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

JUL 15 1970

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

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

Gentlemen:

We have completed our review of the material that you have filed on the loss-of-coolant accident (LOCA) analysis in connection with your application for operating licenses for the Oconee Nuclear Units.

On the basis of our evaluation of the information you have provided, and our evaluations of the LOCA for other PWR designs, we cannot establish reasonable assurance that your methods of analyses are conservative. We are enclosing a list of questions which, if answered satisfactorily, should permit us to conclude that your ECCS design and analyses are acceptable.

Your response to the enclosure should be expedited so that we may complete our review of your ECCS design.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Additional Information
Request

appl

| | | | | | |
|---|---|---|---|---|---|
| OFFICE ▶ | PWR-2/DRL | PWR-2/DRL | PWRs/DRL | DRL | DRL |
| SURNAME ▶ | DFRoss:pt | CGLong | RCDeYoung | FSchroeder | PAMorris |
| DATE ▶ | 7/14/70 7/14/70 | 7/14/70 | 7/14/70 | 7/ /70 | 7/15/70 |

July 15, 1970

ADDITIONAL INFORMATION REQUEST

1. Provide the results of your evaluation of the LOCA using a multinode analysis (such as your FLASH-2.5 code) for a 28-inch ID, double-ended, cold-leg pipe rupture. In addition to providing information on clad temperature, system pressure, etc, also provide the core and hot channel flow rate in detail sufficient to fully characterize the thermal and hydraulic performance during blowdown. These details should include:

   a. core pressure drop, quality, mass velocity;

   b. hot channel pressure drop, quality, mass velocity;

   c. heat flux distribution in hot channel;

   d. flow rates in upper and lower plenums;

   e. flow rate in broken and intact cold-leg and hot-leg piping; and

   f. flow rate out the break.

   Identify the heat transfer correlations used for the various phases of the blowdown and refill period and relate these correlations to the most recent experimental data available.

2. With the same degree of detail, provide the results of your evaluation of a 36-inch ID, double-ended hot-leg pipe rupture.

3. Provide a summary discussion regarding your acceptance criteria for ECCS functional performance. Your discussions should include an identification of any supporting information which has become available as a result of the Commission-sponsored emergency core cooling test programs.

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

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

April 22, 1970

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

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

Gentlemen:

As discussed with you in a meeting at Lynchburg, Virginia on March 31 and April 1, 1970, we require additional information to complete our evaluation of your combined loading stress and deflection analyses for fuel assemblies and reactor internals. The information needed is described in the attached enclosure. The requests are in groups which correspond directly to sections in your Final Safety Analysis Report. You will note that request 3.8.4 forwarded to you by our letter of March 3, 1970 has been revised. This revision is based on the above mentioned discussions.

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

Sincerely,

/S/

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
As stated above

| PWR-2/DRL | PWR-2/DRL | PWRs/DRL | DRL | DRL |
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| 4/20/70 | 4/20/70 | 4/21/70 | 4/22/70 | 4/22/70 |

## REQUEST FOR ADDITIONAL INFORMATION

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

(1)3.8.4 With respect to the response spectrum/modal analysis method discussed in Section 3.1.6 provide:

a. An engineering sketch of the structural configuration represented by the model.

b. A tabulation of masses and flexibility/stiffness factors, preferably in matrix form.

c. A brief discussion of the program used to compute frequencies, mode shapes, etc.

d. The mode shapes, frequencies, and participation factors developed by the analysis.

e. The criteria used to combine the modal contributions in order to arrive at deformations and/or forces on the reactor internals.

*3.8.13 Provide the following information related to the stress analysis:

a. The mathematical models used, the assumed boundary conditions and representative free body diagrams. Identify the component loads, loading sources and resulting stresses for primary load paths, i.e., bolted joints, plenum cylinder, core grids.

b. Sketches or drawings, to supplement Figure 23, showing all critical areas (such as discontinuites, areas of clearance and bolted connections).

3.8.14 We understand that with combined accident loads some of the bolts joining the core barrel and core support shield will be stressed beyond yield strength. Describe the methods of analysis used for these bolts and justify the bases for exceeding yield strength. Also, discuss how you will be assured that these bolts will retain their preload and strength properties throughout the life of the plant.

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(1) This is a revision of request 3.8.4 made by letter of March 3, 1970

* These are in addition to requests 3.8.1 through 3.8.12 made by the March 3, 1970 letter.

3.8.15 Document typical results from either an experiment or a theoretical analysis that considers the shell bell modes (n=0, n=2, n=4, etc) for the core support shield and core barrel during LOCA conditions. These results should include the effect of shell bell mode deformations on bolted joints.

3.8.16 Document typical results of a stress analysis that considers the effects from the lateral pressure maldistribution that occurs across the core support shield and core barrel during a LOCA.

3.9 (The following requests apply to B&W Report BAW-10008, Part 2.)

3.9.14* With respect to the time history/modal analysis discussed in Section 3.3, provide:

a. The thrust vs time function used as the applied force on the assumed model.

b. A brief description of the analytical program used.

c. The modal damping coefficients used.

d. The manner in which the resultant load is combined (magnitude or phasewise) with other LOCA and seismic loads.

3.9.15 With respect to the fuel assembly horizontal seismic analysis, provide:

a. The mathematical models for the Phase 1 and Phase 2 analyses.

b. The engineering basis for and validity of the decoupling assumed between the Phase 1 and Phase 2 models.

c. The analog diagrams for the two phases with accompanying explanations of symbols used on the diagrams.

d. A discussion of damping coefficients to include the basis for their selection, an engineering assessment of the validity and conservation in the computational method used and an example showing how they have been determined.

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*Requests 3.9.1 through 3.9.13 were made by letter of March 3, 1970.

e. A description establishing the basis for the gap and stiffness coefficient values selected.

f. A copy of one analog run giving necessary data for force balance calculations.

g. A discussion of the criteria for the acceptability of the output results of the seismic analyses, and the bases for these criteria.

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

April 15, 1970

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

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 Station, Units 1, 2, and 3, we need the additional information described in the enclosure.

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

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
As stated above

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| 4/14/70 | 4/14/70 | 4/14/70 | 4/15/70 | 4/15/70 | |

April 15, 1970

# REQUEST FOR ADDITIONAL INFORMATION

## 14.5 *REACTOR COOLANT PUMP LOCKED ROTOR ACCIDENT

Provide a qualitative description of the transients caused by a reactor coolant pump locked rotor for each of six possible combinations (i.e., 1 case for 4-pump operation, 2 cases for 3-pump operation, 2 cases for 2-pump operation, and 1 case for 1-pump operation).

For the worst of the above cases, provide the results of calculations of reactor core and coolant leg flows, power, primary system pressure, fuel and clad temperatures, and DNB ratios. Describe the computational procedure and show that conservative assumptions were used for moderator temperature coefficient, initial power, initial temperature, initial pressure, minimum shutdown margin with a stuck rod, hot channel factors, core heat transfer, gap conductance, steam generator heat transfer, and pressurizer response.

## 14.6 REACTOR COOLANT PUMP SHEARED SHAFT ACCIDENT

Provide the same information on the sheared-shaft-accident as requested for the locked-rotor-accident above.

## 14.7 OPERATION WITH LESS THAN FOUR REACTOR COOLANT PUMPS RUNNING

a. Calculate and discuss the flows and temperatures for the reactor core, the two steam generators, and the six primary coolant legs for these modes of partial loop operation: three pumps, two pumps in one loop, one pump in each loop, and one-pump operation. Include subcases corresponding to isolation or nonisolation of one steam generator.

b. Describe the measurements that will be made during the startup program to verify these flows and temperatures.

c. Describe your evaluation of accidents and operational transients which might be initiated during partial-loop operation, especially during single-loop operation.

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* Requests 14.1 through 14.4 were made by letters of February 13 and March 3, 1970.

d. For each mode of partial-loop operation, evaluate the potential for cooling of the loops by the once-through steam generator system. Provide a discussion of the operation of the integrated control system for each mode of partial-loop operation and each mode of control: automatic, manual, load tracking, and startup.

e. For each mode of partial-loop operation, discuss the potential for cold water transients resulting from inadvertent startup of an inactive pump or pumps. Provide an analysis of the consequences of the worst case. Make conservative assumptions such as instantaneous acceleration to full pump flow, most negative moderator temperature coefficient, minimum 1% hot shutdown reactivity margin, minimum stagnant loop temperature, and high initial pressurizer level. Describe the calculational method and give values of all input parameters.

## 14.8 RESTART OF A TRIPPED PUMP

a. Provide an analysis of the worst cold water transient which could result if subsequent to the tripping of a coolant pump, operator and integrated control system actions reduced power and restarted the tripped pump. Make conservative assumptions, especially for secondary side flows and heat transfer.

b. Describe the measurements to be made during the startup program to verify the system behavior and consequences of this transient.

## 14.9 STARTUP ACCIDENT

For the maximum reactivity ramp insertion rate which is slow enough to cause a high pressure reactor trip before a high neutron flux level trip [about 2 x $10^{-4}$ $(\Delta k/k)$/sec, see Figure 14-3], provide curves of pressurizer level and pressure versus time and compare the maximum expansion rate of the primary system with the relief capacities of the pressurizer safety valves.

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March 27, 1970

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

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

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

Gentlemen:

We need additional information to complete our review of your analysis of the loss-of-coolant accident for the Oconee Nuclear Station. The flow rates predicted by the analyses of the system blowdown and core heatup for the spectrum of cold-leg break sizes should be provided and the core heat transfer coefficients used should be identified. A comprehensive summary should be provided of the analytical methods and computer codes used in the analysis of the thermal-hydraulic aspects of the loss-of-coolant accident. This summary should include:

(a) a description of each code (purpose, fundamental assumptions, basic equations, nature of input and output);

(b) a description of the procedures used in applying the output results of one computer code as input information to another code;

(c) identification of the codes used in calculating the results shown in various tables and figures in the FSAR;

(d) a quantitative discussion of the bases for selection of the input parameters used in the FLASH code to model the primary system, and justification that the selected parameters lead to conservative results.

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

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

| PWR-2/DRL | PWR-2/DRL | PWR/DRL | DRL | DRL |
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| ASchwencer:pt | CGLong | RCDeYoung | FSchroeder | PAMorris |
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ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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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,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure: As stated above

| RPB-3/DRL | RPB-3/DRL | RT/DRL | RP/DRL | DRL | DRL |
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| 2/27/70 | 2/27/70 | 2/27/70 | 3/2/70 | 3/3/70 | 3/3/70 |

ADDITIONAL INFORMATION REQUIRED

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 Pressurizer Level

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.

Docket File

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

February 13, 1970

Docket No. 50-269
50-270
and 50-287

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

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

Gentlemen:

In our continuing review of your application for a Provisional Operating License for the Oconee Nuclear Units Nos. 1, 2, and 3, we have identified the need for additional information as described in detail in the enclosure. The requests have been categorized into groups which correspond directly to sections in your Final Safety Analysis Report (FSAR). Most of these requests were discussed with your representatives in meetings held at Bethesda in September and November, 1969.

We recognize that some of the information requested may be available in the public record in the context of our regulatory review of similar features of other facilities. If such is the case, you may wish to incorporate the information by reference.

We have concluded that at least one strong motion accelerograph should be installed in your facility in order to provide information for damage evaluation and a determination of the station's response resulting from an earthquake. We have asked for a description of such instrumentation and its utilization in question 5.15 of the enclosure.

