UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

Docket Nos. 50-269
50-270
and 50-287

March 3, 1970

Duke Power Company
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Attention: Mr. Austin C. Thies
Vice President
Production & Operation

Gentlemen:

In our continuing review of your application for Provisional Operating Licenses for the Oconee Nuclear Stations, Units Nos. 1, 2, and 3, we need additional information as described in the enclosure. The requests are in groups which correspond directly to sections in your Final Safety Analysis Report (FSAR). In most instances, these requests relate to matters discussed with you in a meeting held at Bethesda in January 1970. We understand from that meeting that you intend to submit a revised reactor vessel material surveillance program which includes more capsules. Accordingly, we have not included questions on the surveillance program now referenced in your FSAR.

Some of our questions concern a Babcock & Wilcox proprietary report BAW-10008, Part 2, incorporated in your FSAR by reference. Summarize in the FSAR the nonproprietary aspects of this report including design criteria, design bases, computer codes developed and used, and conclusions reached. The answers to the enclosure may be incorporated in the FSAR or, in the case of proprietary items, be provided as a separate response.

Please contact us if you desire any discussion or clarification of the material requested by this letter.

Sincerely,

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure: As stated above

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RPB-3/DRL	RPB-3/DRL	RT/DRL	RP/DRL	DRL	DRL
ASchwencer:pt	CGLong	RDeYoung	RSBoyd	FSchroeder	PAMorris
2/27/70	2/27/70	2/27/70	3/2/70	3/3/70	3/3/70

3.8 Reactor Internals (The following questions apply to B&W Report BAW-10008, Part 1)

3.8.1 Briefly describe the manner by which Figure 10 of the report "Shear Force on Core for 36-Inch and 28-Inch Rupture," is derived from the pressure differential transients.

3.8.2 The report states that all components will be designed to ensure against structural instabilities, regardless of stress level. We note that the core support shield and core barrel shells were analyzed for unstable collapse due to external pressure. Were the control rod guide tubes analyzed for column buckling effects due to combined LOCA and seismic loadings? Identify any other components of the reactor internals for which buckling is a possible mode of failure under any of the design loading combinations. Provide the bases for using static loads in lieu of the dynamic response loads.

3.8.3 Provide the bases for the dynamic analyses and the associated dynamic load factors which are used in the stress and deflection analysis for horizontal and vertical excitation input, including bell mode responses. Give typical examples of such factors and their effect on the results.

3.8.4 The report states that seismic loads were determined from the response spectra for the design basis and maximum hypothetical earthquakes specified for the Rancho Seco Station site. Discuss how the seismic loads were determined from the response spectra. Give sufficient detail to show the development of the seismic loadings from the ground motion inputs for the containment structure to the final input used for the analysis of the internals structural members. In addition, describe in detail all dynamic analysis methods used in determining stresses and deflections for reactor internals under seismic loadings. Include in the discussion the following:

(a) A detailed description of all mathematical models of the system including a discussion of the degrees of freedom and methods of lumping masses, determining section properties, etc.

(b) A discussion of the analytical methods used including, where applicable, the methods of computing periods, mode shapes, and modal participation factors.

(c) A listing of and the bases for any damping values that were used.

(d) A list of points at which there are changes in stress analysis methods, e.g., dynamic to static, and the bases for such changes.

(e) Indicate the modal responses that were combined, e.g., deflection, acceleration, or stresses, and the procedure for combining these responses.

3.8.5 The discussion of the multimass model, Figure 22, refers to a more detailed multimass model. Describe the more detailed multimass model and discuss the basis upon which the results from this model determined the adequacy of the model used in Figure 22.

3.8.6 It is stated that the plenum cylinder and reinforcement plate were treated as a flat plate with a uniform pressure load in the calculation of stress and deflection. Describe the configuration and similitude of the model and the plenum chamber and reinforcement plate, including the boundary conditions assumed, e.g., edge fixity.

3.8.7 As discussed in the January meeting, the combined stress, $P_b + P_m$ for the control rod guide tube, reported in Section 3.2.2.3 of the report, should be clarified.

3.8.8 In reference to the stress summary of Table 1 of the report, provide the following information:

(a) Examples of how LOCA and seismic stresses were combined to give conservative results for these concurrent loading conditions.

