docket

AUG 2 0 1973

Docket No. 50-269

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President,
Production & Transmission
422 South Church Street
P. O. Box 2178
Charlotte, N. C. 28201

Gentlemen:

The following is a summary of recent items discussed between Duke Power Company and the AEC staff regarding the Oconee Nuclear Station. It is our understanding that you are in basic agreement with these items.

1. Startup Physics Tests for Oconee Unit 1

Technical Specifications 3.11 and 6.6.3.1, "Maximum Power Restrictions" and "Authorization of Changes, Test, and Experiments," respectively, set the requirements for a summary report of plant startup and power escalation test programs and evaluations prior to removal of the current 95% of full rated power restriction (TS-3.11) on Oconee Unit 1.

Included in this report will be a comparison of B&W's PDQ-7 three-dimensional power distribution predictions with the measured power distributions for various startup tests. This information will enable the staff to verify the power peaking predictions used in the safety analysis of reactors of the Oconee design. In addition to other appropriate startup tests, these comparisons will include power maneuvering transients similar to the 100-30-100% design transient.

2. Status Reports of Expected Power Distributions

In addition to the annual report required in the Technical Specifications, a comparison of the power distributions measured during the periodic power maps with B&W's PDQ predictions is required during and shortly after the first fuel cycle of Oconee Unit 1. These comparisons will be provided in a status report submitted in a timely manner after each major repatching of control rods during the first fuel cycle and shortly after the first refueling.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

3. <u>Alarms on Axial Imbalance and Quadrant Tilt</u>

Since it is necessary to maintain the reactor within specified axial imbalance and quadrant tilt limits in order to restrict the peak power to below safety limits, and this will be done by administrative procedures, alarms will be provided to alert the operator prior to exceeding these limits. When these alarms are not available, the pertinent parameters will be logged every two hours.

4. <u>Spent Fuel Storage Area Vent Filters</u>

In order to reduce off-site radiation doses to practicable limits, iodine filters will be installed in the spent fuel storage area exhaust system. These filters will be installed and operable prior to returning to power operation with Oconee Unit 2 following the first refueling of Unit 2.

Sincerely,

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, N. C. 28201

~~XXXXXXXXXX~~
<u>DISTRIBUTION</u>
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PWR Branch Chiefs
RWKlecker
IAPeltier (2)
EIGoulbourne (2)
ACRS (16)

Get TR concurrence (Hendrie, Stello or Ross) before release

OFFICE ▶	PWR-4	L:C/PWR-4	L:AD/PWRs			
SURNAME ▶	IAPeltier:kmf	ASchwencer	RCDeYoung	Per for O.F. Ross		
DATE ▶	8/17/73	8/17/73	8/17/73	8/17/73		

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

Docket No. 50-269

JUL 2 6 1973

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President
Production & Transmission
422 South Church Street
P. O. Box 2178
Charlotte, N. C. 28201

Gentlemen:

We have completed our review of your report "Analysis of Effects Resulting From Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2 & 3" dated April 25, 1973 and the supplement to this report dated June 22, 1973. We find the report acceptable with regard to your analysis of postulated ruptures of high-energy piping external to the reactor building and your proposed station modifications for taking corrective action where warranted.

This letter is to confirm our understanding that all Unit 1 modifications will be complete by November 1, 1973 and that those interim measures described in Section 4.1 of the above report will remain in effect until the station modifications are complete.

R C DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28201

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IAPeltier (2)
EIGoulbourne

JUL 26 1973

Docket No. 50-269

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President
Production & Transmission
422 South Church Street
P. O. Box 2178
Charlotte, N. C. 28201

Gentlemen:

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R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28201

OFFICE ▶	PWR-4	L	L:C/PWR-4	L:AD/PWRs	
SURNAME ▶	IAPeltier:kmf	RWKlecker	ASchwencer	RCDeYoung	
DATE ▶	7/25/73	7/25/73	7/25/73	7/25/73	

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JUL 11 1973

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

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President,
Production & Transmission
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

We have reviewed your April 27, 1973, submittal of the Oconee Nuclear Station Operational Quality Assurance Program which we requested for periodic review by letter dated March 27, 1973. Based on this review and discussion with your representatives on May 29, 1973, and June 12, 1973, it is our understanding that your company is taking measures to strengthen its Quality Assurance Program. Please provide information describing how these changes are being applied to the Oconee Nuclear Station QA Program for Units 1, 2 and 3. Also we require additional information as listed in the enclosure to complete our review. We would appreciate your response to this letter prior to July 31, 1973.