Your design includes actuation of the emergency core cooling system (ECCS) by instrumentation monitoring different variables in order to provide functional diversity. According to your evaluation, reactor trip is required for the ECCS to be effective for some break sizes. Your design, however, does not appear to include reactor trip from diverse variables for these break sizes. We conclude that all of the functions required for effective emergency core cooling, including reactor trip, should be actuated from the sensing of diverse variables. In question 7.22 we have requested that you indicate how you plan to provide this capability.

A

Because the Oconee Unit 1 steam generators will be the first full-scale production units of this design, we have concluded that measurements should be made of the actual vibratory motions of the steam generators during preoperational testing and during initial power operation. Your plans for such measurements should be submitted for our review.

As we discussed with your representatives at the November meeting, since each unit in your complex will be phased into service at approximately yearly intervals, the overall Oconee Station operating organization will undergo several changes during this period. A detailed discussion of the organizational functions and administrative controls during the transition period encompassing the activation of each unit should be provided for our review.

For the initial operation of Oconee Unit 1, we have concluded that a minimum of five men will be required for each shift crew, including one Senior Licensed Operator and two persons with Operator Licenses. After significant operating experience has been obtained, we will consider a smaller shift crew size if it can be shown that fewer men can perform all normal and emergency functions in accordance with established and proven procedures.

For operation of Units 1 and 2 which share a common control room, our current thinking is that a minimum of eight men per shift crew is required, including two Senior Licensed Operators, and three persons with Operator's Licenses.

Our present thinking is that operation of all three units would require a total shift crew complement of twelve men. Assuming overall facility operation is under the direction of a single supervisor, three Senior Licensed Operators and four Licensed Operators would be required in addition to the supporting auxiliary operators. In this case, each licensed operator is assumed to hold a license valid on each unit in order to achieve maximum flexibility. Serious consideration should also be given to providing an Instrumentation & Controls Technician for overall site support on a shift basis.

Before taking final positions on the required staffing for multiunit operation, we would be pleased to meet with you and consider any additional information you have developed which would support a smaller crew size. As indicated during our discussions in November, 1969, we would expect such information to include an assessment of the minimum shift manpower necessary during periods of abnormal or emergency operation.

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

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
As stated above

Distribution:
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| DR | | | | | |
| DRoss:pt | CGLong | RDeYoung | RSBoyd | FSchroeder | PAMorris |
| 2/10/70 | 2/10/70 | 2/10/70 | 2/10/70 | 2/11/70 | 2/13/70 |

February 13, 1970

ADDITIONAL INFORMATION REQUIRED

## 2.0 SITE AND ENVIRONMENT

2.1 Provide data showing the total permanent and transient population within the 6-mile low population zone at present and projected for 2010.

2.2 In Supplement No. 1, dated April 1, 1967, you provided an area map showing the location of the Clemson-Pendleton water intake and the Anderson water intake (to have been completed in 1968). Give the daily water consumption for these intakes and verify the distance in stream miles of each from the Keowee tailrace.

2.3 We understand that you are performing additional meteorological studies at the Oconee site. Provide the data and analyses that justify the valley drainage model presented in the FSAR and used in your dose calculations.

3.0 REACTOR

3.1 Reactivity Calculations

3.1.1 We understand that you use 2 and 3 dimensional PDQ 5 and 7 calculational techniques for flux shape and reactivity eigenvalues in addition to the models discussed in Section 3.2.2.1. Describe the extent that these codes were used in your core design, and discuss the applicability of each utilization.

3.1.2 Describe the methods used to calculate reactivity as a function of core lifetime and to calculate boron reactivity worth. Present experimental verification if available.

3.1.3 Provide comparisons of calculations with experimental data to demonstrate ability to determine power distributions in cores with different enrichment zones. If such experimental data have not yet been obtained, discuss how you will determine such distributions for anticipated operating conditions.

3.1.4 Provide fuel element positions, enrichments and beginning of life (BOL) and end of life (EOL) average and maximum burnups for each zone of the first, second, and equilibrium cycles for all three reactors.

3.1.5 Provide an x-y power distribution at BOL for the unrodded core. In addition provide the x-y power distribution at BOL for the worst case design configuration of part and full length control rod assembly groups, which takes into account transient xenon effects.

3.2 Reactivity Coefficients

3.2.1 Discuss in detail the calculational methods and experimental bases for prediction of Doppler coefficients, including uncertainties in the calculated value of the Doppler coefficient.

3.2.2 Provide information on the temperature dependence of the average moderator temperature coefficient at BOL for an unrodded core. Provide such information at EOL, for the fuel cycle in which the coefficient will be most negative, with the rods in the core (this pertains to possible reactivity insertion in the steam-line-break accident).

3.2.3 Provide information on the spatial variation in the BOL moderator void coefficient for the fuel loading arrangement, enrichment, and largest boron concentrations which will be used. Such variation might lead to a larger maximum reactivity insertion in a depressurization accident than would be the case if the uniform void coefficient is considered. Identify the largest reactivity insertion possible considering the spatial variation of the coefficient and the worst possible configuration of voiding.

3.2.4 Provide details of the calculations predicting reduction in the BOL moderator temperature coefficient as xenon reaches equilibrium as indicated in conditions 5 and 7 of Table 3-7. Describe experimental verification available to support this. This information is needed for our evaluation of the potential for azimuthal xenon instability.

3.2.5 In regard to proposed operation with a positive BOL moderator temperature coefficient not greater than $0.5 \times 10^{-4}$ $\Delta k/k$-°F, identify those startup measurements and analyses you intend to perform and discuss how they will ensure that this coefficient is not actually larger at rated power. Include discussion of uncertainties in the measurements, and how the effects of the coefficients from fuel Doppler effect, axial expansion, and other sources will be treated in predicting the full power moderator coefficient.

3.2.6 Calculate the power coefficients of reactivity, $(\%\Delta k/k)/\%\Delta P$, for constant inlet coolant temperature at BOL with maximum boron for power levels corresponding to two-, three-, and four-pump operation.

3.3 Shutdown Margin and CRA Worths

3.3.1 Additional information is needed on reactivity control requirements and maintenance of a minimum shutdown margin during lifetime of the reactor. The minimum shutdown margin might not occur at BOL or EOL conditions as discussed in FSAR, but rather when the boron concentration reaches its minimum value and the transient xenon control group is inserted in the core. Provide an expected history of the control assembly configuration for each control group, identifying each by position, function, and reactivity worth. Describe under what normal and abnormal operating conditions you expect to reprogram control rod assemblies between groups or alter the functional designation of control groups.

3.3.2 We understand you plan to continue to operate in the event that one CRA is stuck in the withdrawn condition. Because another CRA could fail to insert at shutdown, show how an adequate shutdown margin would then be maintained. Provide calculations of the hot shutdown margin for the worst possible CRA stuck out of the core and for the worst possible pair of CRA's stuck out of the core. When in core life and under what control assembly group configurations do these cases occur? Include the nuclear hot channel peak-to-average factors and the predicted power levels at DNB for these cases.

3.3.3 With reference to the rod ejection accident, what is the maximum possible reactivity worth of an inserted CRA as a function of core life and power level? Describe the bases for these calculations. Is the maximum reactivity rod a stuck rod or is it one of the rods within an inserted control group? Explain how it will be determined during operation that the worth of an inserted CRA does not exceed these calculated values.

3.4 Xenon Stability

What value of the moderator temperature coefficient represents the threshold value for azimuthal xenon instability? Discuss the experimental and calculational bases for the prediction of such thresholds and indicate estimated errors in the prediction. Describe the sensitivity of the predicted threshold to variations in the assumed Doppler coefficient. Using the information supplied in response to 3.2.1 and 3.2.4 above, state the least favorable predictions of Doppler coefficient and moderator coefficient for xenon instability.

3.5 Detection and Control of Power Maldistributions

3.5.1 Describe how the operator will use the out-of-core detector readings to position the part-length control rod assemblies.

3.5.2 State the peaking factors and margins to thermal limits for worst conditions of a CRA left in the core, a misaligned part-length CRA, and one CRA left out of the core when the remainder of a permitted group is fully inserted.

3.5.3 Describe the means available to ensure over the long term that design peaking factors are not exceeded. Discuss the ability to detect x-y power tilts (as from out-of-place control rods), azimuthal xenon oscillations, or fuel loading errors. How can gross errors in fuel loading, such as improper enrichment in a substantial fraction of the fuel be detected? Discuss the effects of misloading of fuel (i.e., wrong enrichment or location) on the margin to DNB during normal and anticipated transient operation.

3.5.4 Describe the calibration of out-of-core neutron detection instruments. Indicate how the need for recalibration will be determined. Show that your method of calibration does not mask axial or azimuthal power maldistributions.

3.5.5 It appears that out-of-core detector readings may not provide an indication of actual incore flux distributions when a control group is inserted in the core, if a reassignment of the control rod assemblies to the xenon transient group were to be made, or if the reactor is returned to full power at the time of maximum xenon buildup. While these conditions do not produce flux tilts, they would change the radial power shape, and therefore the out-of-core detector readings. Further changes also occur until a new equiblibrium is reached. Calculate the magnitude and effect of such changes and show how such changes would affect the adequacy of reactor trip settings.

## 3.6 Thermal-Hydraulic Design

3.6.1 Describe the model, computer code, and primary coolant system input variables used to predict core thermal performance during loss-of -flow accidents, including all-pumps-trip, locked-rotor, and sheared-pump-shaft events.

3.6.2 Describe the method and indicate the results of the analysis that predicts core bypass flow during normal operation, and indicate to what extent this bypass flow rate can be verified during startup and in model tests.

3.6.3 Provide engineering hot spot factors based on measurements from production fuel elements.

3.6.4 We understand from our meetings with you that a mixing code named TEMP is used in your core thermal-hydraulic design. Provide a detailed description of that code, including fundamental assumptions, experimental bases, all input data for normal or design calculations, and output results. The results should include consideration of the various possible modes of operation of the primary pumps.

3.6.5 Justify the continued use of the W-3 correlation in the computation of DNB ratios for operation with less than four pumps, since the lower limit for mass flow rate in the W-3 correlation is $1 \times 10^6$ lb/hr-ft$^2$.

3.6.6 Explain the basis for your selection of the C-factor correlation in the computation of the non-uniform heat flux factor, F, associated with the W-3 correlation.

3.6.7 We understand that your thermal analysis at the design over-power of 114% steady-state power assumes a reactor inlet temperature several degrees cooler than for 100% power. Explain this assumption by discussing the flow rates and temperatures in the primary and secondary coolant systems for the overpower condition.

3.6.8 What is the effect of burnup on the peak linear heat generation rate, maximum fuel temperature, and $UO_2$ melting temperature?

3.6.9 What fuel burnup limit is proposed to limit the fuel clad to 1% plastic strain?

3.7 Internal Vent Valves

The FSAR incorporates by reference a proprietary report, BAW-10005, submitted to the AEC by Babcock & Wilcox. Summarize in the FSAR the nonproprietary aspects of this report including design criteria, design bases, nature of tests performed, production unit tests, and installation and removal provisions. Include the following information in this summary or, in the case of proprietary information, provide a separate response.

3.7.1 The vent valves in the core support shield are designed to provide a flow path to remove steam generated in the core following a loss-of-coolant accident. Provide the following information concerning these valves.

