Docket

Docket Nos. 50-438
and 50-439

JAN 16 1974

Tennessee Valley Authority
ATTN: Mr. James E. Watson
Manager of Power
818 Power Building
Chattanooga, Tennessee 37401

Gentlemen:

In order that we may continue our review of your application for a license to construct the Bellefonte Nuclear Plant, Units 1 and 2, additional information is required. The information requested is described in the enclosure and pertains to Chapter 6 of the Preliminary Safety Analysis Report.

In order to maintain our licensing review schedule, we will need a completely adequate response to all enclosed requests by February 25, 1974. Please inform us within 7 days after receipt of this letter of your confirmation of the schedule date or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our request, it is highly likely that the overall schedule for completing the licensing review for the project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the extension will most likely be greater than the delay in your response.

Please contact us if you have any questions regarding the information requested.

Sincerely,

/S/

A. Schwencer, Chief
Light Water Reactors Br. 2-3
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: See Next Page

LB

OFFICE ➤						
SURNAME ➤						
DATE ➤						

JAN 1 6 1974

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REQUEST FOR ADDITIONAL INFORMATION
TENNESSEE VALLEY AUTHORITY
BELLEFONTE NUCLEAR POWER STATION, UNITS 1 & 2
DOCKET NOS. 50-438 AND 50-439

6.0 ENGINEERED SAFETY FEATURES

6.81 The response to Request 6.5 shows that a carryout fraction of 0.71 was used during core reflood for the CRAFT calculation of the 11.17 ft^2 double-ended cold leg break. Based upon results obtained from FLECHT experiments, we have used a carryout fraction of 0.8 for containment evaluation purposes. Reanalyze the containment response for the limiting double-ended cold leg break using either a 0.8 carryout fraction or an appropriate correlation based on FLECHT data.

6.82 Your response to Request 6.37 indicates that a heat transfer coefficient of 160 Btu/hr-ft^2 - ^oF was utilized for reverse heat transfer in the steam generators based primarily on a forced convection coefficient of 90 Btu/hr-ft^2 - ^oF on the tube side of the steam generator. We believe that a nucleate boiling correlation would be more appropriate than a forced convection coefficient for tube side heat transfer. Since this could increase the overall heat transfer coefficient considerably, determine the effect of tube side nucleate boiling on reverse heat transfer from the steam generators and the resulting increase in containment pressure.

6.83 Your response to Request 6.44 indicates that an interface gap for steel-lined concrete heat sinks would not have a significant effect on the maximum calculated building pressure for the hot leg break design basis accident occurring at 20 seconds. Provide the containment response to the limiting cold leg break including the effect of a gap resistance. Provide those references and experimental data that form the basis for the steel/concrete gap resistance utilized. In addition, provide the results of a sensitivity study showing the containment response assuming that the concrete is not in contact with steel for all steel-lined concrete heat sinks.

6.84 The major contribution to the secondary building pressure change is the primary containment building expansion caused by internal pressure of the DBA-LOCA. Describe in detail the method of analysis and the assumptions used to calculate the expansion of the primary containment building.

6.85 The response to Request 6.62 is inadequate. Provide a detailed description of the method and assumptions utilized in the analysis of the jet forces that can impinge on the reactor cavity and the steam generator cavity structures.

6.86 The response to Request 6.63 is inadequate. You have not provided specifically the conservative assumptions (with respect to low containment pressure) relating to the energy release to the containment, containment heat removal, containment volumes, initial containment conditions, modeling of the heat sinks, heat transfer coefficients to the heat sinks, heat sink surface area and any other parameter assumed in the analysis which established the minimum containment backpressure of 45 psia used in the LOCA analysis. Provide and justify the limiting assumptions for each of the above factors in determining the minimum containment backpressure.

6.87 Provide the bases for the selection of the flow coefficients used in the subcompartment analyses.

6.88 The response to Request 6.66 states that the air cooling system exhaust ducting below the relief panel and the backflow damper is nonessential. Provide justification that acceptable air distribution can be achieved in the event that the exhaust ducting fails.

6.89 Discuss the leak detection capability for lines provided with remote manual containment isolation valves to assure that adequate information is available to the operator to effect isolation of a system if necessary.

6.90 Expand your response to Request 6.10 to include the specific distance criterion utilized in locating the isolation valves with respect to the containment.

6.91 Provide the seismic category and the quality group to which the fuel transfer tube will be designed and constructed. Since this penetration forms an extension to containment it should be designed to the same standards as the containment.

6.92 Provide the rate at which the aluminum components described in Request 6.75 are assumed to corrode for the combustible gas control analyses.

6.93 Provide assurance that the reactor building cooling fans can be used to provide good mixing of hydrogen in the long term following a LOCA as well as in the early stages following an accident.

6.94 Describe the functions of the hydrogen recirculation system (See Figure 9.4-3) during and following a LOCA.

6.95 Provide the following information with respect to the hydrogen monitoring equipment:

a. Describe the process sampling system. Specify the number and identify the locations of the sampling lines within the reactor building to determine local hydrogen concentrations and the degree of atmospheric mixing.

b. Describe the monitoring equipment and readout locations, the hydrogen concentration measurement principle of the equipment, the expected accuracy of the equipment, plans for equipment qualification tests and the constraints on system operation (e.g., radiation level and moisture content of flow stream, and reactor building pressure).