Docket No. 50-269

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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:

A copy of Supplement No. 2 to the Safety Evaluation prepared by the Directorate of Licensing relating to your Oconee Nuclear Station, Unit 1, is enclosed for your information.

Sincerely,

Original signed by
R. C. DeYoung

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

Enclosure:
Supplement No. 2 to the
Safety Evaluation

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

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

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

OFFICE ▶	L:PWR-4	L:PWR-4	L:PWR-4	L:AD:PWRs	OGC	
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SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

OCONEE NUCLEAR STATION UNIT 1

DOCKET NO. 50-269

DECEMBER 19, 1972

APPENDICES

Appendix A - Chronology of Regulatory Review of Oconee Unit 1 Internals Failure and Redesign

1.0 INTRODUCTION

The Duke Power Company (applicant), has requested a license to construct and operate three pressurized water reactors, identified as Units 1, 2, and 3 at its Oconee Nuclear Station in Oconee County, South Carolina.

On June 2, 1969, the applicant filed, as Amendment 7, the Final Safety Analysis Report required by Section 50.34(b) of Chapter 10 of the Code of Federal Regulations as a prerequisite to obtaining an operating license for each unit.

The regulatory staff published its Safety Evaluation Report (original Safety Evaluation Report) for Unit 1 December 29, 1970.

Subsequently a supplemental review of the plant's emergency core cooling systems was performed in accordance with the criteria described in the Interim Policy Statement published in the FEDERAL REGISTER on June 29, 1971 (36 F.R. 12247). Based upon this review the staff issued Supplement 1 to the original Safety Evaluation Report on March 24, 1972.

In March 1972, the Oconee Unit 1 suffered damage to the steam generators and reactor vessel internals requiring significant design modifications. We have reviewed these design modifications and our evaluation is contained in this document. This document is in the same format as the original Safety Evaluation Report for ease of reference. Our evaluation of the reactor internals prior to modification is contained on pages 20-32 of that report.

Failure of and damage to vessel internals are believed to have been caused by flow induced vibration, and damage to the steam generators was caused by loose parts resulting from the vessel internals failures. The Babcock & Wilcox Company (B&W), the system supplier, has assessed the failures and damage and analyzed the cause through extensive examination, laboratory and full scale tests and system mockups. The results of this assessment and analysis have been reviewed by the regulatory staff. In addition, B&W has redesigned the vessel internals and modified them where required to prevent a recurrence of these failures.

The damage to Oconee Unit 1 has been repaired and the modifications required by the new design have been completed. B&W has provided for an extensive vibration and loose parts monitoring program to be carried out during continuation of hot functional tests in Unit 1 to assure that the system response is well understood and is within design predictions and limits. We have reviewed these programs, have found them to be acceptable, and will follow closely the results of the program throughout the hot functional tests now underway.

5.0 REACTOR COOLANT SYSTEM

5.1 General

At the conclusion of the first phase of the hot functional testing program of Oconee Unit 1 on March 11, 1972, an inspection of the reactor coolant system revealed that several reactor internals components had failed and had caused significant damage to hardware within the reactor vessel and to both steam generators. This incident was formally reported to the AEC on April 4, 1972. Since learning of the incident the regulatory staff has been actively involved in reviewing the matter both through visual inspections and the review of data and of corrective engineering design developed by the applicant in order to assess and evaluate the safety implications. Appendix A provides a chronology of the regulatory review including field trips, meetings with the applicant and submittal of important documents. The following sections of this supplement summarize our safety evaluation.

5.2 REACTOR COOLANT SYSTEM COMPONENTS

Damage to the steam generators in the primary system was caused by the impact of portions of the failed internals components from the reactor vessel. This damage was confined to the upper steam generator plenum region and consisted of deformation of the tube ends, the tube sheet clad and the dome wall clad. Since the steam generators could be repaired by approved repair procedures judged to be adequate to restore the generators to their original quality no further safety evaluation was warranted other than to verify the acceptability of the restoration by inspection and test which has been done. This safety evaluation deals with the cause and prevention of flow induced vessel internals failure. The failures experienced were:

a. Incore instrument nozzles (21 broken off and remaining 31 damaged).

b. Incore instrument guide tubes (4 broken off, 4 cracked and remaining 44 no apparent damage).

c. Thermal shield (evidence of movement, wear and mating surface damage).

d. Instrumentation guide tubes (2 broken off and remaining 4 no apparent damage).