3.7.1.1 The material to be used for each component of the valves including the bushings and shaft.

3.7.1.2 The clearances that will be provided between the shaft, bushings, and journals.

3.7.1.3 A preliminary indication of your plans for pre-operational testing, inspection frequency, and evaluation of the long-term effects of the reactor operating environment.

3.7.1.4 Plots of steam generated following a loss-of-coolant accident by all energy sources considered in sizing these valves, including the core, the hot reactor internals including the core shroud, and the hot reactor vessel.

3.7.1.5 An analysis which demonstrates the adequacy of valve sizing, indicating the flow rate that can be passed by the valves assuming a large cold leg break without depressing the coolant level below the core midplane during recovery.

3.7.2 Provide pertinent vent valve design development information. We understand that a design report has been prepared.

3.7.3 Discuss the results of your impact analysis on plastic deformation of the vent valve disc under accident conditions. Indicate the maximum degree of deformation expected, and show why this deformation cannot adversely affect valve performance during the course of an accident.

3.7.4 Discuss the potential for loss, during the plant lifetime, of parts of the jackscrew assembly (shown on FSAR Figure 3-40).

3.7.5 Indicate the scope of the vibration tests performed on the vent valves. Indicate the resonant frequency of the assembly and the basis for concluding that it is not within the range of frequencies expected to be present in the Oconee system.

3.7.6 Discuss how and when you will demonstrate the capability for remote inspection and removal of the vent valves following installation at Oconee.

3.8 Reactor Internals

Provide a complete listing of all non-destructive examinations and inspections to be performed for the reactor internals, and identify the acceptance standards which apply in each case.

4.0 REACTOR COOLANT SYSTEM

4.1 With regard to brittle fracture control of the reactor coolant pressure boundary, discuss the extent to which your design is consistent with the following statement:

a. Those pipes with wall thickness less than 1/2 inch need not have material property tests (such as Charpy V-notch) if (1) they are austenitic stainless steel, (2) the ferritic material is normalized (heat treated), or (3) the ferritic material has been fabricated to "fine grain practice."

b. Pipes with wall thickness greater than 1/2 inch must have a nil ductility transition temperature 60° F below anticipated temperature when the system has a potential for being loaded to above 20% of the design pressure. Ferritic material with an NDTT of -20° F or austenitic stainless steel will also fulfill the requirements.

4.2 The FSAR, Section 4.1.3.3, indicates that the reactor coolant pump casings will meet the intent of ASME Code Section III, Class A vessels, but are not code stamped. Outline briefly the stress analysis procedures used for the pump casing, furnishing references as appropriate, and provide a summary of stress intensities and cumulative damage usage factors obtained. Confirm the absence of deviations from Code requirements other than stamping.

4.3 Amplify the discussions of the supports for the reactor vessel, pressurizer, steam generator, and pump and motor to include:

a. A description of the expected motion of each of the elements of the support structure(s) and how these motions accommodate all normal, emergency, and faulted loading conditions within the allowable stress limits for the supported component, i.e., compliance with paragraph N-473 of ASME Code, Section III;

b. Quantitative stress limits for the support structures for the loading combinations delineated on page 4-4 of the FSAR;

c. Sketches or drawings of the supports which provide sufficient detail to illustrate the information requested in (a).

4.4 Discuss the effect of differential settlement of the foundation in creating relative displacements of the reactor coolant system supports resulting in additional piping reactions at the reactor vessel nozzles and similar effects on other major components of the system. In this discussion state the maximum magnitude of relative support displacement for which the stress intensity limits of ASME Section III will not be exceeded and indicate what assurance exists that these limits will not be violated.

4.5 Indicate how cracks or cracklike defects have been considered in formulating a safe reactor pressure vessel pressure-temperature region which accommodates property changes due to irradiation during the life of this plant, (page 4-25). Specifically discuss how such stress intensifiers were assumed to contribute to potential initiation of a brittle mode of failure.

4.6 Describe in detail those analysis and testing procedures used to determine that the nuclear steam supply system (reactor vessel, steam generators, reactor coolant pumps, etc.) meets Seismic Class I criteria. Include the following:

a. A detailed description and sketch of the mathematical model(s) of the system, including a discussion of the degrees of freedom and methods of lumping masses and determining section properties.

b. A discussion of the analytical procedures used, including where applicable the methods of computing periods, mode shapes, modal participation factors, and the procedures for computing design accelerations, displacements, shears, and moments.

c. A discussion of the possibility and significance of dynamic coupling between the nuclear steam system and the supporting structure (internal structure within the containment building)

d. A listing of the damping values used.

4.7 Identify all electroslag welds incorporated in Class I systems. Describe the process used, its variables, and the quality control procedures employed.

4.8 Reactor Vessel

4.8.1 Describe any requirements imposed on the reactor vessel design by state regulation beyond those specified in Section III of the ASME Code.

4.8.2 Discuss the magnitude of the thermal stress induced in the reactor vessel membrane by radiation.

4.8.3 Identify and locate all ring forgings used for reactor shell sections other than closure flanges for Unit 1, 2, or 3 reactor pressure vessels.

4.8.4 In reference to the summaries of primary plus secondary stress intensitities and cumulative fatigue usage factors for components of the reactor vessel, provide sketches illustrating the points of analysis and a discussion of the results of the transient stress analyses.

4.8.5 Discuss transients, such as loss of flow and loss of load that cause temperature and pressure excursions influencing the cumulative fatigue factor of the reactor vessel in a significant manner.

4.8.6 Specify any nozzle penetrations, in the reactor vessel or heads other than in-core instrumentation and control rod drive nozzles, that are partially penetration welded into the vessel or heads.

4.9 Steam Generator

The FSAR incorporates by reference a proprietary report, BAW-10002 submitted to the AEC Regulatory Staff by Babcock & Wilcox for use in evaluating this R&D item. Summarize in the FSAR the nonproprietary aspects of this development, including design criteria, design bases, nature of model tests performed, and your test program to verify extrapolation of developmental effort to the full-scale production units, the first of which are being installed in Oconee Unit 1. Include the following information in this summary, or in the case of proprietary items, provide a separate response.

4.9.1 Describe the objectives and present an evaluation of the results obtained to date on the 19 tube model tests. Discuss plans for continued testing of these or other models.

4.9.2 Justify extrapolation of model tests (7, 19, and 37 tubes) to predict performance of full-scale steam generators (15,000 tubes). Discuss the basis for confidence in your ability to predict the absence of instabilities in the operation of the full scale production units.

4.9.3 Describe in detail the full-scale verification test program to be conducted at Oconee Unit 1. Discuss parameters to be monitored, transients to be evaluated and conditions (limits) which must be met to validate safety related performance.

4.9.4 Describe the specific method you will use to detect tube fouling. Discuss the consequences of potential flooding of feedwater nozzles. State the fouling factor limits beyond which cleaning procedures are required.

4.9.5 Identify the cleaning process you intend to use (chemicals, temperatures, and cleaning times) in the steam generator to remove fouling deposits and conservatively evaluate metal loss associated with this process based on specific coupon tests or similar test applicable to your situation. State what allowance has been made for loss of tube metal in establishing tube design strength.

4.9.6 Provide transient response curves for the abnormal transient tests performed.

4.9.7 Provide your evaluation of the potential for thermal fatigue due to fluctuation and shifting of the liquid-vapor interface on the tubes.

4.9.8 Describe the several computer programs used to assist in the design of the steam generator and in the transient analyses.

4.9.9 Describe the stress distributions and effective elastic constants obtained under thermal inplane and transverse loadings which the steam generator is designed to withstand. Discuss the detailed analysis of the tube-to-tube sheet complex (as an integral structure).

4.9.10 Provide a summary of the stress intensities and cumulative damage usage factors for the steam generators.

4.10 Describe how flow-induced vibration loads have been considered in the design of the primary system. Indicate the normal and emergency operational modes considered, and the design limits, amplitudes and frequencies applicable to these modes.

4.11 Discuss the possible means of monitoring for vibration and for the presence of loose parts in the reactor pressure vessel and other portions of the primary system during preoperational testing and initial power operation as well as the feasibility of inservice monitoring for this purpose. Indicate your plans to implement such means as are found practical and appropriate for this plant.

4.12 <u>Other Class I Systems and Components</u>

4.12.1 Section 1C of Amendment 8 to the FSAR, System Design Criteria identifies systems and components "designed for seismic loading", but does not identify by seismic classification, i.e., Class I or II. Provide seismic classification for all applicable components and systems.

4.12.2 The FSAR identifies ASME Code, Section III plus code interpretations and code addenda issued through Summer 1967 as being specified for applicable seismically designed components of this plant. Confirm that no earlier editions of Section III or addenda thereto were specified for any applicable Class I components of Units 1, 2, or 3.

4.12.3 Specifically list any systems which contain a seismic classification interface and/or a B&W to Duke system interface responsibility.

4.12.4 With regard to the seismically designed piping within the reactor building provide:

a. The methods utilized to determine the input for the piping analyses.

b. A discussion of the analytical procedures used, including the methods of computing the stiffness and mass matrices, periods, mode shapes, and participation factors, and the procedures for computing design accelerations, displacements, shears, moments, and stresses.

c. Typical mathematical models for several piping systems for the Oconee plant.

4.12.5 State how seismically designed mechanical components have been determined to qualify for service under seismic and other emergency loading conditions. Discuss the means used for the Oconee plant relating the methods used to the frequency spectra and amplitudes calculated to exist at the equipment support and the predicted emergency environment. Indicate whether the components have been tested or analyzed in the operational mode as well as statically. If not so tested or analyzed, explain the basis for assuming that such items as emergency core coolant pumps and drives will start and run, if needed, under these loadings.

4.13 Pipe Whip and Missile Protection

4.13.1 Specify how seismically designed systems are protected against damage by pipe whipping.

4.13.2 Expand your description of the provisions used to protect the reactor primary system, other vital systems, and structural supports for these systems from missile hazards. Describe the design of the missile shields including missile spectrum, missile velocities, and the penetration formulae used.

4.13.3 Provide the results of an evaluation assessing the potential consequences from possible missiles which might be generated in the event of failure of a primary pump flywheel. Describe the program to be followed to minimize the probability for experiencing a flywheel failure, including the consideration given to material selection, design margins, fabrication, failure analyses, acceptance testing, inservice inspection requirements, and other quality assurance measures.

4.13.4 Failure of the bearings on the primary pump motor shaft or of the shaft itself could lead to creation of a missile consisting of the flywheel and part of the motor shaft. Either failure could conceivably lead to creation of missiles through breakup of the flywheel. Provide the results of an analysis of the effects of applicable load combinations, including seismic loads, on the pump motor unit, and indicate the margins against failure of the bearings, the shaft, and other critical components. Provide your assessment of the potential consequences of such failures.

4.14 Inservice Inspection

4.14.1 The bases of your proposed inservice inspection program are not clearly stated in Appendix 4A nor in Technical Specification 15.4.6. Identify, by date of issue, the ASME Draft Code for Inservice Inspection of Nuclear Reactor Coolant Systems that was used for guidance. If any of the design requirements, provisions for access, initial baseline tests or other requirements called for in the Code are to be omitted or modified in your program identify and discuss your reasons for the change.

4.14.2 Describe your inservice inspection program for the Class I Mechanical Systems outside the primary system pressure boundary, including items to be inspected, inspection schedule, and types of inspection. Some items to be considered are primary vessel supports, primary pump flywheels, and all the engineered safety features in the category of Class I Mechanical systems.

