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OCT 15 1973

Docket Nos. 50-438
and 50-439

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 questions by November 26, 1973. 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,

Original Signed

A. Schwencer, Chief
Pressurized Water Reactors Branch 4
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: Mr. R. H. Marquis
General Counsel
629 New Sprakle Building
Knoxville, Tennessee 37902

LB

OFFICE ▶	x7548/L:PWR-4	L:PWR-4				
SURNAME ▶	Don Davis:cjr	ASchwencer				
DATE ▶	10/9/73	10/11/73				

Form AEC-318 (Rev. 9-53) AECM 0240 GPO 643-16-81465-1 445-678

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

OCT 1 5 1973

6.0 ENGINEERED SAFETY FEATURES

6.37 For the most severe hot and cold leg breaks, provide values of the heat transfer coefficients used in the steam generators for forward and reverse heat transfer for all phases of the accident. Discuss the bases for these values including a discussion relating to the conservatism for containment analysis.

6.38 Provide the heat transfer coefficients and mesh spacing used to calculate heat flow from the reactor coolant piping and core internal structures to the reactor coolant for all phases of the accident. Discuss how these values are conservative for containment analysis. Provide the mass and thickness of the metal slabs used to describe the reactor vessel, core internals and steam generator metal.

6.39 Describe the methods employed to calculate the initial reactor coolant system volume which is used to calculate the liquid mass contained in the reactor coolant system. Provide the temperature used to calculate the piping and vessel volumes and the pressurizer level assumed in the calculations. Discuss the conservatism of these values for containment analysis.

6.40 Describe the methods employed to calculate the initial core stores energy used in the containment pressure calculations. Provide the values of the initial and decay power level, burnup, gap conductance, and fuel conductivity used in your calculations. Discuss how these values are conservative for containment analysis.

6.41 For the purpose of performing an assessment of the mass and energy release during the core reflood phase of a loss-of-coolant accident, provide the information related to the hydraulic modeling of the reactor coolant system including the resistances of components (reactor coolant pump, steam generator and piping) the core and downcomer and including the core bypass areas and lengths. If the downcomer area changes as a function of length, these values should be provided. Discuss the assumptions made regarding the water remaining in the reactor vessel at the end of blowdown. A conservativeaapproach would be to assume that the water remaining in the reactor vessel is saturated and at the bottom of the core.

6.42 Provide analyses of the containment pressure response for a spectrum of cold leg pump discharge break sizes using methods and assumptions similar to the hot leg and pump suction break analyses.

6.43 For the calculation of heat flow to the containment heat sink structures, provide justification for the node spacing used in the finite difference equations.

6.44 Provide justification for the lack of an interface resistance (contact or gap resistance) between steel and concrete for the steel-lined concrete heat sinks used in the containment pressure calculation.

6.45 During the post-reflood period following a break at the pump suction when the core has been recovered with water, the core should be cooled by boiling and a two-phase mixture of steam and water would exist in the core. Provide an analysis showing the height that the two-phase mixture will rise above the core. If any mixture is found to enter the steam generators, provide the energy release rate into the containment.

6.46 Discuss the methods and the accuracy of the methods used to calculate the containment free volume. Provide a sensitivity study of the effects of the uncertainty in calculating the containment free volume on the containment pressure response under DBA conditions. Describe any tests to verify the containment free volume.

6.47 Provide an analysis showing the containment response for the cold leg pump suction double-ended rupture (11.17 ft^2 break area) assuming full ECCS operation. The analysis should neglect the quenching action of the incoming ECCS fluid on the exiting steam. Partial containment heat removal systems (i.e., coolers and sprays) resulting from a single active failure should be assumed.

6.48 Provide a time history comparing the condensing heat transfer coefficients used in the containment pressure response for the cold leg pump suction double-ended rupture (11.17 ft^2 break area) of (1) Tagami-Uchida condensing heat transfer coefficients and (2) Kolflat-Chittenden heat transfer coefficients.

6.49 Provide the analysis of a reactor coolant pipe break inside the biological shielding (i.e., pipe annulus). Include a discussion of the model and assumptions used and a flow diagram showing flow connections, the free volume and vent areas considered. Compare the peak calculated pressure differential in the pipe annulus to the design pressure differential.

6.50 Provide a subcompartment differential pressure analysis of the reactor cavity and the steam generator cavity utilizing the CRAFT code. Compare this transient pressure response to that obtained with the multi-node code, BT-129.

6.51 Discuss the results of analyses used to justify that the break locations selected for subcompartment analyses produce the highest calculated subcompartment differential pressures. Include a description of the correlation used for determining subcooled blowdown rates.