(b) A separate summary of stress intensities due to the maximum hypothetical earthquake and the applicable allowable stress intensities.

3.8.9 For loading combination case IV in Appendix A, provide a comparison on an elastic basis between the stated stress limits and a membrane uniform strain for the materials associated with this loading combination.

3.8.10 Equations (5) and (7) of Appendix A should be corrected as discussed in the January meeting.

3.8.11 Appendix C indicates that the case IV loading combination stress limit utilizes ultimate strength curves published by U.S. Steel which are normalized at room temperature to minimum ultimate strength values given by Table N-421 of Section III. These U.S. Steel ultimate strength curves cannot be considered as conservative unless the lower bound value of the ultimate strength of each material at an appropriate design temperature has been established. Indicate how this concern will be resolved.

3.8.12 Amplify the discussion of Appendix D of the report concerning the stress limits and S_m values chosen for load combination cases II, III, and IV. This discussion should consider:

(a) The bases upon which S_m values and stress limits were selected, since code limits are not specified.

(b) The effect upon bolts of preload, pressure, and differential thermal expansion on the stress limits specified, for cases II, III, and IV.

3.9 Fuel Assembly Structural Design (The following questions apply to BAW-10008, Part 2)

3.9.1 Section 2.4 of the report does not sufficiently define the stress and strain limits for the design basis earthquake (DBE) and simultaneous maximum hypothetical earthquake (MHE) and loss-of-coolant accident (LOCA) nor the manner and extent to which the cited limits provide an assured margin against failure for these loadings. Our specific concerns are:

3.9.1.1 DBE Criteria

(a) Confirm that the type of stresses referred to in paragraph 1 are in the primary category as defined in Article 4 of ASME Code, Section III. Describe the basis for establishing 75% of the stress rupture life of the material as a numerical limit and whether that limit is constructed upon the average stress or the minimum stress to produce rupture at the end of 10^5 hours.

(b) Clarify whether stresses of the type referred to in paragraph 2 are in the secondary category in the same context as above. Where stresses exceed yield, are they calculated on an equivalent elastic basis, i.e., pseudo-elastic basis as in Section III? Identify the source of the fatigue curves used for each material of concern (e.g., Article 4, Section III). Where fatigue data are employed which are not included in any codes or standards, specify whether a basic data curve is used or a design curve which incorporates design/correction factors and correction for maximum effect of mean stress. Provide the bases for the statement that strain limits will be set using no more than 90% of the material's fatigue life. Specify the number and type of cycles that have been established for design purposes and indicate the margin of safety that exists over the expected number and type of operation cycles to be experienced.

(c) For the combination of stresses in (a) and (b) above, specify the stress limits that apply (e.g., 3 S_m or S_L).

3.9.1.2 Combined LOCA and MHE

(a) Clarify whether the applied stress referred to in paragraphs 1 and 2 is a primary stress, exclusively. Provide the basis for establishing 85% of ultimate strength of the material as a numerical stress limit. Is the ultimate strength normalized to the minimum tensile strength of the material as specified in the appropriate ASME or ASTM material specification? Is this stress calculated on an elastic basis? Provide the elastic stresses corresponding to this limit for each of the materials of concern. Furnish the corresponding strain limits of each material.

(b) Identify the components referred to in paragraph 2 that contribute to the stability of the control rod guide tubes.

(c) Provide the basis for the allowance of 85% of the critical buckling load as a limit. Identify the theoretical column formulae used (i.e., Euler or other).

3.9.2 Relate quantitatively Figure 3 of this part of the report to the figures of Part 1.

3.9.3 Briefly describe the analytical techniques that the FLASH computer code utilizes and its capabilities in relation to its employment on this problem.

3.9.4 The model used to describe the dynamic behavior of the reactor vessel and internals is not described in sufficient detail to permit an assessment of the accuracy by which the vessel and internals have been analytically described. Provide:

(a) Engineering drawings and/or sketches of the structural features of importance.

(b) A precise description of the location of and basis for computation of masses and section properties/boundary conditions.

(c) Details on the manner in which flexibility coefficients have been computed and the results achieved.

3.9.5 The design loadings and their manner of application to the structure require more precise description. Provide:

(a) The complete digitalized acceleration record that was used in the analysis.