Sincerely,

Original Signed by
Albert Schwencer

A. Schwencer, Chief
Pressurized Water Reactors Br. No. 4
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, N. C. 28201

OFFICE ▶	PWR-4	L:C/PWR-4				
SURNAME ▶	IAPeltier:kmf	ASchwencer				
DATE ▶	7/ /73	7/ 11 /73				

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

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

JUL 11 1973

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

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President,
Production & Transmission
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

We have reviewed your April 27, 1973, submittal of the Oconee Nuclear Station Operational Quality Assurance Program which we requested for periodic review by letter dated March 27, 1973. Based on this review and discussion with your representatives on May 29, 1973, and June 12, 1973, it is our understanding that your company is taking measures to strengthen its Quality Assurance Program. Please provide information describing how these changes are being applied to the Oconee Nuclear Station QA Program for Units 1, 2 and 3. Also we require additional information as listed in the enclosure to complete our review. We would appreciate your response to this letter prior to July 31, 1973.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Pressurized Water Reactors Br. No. 4
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, N. C. 28201

REQUEST FOR ADDITIONAL QUALITY ASSURANCE INFORMATION

OCONEE NUCLEAR STATION

The following additional information is required:

1. Clear description of the specific responsibilities, authorities, and day-to-day duties of those responsible for formulating, establishing, approving, and implementing the QA policies, documented procedures, QA Manuals, and instructions and changes thereto relative to operation, maintenance, repair, modification, and refueling.
2. Provide a listing or tabulation of the titles of the more typical QA procedures that respond to each of the 18 QA requirements of Appendix B to 10 CFR 50. Provide a brief abstract denoting the scope and purpose of each such procedure.
3. Identify those positions or groups responsible for reviewing and approving the QA programs for vendors and contractors relative to the activities of maintenance, modification, repair and refueling.
4. Describe the responsibilities and authorities of the Station Review Committee and Nuclear Safety Review Committee relative to the implementation of the recommendations of AEC Regulatory Guide 1.33. Describe the interaction or role, if any, of these Committees relative to QA staff and its activities.
5. Provide a more indepth description of DPC's inspection and audit program including specific delineation of the onsite and offsite activities to be audited, and provisions of assuring independent acceptance inspection. Provide a specific list of the duties of

those responsible for independent review, inspection and internal and external audit.

6. We require a clear commitment with respect to implementation of AEC Regulatory Guide 1.33. For any items of a QA program that do not satisfy Regulatory Guide 1.33 describe and justify the proposed alternatives.

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ASchwencer
JGallo, OGE

JUL 1 0 1973

Docket No. 50-269

Duke Power Company
ATTN: Mr. Austin C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

Five (5) copies of the Safety Evaluation Supplement No. 3, prepared by the Directorate of Licensing relating to your Oconee Nuclear Station, Unit 1, are enclosed for your information.

Sincerely,

Original Signed by
Albert Schwencer

A. Schwencer, Chief
Pressurized Water Reactors
Branch No. 4
Directorate of Licensing

Enclosure:
Safety Evaluation Supplement No. 3

cc: William L. Porter, Esquire
Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 28201

Troy B. Conner, Esquire
Conner and Knotts
1747 Pennsylvania Avenue, NW
Suite 1050
Washington, D. C. 20006

Mr. J. Bonner Manly, Director
State Development Board
Hampton Office Building
Columbia, South Carolina 29202

Honorable Reese A. Hubbard
County Supervisor of Oconee County
Walhalla, South Carolina 29621

OFFICE	PWR-4	OGC	L:C/PWR-4			
SURNAME	IAPeltier:kmf	JGallo	ASchwencer			
DATE	7/6/73	7/6/73	7/10/73			

UNITED STATES
ATOMIC ENERGY COMMISSION
DIRECTORATE OF REGULATORY OPERATIONS
REGION II - SUITE 818
230 PEACHTREE STREET, NORTHWEST
ATLANTA, GEORGIA 30303

TELEPHONE: (404) 526-4503

MAY 9 1973

In Reply Refer To:
RO:II:RFW
50-269/73-3

Duke Power Company
Attn: Mr. A. C. Thies
Senior Vice President
Production and Transmission
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Gentlemen:

This is in reply to your letter of April 20, 1973, responding to our letter of March 30, 1973.

Concerning item 1, adherence to Technical Specification 6.1.1.6, we have reviewed the changes (effective April 12, 1973) made in the assignment of reactor operations personnel engaged in work on Units 1 and 2. With the assignment of appropriately licensed individuals to Unit 1 and the assignment of other appropriately qualified individuals to supervise the preoperational testing of Unit 2, we have no further questions at this time.

Concerning item 2, our inspector has reviewed the minutes of the Station Review Committee in which test procedure TP-800/5, "Reactivity Coefficients at Power," was reviewed. We have no further questions concerning this item.

Your corrective actions for item 4, failure to have a procedure for draining oil from reactor coolant pumps, will be reviewed during an inspection following the receipt of your oil fire report.

A copy of the report of this inspection is enclosed. In accordance with Section 2.790 of the AEC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the AEC's Public Document Room. If this report contains any information that you believe to be proprietary, it is necessary that you make a written application within 20 days to this office to withhold such information from public disclosure. Any such application

must include a full statement of the reasons on the basis of which it is claimed that the information is proprietary, and should be prepared so that proprietary information identified in the application is contained in a separate part of the document. If the report contains no information which you believe to be proprietary, please inform us in writing within 20 days to this effect. In such case, the report, as enclosed with this letter, will be placed in the Public Document Room.