4.15 Leak Detection

4.15.1 Provide the sensitivity in gallons per minute and the detection time for each of the leak detection systems for the primary coolant pressure boundary. Indicate how information from the systems is provided to the operator, including the control room alarms where provided.

4.15.2 Describe the leak detection systems provided for other Class I fluid systems, and list those Class I fluid systems for which no special leak detection system is provided.

5.0 STRUCTURES

5.1 On page 5-12 it is stated that the finite element mesh for the base slab was extended down into the foundation material to take into consideration the elastic nature of the foundation material and its effect upon the behavior of the base slab. This extension below the base slab is apparently not shown on Figure 5-4, "Reactor Building Finite Element Mesh." Provide a drawing of the mesh used to account for the effects of the foundation material.

5.2 We understand that the tendon access gallery is structurally separated in the vertical direction from the base slab. Describe how the prestress gallery was considered in the design of the base slab.

5.3 The finite element mesh shown for the containment buildings apparently does not include the interior structure. Indicate what influence the interior structure has on the stresses in the base slab computed by the finite element analysis. Describe how the base slab was designed to resist the seismic shear and overturning moment from the interior structure.

5.4 What maximum thermal stresses were calculated for the walls of the spent fuel pool under normal conditions and after prolonged outage of the fuel pool cooling system? State what provisions have been made to control cracking of the concrete structure under these conditions.

5.5 Describe how the fuel storage racks were designed for seismic loadings.

5.6 Submit the containment design report.

5.7 For containment coatings, provide the following information:

a. Identification of material to be used, location, and function.

b. Physical and chemical characteristics.

c. Performance under accident (LOCA) conditions including washdown, radiation, steam, temperature, and jet impingement effects. Performance should demonstrate good adherence with no significant washdown loss that could adversely affect performance of spray nozzles or core and heat exchanger heat transfer surfaces.

5.8 Identify the tendon corrosion inhibitor to be used as tendon filler. If a change has been made from the NO-OX-ID originally indicated, justify in detail by test and performance data, the adequacy of the material selected.

5.9 The containment proof test plans and containment monitoring accomplished to date have not been described in sufficient detail to permit us either to evaluate the adequacy of the planning for conduct of the test or to assess the meaning of test results in terms of structural adequacy. Provide the following:

a. An updated description of the instrumentation to be used to monitor the structure during the proof test. Emphasize the extent to which the embedded instrumentation is expected to remain operable and describe the degree to which failed instrumentation can be tolerated in judging structural adequacy from the test; if not tolerable, describe provisions for replacement prior to the pressurization of the structure.

b. The final procedures (in sequence) of structural proof testing. Include the extent of observation of structural behavior during pressurization and depressurization of the structure. Discuss the extent of the internal containment temperature control and the basis for this control.

5.10 For Class II components, systems, and structures provide a detailed description of the design procedures used, the constants selected and an example of their application to a component, a structure, and a system.

5.11 Discuss the possibility and significance of dynamic coupling between the nuclear steam system and the supporting structure (internal structure within the containment building).

5.12 Describe the provisions made to transfer seismic and wind shear forces across construction joints.

5.13 It is understood that spectra from the highest piping system anchor point in the Auxiliary Building are used for both the Auxiliary Building and the Turbine Building piping. Explain why the spectra for the two buildings are not expected to be different and exhibit different amplifications at different frequencies. Describe how rocking of the Turbine Support Structure has been considered. Demonstrate that use of the spectra from the Auxiliary Building for pipes in the Turbine Building results in conservative seismic stresses.

5.14 We understand that the Turbine Building has been designed to resist the earthquake loadings postulated for the site in order to protect the Seismic Class I equipment and piping located within the Turbine Building, and that the structure has been designed for a uniform static lateral coefficient of 0.22 g for the maximum hypothetical earthquake, which corresponds to the peak spectral acceleration for 2% damping. Demonstrate that this method is conservative as stated. If contributions from the various modes of response can result in an acceleration higher than 0.22 g at the roof, show how the structure can withstand this loading.

5.15 Describe the instrumentation that will be installed at the facility to provide information for damage evaluation and determination of the plant's response resulting from an earthquake. Include the type of instruments to be used, their location, the type of information that will be obtained from each, and how the information will be utilized.

## 6.0 ENGINEERED SAFETY FEATURES

6.1 With regard to the reactor building penetration room and its associated exhaust air treatment system:

a. Provide elevation and plan views as necessary to show the locations of: all openings that must be sealed under accident conditions, location of both filter intake ducts, valve PR-V11, vacuum relief valve, all pressure sensor(s) and associated penetration room connections used to control room vacuum, and any deliberate inleakage paths.

b. Describe the instrumentation system used to control the penetration room vacuum and indicate its ability to withstand single failures. Discuss both pressure control, and to the extent necessary, filter face velocity control. Indicate the system parameters which are monitored and alarmed in the control room. Provide the flow vs head characteristic of one filter fan and total pressure drops calculated at design vacuum conditions (indicate design vacuum) assuming one fan inoperative, for both clean and dirty filter conditions.

c. Discuss how an adequate vacuum will be maintained throughout the penetration room and connected areas taking into account the number and location of pressure sensors, the constriction at the fuel pool location and any locations of significant inleakage including leakage of equipment hatch seals.

d. Discuss the effects of high winds on the exterior walls of the penetration rooms in terms of potential for unfiltered leakage.

e. From Figure 6-5 it appears that a potential exists for loss of air flow (cooling) in one of the two filter trains that could cause filter heating and potential desorption. Evaluate the ability of the present design to either preclude this condition or evaluate the consequences. The heat load calculation involved in this evaluation should be based on activity buildup due to maximum proposed containment leakage.

f. Provide final design information on the filters. Include charcoal type, mass, flow cross section, bed depth and iodine contact time. Give rated flow and provisions to continue a specified cooling flow in the event of fan failure. Give provisions for moisture control or evaluate the consequences of not providing such control.

g. Describe the factory and in-place efficiency tests that will be performed on the penetration room filters for particulates and iodine. Include general method of test, materials used, in-place test connections, test material injection time and effect of test materials on subsequent performance of filters.

6.2 Show how the design of the reactor protection system and of the electrical and mechanical equipment associated with engineered safety features located in the containment, or elsewhere in the plant, takes into account the potential effects of radiation on these components due to normal and accident conditions (superimposed on long-term normal operation). Describe the analysis and testing performed to verify compliance with design requirements.

6.3 Identify all equipment and components (e.g., motors, cable, pump seals) located in the primary containment or elsewhere in the plant which are required to be operable during and subsequent to a loss-of-coolant or a steam-line-break accident. Describe the qualification tests which have been or will be performed on each of these items to ensure their availability in a combined high temperature, pressure and humidity environment.

7.0 INSTRUMENTATION AND CONTROL

7.1 Provide your seismic design bases for the reactor protection system (RPS), the emergency electric power system and its controls, and the instrumentation and controls for both the engineered safety features (ESF) and the decay heat removal system. Include consideration of the ability of the systems to actuate reactor trip or engineered safety feature action if called upon during and following the maximum peak acceleration. If a seismic disturbance occurred after a major accident, evaluate the likelihood and consequences of possible interuptions of engineered safety features functions. Identify the seismic specifications employed in the instrumentation and control purchase orders and describe what tests and analyses will be required to assure that the seismic design bases are met.

7.2 Describe the quality control procedures which apply to the equipment in the RPS, the ESF and containment isolation systems, and associated emergency power systems. This description should include quality control procedures and records used during equipment fabrication, shipment, field storage, field installation, and system component checkout.

7.3 Pages 7-8, 7-10, 8-9, and 8-10 of the FSAR do not present sufficient information on the installation of the reactor protection systems. Submit your cable installation design criteria for independence of redundant RPS and ESF circuits (instrumentation, control and power). (The protection system circuits should be interpreted to include all sensors, instrument cables, control cables, power cables, and the actuated devices, e.g., breakers, valves, pumps.) Include the following:

(a) Separation of power cables from control and instrument cables. (Describe any intermixing within a tray---conduit, ladder, etc--of control and instrument cables, of different protection channel cables, or of nonprotection cables with protection cables.)

(b) State how your design accomplishes separation of electrical penetration assemblies within the penetration rooms into areas, grouping of these assemblies in each area, and the separation of assemblies with mutually redundant circuits.

(c) Describe cable tray loading, insulation, derating, and overload protection for the various categories of cables.

(d) Describe your design with respect to fire stops, protection of cables in hostile environments, temperature monitoring of cables, fire detection, and cable and wireway markings.

(e) Describe the administrative responsibility and control provided for the foregoing (a-d) during design and installation.

(f) Describe how the location of RPS and ESF process instrumentation inside containment has been designed to include separation of redundant sensors and sensing lines, protection for cable runs between sensors and their electrical penetrations.

7.4 Provide the basis for assurance that loss of the air conditioning and/or ventilation system will not adversely affect operability of safety related control and electrical equipment located in the control room and other equipment rooms. Describe the analysis performed to identify the worst case environment (e.g., temperature, humidity). Identify the limiting temperature and associated conditions that would require reactor shutdown, and state how this was determined. Describe what factory and onsite testing has been or will be performed to verify satisfactory performance under extreme environmental conditions.

7.5 Describe how RPS and ESF equipment will be physically identified as safety equipment in the plant.

7.6 In your FSAR, Sections 1A.11 and 7.4.5, you have discussed the capability of maintaining a safe shutdown if access to the control room is lost. Describe your capabilities and intentions for going to a safe shutdown from the various operating modes, from outside the control room, (refer to General Design Criterion 11).

7.7 Describe the communication systems available to the control rooms for special purpose use (e.g., sound powered phones) and emergency use (e.g., the Duke microwave system).

7.8 Describe your emergency lighting facilities and areas of coverage (e.g., control room, operating stations, passageways, equipment rooms).

7.9 With regard to the bypass of the reactor coolant pressure actuation signal in the HP and LP Injection Systems, supply the following additional information:

(a) The conditions prerequisite to permitting initiation of bypass, including status of diverse protection instrumentation.

(b) The number, type, and activation sequence of switches used to initiate bypass in each system.

(c) The indication available to the operator that each bypass has been actuated and/or is capable of being actuated.

(d) The justification for manually bypassing automatic actuation of the HP Injection System 400 psi above its actuation setpoint.

(e) The provisions available to the operator to readily remove each bypass below its respective automatic removal setpoint.

7.10 Describe what information is available to the operator to identify all RPS and ESF channels that are in test or maintenance. State what prevents more than one redundant channel to be in test or maintenance at the same time. Describe the indication available, down to the channel level, to identify which instruments initiate a protective action. These descriptions should be in sufficient detail to permit a determination of the system's compliance with Sections 4.13 and 4.19 of IEEE 279.

7.11 Clarify or correct the following items in the FSAR.

(a) Table 7-1, Figure 7-1, and Section 15.2.3 show different setpoints and conditions for the Power/Flor and Power/RC Pump reactor trips.

(b) Figure 7-6 and 7-7 disagree as to how the SCR gating circuits are disabled on a reactor trip. Figure 7-6 indicates that power from the programmers to the group power supplies is interrupted. Figure 7-7 indicates that 120 vac input power to the programmer, which is shown as part of the group power supply, is interrupted.

7.12 Page 7-22 states there are asymmetric rod pattern monitors. Provide a description of these monitors to include detection circuitry, alarm logic and alarm setpoints, control or protective actions served and design bases.