(b) Discuss the stress limits applicable to the simultaneous LOCA and seismic loads and the basis therefor.

(c) A general description of the manner of digital-to-analog conversions of data, an estimate of the accuracy of the process and a description by which the acceleration was inserted into the electronic differential analyzer.

(d) A complete acceleration response spectrum comparison at 1 and 10 percent critical damping.

(e) The manner in which the vertical seismic component has been factored into the analysis and the importance of the stresses and deflections therefrom with respect to the horizontal seismic and LOCA loadings.

3.9.6 The manner in which analog computations have been performed is not presented. Provide a detailed description of the manner in which these computations have been performed. In addition, provide strip chart recorder output results for several typical runs and a tabulation of significant stress, strain and deflection results at critical locations for these same runs.

3.9.7 Provide a sketch of the second model segment (as discussed in Section 4.1.4 of the report) and discuss its interaction with the first model segment.

3.9.8 In reference to Figures 7 and 8 of the report which show the mathematical model for the vertical contact analysis and its load-deflection curve, specify the spring constant variation for the fuel assembly in relation to its location within the core for that part of the load-deflection curve which occurs after the gap is closed.

3.9.9 Section 5.1 of the report discusses the frequency and damping tests performed for full-size and subsized specimens. Further detailed information is required to complete our review. Provide discussion of the following:

(a) The basis for test amplitudes and frequencies used.

(b) A description of and bases for the type of loadings used, including test fixtures employed.

(c) A detailed description of the full-size and subsized specimens used including the identification of specimen materials.

(d) Description of test data obtained.

(e) Interpretation and analysis of results.

3.9.10 In reference to the spacer grid compression tests described in Section 5.3, provide a sketch showing the test specimen, its orientation in the loading fixture, and the direction of loading. Explain how corrections were made for temperature effects. Provide elaboration on the load cycling phenomenon noted in paragraph 2 and show graphically how this occurs.

3.9.11 Horizontal contact analysis results are given in Section 6.1 in terms of margins of safety calculated on the basis of allowable and applied loads. Provide the maximum stresses that were calculated from the applied loads for the applicable components in both Sections 6.1 and 6.2. Specify how LOCA and seismic stresses are combined.

3.9.12 Section 6.2.1.2 shows the margin of safety for guide tube buckling under LOCA loadings only. Indicate the margin of safety for combined LOCA and seismic loads. Confirm that seismic loads are included in the reported results of Section 6.2, vertical contact analysis.

3.9.13 Provide a detailed explanation for the conclusion in Section 6.2.2.1 that loads due to LOCA and/or earthquake are not additive to those due to normal operation because the maximum loads are limited by the available friction loads between the end grids and the fuel rods.

3.10 Control Rod Drive System

3.10.1 Identify in the FSAR or in B&W Report BAW-10007 the design codes which are applicable and applied to the rod drive system. For non-code items indicate the stress, deformation and fatigue limits used. Discuss the analytical approaches taken in a format which will include the above items and which will demonstrate the margins of safety provided under normal operating conditions and hypothetical accident conditions.

3.10.2 Provide descriptive information and a discussion of the function of the springs which release the roller nuts. Include information on spring material and material specification, fabrication techniques, and design stresses.

3.10.3 We understand that, in addition to the motor torque tests referenced in BAW-10007, tests have been performed to assess the ability of the control rod drive mechanism to drive-in a stuck rod. Describe these tests and provide the results.

3.10.4 All tests reported in BAW-10007 have been performed on a prototype unit. Indicate any significant differences in design, materials, tolerances, and fabrication techniques between the prototype units and the production units, and discuss their importance in determining the need to repeat the basic tests with production units. Discuss the test program contemplated for the production units and the acceptance criteria to be applied.

3.10.5 Discuss the tests and/or analyses that have been employed to assess the damage which would result from operator errors or minor malfunctions, such as over-driving a limit switch.

3.10.6 Provide a list of the metals, lubricants, insulation materials, etc., which were tested in the prototype unit and discuss their long-term reliability in the reactor environment.

11.8 We understand that you intend to rely on the RIA-36 reactor coolant letdown radiation monitors for detection of prompt fuel failures. Describe the sensitivity and response time of these monitors. Indicate the smallest number of failed fuel elements that the monitors can detect as well as the highest activity they can withstand without loss of function. Discuss the effects of crud buildup and provisions for decontamination of the section of letdown line being monitored.