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Very truly yours,

Norman C. Moseley

Norman C. Moseley
Director

Enclosure:
Inspection Report No.
50-269/73-3

Ltr to Duke Power Company fm N. C. Moseley
dtd MAY 9 1973
Ltr to N. C. Moseley fm A. C. Thies, DPC,
dtd 4/20/73

cc w/o report:
J. G. Keppler, RO
J. B. Henderson, RO
RO:HQ (4)
Directorate of Licensing (4)
DR Central Files
PDR
Local PDR
NSIC
DTIE, OR
State

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. G. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2178

April 20, 1973

Mr. Norman C. Moseley
Directorate of Regulatory Operations
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: RO:II:RFW
50-269/73-3

Dear Mr. Moseley:

The following is submitted in response to Items 1, 2, and 4 of the enclosure to your letter dated March 30, 1973.

1. Alleged violation of Technical Specification 6.1.1.6

Technical Specification 6.1.1.6 requires that the minimum shift staffing for Unit 1 consist of the following membership: One senior reactor operator, two reactor operators, and two non-licensed operators. The technical specification also permits the minimum operating staff to assist in the prelicensing activity of Units 2 and 3 to the extent that it does not affecttheir full availability for Unit 1 operation.

One member of the minimum shift requirement who has a reactor operator license has been assigned responsibility for aiding in the startup of Unit 2. This individual's primary responsibility is to be a backup control operator for Unit 1. A majority of his time is spent in the Unit 1 control room and he is fully cognizant of Unit 1 activity. His Unit 2 responsibilities in no way impair his ready availability for Unit 1. We believe that this utilization of a licensed relief reactor operator as a backup to the licensed control operator meets the intent of both our technical specifications and AEC guidelines.

In order to confirm our understanding of the AEC's position on shift staffing, Mr. Dick VanNeil, AEC/DOL, was contacted by telephone. Attached is a February 21, 1973 memorandum for file which documents that conversation with Mr. VanNeil. We believe that memorandum confirms our understanding of the technical specification and the AEC's guidelines.

In order to assure there is no future misunderstanding on this issue, a supervisor has been appointed to direct the startup activities of Unit 2. Because the relief reactor operator on Unit 1 has valuable experience in startup activities, the Unit 2 supervisor will consult with him for guidance on Unit 2. The Unit 1 reactor operator's primary assignment will be to Unit 1 and he will be readily available to assist the control operator on Unit 1 whenever necessary. Howeve_, it is expected that some of his time can be devoted to consultation on Unit 2 startup.

2. Violation of Technical Specification 6.1.2.1

Through an oversight, test procedure TP-800/5, "Reactivity Coefficients at Power," was not received by the Station Review Committee (SRC) for its review prior to approval. The proper review of several hundred procedures has been performed, with checks by several individuals for completeness of the review process. This is the first identifiable failure of this process. This matter has been discussed with those responsible for assuring completeness of reviews. The test procedure has been reviewed by the SRC prior to its use.

4. Failure to have a written procedure for draining oil from reactor coolant pump (RCP)

The piping system for draining oil from the RCP had been recently installed, and a procedure had been provided for aligning the valves for operation. A procedure for draining the oil had not been written as there are numerous bearings in the station which are drained at periodic intervals to replace the oil. It has not been our practice to provide written procedures for these routine maintenance activities. In this instance, a written procedure may have been of value had we recognized the potential for backing oil up through the lower bearing overflow in the extremely unlikely event that both bottom drain paths were closed off while an upper bearing was being drained.

The oil drain piping on the RCP's has been revised to remove the potential for a similar occurrence, and a written procedure for draining oil from the RCP's has been provided. A meeting of the station supervisory personnel has been held and emphasis placed upon the necessity for written procedures or checklists, which have been properly reviewed.

Very truly yours,

A C Thies

A. C. Thies

ACT:vr

Memorandum for File February 21, 1973

Subject: Oconee Nuclear Station
Telecon with AEC Concerning Shift Staffing

On Friday, February 16, 1973, J. E. Smith and K. S. Canady called Dick VanNeil of the AEC to discuss the responsibilities of the shift staffing as defined in Unit 1 technical specification 6.1.1.6 (Al Schwencer had been previously contacted and suggested that we call Dick VanNeil directly and not involve Projects).

Ed Smith requested interpretation of the use of the assistant control operator. It had been our previous understanding that the reason for adding a third licensed operator was to provide relief for the control operator should he need it. Consequently, Ed Smith was proposing that even though the assistant control operators' duty would primarily be to Unit 1 that he be allowed to spend a significant portion of his time when not needed on Unit 1 to aid in the startup and checkout of Unit 2. This problem is specifically acute on one shift where the shift supervisor is licensed as an RO only and the assistant shift supervisor is an SRO. In this case, the assistant shift supervisor is responsible for Unit 1 and the shift supervisor is being used as the relief control operator for Unit 1 and aiding in the checkout of Unit 2.

Dick VanNeil informed us that although the primary responsibility of the relief control operator is backup for the control operator, the AEC expects that the relief control operator would spend approximately 75 percent of his time in the Unit 1 control room. He agreed, however, to check this out with his management and advise us.

