Dockets

JUL 16 1973

Docket No. 50-346

Richard C. DeYoung, Assistant Director for Pressurized Water Reactors, L.

REQUEST FOR ADDITIONAL INFORMATION FOR DAVIS-BESSE

Plant Name: Davis-Besse
Licensing Stage: OL
NSSS Supplier: Babcock and Wilcox
Architect Engineer: Bechtel
Containment: Dual
Docket No.: 50-346
Responsible Branch & Project Manager: PWR #4, I. Peltier
Requested Completion Date: July 6, 1973
Applicant's Response Date: October 12, 1973
Description of Response: Additional Information
Review Status: Awaiting Information

The enclosed request for additional information for the Davis-Besse Nuclear Power Station operating license review has been prepared by the Containment Systems Branch after having reviewed the applicable sections of the FSAR.

The following comments are based on our review:

1. The applicant has not performed a complete pipe break spectrum analysis which would identify the break size and location that results in the highest calculated containment pressure.

2. The applicant has not discussed the conservatisms in the analysis of the core flooding rate or presented curves of core flooding rate as a function of time.

3. The applicant has not described the core reflood model.

4. The applicant has not provided sufficient information to permit us to perform confirmatory containment response analyses.

5. The applicant proposes to repressurize the containment following a loss-of-coolant accident as a means for diluting possible hydrogen evolution from metal-water reactors or radiolizers. Since the

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Davis-Besse plant predates the guidelines of RG 1.7, the concept of a purge system could be acceptable as a means of hydrogen control based on the Supplement to RG 1.7.

Original signed by:
Robert L. Tedesco

Robert L. Tedesco, Assistant Director
for Containment Safety
Directorate of Licensing

Enclosure:
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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1973—

REQUEST FOR ADDITIONAL INFORMATION
DAVIS-BESSE NUCLEAR POWER STATION
DOCKET NO. 50-346

6.1 The FSAR indicates that cold leg, pump suction and pump discharge breaks have not been analyzed, and it is not apparent that the 3 ft^2 hot leg break results in the highest calculated containment pressure; therefore, provide the results of containment pressure response analyses for a spectrum of break areas for a cold leg (pump suction) pipe and a cold leg (pump discharge) pipe to identify the break size and location that results in the highest containment pressure. Include the following information for each case analyzed: break area, break location, peak containment pressure, time of peak pressure, and energy released to the containment up to the time of peak pressure. For the loss-of-coolant accident at each of the assumed break locations, i.e., the hot leg and cold leg, pump suction and pump discharge pipes, that results in the highest calculated containment pressure, provide a table of mass release rate (lb_m/hr) and enthalpy (Btu/lb_m) as a function of time (hr) throughout the blowdown and core reflood phases of the accidents.

6.2 Provide an analysis of the containment pressure response for a spectrum of steam generator, steam line and feedwater pipe ruptures. Specify the postulated break sizes and locations and initial plant conditions. Provide justification for the assumed initial plant conditions. Describe the analytical model used in the analysis. Discuss the conservatism in the analysis with regard to maximizing the energy release to the containment. Provide a table of mass release rate (lb_m/hr) and enthalpy (Btu/lb_m) as a function of time (hr) for the secondary system pipe rupture that results in the highest containment pressure.

6.3 The FLASH computer code is used to predict the mass and energy release to the containment during blowdown. Discuss the assumptions made to obtain conservatively high energy release rates from the core for containment evaluation studies. Discuss the criterion used to establish the time to DNB considering that a conservative approach would be to delay DNB until the core was voided by steam.

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6.4 During blowdown, energy may be transferred from the steam generators to the primary coolant by conduction through the tube walls. Discuss the heat transfer correlations used for both the primary and secondary sides of the steam generator during blowdown. Give the additional energy that could be released to the containment if DNB was delayed on the primary side of the steam generator tubes.