As amended through Amendment 37 the Duke Power Company application for an operating license for Oconee Unit 1 references four B&W topical reports dealing with the reactor internals, their failure and redesign. They are:

BAW-10037, Revision 2, November 1972, "Reactor Vessel Model Flow Tests"

BAW-10038, Revision 2, November 1972, "Prototype Vibration Measurement Program For Reactor Internals"

BAW-10050, Revision 1, November 1972, "Evaluation of Oconee Reactor Failure"

BAW-10051, Revision 1, November 1972, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibration"

Our safety evaluation is based principally upon these reports; visits to the reactor site and vendor's facilities; and meetings held with the applicant and B&W.

In the process of reviewing the failure, we have evaluated the above reports. We concur with the conclusions set forth in BAW-10050 Revision 1 that the recommended design modifications on the internals have been based upon a conservative application of the response and failure data from Oconee Unit 1. We concur with the conclusions set forth in BAW-10037 Revision 2 that the reactor vessel scale model flow test approach used will verify the core flow distribution, However, due to a lack of valid flow forcing functions, B&W has not yet demonstrated a dynamic analysis to predict the structural behavior of reactor internals when subjected to transient loadings. The redesigned internals are accepted for Oconee Unit pending satisfactory completion of the new hot functional preoperational tests. The results obtained from the preoperational tests will be evaluated prior to permitting significant power operation to confirm this acceptability. The lack of valid vibration predictions precludes our acceptance at this time of BAW-10038 and BAW-10051 with respect to the designation of Oconee 1 as a prototype for follow-on plants.

In BAW-10050 Revision 1, B&W describes its investigation on the cause of the preoperational test failure. On the basis of the metallographic examination of the failure surfaces B&W concluded that fatigue due to flow induced vibratory motion was the major failure mode. Component redesign was based upon (a) separation of structural frequencies further from vortex shedding frequencies, and (b) reduction of the stresses to a level further below the material endurance limit. We concur with B&W that such design modifications will improve the structural integrity of the reactor internals.

In BAW-10037, Revision 2, B&W describes the reactor vessel flow testing conducted on a one-sixth scale model to investigate flow distribution, pressure loss and the pattern of flow mixing from the various inlets. The flow characteristics inside the core and vent valve testing were emphasized. Both the original and the modified designs were tested. The results of the tests showed that the modified design provides more uniform flow distribution with acceptable pressure loss. The flow rate was slightly higher at certain portions of the core and required further minor modifications in design. We concur with B&W on the approach used to verify the core flow distribution.

In BAW-10051, Revision 1, B&W describes the attempts made to justify the reactor internals design modifications by computing responses of modified components to flow induced vibration. However, the actual flow forcing functions may not be verified until completion of the new preoperational vibration test program for Oconee Unit 1. Until the required verification is available, we cannot concur with the applicant's conclusion on this matter. The applicant has stated that further steps will be taken, including component testing of instrument guide tubes and incore nozzle assembles, to provide a better understanding of the vibration behavior. In addition a more definitive understanding of the thermal shield vibration response characteristics will be sought through further evaluation of the Oconee Unit 1 response and failure data.

In BAW-10038, Revision 2, B&W describes its prototype preoperational vibration testing program for reactor internals. The applicant cannot provide valid vibration predictions as required by Safety Guide 20, "Vibration Measurements on Reactor Internals" for prototype qualification because of the lack of conclusive dynamic analysis. Therefore, we cannot complete our prototype qualification evaluation of Oconee Unit 1 at this time.

5.3 Conclusion

Based on the above evaluation and the information presented in our original Safety Evaluation Report we conclude that there is reasonable assurance that the redesign of the Oconee Unit 1 reactor internals is acceptable pending confirmation by the vibration testing to be conducted during the preoperational tests. Operations of Unit 1 will be restricted to no greater than 5.0% full rated power until the results of preoperational testing have been evaluated by the regulatory staff.

APPENDIX A

CHRONOLOGY OF REGULATORY REVIEW OF OCONEE UNIT 1 INTERNALS FAILURE AND REDESIGN

1.	March 29-30, 1972		Site visit to Oconee to view Unit 1 internals and steam generator damage.
2.	April 6, 1972		Meeting at Barberton with B&W and Duke to discuss steam generator repairs and vibration testing of internals for Oconee Unit 1.
3.	May 24, 1972		Meeting at Bethesda with B&W and Duke to discuss repair of steam generators and vessel internals for Oconee Unit 1.
4.	August 7, 1972		Meeting at Bethesda with B&W and Duke to discuss vibration monitoring of vessel internals.
5.	September 15, 1972		Application Amendment No. 35 provided B&W topical reports, BAW-10037, BAW-10038, BAW-10050, BAW-10051.
6.	October 25, 1972		Meeting at Bethesda with B&W and Duke to discuss reactor vessel internals redesign and vibration monitoring.