On Tuesday, February 20, 1973, Dick VanNeil called to inform us of the following. The utilization of manpower for Oconee is essentially up to us as long as the technical specification requirements are fulfilled. Dick VanNeil made the following points:

1. The man should be assigned to Unit 1
2. He must be familiar with Unit 1 operations
3. He probably should spend much of his time in the Unit 1 control room
4. He should be readily available for Unit 1 relief if needed

J. E. Smith and K. S. Canady agreed to the above and thanked Dick VanNeil for checking into this matter.

K S Canady

K. S. Canady

KSC:vr
cc: Mr. P. H. Barton
Mr. J. E. Smith

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

APR 20 1973

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

In response to your letter of April 4, 1973, we have taken the following positions concerning the Technical Specifications for Units 2 and 3:

Item 14 Specification 6.1.1.6.b
Our criterion for Senior Reactor Operators (SRO) is that, for any station with more than one reactor containing fuel, the number of Senior Reactor Operators onsite at all times shall not be less than the number of control rooms from which the fueled units are monitored. This provides for an SRO who is available to follow shutdown maintenance and take emergency action if needed, while the other control room is under the supervision of another SRO. The requirement of 3 SRO's when all three units are at other than cold shutdown conditions would still apply. Therefore, the specification for Units 1, 2 and 3 should be revised as follows: "At least one licensed Reactor Operator per unit and two Senior Reactor Operators shall be at the station at all times when there is fuel in two reactor vessels. One licensed operator per unit shall be in the control room for that unit. At no time will the shift crew be less than three persons per unit."

Item 16 Proposed Specification 6.1.1.6.f
In previous discussions on minimum shift staffing, we have stated the AEC position for 2-unit and 3-unit operation, i.e. 8 and 12 operators respectively. Table 6.1-1 of your Technical Specifications, as presently worded, takes credit for an operational computer by requiring one less operator in each case. Therefore, the following new specification is required:

"f. If the computer for a reactor is inoperable for more than eight (8) hours, an additional operator shall be called in to supplement the shift crew."

OFFICE ▶						
SURNAME ▶						Appl
DATE ▶						

APR 20 1973

If you have questions regarding the above, please contact us.

Sincerely,

Original signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
As stated

cc: William L. Porter, Esquire
Duke Power Company
P.O. Box 2178
422 South Church Street
Charlotte, North Carolina 28201

DISTRIBUTION:
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Docket File (3)
R. C. DeYoung
J. Hendrie
A. Kenneke
R. W. Klecker
OGC
RO (3)
I. A. Peltier
A. Schwencer
E. I. Goulbourne
R. Houston

OFFICE ▶	L:PWR-4	L:PWR-4	L:OSB	L:PWR-4		
SURNAME ▶	IPeltier:dt	ASchwencer	RHouston	RCDeYoung		
DATE ▶	4/20/73	4/20/73	4/20/73	4/20/73		

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EIGoulbourne (18)
PWR BC's
ASchwencer

MAR 27 1973

Docket No. 50-269

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

Section 50.34(b) (6) (ii) and Appendix B of 10 CFR 50 require that operating plants have a Quality Assurance Program for Operation. To provide guidelines for licensees in establishing adequate programs in this regard, Safety Guide No. 33, Quality Assurance Program Requirements (Operation) was issued on November 3, 1972.

As a part of our periodic review of quality assurance programs for operating plants, we request that you submit a description of your present Quality Assurance Program for Operation for Oconee Station Unit 1. To assure that your program meets the requirements of Appendix B of 10 CFR 50, we also request that you compare your program to the guidelines expressed in Safety Guide No. 33. In your submittal, identify any areas in which the program does not meet the guidelines in Safety Guide No. 33 and the extent to which the guidelines are not met.

The information which we are requesting should be submitted within thirty (30) days after receipt of this letter.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: William L. Porter
Duke Power Company
P. O. Box 2178
422 So. Church St.
Charlotte, N. C. 28201

OFFICE ▶		PWR-4	L:C/PWR-4	L:AD/PWRs	
SURNAME ▶		IAPeltier:kmf	ASchwencer	RCDEYoung	
DATE ▶		3/ 26 /73	3/ 26 /73	3/ 26 /73	

Docket File

Docket Nos. 50-269/270/287

MAR 14 1973

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

Our November 20, 1972 letter to you requested that you provide us with the necessary analysis and other relevant data for determining the consequences of fuel densification in the Oconee Nuclear Station, Units 1, 2 and 3. In response to our requests you have provided B&W topical reports BAW-10055 and BAW-1388 which report the results of your study and analysis. Our review of these documents has revealed a number of areas where additional information will be required for us to complete our evaluation. These areas are listed in the enclosure to this letter. Also included in the enclosure are statements of our positions on conservative assumptions to be used regarding vent valve malfunction and departure from nucleate boiling.

