Dockets

JAN 11 1974

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 Mr. J. E. Gilleland's December 14, 1973 letter to Mr. J. F. O'Leary, the Tennessee Valley Authority identified several nuclear steam supply system (NSSS) and fuel design modifications that will shortly be incorporated in your application for a license to construct the Bellefonte Nuclear Plant, Units 1 and 2. In addition, it was indicated that these design modifications would make the Bellefonte NSSS and fuel design similar to that of the Detroit Edison Company's Greenwood facility. In order to minimize any delays in our review of your application which may result from the late incorporation of these modifications, we are providing Regulatory staff positions and requests for additional information that were developed in the staff's review of the Greenwood application. These positions and requests for additional information are listed in Enclosures 1 and 2, respectively. We request that you state your intent regarding compliance with each of the positions and where applicable provide the requested additional information. In addition, we request that you provide a tabulation of the differences between the Bellefonte NSSS and fuel design and that of the Greenwood facility. We are prepared to meet with you to facilitate a complete understanding of these safety matters and the bases for our positions.

Since these design modifications introduce significant changes to your application late in our review process, it may be necessary to adjust the review schedule that was indicated in our July 27, 1973 letter. In order that we may assess the degree, if any, of this adjustment, we request that you inform us within 7 days after receipt of this letter of your schedule for responding to all of these staff positions and requests for information, as well as your schedule for amending your application to completely incorporate these design modifications.

Appl. CB

OFFICE ▶						
SURNAME ▶						
DATE ▶						

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

Sincerely,

Original Signed by
Albert Schwencer

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

Enclosures:
1. Positions
2. Request for Additional Information

ccs: Mr. R. H. Marquis
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ENCLOSURE 1

POSITIONS REGARDING CONSTRUCTION PERMIT
TENNESSEE VALLEY AUTHORITY
BELLEFONTE NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 50-438 AND 50-439

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 It is the staff's position that the applicant shall commit to the completion of all research and development programs now underway or planned for the Mark C (17 x 17) fuel assembly prior to application for an operating license. Furthermore, the required output of these programs shall be described in detail and listed in Section 1.5 of the PSAR. The programs include the following:

A. Assembly Flow Tests
   1. Rod and Assembly Vibration
   2. Resistance to Fretting and Wear
   3. Pressure Drop Characteristics
   4. Lift Forces

B. Assembly Mechanical Tests
   1. Vibration and Damping Characteristics
   2. Load-Deflection-Stress Response
   3. Impact Behavior

C. Control Rod Tests
   1. Scram Times
   2. Control Rod Wear
   3. Control Rod Drive System

D. Component Mechanical Tests
   1. Spacer Grid
      a. Grid Spring Characteristics
      b. Seismic Capability
      c. Structural Adequacy
   2. End Fittings
      a. Load, Deflection and Stress Characteristics
      b. Guide Tube Attachment Verification
      c. Holddown Spring Behavior

E. Critical Heat Flux Tests
   1. 6 x 6 Array
   2. Non-uniform Flux Distribution

F. Manufacturing Feasibility Tests
   1. Identification of Fabrication Problems
   2. Component Inspection
   3. Assembly Inspection

G. One-Sixth Scale Model Flow Tests for 205 Fuel Assembly Reactor and Internals

1.2 It is the staff's position that the applicant shall commit either to demonstration that tests and analyses previously used for accident analysis of the Mark B (15 x 15) fuel assembly are applicable to the Mark C fuel assembley or to perform additional tests that may be necessary for the Mark C fuel assembly.

ENCLOSURE 2

REQUESTS FOR ADDITIONAL INFORMATION
DETROIT EDISON COMPANY
GREENWOOD ENERGY CENTER, UNITS 2 & 3
DOCKET NOS. 50-452 AND 50-453

4.0 REACTOR

4.3 Provide the nominal helium fill gas pressure in the fuel rod, and include the upper and lower tolerance limits.

4.4 Explain the basis for the 10% maximum allowable plastic diametral strain. What is the maximum total strain?

4.5 Provide a drawing of the fuel rod, simular to Figure 4.2-2, with dimensions. A drawing of the fuel pellet with dimensions should also be provided.