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50-270
and 50-287

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

Gentlemen:

The Regulatory staff's continuing review of reactor power plant safety indicates that the consequences of postulated pipe failures outside of the containment structure, including the rupture of a main steam or feedwater line, need to be adequately documented and analyzed by licensees and applicants, and evaluated by the staff as soon as possible. Criterion No. 4 of the Commission's General Design Criteria, listed in Appendix A of 10 CFR 50 requires that:

> "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit."

The previous version of the Commission's General Design Criteria also reflects the above requirements.

Thus, a nuclear plant should be designed so that the reactor can be shutdown and maintained in a safe shutdown condition in the event of a postulated rupture, outside containment, of a pipe containing a high energy fluid, including the double ended rupture of the largest pipe in the main steam and feedwater systems. Plant structures, systems, and components important to safety should be designed and located in the facility to accommodate the effects of such a postulated pipe failure to the extent necessary to assure that a safe shutdown condition of the reactor can be accomplished and maintained.

Based on the information we presently have available to us on the Oconee plant, Units 1, 2 and 3, we understand that one steam line passes through the containment wall clear of all other buildings and into the turbine building. The other steam line passes through the penetration room and auxiliary building into the turbine building. From this it appears that failure due to pipe whip or overpressure of the closed compartment may be possible and some modifications of the facility may be necessary.

We request that you provide us with analyses and other relevant information needed to determine the consequences of such an event, using the guidance provided in the enclosed general information request. The enclosure represents our basic information requirements for plants now being constructed or operating. You should determine the applicability, for the Oconee facility, Units 1, 2 and 3, of the items listed in the enclosure.

If the results of your analyses indicate that changes in the design of structures, systems, or components are necessary to assure safe reactor shutdown in the event this postulated accident situation should occur, please provide information on your plans to revise the design of your facility to accommodate the postulated failures described above. Any design modifications proposed should include appropriate consideration of the guidelines and requests for information in the enclosure.

We will also need, as soon as possible, estimates of the schedule for design, fabrication, and installation of any modifications found to be necessary. Please inform us within 7 days after receipt of this letter when we may expect to receive an amendment with your analysis of this postulated accident situation for the Oconee facility, Units 1, 2 and 3, a description of any proposed modifications, and the schedule estimates described above. Sixty copies of the amendment should be provided.

A copy of the Commission's press announcement on this matter is also enclosed for your information.

Sincerely,

Original Signed by
A. Giambusso

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

Enclosures:
As stated

RW

cc: See next page

cc: Mr. William L. Porter, Esq.
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General Information Required for Consideration of the Effects of a Piping System Break Outside Containment

The following is a general list of information required for AEC review of the effects of a piping system break outside containment, including the double ended rupture of the largest pipe in the main steam and feedwater systems, and for AEC review of any proposed design changes that may be found necessary. Since piping layouts are substantially different from plant to plant, applicants and licensees should determine on an individual plant basis the applicability of each of the following items for inclusion in their submittals.

1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:

   (a) Both of the following piping system conditions are met:

      (1) the service temperature is less than 200° F; and

      (2) the design pressure is 275 psig or less; or

   (b) The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained from whipping by plant design features, such as concrete encasement; or

   (c) Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety; or

(d) The internal energy level[1] associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

2. The criteria used to determine the design basis piping break locations in the piping systems should be equivalent to the following:

(a) ASME Section III Code Class I piping[2] breaks should be postulated to occur at the following locations in each piping run[3] or branch run:

(1) the terminal ends;

(2) any intermediate locations between terminal ends where the primary plus secondary stress intensities S_m (circumferential or longitudinal) derived on an elastically

---

[1]The internal fluid energy level associated with the pipe break reaction may take into account any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe may be considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

[2]Piping is a pressure retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

[3]A piping run interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movement beyond that required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection, as a branch of the main pipe run.

calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions[4] exceeds 2.0 S_m[5] for ferritic steel, and 2.4 S_m for austenitic steel;

(3) any intermediate locations between terminal ends where the cumulative usage factor (U)[6] derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1; and

(4) at intermediate locations in addition to those determined by (1) and (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:

(1) the terminal ends;

---

[4] Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences) and testing conditions.