The bulk of the enclosed material has already been discussed with you and therefore we anticipate you will be able to respond in a short period of time. In order to meet our May 1, 1973 target date for completing our review of BAW-10055 and BAW-1388 we will need your complete response by April 13, 1973. If you have questions regarding the enclosure please contact us.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Request for Additional Information

cc: See attached page

RG

cc: William L. Porter
Duke Power Company
P. O. Box 2178
422 So. Church Street
Charlotte, North Carolina 28201

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RWKlecker
OGC
RO (3)
IAPeltier (2)
EIGoulbourne (2)

OFFICE ▶	PWR-4	L:C/PWR-4	L:AD/PWRs			
SURNAME ▶	IAPeltier:kmf	ASchwencer	RCDeYoung	H Schierling		
DATE ▶	3/ 14 /73	3/ 14 /73	3/ 14 /73	3/14/73		

REQUEST FOR ADDITIONAL OCONEE 1 FUEL DENSIFICATION INFORMATION

1. Provide the values for the following physical properties and dimensions of the Oconee Unit 1 fuel pins:

   a. fuel pellet length, diameter, dished end volume and chamfer volume

   b. fuel pellet density

   c. clad inside and outside diameter, initial ovality, and wall thickness.

   For each of these parameters, provide the nominal value, the specified value and tolerances, and the average value (as built) with standard deviation as well as maximum and minimum measured values.

   Describe the method of measuring that was used, the frequency of measuring samples, and the production step at which the measurements are performed in the production process of the fuel assembly. If any of these parameters is measured at different times in the production process, e.g., at the fuel pellet manufacturer and at B&W, with identical or different measuring techniques compare the results and discuss any differences.

2. In order to assess the B&W evaluation model, TAFY, for stored energy, fuel pellet to clad gap conductance, and clad temperature of a fuel pin for Oconee Unit 1, a detailed description of the following items is required. Where applicable, equations and empirical formulations should be provided.

   a. The amount and composition of the sorbed gases assumed to be present in the fuel including the analytical methods used to describe the release rate of the sorbed gases.

b. The amount and content of the gas in the gap between fuel pellet and clad.

c. Description of the gas mixture conductivity model for the gas in the fuel pellet-clad gap, and the thermal expansion model for the fuel pellet and the clad. Discuss how the fuel cracking is treated.

d. A listing of input values used for the TAFY code, including fuel and clad surface roughness and the fuel pin plenum volume.

e. A listing of the following parameters calculated with TAFY: hot gap size, fuel pellet diameter, conductivity of gas mixture, temperature jump distance, gap conductance and the contribution of each of the additive terms in the gap conductance. The information should be provided as a function of linear heat generation rate (kw/ft) and as a function of fuel burnup.

f. A comparison of TAFY calculated gap conductance (see item e) with applicable fuel performance data.

3. Provide additional information on the fuel cladding creep tests that were performed in the BAWTR and that form the basis for the B&W clad creep model, CRECOL, which is used to calculate the expected collapse time for the Oconee Unit 1 fuel cladding. The requested information should include:

a. Physical properties of samples including yield stress, Young's modulus, Poisson's ratio and cold work.

b. Physical dimensions of samples including measured outside diameter, ovality, and thickness

c. Three profilometer measurements of the pellet diameter (one typical and two extremes).

4. In order to assess the B&W collapse model, CRECOL, that is used to calculate expected collapse time for the fuel pins in Oconee Unit 1, the following additional information is required:

a. The equation used to calculate collapse ovality and a justification for not using the creeprate of irradiated fuel in that calculation.

b. A detailed description and justification for the extrapolation of BAWTR test data to the collapse time by use of the Larson Miller Parameter (LMP) including specific literature references to this method of extrapolation.

c. A justification for not including in the cladding stress analysis such axial forces as caused by pellet hangup, rod interference on the grid plates, and rod bending at the spacer grids.

d. A comparison of CRECOL calculated critical ovality and collapse time with experimental clad performance data.

e. A discussion of how flow induced fuel pin vibrations could affect the fuel pin collapse time.

f. A discussion of the clad temperature used in the CRECOL calculation.

g. A comparison of the B&W CRECOL code and the CRECOL code described in USAEC Report GAMD 9623, GGA, 1969. The comparison should identify any changes made to result in the present B&W version.

5. In order to perform an independent staff evaluation of the clad integrity for Oconee Unit 1, the following information is requested:

a. Detailed discussion of the 0.9 value for the usage factor and of the damage catagories included in the analysis.

b. Collapse time calculated with CRECOL with a comparison to one cycle and three cycle operating times.

c. A list of operating conditions and physical properties including:

   - maximum operating time
   - maximum external fuel pin pressure
   - clad temperature
   - clad outside diameter
   - clad thickness
   - initial ovality
   - yield stress vs temperature and fast flux
   - elastic modulus vs temperature
   - Poisson ratio vs temperature
   - internal fuel pin pressure vs time
   - fast flux

d. Discussion of the assumptions for the internal fuel pin pressure vs time, including the cold and hot BOL pressure with and without fuel densification.

6. In order to assess the B&W evaluation of transients and accidents, provide a complete and consistent set of design values and operating parameters for conditions with and without fuel densification. Where applicable, appropriate information in the Final Safety Evaluation Report, FSAR, for Oconee Unit 1 should be referenced. The information requested should include:

a. core wide radial power map

b. radial local peak for hot assembly

c. axial flux shape

d. local flux distribution in hot assembly

e. mass inlet velocity to hot assembly (with and without one vent valve assumed open)

f. loss coefficients for spacer grids and upper and lower end fittings.