7.13 Page 7-23 lists rod drive control system faults. Describe the circuits used to monitor for these faults, the basis for automatic correction and the nature of the corrective action taken. Include the circuitry involved in the stuck CRA accident discussed in Section 14.1.2.7 of the FSAR.

7.14 Pressure switches used for ESF Channels 7 and 8 are shown on FSAR Figure 7-3. Describe your provisions for sensor checks, channel testing and calibration to show your ability to meet the intent of Sections 4.9 and 4.10 of IEEE-279 during power operation.

7.15 Page 7-17 of the FSAR states that "The drive controls, i.e., the drive mechanism and rods combination, have an inherent speed-limiting feature." Describe this feature and show how it prevents rod speeds of other than 30 inch/minute. We understand that this speed-limiting function is accomplished by the use of synchronous programmer motors. Identify the sources of power to the programmer motors. For each of these sources, describe your evaluation of the features which affect frequency and how they can be depended upon to limit frequency changes to acceptable values.

7.16 Only a portion of the control rod drive system is shown on Figures 7-6 and 7-7 of the FSAR. Missing are the mode control portion which automatically or manually selects rods or groups of rods) the regulating rod sequencer, relay logic, relay logic monitors, and interlock inputs from the reactor protection system. Provide suitable schematic and logic diagrams to correct this deficiency. In addition, provide the following:

a. Show how the regulating rod group sequencer and "enable" curcuits are electrically independent of means used to move the safety rods.

b. If the auxiliary power supply can be used to move rods in more than one group explain how two-group movement is thus controlled.

c. Describe the conditions under which the regulating rod groups "sequence" mode is bypassed. This bypass mode or the manual control mode will permit operation of more than one group movement in the 25 to 75% withdrawal (high reactivity insertion rate) region; show how this was evaluated.

d. Identify which rod groups are automatically inhibited from movement or are automatically caused to be inserted by specific ICS or RPS conditions.

e. Clarify the manner in which the part length Group 8 CRA's are moved as regulating CRA's (e.g., manual, automatic) also discuss whether these Group 8 CRA's should be tripped by RPS logic as shown on FSAR Figure 7-7 or should not be tripped as noted on FSAR page 3-6.

f. A design feature common to all CRA drives is that they can be held in a withdrawn position with dc voltage applied to one of the six motor windings. Identify the minimum and maximum applied voltages that can do this and discuss the potential for such a voltage being applied downstream from the reactor trip points.

7.17 The Integrated Control System (ICS) and its design bases are discussed in Section 7.2.3 of the FSAR. This discussion does not identify which, if any, of the functions provided by this system are required for reactor protection or for actuation of the ESF. For example it appears that the ICS is required to limit the consequences of a steam line break event. Please supply the following information:

a. Identification of the safety related functions provided by the ICS;

b. The limitations placed on reactor operation if the ICS or any of its subsystems (unit load demand, integrated master, steam generator control, and reactor control) is not operating properly.

17.18 For the process instrumentation channels which provide signals to the RPS and ESF actuation circuitry, provide a table which lists the following information: (1) parameter sensed; (2) sensor type (e.g., Bailey pressure), manufacturer's specified accuracy, repeatability and expected failure mode(s); (3) type of readout (e.g., indicating, blind); (4) the type of power required (e.g., external, self); (5) use of channel (RPS or ESF); (6) identification of sensors connected to a common sensing line (e.g., a common pressure tap).

7.19 Identify the type and manufacturer of the out-of-core nuclear detectors. Cite prior experience with these detectors in operating power reactors. Provide an evaluation showing that the detector design capabilities are compatible with application requirements. Include ambient pressure and temperature and gamma and neutron levels (instantaneous and integrated) in this evaluation. If integral cables are not used, discuss the reliability of the connector at the detector.

7.20 Briefly describe the design concepts utilized for the signal conditioning and readout circuitry for the process and nuclear instrumentation.

7.21 The information as now contained in the FSAR is not sufficient to warrant a conclusion that the reactor coolant flow sensing scheme complies with the requirements of IEE-279 (Sections 4.2, 4.6, and 4.7 in particular). Examination of Figures 4-2 and 7-17 show that all four RPS dP cells and the control dP cell are taken from the same flow nozzle in each loop. Provide an analysis to show the ability of the reactor protection system to withstand failure (e.g., severance) of any one of the 1-inch flowmeter connections. Indicate what effect the loss of one such connection will have on the remaining connection to that flow nozzle.

7.22 Provide a description of the actuation of both the ECCS and a reactor trip from diverse signals. Evaluate this design for the full spectrum of breaks in the primary coolant system. This evaluation should include the time dependent sequence of important events, such as reactor trip, reaching pressure trip setpoints, ECCS actuation.

## 11.0 RADIOACTIVE WASTES AND RADIATION PROTECTION

11.1 Describe the factory and in-place efficiency tests to be performed on the gaseous waste system filters for particulates and iodines. Indicate general test method, materials, and test times.

11.2 We require information to show that malfunctions of bleedback valve WD-V66 cannot overpressure the liquid waste tanks. Provide details of the maximum pressure that might be reached in these tanks upon failure of this valve or its control causing it to become instantaneously full open. Relate this pressure to the relief valve discharge capacity.

11.3 Figure 11.3 shows a single control room switch common to both waste gas exhauster line valves, WD-V62 and WD-63. These valves are also shown commonly interlocked with radiation monitors. Provide sufficient details on operation of these valves to show that no single failure can cause an unintentional activity release through this line.

11.4 Provide sufficient detail to show that activity from Waste Gas Tank A (or B) cannot be inadvertently released through failure of control or protection instrumentation connected to valve WD-V65A (or WD-V65B). As a minimum, there should be an additional valve in the outlet line of each of these tanks to provide for single valve failures.

11.5 Regarding operation of the purge system, provide the normal conditioning flow and the flow design pressure at accident pressure prior to isolation.

11.6 Demonstrate the suitability of ranges of radiation monitors for the following conditions:

a. Those channels monitoring routine releases should remain on scale for releases up to technical specification limits.

b. Those channels monitoring the consequences of accidents should remain on scale during the postulated accident.

c. Those channels providing a control function for an engineered safety feature should not have their function denied by the dose consequences of an accident.

11.7 Provide verification that the minimum dilution flow from the Keowee tailrace with no hydro units operating is 30 cfs or greater as assumed.

## 12.0 CONDUCT OF OPERATIONS

12.1 The information describing the plans for dealing with emergencies of the Oconee site is insufficient to permit evaluation. Please provide the overall Emergency Plan including: basis and objectives; emergency organization including specific assignments of authority and responsibility; identification of emergency conditions considered; designation of protective measures to be taken when specific predetermined action levels are reached; technical bases for applicable portions of the plan; emergency communication networks; notification responsibility and authority of offsite agencies and support groups, medical arrangements for contaminated and/or injured personnel; training requirements; and provision for periodic review and updating. The plan should also include provisions for possible multi-unit interaction, particularly while Units 2 and 3 are under construction.

12.2 Provide information describing how the security of the Oconee site will be ensured against acts of industrial sabotage. Indicate the extent of perimeter fences, security lighting, guards, control room access, visitor accountability and other site surveillance methods which may be employed. Indicate what review of critical plant features has been made to ensure suitable protection in regard to the above. State the provisions to be taken to limit the access of construction personnel and prevent inadvertent operation of equipment during the construction of Units 2 and 3.

12.3 Indicate the organizational structure and relationship between Duke Power Company and B&W for each phase of operation from preoperational testing through the power ascension program. Include assignments of responsibility and authority for approval and conduct of tests and procedures, evaluation of results, and resolution of system and equipment anomalies. Provide resumes indicating experience and qualifications of all supplemental personnel expected to be utilized for technical and operational support during this period of initial operation. Indicate by position which personnel are expected to possess operator licenses prior to fuel loading.

## 14.0 SAFETY ANALYSIS

14.1 Provide analyses of the potential hydrogen evolution in the containment volume following a LOCA, as a result of radiolysis of emergency coolant, clad-water reaction, and chemical reactions of materials subject to corrosive attack in the post-accident environment. Evaluate the potential hazards to containment and other engineered safety features that may be associated with the accumulation of combustible gases. Describe the provisions you will make for controlling the post-accident concentration of combustible gases and indicate the nature of and plans for any development and testing required to demonstrate the performance and reliability of associated equipment.

### 14.2 Design Basis Loss-of-Coolant Accident

14.2.1 Indicate the time of occurrence for the following events following initiation of a design basis loss-of-coolant accident: start of injection from core flooding tanks, peak containment pressure, blowdown over, core flooding tanks empty, ECCS starts, containment spray starts, containment heat removal fans start, and containment spray water storage tank empties. Assume only emergency power is available.

14.2.2 If removal of energy by the steam generators is included during blowdown, present a detailed analysis of the method used to calculate the heat removal and provide assurance that the required heat sink will always be available.

14.2.3 In order to evaluate the active containment heat removal systems, i.e., the emergency fan cooling units and the sprays, the following parametric data are required: (1) the effects of inlet water temperature and vapor flow rate on the heat removal capability of the fan coolers when the containment is at the peak pressure following the design basis accident and (2) the heat removal capability of the fan coolers as a function of the steam-air mixture temperature. Provide similar information for the heat removal capability of the spray systems.

14.2.4 Describe the model and assumptions used to calculate the pressure buildup in different containment compartments during the design basis accident.

14.2.5 List the thermal diffusivities of the structural heat sinks that were used in the containment pressure transient analysis. If various surfaces are painted or treated in a manner that might affect their heat transfer characteristics, describe how this is accounted for in the analysis.

14.2.6 Provide an energy balance table showing the energy stored prior to the design basis LOCA, the energy generated and absorbed from t = 0 seconds to the time of the peak pressure, and the energy distributed at the time of peak pressure for at least the following items:

- Primary coolant internal energy
- Core flooding tanks internal energy
- Energy stored in fuel and clad
- Energy stored in core internals
- Reactor vessel metal energy
- Shutdown energy and decay heat
- Energy transferred to steam generators
- Energy in piping, pumps, and valves
- Steam generator metal energy
- Secondary coolant internal energy
- Containment air energy
- Containment steam energy
- Energy transferred to steel structures
- Energy transferred to concrete structures

14.2.7 In order to show a mass balance at any time in the pressure transient, plot the mass of water entering or leaving the containment free volume from such systems as the core flooding tanks, primary coolant system, refueling water storage system, and the containment cooling system. Similarly, plot the pounds/hour of steam evolved from the design break area into the containment versus time. These plots should be for cases assuming the minimum containment heat removal rates, minimum ECCS capability, and the design basis accident break area.

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November 28, 1969

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

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

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

Gentlemen:

Amendment No. 8 to your application for licenses for the Oconee Nuclear Station incorporated by reference Babcock & Wilcox Topical Report BAW-10014 "Analysis of Sustained Departure from Nucleate Boiling Operation" dated August 1969. As discussed with you at our meeting on September 24, 1969, we require additional information in order to evaluate the potential for operation at sustained DNB in the Oconee reactors and the potential for DNB conditions to propagate with associated cladding failures.

The additional information should be provided as an amendment to your application or as a revision to BAW-10014 incorporated into your application by reference. Your response should include the information requested in the attached enclosure, "Information to Be Supplied on Sustained DNB."