6.5 Provide a description of the core reflood model. Discuss the conservatism in the model with respect to maximizing the energy release to the containment. Include the following in your discussion of the core reflood model:

(1) Discuss the assumptions made regarding the water remaining in the reactor vessel at the end of blowdown. We believe a conservative approach for containment analyses would be to assume that the water remaining in the reactor vessel is saturated and at the bottom of the core.

(2) Discuss the assumptions made regarding the core flooding rate. We believe a conservative approach for containment analyses would be to assume full ECCS operation.

(3) Discuss the assumptions made regarding the core quench height and carryout fraction. We believe a conservative approach for containment analyses would be to assume a carryout fraction of 0.8 and that the core would be quenched at the 10-foot level.

(4) Provide a tabulation of the system resistances used in the reflood analysis. If these resistances were determined for normal system operating conditions, describe the method used to extrapolate them to reflood conditions.

6.6 After the core has been recovered with water following a pump suction break, boiling will occur to cool the core, and a two-phase mixture of steam and water will be generated. Provide an analysis showing the height that the two-phase mixture will rise above the core. If any water is calculated to enter the steam generators, provide the energy release rate to the containment as a function of time.

6.7 With respect to the heat sinks listed in Table 6-1 of the FSAR, identify the heat sinks that are exposed to the containment atmosphere on both sides, and specify whether the exposed

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surface areas represent the surface area of one side or both sides. Also, provide the exposed surface area of the miscellaneous sheathed concrete (item 7 in Table 6-1).

6.8 Discuss the method(s) and the accuracy of the method(s) used to determine the free containment volume. Provide a sensitivity study of the effect of the uncertainty in calculating the full volume on the containment vessel pressure response under loss-of-coolant accident conditions. Discuss how the containment full volume will be verified.

6.9 For the subcompartment analyses, provide assurance that there are no flow restrictions within a subcompartment that could cause pressure differences. Discuss the difference between the orifice flow area and the miscellaneous flow area that are given for each subcompartment, and how the areas are treated by the computer code COPRA.

6.10 The arrangement drawings of the plant indicate that the containment emergency sump is not at the lowest elevation in the plant, and that a significant amount of water could be retained below the elevation at which water would begin to overflow into the emergency sump. The reactor vessel cavity, normal sump, refueling canal, incore instrumentation tunnel, pipe tunnel, and value pit are some of the areas that lie below the emergency sump. Also Figure 6-17 indicates that the refueling canal drains to the reactor vessel cavity which drains to the reactor vessel cavity which drains to the normal sump, and the emergency sump also drains to the normal sump. Specify the water level in the containment following a LOCA assuming the containment volume below the elevation of the emergency sump is uniformly filled with water. Discuss the adequacy of available NPSH to the containment spray pumps in the context of Safety Guide 1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps".

6.11 The intake screen installed over the containment emergency sump does not appear to be structurally adequate. For example, only a single, completely exposed wire mesh screen is provided, and if the screen was damaged debris could enter both recirculation lines. Provide the following information:

(1) a more detailed drawing of the intake screen which shows how the screen is attached to the containment vessel wall and floor,

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(2) assurance that the failure of a portion of the intake screen will not negate the effectiveness of the entire screen, and

(3) assurance that the screen cannot be readily damaged by a missile or large debris that could be carried in the water following a LOCA.

6.12 Specify the manufacturer of the containment air cooler units. Describe the qualification test program that was conducted to determine the performance capability of an air cooler unit. Provide a curve of air cooler performance showing energy removal rate as a function of containment atmosphere temperature. Since lake water will be circulated through the air coolers and since the air coolers will be used under both normal and accident conditions, discuss how fouling of the secondary side of the cooling coils was factored into the analysis of the heat removal capability of an air cooler. Specify the service water (lake water) temperature used in the analysis, and provide a table of the maximum and minimum, and monthly average temperature of the lake water at the service water system intake.