7. Provide an analysis of the (1) loss of flow transient and (2) Locked rotor accident for Oconee Unit 1 without and with the assumption of densified fuel. The information provided should include the following:

a. nuclear power decay

b. core coolant flow decay

c. core inlet pressure

d. DNBR vs time

e. peak clad temperature vs time

f. peak fuel centerline temperature vs time

g. average heat flux vs time

h. gap conductance vs time

i. clad to coolant heat transfer coefficient

8. Provide the technical bases and supporting analyses for your conclusion that the 8.55 ft^2 split break would remain the worst break for a loss of coolant accident within the break spectrum considering the effects of fuel densification. For the worst break size provide curves showing:

a. Hot rod axial flux distribution for the steady state condition

b. Maximum clad temperature and local hot rod heat transfer coefficient as a function of time

c. Hot channel flowrate as a function of time.

9. Provide details of the assumptions and justification for establishing the design transient 100% - 30% - 100% power as the limiting transient. Discuss the axial xenon oscillations that are included in the design transient analysis.

10. Discuss, in detail, and justify how the effects of fuel densification are included in the rod ejection accident analysis and compare this analysis to the one in the FSAR without fuel densification. In particular, for full power cases, provide the initial peaking factors and their relationship to the power spike model and to design limits in the Technical Specifications for Oconee Unit 1. Describe how the initial pellet density variation has been accounted for if other than by a 2 variation on heat flux and gap increase. Provide the peak fuel temperature (average and centerline) and clad temperature as a function of time during the accident. Indicate the number of fuel rods that experience DNB and will fail during the course of the accident.

11. The appropriate conservative assumptions for an assumed vent valve malfunction and the criteria for determining departure from nucleate boiling and subsequent degraded heat transfer are:

(a) Since the status of vent valves (open or closed) are not directly monitored and there is limited operating experience with these devices one vent valve should be assumed to malfunction in either the closed or open position (whichever is more conservative) for safety analyses.

(b) Departure from nucleate boiling (DNB) and a subsequent degraded heat transfer condition should be assumed whenever a DNBR equal to or less than 1.3 is predicted. For this purpose the use of the TEMP code and the B&W-2 correlation is acceptable.

Docket File

Docket Nos. 50-269/270/287 MAR 14 1973

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production & Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

This letter is in regard to the establishment of a new schedule for our review of your application for an operating license for your Oconee Units 2 & 3 located in Oconee County, South Carolina.

We have evaluated our workload, and based on this evaluation we have targeted the issuance of our Safety Evaluation Report for June 26, 1973. A summary of the more important milestones are provided as an enclosure for your information and review.

It should be realized that the enclosed dates are target milestones which may require some changes as the review process proceeds, and better estimates of the time required are developed. While we will not formally inform you (by letter) of minor changes, we will confirm, in writing, significant changes to the schedule.

Many of these milestones require specific input, preparation, or participation from you, such as responses to questions, site visits and ACRS meetings. For this reason, we believe that you

Ry

should critically review the dates associated with these important milestones and advise us as to whether you believe the dates are realistic or should be adjusted. After you have reviewed these milestones and associated dates, I suggest that we meet with a representative of your corporate organization to discuss the schedule if these are significant matters of concern to you. This meeting should be scheduled sometime within the next two weeks.

Sincerely,

Original Signed By
A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosure:
As Stated

cc: William L. Porter
Duke Power Company
P. O. Box 2178
422 So. Church St.
Charlotte, N. C. 28201

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ENCLOSURE I

Milestone	Date	Comment
Complete Applicant Response	5/1/73	This is the latest date for applicant input on outstanding issues such as ECCS, Fuel Densification and High Energy-Line Rupture.
Safety Evaluation Complete	6/26/73	Based on July ACRS Meeting
Tech Spec Approval	6/30/73	
ACRS Meeting	7/5-6/73	
Prospective Decision Date	8/15/73	Earliest date for Licensing Unit 2

Docket
File

MAR 12 1973

Docket Nos. 50-269/270/287

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

We have received your Oconee Nuclear Station Industrial Security Plan, Revision 2, forwarded by your letter of February 14, 1973. It is being withheld from public disclosure. Future correspondence on the plan should be withheld from public disclosure and handled in the same manner as the original plan and its revisions.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: William L. Porter
Duke Power Company
P. O. Box 2178
422 So. Church St.
Charlotte, N. C. 28201

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MAR 12 1973

Docket Nos. 50-269/270/287

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

We have completed our review of your Oconee Nuclear Station Industrial Security Plan, Revision 2, forward by your letter of February 14, 1973, and find it satisfactory. The plan is being withheld from public disclosure. Any future revisions, amendments or correspondence on the Plan should consider the proprietary status of the Plan and be treated in the same manner as the original plan and its current revision.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

cc: William L. Porter
Duke Power Company
P. O. Box 2178
422 So. Church St.
Charlotte, N. C. 28201

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

FEB 1 2 1973

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

It is our understanding that the B&W topical report, BAW-10001, dated August 1969, "Incore Instrumentation Test Program," is referenced in your Oconee application. Enclosed for your information is a copy of our letter to B&W concerning our review of this report.

