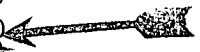# UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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 measure-
ments 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 approx-
imately yearly intervals, the overall Oconee Station operating organi-
zation 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 con-
sider a smaller shift crew size if it can be shown that fewer men can
perform all normal and emergency functions in accordance with estab-
lished 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 addi-
tional 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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2/10/70	2/10/70	2/10/70	2/10/70	2/11/70	2/13/70

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 0.5×10^{-4} $\Delta k/k^\circ F$, 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 1×10^6 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 overpower 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 intensities 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   Other Class I Systems and Components

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 pene-
tration 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).

23

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.