4.6 Provide information as a function of burnup for 100% power and nominal dimensions for (a) gap conductance, (b) hot gap size, (c) clad I. D., (d) gas pressure and thermal conductivity, (e) volumetric average temperature, and (f) centerline temperature. State whether the effects of densification are included in these calculations.

4.7 Provide fuel centerline temperature, volumetric average fuel temperature, and gap conductance as a function of Kw/ft for beginning of life, end of cycle 1, and 55,000 MWd/mtU burnup.

4.8 Describe the power history used to calculate fission gas released.

4.9 Give limits on specifications of absorbed gases and moisture in fuel rods.

4.10 Provide values used in analyses for Zr-4 irradiation growth and supply supporting data or references.

4.11 Justify use of 425^{o}F as the limit at which "all significant hydrides have had a chance to precipitate under a favorable stress field." Define what is meant by "all significant hydrides."

4.12 Provide the burnable poison concentrations of the initial loading.

4.13 Provide details of the axial power distribution for a period of one week with daily load swings of 100% to 50% and return to 100% power.

4.14 The values given in Tables 4.3-3 and 4.3-4 for power peaks and Doppler coefficients are exactly the same values as given for the 15 x 15 core. Verify these values.

4.15 Explain why the uniform void coefficient (Table 4.3.8) changed by a factor of 10 for one case and other coefficients, particularly the moderator coefficient, did not change from the 15 x 15 to the 17 x 17 design.

4.16 In Table 4.3-11 the k_{eff} values are not consistent. Provide correct values for k_{eff} at BOL boron levels given for 70°F for all CRA in and one stuck CRA.

4.17 Table 4.3-12 lists the exact values of moderator coefficient as were given for the 15 x 15 design. However, the values given for the stability index are quite different. Describe, in detail, the methods used to calculate the stability index.

4.18 Figure 4.3-3 is given in terms of percent of total Pu. Explain what is meant by total Pu.

4.19 In Figure 4.3-4 the text gives a value of B_{eff} = .00691 at BOL. If Figure 4.3-4 is to show B_{eff} versus core burnup, it should show B_{eff} as a function of burnup early in life.

4.20 In Figure 4.3-24 the value on this curve for 100% power does not agree with the value given in Table 4.3-5. Explain the difference.

4.21 The values in 4.3-27 do not agree with values given in Table 4.3-11. Explain the difference.

4.22 Does Figure 4.3-29 give rod worth versus rod index for Greenwood? Why is there no difference between it and corresponding curve for the 15 x 15 design which had a different rod bank structure?

4.23 Correct the reference to Table 4.4-1 which occurs in the last sentence on page 4.4-1.

4.24 The radial and axial peaking factors used in calculating fuel cladding temperatures (at page 4.4-2) are not consistent with one another nor with Table 4.4-1, nor with "the most probable design condition" defined on page 4.4-25. Explain or correct these inconsistencies.

4.25 Define, describe, and justify the coolant quality and void limits spoken of in the second paragraph of Section 4.4.2.5.

4.26 Provide a basis for the assumption at page 4.4-4 that the "flow in the hot bundle position is 1 percent less than average bundle flow under isothermal conditions." Describe or reference the method of calculation of hot assembly flow diversion which yields the 85.9 percent of nominal value given on page 4.4-4.

4.27 Four operational transients are summarized at page 4.4-19. With regard to this summary and Figures 4.4-12 and 4.4-13:

a. The minimum hot channel DNB at 90 percent power is shown to have three valves; viz., 3.07, 2.54, and 2.0. Explain and correct these inconsistencies.

b. The minimum hot channel DNB ratio at 100 percent power is shown to have two values; viz., 2.54 and 2.0 (as compared to values of 1.82 in Table 4.4-1 and a minimum of 2.11 in Table 4.4-2). Explain and correct these inconsistencies.

c. Oscillations in DNBR and characteristic of flow instability are shown in Figure 4.4-12. Describe or reference the method of calculation which shows damping of these oscillations.