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

Sincerely,

Original Signed By
A. Schwencer

A. Schwencer, Chief
Pressurized Water Reactors Branch No. 4
Directorate of Licensing

Enclosure:
Ltr to B&W

cc w/ encl:
William L. Porter, Esquire
Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 28201

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Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE : 1970 O - 405-346

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

FEB 12 1973

Mr. James F. Mallay
Manager, Licensing
Nuclear Power Generation
Babcock & Wilcox Company
P. O. Box 1260
Lynchburg, Virginia 24505

Dear Mr. Mallay:

We have completed our review of your topical report, BAW-10001, dated August 1969, "Incore Instrumentation Test Program." In the context that the instrumentation is used only for physics calculation verification and fuel management, we consider the report adequate. However, if credit for power flattening beyond the capability of the excore instrumentation becomes necessary, the detection characteristics of this system would require further review.

Sincerely,

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

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Docket No. 50-269

JAN 24 1973

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

We have completed our review of Revision 1 of your Industrial Security Plan submitted by your January 5, 1973 letter. Enclosed are a number of deficiencies and requirements for additional information which we have noted. Please provide the additional information and correct the noted deficiencies in a revised plan and submit it in three (3) copies for our approval by February 15, 1973. If you have any questions regarding the enclosed, please contact us. If you cannot meet the above submittal date please inform us of the date you will meet within seven (7) days of receipt of this letter.

Sincerely,

Original Signed by
Albert Schwencer

A. Schwencer, Chief
Pressurized Water Reactors Branch No. 4
Directorate of Licensing

Enclosure:
Request for Additional Information

cc w/encl:
William L. Porter, Esquire
Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 28201

RW

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SURNAME ▶	IAPeltier:emp	ASchwencer				
DATE ▶	1/24/73	1/24/73				

INDUSTRIAL SECURITY PLAN

REQUESTS FOR ADDITIONAL INFORMATION

	Section	
1.	3.2	The fence described is not consistent with that specified by paragraph 2.2.4 of ANS-3.3 draft 4.
2.	3.3	Locate on Drawing O-3 the location of the CCTV for the intake structure.
3.	3.3	Does the phrase "locked exterior doors on the west side of the station" include the several shipping and receiving doors, and the two fuel transfer area doors? The security measures for these doors are not shown on Figure 1-5.
4.	3.3	Provide locks on the two main doors to each of the two control rooms (Figure 1-7).
5.	3.3.1	Provide for the surveillance of vital equipment and facilities inside the vital area boundary by operating personnel. See ANS 3.3, Draft 4, Subsection 3.4.3.
6.	4.0	Corporate office responsibility for the security program is not discussed as specified in paragraph 4 of ANS-3.3 draft 4.
7.	4.1	Periodic review, control and dissemination of procedures is not discussed as delineated in ANS-3.3 draft 4.
8.	4.4	State the average time required for the Oconee County Sheriff's Department to respond to a request for assistance.

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

JAN 24 1973

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

It is our understanding that the Babcock & Wilcox topical report, BAW-10029, "Control Rod Drive Mechanism Test Program," is referenced in the Final Safety Analysis Report for Oconee Nuclear Station Units 1, 2, and 3. Enclosed for your information is a copy of our letter to B&W presenting the results of our evaluation of this report. Please contact us if you desire any discussion or clarification of this matter.

Sincerely,

Original Signed by
Albert Schwencer

A. Schwencer, Chief
Pressurized Water Reactors Branch No. 4
Directorate of Licensing

Enclosure:
Ltr to B&W dtd 1/22/73

cc w/encl:
William L. Porter, Esquire
Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 28201

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

JAN 22 1973

Mr. James F. Mallay
Manager, Licensing
Nuclear Power Generator
Babcock & Wilcox Company
P. O. Box 1260
Lynchburg, Virginia 24505

Dear Mr. Mallay:

The Regulatory staff has completed its review of Babcock & Wilcox topical reports BAW-10007, Revision 1 (Proprietary) and BAW-10029 (Non-proprietary) entitled "Control Rod Drive Mechanism Test Program."

A summary of our review is enclosed for your information.

As a result of our review we have concluded that the non-proprietary report, BAW-10029, is adequate and acceptable as a reference in lieu of the proprietary report, BAW-10007, in application for construction permits and operating licenses provided that the application satisfies these conditions:

1. The purpose of the reference to BAW-10029 should be limited to control rod drive mechanism (CRDM) operating characteristics.

2. The application should discuss the analytical methods, criteria and stress limits used in the design of the CRDM.

3. The application should address the difference between the 20-year design life of the CRDM (as presented in BAW-10029) and the 40-year design life of the typical plant.