6.52 Provide a description of the correlations used in the calculation of sonic flow through the compartment. Provide a description of any conservatism applied to the sonic flow correlation. A multiplier of 0.6 should be used with the Moody correlation.

6.53 Relate the compartments (rooms) described in Table 6.2-7 and Figure 6.2-7 to a simplified physical arrangement drawing. On this drawing, show the walls, components, structures, etc. which separate the compartments.

6.54 Justify the selection of the subcompartment subdivisions. Discuss considerations given to the effects of flow blockage by components.

6.55 It is not clear how the calculated room pressure differentials provided in Table 6.2-8 along with a multiplying margin factor of 1.1 will be used to arrive at design pressure differentials for the subcompartment and their associated structures. Provide the design pressure differential to be applied to each structure or component separating the rooms. Currently for our analytical assumptions, a 40% margin is being applied between the calculated and design subcompartment pressure differentials.

6.56 Define the term "equivalent area" in Table 6.2-8 - Flow Path Description.

6.57 Describe any components (grating, air ducts, shield plugs, etc.) that could obstruct the vent area between subcompartments. If all of the vent areas given in Table 6.2-8 are not immediately available for venting the compartment, provide an analysis justifying this increase in vent area.

6.58 Provide a simplified two volume subcompartment benchmark calculation comparing the three codes used to analyze the subcompartments (i.e., BT-129, CONTEMPT-PS, and the CRAFT code). To allow us to perform a confirmatory analysis, provide a complete description of the benchmark problem, including all input (i.e., initial conditions, blowdown data, loss factors, compartment volumes, vent area, critical flow correlation, etc.). Provide the transient response of the compartment absolute and differential pressures using each of the codes.

6.59 Provide and justify the basis for the external design pressure of both the reactor building and the dual containment building.

6.60 Describe in more detail the method of analysis and assumptions used to calculate the secondary containment pressure response following a loss-of-coolant accident. Clarify the assumptions used for heat transfer through the primary containment building and expansion of the primary building. Provide the component pressure increase resulting from (1) building expansion, (2) inleakage, and (3) heat transfer from primary containment.

6.61 Describe the pressure or vacuum relief system referred to on page 6.2-6 in a manner similar to other system descriptions.

6.62 Provide the method and results of analysis of the jet forces which can impinge on the reactor cavity and steam generator cavity structures. Discuss the structural design capability of each compartment to withstand the differential pressure and jet forces resulting from postulated loss-of-coolant accidents.

6.63 State the minimum containment backpressure that has been used in the analysis of the emergency core cooling systems. Describe the conservatism (with respect to low containment pressure) relating to the assumptions of 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. Provide the containment pressure, temperature and sump temperature response for the most conservative assumptions (i.e., low containment backpressure) with respect to these parameters.

6.64 Provide the instrument pressure actuation setpoint for initiation of the reactor building spray trains.

6.65 Identify those areas within the reactor building whose relative elevation is below the emergency sump (e.g., refueling cavity and reactor cavity) and which could prevent recirculation of the spray water. Discuss the provisions which permit the spray water that may enter these areas following a loss-of-coolant accident to be drained to the containment sump.

6.66 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. Describe provisions made in the reactor building emergency cooling system fans and ductwork to assure that the system can withstand, without loss of function, the differential pressures imposed under the most limiting loss-of-coolant accident pressure conditions.

6.67 Provide the thermal efficiency of the reactor building spray as a function of the steam-to-air mass ratio.

6.68 Describe the procedure for transferring the spray system pump suction from the BWST to the reactor building sump. Describe the information that would be available to the operator to guide him in making a timely decision in this action.

6.69 Describe the qualification test program to determine the performance capability of the air cooler units.

6.70 On page 6.2-31, it is stated that "Lines which must remain in service subsequent to accidents that activate containment isolation will be provided with at least one remote-manual isolation valve." Identify any valves in the above classification which are exceptions to the General Design Criteria and justify these exceptions.

6.71 Provide a listing of all penetrations of the secondary containment building which penetrate the secondary containment but not the primary containment building. Describe means for isolating the secondary containment building.

6.72 Provide the design pressure for the sleeves around the main steam lines, the feedwater lines and any other similar lines that penetrate the secondary containment annulus.

6.73 Provide the mass of zirconium in the fuel cladding and describe the methods utilized in the determination of the hydrogen release from zirconium-water reactions with regard to the parameters given in Regulatory Guide 1.7.

6.74 Identify any additives that will be utilized in the reactor building spray system.

6.75 Provide a listing of the aluminum components within the reactor building, with the mass and surface area of these components. Identify the component surfaces that have been painted. Discuss the assumptions made regarding the corrosion of the aluminum components.

6.76 Provide the basis (time or hydrogen concentration) for initiation of operation of the hydrogen recombiners.