4.28 Correct sentences f.1 and f.3 on page 4.4-26.

4.29 Correct the first equation on page 4.4-18.

4.30 Table 4.4-2 provides a comparison of the thermal margins as computed by three different DNB correlations. The implication of this table is that the B&W-2 correlation allows more than 30 percent higher power than the W-3 correlation for the same minimum DNBR. Conversely, at a given power level, the B&W-2 correlation indicates greater than 30 percent more margin to DNB than does the W-3 correlation. This increase in margin or power capability was not indicated or substantiated in the Staff's review of the B&W-2 correlation (SER for North Anna 3/4, Supplement 1, February 21, 1973). Explain and justify these differences with specific reference to the fact that both W-3 and B&W-2 have been shown to correlate DNB data from uniformly heated vertical tubes.

4.31 Table 4.4-3 indicates that the Greenwood reactors will operate with hot assembly exit quality of 3.3 percent at 100 percent power and 11.1 percent at 112 percent power. What methods are used by B&W to couple local coolant void and fuel power and thus to calculate changes in axial power profile with increasing void? What experimental information (test loop or reactor) is planned to confirm that analysis method? The hot unit cell approaches the limit of applicability of the B&W-2 correlation for the design overpower case (i.e., 14.6 percent exit quality versus 15 percent quality). Lacking definition of the B&W method for calculating axial power shift due to voids and knowing that the design approaches the limit of the DNB correlation, our concerns are:

(a) the accuracy of the DNB calculation
(b) the proximity to the limit of applicability of the DNB correlation
(c) the proximity of the hot unit cell exit qualify to limits which define two phase flow regime changes and hence flow instability

Please expand the description of the thermal/hydraulic design methods to address these concerns.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.22 Describe or reference the methods used to derive Table 5.3-1, loop flow distribution as a function of operating pumps.

5.23 On pages 5.3-2 and 5.3-4 in the discussion of the Integrated Control System (ICS), a table of limiting power levels is presented for various RC pump operating conditions. On what bases were these power levels chosen? Since the LOCA and other accident analyses assume full power, 4 pump operation as initial conditions, what analyses have been performed to show that other permissible initial conditions (such as given in the table on page 5.3-3) do not yield more severe results in the event of an accident or transient?

5.24 The reactor coolant pump drawing (Figure 5.5-1) is insufficient for understanding the internal functioning of the pump, the location and arrangement of the seal mechanisms, etc. Provide a drawing sufficient to illustrate all of the pump internal mechanisms. Name the pump manufacturer; compare the pump to those used on other B&W reactors; list the forward and reverse K-factors for this pump and for other pumps typical of B&W reactors.

5.25 The potential for pump overspeed during a LOCA is discussed at pages 5.2-21, 5.2-22, and 5.5-3. The application alludes to a variety of testing programs and in essence says that if data becomes available it will be used in the FSAR. The application should be amended to contain a clear commitment to resolve this issue before the FSAR through specific test programs or through specific, but alternative, design changes.

5.26 At various places in Section 5.5 (e.g., pages 5.5-3 and 5.5-9) the application states that consideration is given in the design of the RCS to the effects of vibration. In this regard, what hydraulic analysis methods, tests in models or operating reactors, or other means have been used to determine the character of the reactor coolant flow and its tendency to become a source of flow induced vibration?

5.27 On page 5.5-6 there is a discussion of the temperature difference between steam generator tubes and secondary shell during a LOCA blowdown and the resultant loading of the tube sheet. The discussion does not deal with the blowdown induced differential temperature across the tube sheets (approximately 470 F on the secondary side and approximately 270 F on the primary side). Comment on the effect of this differential temperature loading of the tube sheet.

6.0 ENGINEERED SAFETY FEATURES

6.40 Provide a more detailed characterization of the core barrel vent valves by referencing the design to earlier plants (e.g. Oconee). Are all the tests and analyses done for the Oconee class vent valves equally applicable to the Greenwood design?

6.41 Provide or reference an analysis of the CFT line break in the Greenwood class reactor. What are the effects of small breaks in the CFT line (i.e., not offset shear)?