Sincerely,

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Topical Report Evaluation

TOPICAL REPORT EVALUATION

Report Identification: BAW-10007, Rev. 1;
BAW-10029
Report Title: Control Rod Drive Mechanism Test Program
Report Dates: June 1971 and January 1972
Originating Organization: Babcock & Wilcox
Reviewed By: Mechanical Engineering Branch, AEC Directorate of Licensing
November 1972

SUMMARY OF TOPICAL REPORT:

Control rod drive mechanism tests were carried out under operating coolant flow, temperature, pressure and water chemistry conditions and varying the equipment alignment from optimum to the maximum amount of misalignment expected with the most adverse combination of design tolerances. In addition, a test was completed to confirm the motor's ability to provide for forced insertion in the event of a stuck rod. Life testing was carried out under the misalignment conditions and duplicating the stroke, travel and reactor trips expected during 20 years of service life. Trip times, deacceleration rates and component wear were measured in conjunction with the life tests, and insertion force was measured under the stuck rod conditions. All test results were judged to comply with the performance criteria specified and were considered satisfactory.

SUMMARY OF REGULATORY EVALUATION:

We have reviewed the subject reports and find that the three specific versions of a basic roller nut drive mechanism designated "A", "B" and "C" by B & W exhibit satisfactory trip, wear, acceleration, and forced insertion characteristics in operation. However, the reports are considered acceptable only if the provisions stated below in the regulatory position are satisfied.

Pressure boundary design criteria and stress limits except for the OBE and DBE under various operating conditions are not presented for the type "A" mechanism. Zero period ground accelerations of 0.25 g horizontal and 0.3 g vertical equivalent static accelerations were used for seismic analysis. Type "C" components are designed and manufactured to the Class A vessel requirements of Section III. However, specific stress limits for emergency and faulted operating conditions, and the seismic design input loadings are not specified. The design criteria used for the design of type "B" mechanisms which are hybridized version of types "A" and "C" mechanisms are also not described.

REGULATORY POSITION:

The subject reports provide a satisfactory basis for accepting the operating performance of B & W control rod drive mechanisms Type "A", "B", or "C" and may be referenced in future case applications provided that: (a) the purpose of reference is limited to only mechanism operating characteristics and (b) the analytical methods, criteria and stress limits used in the design of the mechanisms and (c) life testing or equivalent procedures are used to confirm service life beyond the 20 years demonstrated by the test program are submitted. Items (a), (b) and (c) must be evaluated on a case-by-case basis.

JAN 1 9 1973

Docket Nos. 50-269/270/287

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

In your letter of January 5, 1973 you transmitted to us Revision No. 1 to the Industrial Security Plan for the Oconee Nuclear Station, Units 1, 2 and 3. In view of this, the two signed original copies (No. 003 and 004) of the Oconee Nuclear Station Security Plan transmitted by your November 17, 1972 letter are hereby returned.

Sincerely,

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
Industrial Security Plan
Copies #003 and #004

cc: William L. Porter, Esquire
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28201

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Docket Nos. 50-269
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JAN 17 1973

Duke Power Company
ATTN: Mr. A. C. Thies
Senior Vice President
Production and Transmission
P. O. Box 2178
Charlotte, North Carolina 28201

Gentlemen:

Please refer to our letter to you, dated December 15, 1972, regarding the analysis of postulated pipe failures outside of the containment structure. Enclosed are errata items for your additional guidance. If you have questions regarding this new material, please contact us.

Sincerely,

Original Signed by
Albert Schwencer

A. Schwencer, Chief
Pressurized Water Reactors Branch No. 4
Directorate of Licensing

Enclosure:
Errata Sheet

cc w/encl:
William L. Porter, Esquire
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28201

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SURNAME ▶	IAPeltier:emp	ASchwencer				
DATE ▶	1/15/73	1/16/73				

ERRATA SHEET FOR "GENERAL INFORMATION REQUIRED FOR CONSIDERATION OF THE EFFECTS OF A PIPING SYSTEM BREAK OUTSIDE CONTAINMENT"

The following lists the changes that have evolved on our initial information request:

1. Page 2, Item 2--Insert the following in 2. to precede the existing first sentence:

   "Design basis break locations should be selected in accordance with the following pipe whip protection criteria; however, where pipes carrying high energy fluid are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of those structures and systems shall be provided to cope with the environmental effects (including the effects of jet impingement) of a single postulated open crack at the most adverse location(s) with regard to those essential structures and systems, the length of the crack being chosen not to exceed the critical crack size. The critical crack size is taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width."

2. Page 2, Item 2(a)(2)--Change nomenclature to read "any intermediate locations between terminal ends where the primary plus secondary stress intensities Sn ... "

3. Page 4, Item 2.(b)(2)--Change 0.9 $(S_h + S_A)$ to 0.8 $(S_h + S_A)$.

4. Page 6, Item 7 --Add "structural" to read "The structural design loads..."

5. Page 7, Item 11.(a)--Add "required" so as to read, "Loss of required redundancy..."

6. Page 7, Item 11.(a)--Delete "the steam line break" and replace with "that" to read "...the consequences of that accident..."

7. Page 8, Item 11.(b)-- Replace (b) with the following: (b) "Environmentally induced failures caused by a leak or rupture of the pipe which would not of itself result in protective action but does disable protection functions. In this regard, a loss of redundancy is permitted but a loss of function is not permitted. For such situations plant shutdown is required."

8. Page 8, Item 13--Change wording in the first sentence to read "Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a high energy fluid line break."