6.13 Identify the ductwork of the containment air cooling system that must remain intact following a loss-of-coolant accident to assure that the functional capability of the system is not impaired. Discuss the design provisions to assure that the air cooler unit housings and system ductwork can withstand the differential pressures resulting from a loss-of-coolant accident.

6.14 Describe how the fusible dropout register(s) associated with the containment air cooling system (as shown on Figure 9-12A) will function.

6.15 Provide the following information in Table 6-8, Containment Vessel Isolation Valve Arrangements:

(1) the type of valve and valve operator,

(2) the valve location with respect to the containment vessel,

(3) the method(s) of valve actuation,

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(4) the valve operator power source,

(5) the valve position on motive process failure,

(6) the line size, and

(7) the FSAR figure on which the isolation valve arrangement is shown.

The penetration numbers listed in Figure 6-12 as spares do not correspond to those listed in Table 6-8 as spares; provide clarification.

6.16 The containment vessel penetrations that are exceptions to General Design Criterion 56 are listed on page 6-46 of the FSAR. With respect to items 6 and 7; i.e., the isolation valve arrangements for the containment vessel hydrogen dilution and purge system, and the containment vessel air sample inlet and outlet lines, the rationale for exempting them from the requirements of GDC 56 was not presented. Therefore, discuss why these penetrations are being considered to be exempted from the requirements of GDC 56. Include the containment vessel spray lines in the discussion.

6.17 Table 6-8 in the FSAR indicates that the core flooding tank sample and vent lines are each provided with a single isolation valve outside containment. The core flooding tanks are not considered closed systems inside containment and, therefore, General Design Criterion 57 does not apply to these lines. Discuss any other basis that you may have which would demonstrate that the valve arrangement meets the intent of the GDC.

6.18 Table 7-5, SFAS Actuation Summary, indicates that the containment valves are grouped into three systems. Provide a tabulation of the isolation valves in each system and specify the trip setpoints.

6.19 Describe the qualification test program that was conducted to assure the operability of containment isolation valves, valve drives, position indicators, sensing elements, cables, etc. following a LOCA or steam line break accident. Identify the equipment that was tested. Graphically show the environmental test conditions as a function of time.

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6.20 Identify all lines penetrating the containment that do not terminate within areas served by the emergency ventilation system. Provide an estimate of the total amount of containment leakage which can bypass the areas served by the emergency ventilation system.

6.21 Provide the following information with respect to the plant combustible gas control systems, i.e., the hydrogen dilution system, the hydrogen purge system, and the containment air recirculation system:

(1) Provide an analysis of the differential pressures that may occur following a LOCA for the fan housings and ductwork of the containment air recirculation system.

(2) On page 3-3 of the FSAR, the hydrogen purge - dilution system is identified as being seismic Category I. However, the purge line is not seismic Category I (as indicated on Figure 9-12A), and is subject to a single active failure. Since the proposed method of hydrogen control for the plant involves repressurizing the containment, the purge line should be designed to engineered safety feature standards to assure that continuous hydrogen control capability will exist. Therefore, provide a hydrogen purge system that meets the design criteria for an engineered safety feature.

(3) Specify the maximum allowable pressure that the containment will be repressurized to using the hydrogen dilution system before hydrogen purge system operation becomes necessary.

(4) Specify the power source for each isolation valve in the hydrogen dilution system (HV 5064, HV 5065, HV 5090, and HV 5091) to assure that the hydrogen dilution system is not subject to a single active failure.

6.22 Provide a P and I drawing of the containment gas monitoring system. Discuss the accuracy of the hydrogen analyzer.

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the "failure of an interlock" was assumed to occur as the operator or the control system was imposing a demand signal which would normally be limited by the subject interlock. Satisfactory clarification has been obtained from the applicant. This concern has been satisfied. Reference IIC

4. Onsite Power

The response is acceptable on the basis that the applicant understands that the continuous rating is the 8,000-hour rating. This statement was not included in the response. Reference IIIB

5. Environmental Testing (Valves in Containment)

Response is acceptable. Reference IVC

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