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Additional
Information

See Attached for Previous Concurrences

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Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Additional Information

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INFORMATION TO BE SUPPLIED
ON SUSTAINED DNB

1. Provide a discussion of relevant experimental and operational data on sustained DNB available in the literature and in B&W's experimental programs.

2. Provide a list of the mechanisms that might cause high heat flux leading to sustained DNB operation in the Oconee reactors (e.g. misplaced fuel assemblies). Include a discussion of the location and magnitude of these high heat fluxes and an analysis of the precautionary measures and design features of the Oconee reators which limit the possibility and consequences of high heat fluxes leading to sustained DNB.

3. Provide additional analysis and documentation of the local consequences to irradiated fuel elements caused by sustained DNB operation (e.g. accelerated cladding corrosion, fatigue, rod swelling, rod bowing). Include an analysis of the possibility and extent of DNB propagation caused by associated cladding failure.

4. Provide an analysis of the methods available in the Oconee reactors to detect operation at sustained DNB.

5. Provide a discussion of how accidents, transients, and the function of engineered safeguards would be affected by operation at sustained DNB.

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The additional information should be provided as an amendment to your application or as a revision to BAW-10014 incorporated into your application by reference. Your response should include the information named in the attached enclosure, "Information to Be Supplied on Sustained DNB."

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

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

MAY 2 1969

Mr. William S. Lee
Vice President, Engineering
Duke Power Company
Power Building, Box 2178
Charlotte, North Carolina 28201

Dear Mr. Lee:

We have completed our review of the information submitted in Amendment 6 to the Oconee application regarding the design of the submerged weir in the cooling water intake canal. We and our consultants conclude that the design is satisfactory and that the weir should withstand the effects of any high velocity flow over the crest or hydrostatic uplift pressures associated with rapid drawdown of Lake Keowee.

Our review of the other matters contained in Amendment 6 reveals no apparent significant departures from currently acceptable designs, however, we will review these matters in detail during the operating license review. We wish to acknowledge, however, and commend your willingness to voluntarily modify the Oconee design in an attempt to conform with more recent licensing actions.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

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50-270
and 50-287

AUG 9 1967

Duke Power Company
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Attention: Mr. W. S. Lee
Vice President

Gentlemen:

Three copies of a Safety Evaluation by the Division of Reactor Licensing dated August 4, 1967, were sent to you on August 7, 1967 under separate cover.

The Safety Evaluation relates to the staff's review of the application filed by Duke Power Company for licenses to construct the Oconee Nuclear Station Units 1, 2, and 3 near Clemson, South Carolina.

Sincerely yours,

Original Signed by
Frank W. Karas

for Roger S. Boyd, Assistant Director
for Reactor Projects
Division of Reactor Licensing

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| OFFICE ▶ | RPB-3/DRL | RP/DRL | | | | | |
| SURNAME ▶ | FWKaras/ltj | for RSBoyd | | | | | |
| DATE ▶ | 8-8-67 | 8- -67 | | | | | |

Form AEC-318 (Rev. 9-53)

U.S. GOVERNMENT PRINTING OFFICE : 1966—O-214-629

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50-269

JAN 9 1969

Mr. C. J. Wylie
Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 28201

Dear Charlie:

I am voting for approval of Draft #5 of the Electrical Systems Standard, and wish to submit the following comments for your consideration:

1. The concept of "independence" is not defined precisely; (see paras. 5.2.2.c and 5.2.4.c.2). In my opinion, independence should preclude the automatic connection of redundant standby sources onto the same bus simultaneously, or in sequence upon loss of the first source.

2. Fires should be included in Table 1.

3. Para. 5.24.g should specifically require a seven day fuel supply as a minimum, assuming realistic loading.

4. The surveillance requirements should be more specific in terms of the parameters to be monitored.

5. "Engineered Safety Features" should be defined.

6. The second sentence of Para. 5.2.1 (General) should be rewritten as: "Sufficient physical separation, electrical isolation, independence, and redundancy shall be provided to ensure the continued functional capability of the station's class 1E systems under design basis event conditions".

7. Redefine "Acceptable" (see Para. 3.10).

8. Delete Para. 5.1.2 since its provisions are implicit in the preceding paragraphs.

9. Under 5.3.3, include a requirement that redundant station batteries be in separate rooms.

Very truly yours,

Original signed by

Donald F. Sullivan

| | | | | | | |
|---|---|---|---|---|---|---|
| OFFICE ▶ | I&PTB:DRL | | | | | |
| SURNAME ▶ | DFSullivan:eks | | | | | |
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W. B. Cottrell,
A. Wells, AS&LB

MAY 15 1967

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

avp

Duke Power Company
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Attention: William S. Lee
Vice President, Engineering

Gentlemen:

This is to inform you that your application for a construction permit and an operating license for a third pressurized water reactor to be known as "Oconee Nuclear Station, Unit 3," which was contained in Amendment No. 3 dated April 29, 1967, has been a assigned Docket No. 50-287.

A copy of a Notice of Filing of Application for Construction Permit and Facility License which has been submitted to the Office of the Federal Register for filing and publication is attached for your information.

You will be informed when action is taken on your new application or if additional information is required.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Federal Register Notice

| OFFICE | RPB #3-DRL | RPB #3-DRL | OGC | RP/DRL | DRL |
|---|---|---|---|---|---|
| SURNAME | FWKaras/ltj | CGLong | | RSBoyd | PAMorris |
| DATE | 5-8-67 | 5-11-67 | 5-13-67 | 5-15-67 | 5-15-67 |

Form AEC-318 (Rev. 9-53) U. S. GOVERNMENT PRINTING OFFICE 16—62761-3

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NOS. 50-269, 50-270 AND 50-287

DUKE POWER COMPANY

NOTICE OF FILING OF APPLICATION FOR CONSTRUCTION PERMIT AND FACILITY LICENSE

On November 28, 1966, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28201, filed an application for a construction permit and facility license to authorize construction and operation of a two-unit pressurized water nuclear reactor power plant at its Oconee Nuclear Station located in eastern Oconee County, approximately eight miles northeast of Seneca, South Carolina. A notice of receipt of this application was published in the Federal Register on December 20, 1966, 31 F.R. 16286.

Please take notice that Duke Power Company has filed an amendment to its application dated April 29, 1967, requesting authorization to construct and operate a third pressurized water nuclear reactor at the applicant's Oconee Nuclear Station described above. The third unit, identified as Unit 3, will be identical to Units 1 and 2. It will have a design capacity of approximately 874 megawatts electrical.

Copies of the original application and this amendment are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Dated at Bethesda, Maryland
this 15th day of May, 1967.

SUPPLEMENTAL FILE COPY

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

MAY 11 1967

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F. W. Karas
bcc: W. B. Cottrell, ORNL

IN REPLY REFER TO:
Docket Nos. 50-269
50-270
and 50-287

Duke Power Company
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Attention: Mr. W. S. Lee
Vice President

Gentlemen:

This is a request for supplemental information to your application for a construction permit and operating license for the Oconee Units 1, 2 and 3 to be located in Oconee County, South Carolina.

During a meeting on April 27 and 28, 1967, between representatives of your company and the regulatory staff, a number of technical areas were discussed and it was concluded that additional written information would be required to continue our review. In this regard you are requested to provide the information listed in the enclosure.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Requested Additional Information

| RPB #3-DRL | RPB #3-DRL | RT/DRL | RP/DRL | DRL |
|---|---|---|---|---|
| BGrimes | CGLong | SLevine | RSBoyd | PAMorris |
| 5-10-67 | 5-10-67 | 5-10-67 | 5-10-67 | 5-11-67 |

50-269
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-287

REQUESTED ADDITIONAL INFORMATION

dispatched w/5-11-67
elr.

## 11.0 General

11.1 As a result of recent discussions you have indicated that the turbine missile analysis will be reanalyzed for 180% turbine overspeed. Please include the following points in the analysis:

11.1.1 Provide missile area and energy absorption assumptions and give the energy partition and impact velocities.

11.1.2 Discuss the effect of the missile on the tendons and the number of tendons that could be damaged locally in the dome without endangering containment integrity.

11.1.3 Discuss the physical separation, redundancy and protection of vital shutdown components including protection of the control room.

11.2 In recent discussions with the staff the emergency power proposal has been further elaborated to include power separate from the grid which could be supplied during hydro outages or at other times when grid power and hydro power would not be available. An alternate water source was also outlined. Please provide documentation of these proposals. In addition, address yourself to the requirement for shutdown cooling when there is an equipment leak in the primary system during "blackout" conditions.

11.3 Our consultants note that the response spectrum used for the seismic analysis is less conservative than the scaled El Centro spectrum. Please modify the proposed spectrum.

11.4 Discuss the capability of the hydro plant equipment to operate during and after the maximum earthquake.

11.5 Provide the criterion for location of isolation valves with respect to the containment penetration and the strengthened piping in this area.

11.6 Please submit a statement indicating the ability of the anchoring pipe guides to limit forces transmitted to the penetration.

## 12.0 Site

12.1 Describe the foundation investigation for the dams which will assure that there are no zones of poor material in which "piping" could occur and which will assure that no strata of unsuitable material will be present in the unremoved overburden. Also discuss provisions to detect excessive leakage through the dams and remedial action that could be taken.

12.2 Provide information for the intake canal dike which verifies your statement that it is more conservatively designed than the major dams which have been analyzed from a seismic standpoint.

12.3 Will the future Jocassee Dam be designed for the maximum earthquake?

12.4 We believe that a peripheral tangential tornado design wind speed of at least 300 mph rather than 225 mph should be used in design basis for Class I structures. Also, a pressure differential of 3 psi developed over 5 seconds should be considered. If a lower value is to be justified it will be necessary to present data which indicate lower tornado wind speeds in mountainous versus flat terrain and to justify any assumed variation of wind speed with elevation.

12.5 Discuss the potential for reconcentration of liquid wastes in downstream industrial plants or public water supplies for normal and accidental discharges. Also account for all liquid wastes after a loss-of-coolant accident.

13.0 <u>Thermal Analysis</u>

13.1 Give the DNB ratios for the nominal and worst hot channels at 114% power for unit, wall and corner cells using the W-3 correlation with the non-uniform axial and unheated wall corrections. Provide enthalpy and quality at burnout conditions and the axial location of the calculated burnout. Provide the dimensions of the corner and side cells.

13.2 Please indicate which fuel conductivity model was used in the various calculations for fission gas release, center fuel temperature, average fuel temperature and transient analyses for accident and normal conditions.

14.0 <u>Instrumentation</u>

14.1 Will the circuits which remove the "low reactor coolant pressure trip" bypass be designed to protection system standards?

14.2 Discuss the portions of the rod drive control system which act to limit rod speeds to safe values and the inherent speed limitation of the equipment.

14.3 How are the set points on the power/flow instrumentation calibrated as rod positions change?

15.0 Core Cooling

15.1 Discuss the possibility of a water leg remaining in the steam generators for the spectrum of pipe breaks which could trap a steam bubble as the core was flooded and result in the safety injection water bypassing the core.

15.2 Outline the action which would be taken in case of a leak through the check valves in the core flooding system and the conditions under which reactor operation would be continued.

15.3 State the level to which one core flooding tank will fill the core and provide an analysis of the degraded system case in which one core flooding tank and minimum injection flow only is available.

