action is not acceptable unless analyses show that sufficient time exists for the operator to recognize the initiating event, ascertain the proper response, perform the appropriate manual action(s), and that the required actions are clearly defined in the operating procedures. Identify the exact manual operations required by the operator during the short term and long term. Specify: (1) the information available to the operator, (2) the time delay during which his failure to act properly will have no unsafe consequences, and (3) the consequences if the action is not performed at all.

If credit for sufficient flow splits using line orifices is proposed, provide a complete description of the design basis of this feature and justify its capability to achieve this design basis. Describe all testing that has been conducted to confirm the expected flow rates. In addition, discuss the preoperational tests which are planned to observe the flow splits for Davis-Besse.

14. Provide a plot similar to PSAR Figure 14-48, "ECCS Capability to Meet Fuel Clad Temperature Design Limit." Comment on any differences between the PSAR and FSAR figures. Show also the capability of the HPI pump + 1 core flooding tank combination (CFT line break).

15. With regard to the response to question 15.3.1, the statement is made that for a hot leg break all the fluid injected by the core flooding tanks, the HPI pump, and the LPI pump must enter the core before being lost out the break. For the 14.1 ft^2 hot leg break, discuss the potential for a portion of the ECC water to flow into both cold legs, through the steam generators, and out the hot leg break.

16. Provide a plot of peak clad temperature versus break size for the complete break spectrum.

17. Provide a sensitivity study of various axial power shapes to justify the appropriateness of the shape selected for the LOCA analyses. Specify the positions of all control rods. Identify the shape utilized for the LOCA and discuss the rationale for its selection.