6.42 Refer to Figure 6.1-1, the ECCS P&ID. Starting at either ECC vessel injection nozzle, trace back along the piping towards the low pressure system, through two check valves, the reactor building boundary, and a normally open motor operated valve. At this last valve there is a transition from high pressure to low pressure piping followed by a safety valve. Our concern is that no means are provided to detect leakage from the reactor coolant system back through the first (relative to the RCS) check valve, or from the CFT back through the second check valve. In the latter case, a decreasing CFT level or a discharge from the safety valve may be sufficient indication of leakage for the operator. However, in the former case, undetected leakage from the RCS could pressurize the line between the two check valves for an undetermined period of time. Subsequent failure of the second check valve or the CFT check valve would result in a LOCA (outside containment in the first case, inside in the second) with diminished ECCS capability. Thus, the failure of one check valve could lead to a LOCA and a degraded ECCS.

A change in design or monitoring should be made so that full credit can be taken for both check valves as protection against back leakage from the RCS. Such a change could take the form of a pressure indicator between the check valves, use of high pressure piping throughout, additional valves, increased relief valve capacity, different valve administrative aligument, or a combination of these.

6.43 The ECCS is required to be designed for all break sizes from small leaks within the capability of the makeup system up to the double ended rupture of the largest pipe. Correct the first statement in Section 6.3.1.1 in this regard. Provide or reference small break analyses applicable to the Greenwood reactors.

6.44 It is our position that the normally closed valves in each of two lines between the HP pump suction header and the DH pump discharge lines are not acceptable as proposed. These two valves should be remote manual valves with indication and control from the control room to facilitate system alignment as reguired for intermediate or small breaks.

6.45 Section 6.3.2.2.2 does not state that the DH and HP pump seals and other components will be designed for radioactivity that could be present for the recirculation cooling mode. Correct this deficiency.

6.46 Refer to Section 6.3.2.17, Manual Actions. Specify in complete detail the type and location of information available to the operator and all actions (e.g., push button, read meter, etc.) required of the operator to accomplish (a) the switching of ECC suction from the BWST to the sump and (b) the alignment of HP and DH systems for high pressure recirculation. Include in your discussion the response to request 6.44 , above. The information supplied in 6.3.2.17 is not now complete enough to construct the complete scenario of these actions or to judge the reliability of the proposed system.

6.47 On pages 6.3-12 and 13 the cross connection between low pressure injection piping is described. That description is not sufficiently complete. Provide drawings, analysis, plans for required testing, bases for instrumentation, and any other information pertinent to the specification of the cavitating venturis. How will the as-built flow split performance of the low pressure injection system be evaluated during preoperational testing?

6.48 What size are the vessel nozzles through which CFT injection occurs? What is the basis for this size (refer to response to Request 6.44, above)?

6.49 Section 6.3.4 should be amended to include periodic testing of the capability to realign pump suction and discharge by remote valve operations from the control room (refer to Response 6.44, and 6.46 above).

6.50 Certain inservice inspection capabilities could he precluded by design decisions, therefore, the applicant's commitment to supply inservice inspection information at the FSAR stage (page 6.3-20) is inadequate. What design provisions will be made to facilitate inservice inspection in the following regards?

a. CFT vessel injection nozzles (normal operation).
b. ECCS pumps, valves, and heat exchangers, and piping runs in the long term post LOCA mode of operation (including consideration of radioactive coolant).
c. ECCS valves, pumps, heat exchangers, and piping runs (normal operation).
d. Containment sump condition (normal operation and post LOCA).

What design provisions have been made to facilitate long term maintenance of the ECCS following a LOCA?

6.51 The SAR in Section 6.3 describes design considerations given to the possible failure of single active components in the ECCS. No consideration is given to failures of passive components. For long term core cooling following a LOCA, what information is available to the reactor operator in the event that coolant delivery to the core is interrupted for whatever reason; i.e., active or passive failure? What remote isolation capability is available to the reactor operator for use in switching from a disabled ECC delivery chain to an intact chain?