15.4 Provide an analysis to show that the reactor vessel and thermal shield will accommodate without failure the transient loading, close to the end of its design life, due to safety injection of cold water up to the level of the main coolant nozzles. Assume the maximum deluge rate starting from an empty vessel. State the initial vessel temperature used and the assumed failure criterion. Also, estimate the limit of initial vessel temperature which could cause failure upon injection and relate this temperature to a delayed injection time.

16.0 Accident Analysis

16.1 Consider the long term effects of continuing the feedwater supply to a steam generator after a steam line break.

16.2 Provide a study of the core reactivity effects after a steam line break in which there have been generator tube ruptures. What additional fuel failure could result from the blowdown and potential secondary criticality?

16.3 Give the sequence of events after tube breaks have occurred in the steam generator and state the signals which the operator will have available so that the proper steam generator can be isolated.

16.4 Provide an analysis of a pump seal failure in the safeguards equipment after a loss-of-coolant accident. Give the fission product source and the iodine partition factor assumed. The maximum containment water temperature should be used in the analysis. Discuss the route of fission product release to the environment.

16.5 Discuss the heatup of the penetration room filter after inadvertent closure of the outlet filter valve after fission products have been deposited on the filter.

17.0 Primary System

17.1 Provide the justification for the Class C classification of the letdown cooler.

17.2 Describe the design of the bypass valves on the secondary system which act as both control and safety valves.

17.3 Discuss the physical availability of the external surfaces of the reactor vessel if inspection should be found necessary during the plant life.

17.4 Estimate the sensitivity of the primary system leakage detection methods to be used and state the criterion for corrective action.

18.0 Materials and Construction

18.1 Please provide the standards for acceptance of mechanical splices of reinforcing steel and the extent to which your quality control program assures that the standards are being met.

18.2 Provide information on the welding of structural steel reinforcing bars. Indicate the type, size and locations of reinforcing bars that are to be butt, lap or tack welded. Indicate the quality control measures to be employed for the welding.

19.0 Control Rod Drives

19.1 Discuss the design of the drive housings with respect to forces imposed in the case of a thimble rupture and the need for a holddown mechanism to prevent rod ejection.

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WASHINGTON, D.C. 20545

MAR 23 1967

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W. B. Cottrell, ORNL

IN REPLY REFER TO:
Docket No. 50-269/270

Duke Power Company
422 South Church Street
P. O. Box 2178
Charlotte, N. C. 28201

Attention: Mr. W. L. Lee
Vice President

Gentlemen:

This is a request for supplemental information to your application for a construction permit and operating license for the Oconee Units 1 and 2 to be located in Oconee County, South Carolina. During a meeting on Fegruary 14 and 15, 1967, between representatives of your company and the regulatory staff a number of technical areas were discussed and it was concluded that additional written information would be required to continue our review. In this regard you are requested to provide the information listed in the enclosure.

We understand that changes will be made in the core cooling systems of the proposed reactors in response to increased emphasis in this area as reflected in recent licensing actions. Appropriate submittals reflecting these changes should be made so that we can continue our review of these systems.

In order to facilitate our technical review, we urge that you provide full and complete answers to the attached questions so that further questions covering the same material will not be required. We will be available to amplify the meaning of any of the questions.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
As stated above

| DRL | DRL | DRL | DRL |
|---|---|---|---|
| FWKaras | CGLong | RSBoyd | PAMorris |
| 3-21-67 | 3-20-67 | 3-21-67 | 3-22-67 |

## 1.0 GENERAL

1.1 Supply specific ASME Code vessel classifications for all components, including heat exchangers, in the systems which handle reactor coolant.

1.2 Your calculations indicate that xenon oscillations might occur in this core. Please describe the method by which xenon oscillations would be controlled should they occur.

1.3 Discuss the use of aluminum components in the primary system from the standpoint of experience with these components in service and state the criteria to which these components will be designed and fabricated including corrosion and fit-up considerations.

1.4 Discuss the inspectability of the primary system and the reactor vessel during their service life. Will representative longitudinal and radial welds, all nozzles and dissimilar metal welds be inspectable? Supply a tentative schedule for inspection of the primary system and reactor vessel during their service life.

1.5 Discuss the containment tests which will be performed initially and over the service life of the containment which will assure that at least the specified 50% of containment leakage will exit through filters via the penetration room.

1.6 Discuss the frequency and type of maintenance likely to be performed on the hydro plants. What is the time required to restore the hydro plants to operation for these various types of maintenance? This should include maintenance that might be performed on the penstocks.

1.7 Please provide a discussion of how the larger water gap and thinner thermal shield in this proposal affect, (as compared to currently licensed plants), (1) the neutron irradiation and (2) the thermal stresses in the pressure vessel wall.

1.8 Discuss the limitations on frictional contact between control rods and guide tubes with repsect to the life of the rods and give the inspection criterion.

## 2.0 SITE

2.1 Provide a drawing indicating the location of all areas within the site boundary which will not be owned by Duke Power and those that will be leased or otherwise used for purposes other than power generation. State the control that will be exercised by Duke Power over these areas.

2.2 Estimate the expected transient population around the future Lake Keowee as a result of summer cottages, boat access and any commercial activities.

2.3 Locate the water intake for the town of Seneca with reference to the reactor and also indicate the distance to the proposed intake point on Lake Hartwell for the city of Anderson and the towns of Clemson and Pendleton. Provide stream flows, travel times, and estimated dilution to these intakes. Estimate the length of time that these muncipalities could suspend use of these intakes.

2.4 Discuss the reasons for discharging liquid radioactive waste into the tailrace of the hydro plants rather than into Lake Keowee. In this regard provide the following information: (a) What is the effective transit time and dilution factor from the plant discharge canal through the lake to the intake canal and how would these be affected by various flow conditions in the rivers? (b) What are the corresponding factors between the discharge canal and the tailrace of the hydro plants? (c) How will the flow through the hydro station be affected by low flow in the rivers feeding the lake?

2.5 Please provide the following information with respect to site meteorology:

2.5.1 What is the average wind speed for Type F stability conditions, including the calms? Considering that this site is an area having a dilution climate which is below average, how can the use of a higher than usual wind speed for site evaluation be justified? Also, why is the persistence of inversion conditions less than 24 hours? Similar sites have shown much longer persistence of inversions. Re-examine the assumed 20% wind direction persistence in 24 hours in light of Weather Bureau data indicating the persistence at most sites is approximately 15 hours.

2.5.2 Please re-examine the assumed 30-day meteorology as compared to that presented for the yearly average.

2.6 Describe the scope of the preoperation and postoperation environmental monitoring program including the type and frequency of sample collection.

2.7 It is our consultants' tentative opinion that the maximum hypothetical earthquake should be about 0.10g for those Class I structures which are founded on bedrock and 0.15g for any Class I structures located on overburden. Please provide your structural design criteria for the maximum hypothetical earthquake.

## 3.0 THERMAL ANALYSIS

3.1 Please provide a numerical breakdown of the following factors for the hot channel of this reactor:

3.1.1 Integrated power effects due to rod pitch, bowing, pellet diameter and enrichment variations.

3.1.2 Flow distribution effects due to (a) the inlet plenum, (b) redistribution in adjacent channels of dissimilar coolant conditions, (c) physical mixing of coolant between channels.

Discuss the effect of variation of the above factors on exit quality in the hot channel.

3.2 Please discuss the effect of variation of the mass flow rate (G) on the DNB ratio of the hot side cell and the hot corner cell.

3.3 Discuss the degree of confidence which you have in the flow instability analysis. What margin above slug flow exists in the corner and side flow channels? What are the consequences of locally operating in the slug flow regime (e.g. due to unexpectedly low mass flow rates in the corner channel?)

3.4 Provide a description of the methods used to calculate core void fractions.

3.5 What is the effect on the calculated fuel rod internal pressure due to fission gas release if the voids within the fuel are not utilized in the calculations? (It appears that these voids have been used twice; once for fuel expansion and again for fission gas release).

3.6 Please provide information on current burnout experimental studies with multirod geometries and non-uniform heat generation for the configuration and service conditions of the proposed reactor.

3.7 Provide the basis for the conductivity curve used and describe the calculational procedures and assumptions used to calculate the center line fuel temperature.

## 4.0 INSTRUMENTATION

4.1 Please discuss the reliability of those power generation sources and associated circuitry which will provide emergency power in the event of an accident and simultaneous loss of the external grid. The discussion should include considerations of redundancy and independence of the sources, and the degree of immunity to "single failures" of the total emergency power system (including load-shedding subsystems, d.c. sources feeding breaker control circuits, undervoltage circuits, etc).

4.2 Please re-submit a revised version of Figure 7-2, incorporating your present intentions relating to the design of the nuclear instrumentation and protection systems.

4.3 Please describe the power/flow scram channels.

4.4 Please submit a schematic diagram (similar to the format of Fig. 7-2) showing your proposed three-wire d.c. system. Please include a failure analysis which shows that no single fault within this system (e.g., short, ground, failed breaker, faulted charger. . ., etc.) can preclude the actuation of protection and safeguards devices under accident conditions.

4.5 Does the design of your protection system conflict in any way with the proposed IEEE Standard for Nuclear Power Plant Protection Systems? If so, please state reasons justifying your position.

4.6 Please discuss your criteria relating to the qualification testing of instrumentation and associated circuits to ensure their ability to survive an accident environment.

4.7 Please discuss in further detail the development program of the nutating rod drives, including experimental data which will confirm that the drives will meet design requirements.

4.8 Please list those portions of the containment isolation system which are not fail-safe upon loss of voltage. Provide justification for your design basis.

4.9 Please perform similar analyses to that in Section 14.1.2.3 assuming a reactivity insertion rate equivalent to simultaneous all-rod withdrawal, commencing from various initial power levels sufficient to show that, in no case, does fuel damage occur.

5.0 CORE COOLING

5.1 Provide a plot of the coolant flow within the reactor as a function of time after hot leg and cold leg major coolant line breaks. How does the injection location of the deluge system affect this flow transient?

5.2 Discuss the mechanism of clad failure during heatup and quenching. Could the rods swell and block coolant channels? Could fuel integrity be lost as a result of rapid quenching? How many adjacent channels could be blocked and local melting still be prevented by the core flooding system?

5.3 What temperature transient does a control rod experience during the core heatup. (Consider heat transfer from bowed fuel rods as well as radiant heat). What eutectics might be formed between the dissimilar core materials and could these be formed before either component was molten? Can the core remain subcritical after flooding without control rods in the core?

5.4 Discuss design of the vessel internals to withstand blowdown forces from a hot leg or cold leg break. In particular, provide information on the method used to calculate core pressure drops during the subcooled blowdown phase and compare the results to experimental data such as the LOFT tests. How are assemblies held in place during the calculated transients?

5.5 Justify the use of the steam-limited zirconium water reaction assumption considering that the core deluge system may be partially effective in providing water to the core.

5.6 We understand that the engineered safety features are being redesigned and that stored energy flooding tanks will be provided. Please include the following points in your description and analysis of these systems:

5.6.1 Justify the capacity of the systems including single failure considerations.

5.6.2 Provide the analysis by which injection above rather than below the core was chosen. What experimental information substantiates the ability to flood the hot core from the upper plenum?

5.6.3 Provide the analysis by which the design pressure of the core flooding tanks was chosen from a performance viewpoint, including the variation of significant parameters.