14.3 Steam-Line-Rupture Accident

14.3.1 We understand that the main turbine stop valves serve to isolate the unaffected steam generator in the event of a steam-line-rupture accident. Describe the design, operation, and inspection of the main turbine stop valves. Discuss the capability of a turbine stop valve to close against reversed critical flow.

14.3.2 Describe the extent that the system which trips the turbine stop valves by a reactor trip signal meets IEEE-279.

14.3.3 We understand that in your analysis of the steam-line-rupture accident you have assumed that portions of the Integrated Control System (ICS) function (e.g. closing the main turbine stop valves, and the feedwater valves). For those portions of the ICS which you have assumed to function properly, either provide an evaluation for our review to show that the system design conforms to IEEE-279 Criteria or analyze the steam-line-rupture accident at 100% power with an end-of-life moderator coefficient, minimum shutdown margin and a stuck rod condition, assuming that the ICS and the operator fail to function or function in an adverse manner.

14.3.4 Describe the hybrid analog-digital computer program used for analysis of the steam-line rupture including physical models, equations, assumptions, numerical approximations, and input parameters.

14.3.5 For the analysis of the worst case steam-line-rupture accident provide the following:

(a) All input quantities including fluid inventories, time delays and constants, instrumentation time delays, negative reactivity insertions, flow rates, and heat transfer coefficients. Justify each and explain why each is a conservative assumption.

(b) The time sequence of important events including reactor trip, turbine stop valve trip, main feedwater valve and pump operation, main feedwater startup valve operation, emergency feedwater valve and pump operation, bypass and relief valve operation, high pressure injection actuation.

(c) Results in the form of process variables as a function of time for all important quantities including:

(i) steam flow, pressure and temperature for both the affected and normal steam generators;

(ii) feedwater flow pressure, temperature, and liquid level for each steam generator;

(iii) liquid and vapor mass inventories in each steam generator;

(iv) heat transfer rate in each steam generator;

(v) maximum shell and tube temperature and pressure difference and maximum thermal stresses;

(vi) primary system pressure;

(vii) pressurizer level;

(viii) primary system coolant temperatures;

(ix) enthalpy peaking factors with a stuck-out rod;

(x) reactivity;

(xi) average and maximum fuel temperatures;

(xii) average and maximum cladding temperatures;

(xiii) thermal power or heat flux;

(xiv) DNB ratios, including correlations and justifications for use;

(xv) primary containment pressure for the break occurring within the primary containment.

14.3.6 We understand that you assume the most adverse steam-line-rupture accident consequences would occur at 100% reactor power with no loss of offsite electric power. Provide a discussion of why this assumption is conservative and why it is not necessary to consider other initial conditions such as:

(a) concurrent loss of offsite power
(b) break occurring at hot shutdown
(c) break occurring at less than 100% power.

14.3.7 Under the assumption that only protective systems function for the first 10 minutes of the worst case steam-line rupture accident, determine the additional amount of time the operator has to isolate the affected steam generator and provide for an orderly plant cooldown. Describe the actions the operator must take in order to terminate the accident.

14.3.8 For the worst case steam-line-rupture accident analyzed above, evaluate the possibility of steam generator tubes being ruptured concurrently. Consider blowdown loads, pressure and temperature induced stresses, and tube degradation caused by long-term erosion, vibration, corrosion, and leakage.

14.3.9 Discuss the need for and capability of the steam generator level indication system to function during a steam-line-rupture accident.

14.4 <u>Pressurizer Level</u>

14.4.1 Either demonstrate that pressurizer level need not be considered by providing a sensitivity analysis of the effects of pressurizer level on the consequences of the startup accident, the rod ejection accident, and the steam-line-rupture accident (consider the complete range of initial pressurizer level, from empty to full), or provide and describe a system that detects and alarms at high and low pressurizer levels and meets the criteria of IEEE-279.

14.4.2 Provide the following information on the pressurizer heaters: either an analysis of the consequences of uncovering energized heaters or a description of a protection system meeting the criteria of IEEE-279 which would prevent energizing the pressurizer heaters unless they are submerged.