6.52 How were the core barrel vent valves treated in the thermal hydraulic analysis for normal operation? What assumptions were made in this regard for the analyses of accidents and transients? It is our position that one valve less than the minimum detectable number of stuck open vent valves should be assumed to be open in the analyses for the thermal hydraulic design of the reactor coolant system and core and for all transients. What is that minimum number? How is detection accomplished?

15.2 What is the basis for the statement given on pages 15.1-2A to 15.1-3 that if DNBR is less than 1.32 for a fuel element then its gap activity is assumed released, except for the locked rotor event where the DNBR is 1.0? What qualifies the locked rotor event as unique in this regard?

15.3 It is our position that for consistency in the description of safety margin for accidents, all analyses presented in Chapter 15 should assume 102 percent of rated core power (3600 MWt in this case) as an initial condition to allow for calorimetric error. We do not require at the PSAR stage that all analyses in Chapter 15 be revised to meet this assumption. Rather the applicant should describe the effect of this assumption on the analyses which have been presented, and if in specific accidents it is not possible to clearly demonstrate the margin available to accomodate this assumption, then the analyses for such accidents should be revised.

15.4 It is not clear from the information presented in Chapter 15 which equipment is assumed to function for each event analyzed. Thus, for example, it is impossible to determine how the single failure criterion was applied, if at all, for any event, except the LOCA. Provide a table for each event analyzed to show which equipment is assumed to operate and which is assumed to be inoperative. Include the assumptions for offsite and onsite power sources. For each event for which the single failure criterion or other measure of reliability margin has not been incorporated in the design, provide an assessment of the sensitivity of the plant response to the assumed level of performance of the safety or auziliary systems.

15.5 Describe in detail the method of computing the maximum dilution flow rate of 200 gpm given in Table 15.1.4-1. Include and justify the assumed flow split between RCS makeup and reactor pump seals. Specify the assumed number of operating makeup pumps and relate that number to the maximum possible number under normal plant operations.

15.6 Justify that startup of two inactive pumps from a plant operating power level of 61 percent is the worst case of the possible pump startup accidents, (Section 15.1.6). Include consideration of all idle pump operating modes given in the table on page 5.3-3 and allow for two percent increase in power level for calorimetric error.

15.7 Pump acceleration in 15 seconds to full flow is assumed for the pump startup accident. Justify that this is a realistic assumption. Contrast this assumption to Bellefonte 1 and 2 where instantaneous change from 50 to 100 percent flow was assumed for this accident.

15.8 Provide or reference a core thermal and hydraulic analysis (e.g., provide and describe a DNB analysis) for the "Loss of External Electrical

Load and/or Turbine Trip" to justify the statement at page 15.1-44 that "the loss of load transient...will induce...[No] increase in fuel cladding perforations."

15.9 Provide a response similar to 15.8 above for the case of "Loss of Normal Feedwater" to justify that the consequences of this event are limited to the release of coolant activity.

15.10 Expand Figure 15.1.9-1 to include a plot of minimum DNBR versus time.

15.11 For the Feedwater System Malfunction analysis described in this Section describe the analytical coupling between the analysis with the code POWER TRAIN and the conclusion that the minimum DNBR exceeds 1.32. That is, describe or reference the sequence of analyses. Is this coupling identical for all other Sections of Chapter 15 in which POWER TRAIN is used and in which the fuel thermal-hydraulic performance is calculated?

15.12 The third sentence of paragraph three on page 15.1-60 refers to "end-of-life conditions." This sentence is apparently in error and should refer to "rated power conditions." Please clarify.

15.13 Does the minimum subcritical margin of 1.2 percent $\Delta k/k$ given on page 15.1-88 for the steam line break include allowance for the highest worth rod stuck out of the core?

15.14 What system transient analysis method was used for the Steam Generator Tube Rupture accident? Reference of describe.

15.15 What instrumentation would be relied on to single out steam generator tube failure as the cause of an event so that the reactor operator would know that the required action at 15 minutes must be accomplished (see table 15.1.17-2)? Our concern is that a number of other possible events, e.g., a small pipe break LOCA for which no operator action is required, would be incorrectly diagnosed by the operator. The operator could then fail to achieve the proper manual action at 15 minutes.