Docket File

OCT 1 1976

Docket No. 50-346

MEMORANDUM FOR: D. Vassallo, Assistant Director for Light Water Reactors, DPM

FROM: R. Tedesco, Assistant Director for Plant Systems, DSS

SUBJECT: REVISION TO THE DRAFT SAFETY EVALUATION REPORT - DAVIS-BESSE NUCLEAR POWER STATION, UNIT 1

Plant Name: Davis-Besse Nuclear Power Station, Unit 1
Docket Number: 50-346
Licensing Stage: OL
NSSS Supplier: Babcock & Wilcox
Containment Type: Dry Dual
Architect Engineer: Bechtel
Responsible Branch and Project Manager: LWR Branch 4; L. Engle
Review Status: Incomplete
Requested Completion Date: N/S
Applicant's Response Date: N/S

Enclosed are revisions to the draft Safety Evaluation Report for the Davis-Besse Nuclear Power Station, Unit 1. This report has been prepared by the Containment Systems Branch after having reviewed the applicable portions of the FSAR as amended (through Amendment 36). In addition, we have prepared comments on the Technical Specifications concerning containment leak testing which are included in an enclosed request for additional information.

The following items describe the status of the draft Safety Evaluation Report (issued February 12, 1975 and revised April 30, 1976) and the Technical Specifications:

1. Subcompartment Analysis

   The subcompartment analysis is still identified as an open item in the draft Safety Evaluation Report, as revised. For the reactor cavity and steam generator compartments, we are calculating differential pressures that exceed design values. It should be noted that our confirmatory analysis of the reactor cavity is based on the use of modified inertia (L/A) terms from a similar B&W plant.

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Upon examination of plan and elevation drawings of the reactor cavity, we determined that certain vent areas were calculated incorrectly by the applicant. We have discussed this matter with the applicant and the applicant has agreed to submit a revised reactor cavity analysis using the correct vent area data, and has agreed to provide the inertia term data which we have previously requested from them. We feel certain, however, that peak calculated differential pressures for the reactor cavity will exceed design conditions unless the basis for assuming a 14.14-ft^2 longitudinal split in the hot leg is re-examined. We have discussed this with the applicant, and the applicant plans to submit additional information regarding postulated pipe break configurations and sizes for the reactor cavity analysis.

The applicant has indicated his intentions to adopt a similar approach for the steam generator compartment analysis, if necessary.

2. Shield Building Depressurization Time

Amendment 36 presented a revised thermal analysis of the shield building following a LOCA which increased the shield building depressurization time from 65 seconds to 12.33 minutes. Before we can conclude on the acceptability of the revised shield building analysis, additional information will be required. The attached questions have been discussed with the applicant. We have also notified AAB of the increased depressurization time.

3. Containment Purge System

We have reviewed the applicant's plans for operation of the containment purge system during normal plant operation. In the attached request for additional information, we are requesting the applicant to provide the analyses identified in Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations," to justify operating the purge system during normal plant operation.

4. Containment Leak Testing (Technical Specifications)

(a) System Venting and Draining

We will require that the systems to be vented and drained during the containment integrated leak rate (Type A) test be identified in the plant Technical Specifications. This is included in the attached request for additional information.

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(b) Fuel Transfer Tube

Revision 1 to the draft Safety Evaluation Report states that the applicant intends to demonstrate zero leakage through the fuel transfer tube and thereby eliminate it as a potential bypass leak path. Since the test method employed during the preoperational leak testing has been shown to be quite sensitive, the applicant can no longer commit to demonstrating zero leakage but proposes to demonstrate negligible leakage. Rather than attempt to define what constitutes negligible leakage, we will require that the fuel transfer tube leak rate be included in the total for all potential bypass leak paths.

(c) Personnel Air Lock

The draft Technical Specifications allow a maximum rate for an airlock of 0.05 La at Pa, (38 psig). Since the air lock is identified as a potential bypass leak path, its individual leakage limit is in conflict with the maximum allowable bypass leak rate of 0.015 La. We have asked the applicant to propose a leak rate limit for the airlock that will not conflict with the maximum allowable bypass leak rate.

Also the draft Technical Specifications state that periodic door seal leak testing must demonstrate no detectable seal leakage at peak calculated accident pressure Pa. The applicant has informed us in a telecon that the door seals cannot be leak tested at Pa without the use of strongbacks. Furthermore, the applicant stated that they have detected some leakage when pressurizing between the seals at a reduced pressure. The applicant, therefore, proposes to specify a leak rate for a reduced pressure which, when extrapolated to Pa, would not exceed a maximum allowable leak rate. We have asked the applicant to propose a test method for the air lock door seals including an acceptance criterion.

Original signed by
Robert L. Tedesco

Robert L. Tedesco, Assistant Director
for Plant Systems
Division of Systems Safety

Enclosures:
As Stated

cc: See Page 4

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