6.0 <u>ACCIDENT ANALYSIS</u>

6.1 Please provide additional information concerning the effect of the positive moderator temperature coefficient on the reactivity insertion and fuel heat-up during a loss of coolant accident. This should include a thorough discussion of the work done to date and the major areas (if any) which remain to be resolved. Include curves illustrating the coolant condition (e.g. flow and density) as a function of time. Also provide plots of the various reactivity components, power, and integrated energy as a function of time for the various break sizes and break locations studied. Provide information in the following areas in conjunction with your consideration of the above problem:

6.1.1 Discuss your analysis of heat transfer during the blowdown including experimental information which might support the heat transfer coefficients assumed. Discuss the effect which a significantly larger or smaller heat transfer coefficient might have on the above analysis.

6.1.2 Show that the positive moderator temperature coefficient could be eliminated if this is found necessary (e.g., by fixed shims). Is there a practical limit to the size of the positive coefficient that could be negated in this manner? Provide the bases for determining the maximum acceptable positive moderator temperature coefficient from an operational viewpoint.

6.1.3 Discuss the method used to calculate the maximum reactivity insertion including (1) the variation of the spatial density distribution as a function of time, and (2) the nuclear calculations required to estimate the effect of this variation on the reactivity of the system.

6.1.4 Discuss the method used to calculate the energy generated in the reactivity transient.

6.1.5 Discuss the accuracy you believe may be assigned to (c) and (d) above, including experimental corroboration of the accuracy of the calculations.

6.2 Please provide the following information concerning the rod ejection accident:

6.2.1 Justify the assumed rod worth values of 0.2% reactivity from full power and 0.5% reactivity from zero power. On what basis was a rod ejection accident from hot standby not considered?

6.2.2 Discuss the thermal-hydraulic assumptions used in the transient calculations.

6.2.3 Provide plots of the various reactivity components, power and integrated energy as a function of time.

6.2.4 Justify the use of the point kinetics model in this analysis. We understand that some comparisons of the point kinetic results with explicit space-time calculations (WIGL) has been made. A presentation and discussion of these results would be useful.

6.2.5 Discuss the margin which exists between the calculated transient and those transients which could (1) cause major damage to vessel internals and (2) cause primary system rupture.

6.3 Provide a plot of the temperature of the primary system water after a steam line break as a function of time and justify the 60°F cooldown figure used in the analysis. Provide a plot of the power and reactivity as a function of time for this condition.

6.4 Consider the case of a steam line break accident in which feedwater continues to be fed to the steam generator. What is the temperature response of the steam generator shell and what stresses are imposed on the tubes?

6.5 Provide an analysis of the effects of steam generator tube ruptures coincident with (precipitated by) a steam line break with respect to (1) reactivity effects on the primary system and (2) release of fission products to the environment, as a function of number of tubes ruptured.

6.6 Discuss the need for isolation valves on the secondary system particularly with reference to leakage from the primary system after a steam line break. Could safety valves on the secondary side be run through separate penetrations and an isolation valve be located inside containment?

6.7 Provide the method used to calculate the reactor coolant activity from 1% failed fuel. Include assumptions on the release rate of noble gases and the cleanup rate. Provide a definition of "equivalent curies of iodine-131."

6.8 Justify the use of a $10^4$ reduction factor for fission products in the event of a steam generator tube rupture and release through secondary system safety valves.

6.9 Justify the assumption that hydrogen evolved in a metal-water reaction would be above the ignition temperature in all cases. Particularly consider partial effectiveness of the cooling systems. What effect would delayed burning of hyrdogen have on containment design margins?

6.10 What mixing depth and deposition areas in the lake are assumed in calculating iodine intake from rainout after an accident during the first 5 or 10 days? For how long is the intake of water assumed in the dose calculation?

## 7.0 CONTAINMENT COOLING

7.1 Describe the capability to flood the reactor cavity after an accident. How does the volume required to flood the cavity compare with the primary system volume?

7.2 Discuss the requirements for cooling the water recirculated from the containment sump.

7.3 Discuss the NPSH requirements of the recirculating pumps with respect to the minimum beight of water required in the sump. To what water volume inside containment does this correspond? Discuss the sump location considerations, intake design details and the criterion for redundancy in sump outlet capacity.

7.4 Provide an analysis to show the amount of time available to isolate the service water in case of a break in the containment cooler tubular heat exchanger (resulting in the injection of unborated water).

7.5 Provide an analysis showing the minimum containment safeguards required to handle the design basis accident and illustrate the margin, in terms of metal water reaction, provided by the proposed system capacities.

7.6 Discuss the separability and location of the recirculation system pumps to avoid flooding of the pumps in case of a major system leak.

## 8.0 STRUCTURAL DESIGN

8.1 Provide the loading combination considering the design basis accident-maximum earthquake combination, and the design basis accident-maximum wind combination.

8.2 Please provide a complete, detailed description of how the design wind and tornado wind are translated into static loadings on the structure. Estimate the ultimate capability of the containment and other structures to withstand tornado differential pressure and wind loadings. Justify use of 225 mph as the tornado design wind load and the differential pressure associated with this tornado.

8.3 Clarify the design approach in the PSAR allowing limited plastic yielding in a working stress design.

8.4 Provide the following information relative to seismic design in light of question II.7.

8.4.1 The damping factors to be used in the various loading combinations that include a seismic contribution.

8.4.2 A statement of the intent of the designer with regard to combination of maximum vertical and horizontal earthquake components in conjunction with the other applicable loadings.

8.4.3 The mathematical model to be used in the seismic design analysis.

8.4.4 The stiffness factors to be used in the design analysis and a detailed basis for the selections.

8.4.5 The design criteria and procedures for design of the piping systems and supports for Class I components for seismic loadings in combination with the other applicable loads.

8.5 With regard to earthquake response spectra provide the following:

8.5.1 A response spectrum for the maximum earthquake.

8.5.2 The basis for the shape of the proposed response spectra.

8.5.3 Identify, explain and justify the scaling of the response spectra with respect to displacement, velocity, acceleration and frequency on the plots presented.

8.6 Provide analytical studies to support the safety of the dam against failure under earthquake loading. If such a failure were to occur, what effect would it have on the capability for safe plant shutdown specifically in the areas of shutdown cooling and emergency power.

8.7 Provide the following information with regard to shear design:

8.7.1 An analysis of recent test experience (such as the University of Washington data) under combined shear and tensile loading and an evaluation of the extent to which this experience influences the radial shear design criterion.

8.7.2 A revised shear criterion incorporating clarified principal stress limits for loadings both including and excluding thermal effects.

8.7.3 An analysis of recent test experience on strength in shear of sections with large sized bars and low longitudinal steel percentages and an evaluation of how such experience influences the shear design criterion.

8.8 Provide the following information with regard to penetration design:

8.8.1 A concise criterion with regard to prevention of failures at leakage barriers due to all conceivable loading conditions during accidents and typical details illustrating how the criterion will be implemented.

8.8.2 The basis for providing reinforcing in the containment concrete wall around penetrations and typical details indicating implementation of this criterion.

8.9 Provide justification for the use of lapped splices in large sized reinforcement bars under biaxial tensile loading.

9.0 <u>MATERIALS AND CONSTRUCTION</u>

9.1 With regard to construction practice indicate:

9.1.1 The extent to which ACI 301 will be adhered to in construction.

9.1.2 The construction procedures to be used to achieve bonding between lifts.

9.1.3 The pattern of construction joints that will be used in the structure and the degree to which joint staggering will be accomplished. Where joint stagger is not accomplished justify in detail its elimination.

9.1.4 The extent to which liner plate radiography will be accomplished.

9.2 Provide details on the prestressing system to be used.

9.3 Provide details on what user testing of liner plate, reinforcing steel, corrosion inhibitors, cement, prestressing wire (or strand) and anchorage hardware will be conducted.

9.4 Define the amount of concrete testing to include more specifics on slump testing, justification of the amount of strength testing, and how this testing will be factored into an overall statistical sampling program.

9.5 Define in more detail the program for construction inspection by identifying the organization, responsibilities, authority and independence of the quality control group and by indicating how supervision and review by the design group will be achieved.

9.6 With regard to design details on load transfer through the leakage barrier provide:

9.6.1 A typical crane support bracket detail.

9.6.2 A typical detail on equipment support load transfer through the base liner.

9.7 With regard to corrosion protection provide the cover provision for reinforcing steel for the dome, base slab and cylinder. Also justify the selected cover requirements on the basis of code practice and field experience.

10.0 CONTAINMENT INSPECTION AND INSERVICE SURVEILLANCE

10.1 Provide an enlarged detail of instrumentation planned to monitor the equipment opening during the proof test, indicate the purpose of the instrumentation provided, and state the interpretation that will be placed on these measurements.

10.2 Provide a detailed comparison of the stresses in the structure and liner under combined accident conditions and under the proof test loading.

10.3 Provide a detailed prediction of strains around the equipment hatch, in the cylinder-dome region, at the base-cylinder junction and in the liner. Also provide detailed predictions of overall shell and dome growth. Provide an estimate (plus and minus) of the prediction accuracy, a description of the instrumentation that will monitor structural performance at the prediction location during the proof test, and an estimate of instrumentation accuracy.

10.4 List the quality control records that will be in the possession of Duke Power Company after construction of the plant has been completed and indicate the length of time these records will be maintained. Include both reactor system and containment records.

Distribution:
Doc. Room
Formal
Supplemental
C. Henderson
E. G. Case
H. Shapar
L. Kornblith (2)
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bcc: H. J. McAlduff, ORO
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J. A. Harris, PI
E. Tremmel, IP
R. Leith, OC
W. B. Cottrell, ORNL, encl. only
DRL Reading
R&PRSB Reading
Orig: HSteele

Docket Nos. 50-269 &
50-270

DEC 1 2 1966

Duke Power Company
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Attention: Mr. W. S. Lee
Vice President, Engineering

Gentlemen:

Enclosed is a copy of a notice relating to your application dated November 28, 1966, for authorization to construct and operate the Oconee Nuclear Station Units 1 and 2. The notice has been transmitted to the Office of the Federal Register for publication.

Docket numbers 50-269 and 50-270 have been assigned to this application. All future related correspondence should bear these docket numbers.

Sincerely yours,

Original signed by
E. G. Case

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
As stated above

| OFFICE ▶ | DRL | OGC | DRL | DRL | DRL |
|---|---|---|---|---|---|
| SURNAME ▶ | HSteele/dj | | RSBoyd | EGCase | PAMorris |
| DATE ▶ | 12/7/66 | 12/9/66 | 12/9/66 | 12/12/66 | 12/12/66 |

Form AEC-318 (Rev. 9-53)
U. S. GOVERNMENT PRINTING OFFICE 16—62761-3

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NOS. 50-269 AND 50-270

DUKE POWER COMPANY

NOTICE OF RECEIPT OF APPLICATION FOR CONSTRUCTION PERMIT AND FACILITY LICENSE

Please take notice that Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28201, pursuant to Section 104(b) of the Atomic Energy Act of 1954, as amended, has filed an application, dated November 28, 1966, for authorization to construct and operate a two-unit nuclear power plant at its Oconee Nuclear Station located in eastern Oconee County, approximately eight miles northeast of Seneca, South Carolina.

The proposed nuclear power plant will consist of two pressurized water reactors, designated by the applicant as the Oconee Nuclear Station Units 1 and 2, each of which is designed for initial operation at approximately 2452 thermal megawatts with a net electrical output of approximately 839 megawatts.

A copy of the application is available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
E. G. Case

Peter A. Morris, Director
Division of Reactor Licensing

Dated at Bethesda, Maryland
this 12th day of December, 1966.