[5] S_m is the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Plant Components."

[6] U is the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components."

(2) any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.9 $(S_h + S_A)$[7] or the expansion stresses exceed 0.8 S_A; and

(3) intermediate locations in addition to these determined by (2) above, selected on reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

3. The criteria used to determine the pipe break orientation at the break locations as specified under 2 above should be equivalent to the following:

(a) Longitudinal[8] breaks in piping runs and branch runs, 4 inches nominal pipe size and larger, and/or

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[7] S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1972 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

[8] Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

(b) Circumferential[9] breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

4. A summary should be provided of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break including:

(a) The locations and number of design basis breaks on which the dynamic analyses are based.

(b) The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.

(c) A description of the forcing functions used for the pipe whip dynamic analyses including the direction, rise time, magnitude, duration and initial conditions that adequately represent the jet stream dynamics and the system pressure difference.

(d) Diagrams of mathematical models used for the dynamic analysis.

(e) A summary of the analyses which demonstrates that unrestrained motion of ruptured lines will not damage to an unacceptable degree, structure, systems, or components important to safety, such as the control room.

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[9] Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially, and cause whipping in any direction normal to the pipe axis.

5. A description should be provided of the measures, as applicable, to protect against pipe whip, blowdown jet and reactive forces including:

   (a) Pipe restraint design to prevent pipe whip impact;

   (b) Protective provisions for structures, systems, and components required for safety against pipe whip and blowdown jet and reactive forces;

   (c) Separation of redundant features;

   (d) Provisions to separate physically piping and other components of redundant features; and

   (e) A description of the typical pipe whip restraints and a summary of number and location of all restraints in each system.

6. The procedures that will be used to evaluate the structural adequacy of Category I structures and to design new seismic Category I structures should be provided including:

   (a) The method of evaluating stresses, e.g., the working stress method and/or the ultimate strength method that will be used;

   (b) The allowable design stresses and/or strains; and

   (c) The load factors and the load combinations.

7. The design loads, including the pressure and temperature transients, the dead, live and equipment loads; and the pipe and equipment static, thermal, and dynamic reactions should be provided.

8. Seismic Category I structural elements such as floors, interior walls, exterior walls, building penetrations and the buildings as a whole should be analyzed for eventual reversal of loads due to the postulated accident.

9. If new openings are to be provided in existing structures, the capabilities of the modified structures to carry the design loads should be demonstrated.

10. Verification that failure of any structure, including nonseismic Category I structures, caused by the accident, will not cause failure of any other structure in a manner to adversely affect:
    (a) Mitigation of the consequences of the accidents; and
    (b) Capability to bring the unit(s) to a cold shutdown condition.

11. Verification that rupture of a pipe carrying high energy fluid will not directly or indirectly result in:
    (a) Loss of redundancy in any portion of the protection system (as defined in IEEE-279), Class IE electric system (as defined in IEEE-308), engineered safety feature equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of the steam line break accident and place the reactor(s) in a cold shutdown condition; or

(b) Loss of the ability to cope with accidents due to ruptures of pipes other than a steam line, such as the rupture of pipes causing a steam or water leak too small to cause a reactor accident but large enough to cause electrical failure.

12. Assurance should be provided that the control room will be habitable and its equipment functional after a steam line or feedwater line break or that the capability for shutdown and cooldown of the unit(s) will be available in another habitable area.

13. Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a steam line or feedwater line break. The information required for our review should include the following:

(a) Identification of all electrical equipment necessary to meet requirements of 11 above. The time after the accident in which they are required to operate should be given.

(b) The test conditions and the results of test data showing that the systems will perform their intended function in the environment resulting from the postulated accident and time interval of the accident. Environmental conditions used for the tests should be selected from a conservative evaluation of accident conditions.

(c) The results of a study of steam systems identifying locations where barriers will be required to prevent steam jet impingment from disabling a protection system. The design criteria for the barriers should be stated and the capability of the equipment to survive within the protected environment should be described.

(d) An evaluation of the capability for safety related electrical equipment in the control room to function in the environment that may exist following a pipe break accident should be provided. Environmental conditions used for the evaluation should be selected from conservative calculations of accident conditions.

(e) An evaluation to assure that the onsite power distribution system and onsite sources (diesels and batteries) will remain operable throughout the event.

14. Design diagrams and drawings of the steam and feedwater lines including branch lines showing the routing from containment to the turbine building should be provided. The drawings should show elevations and include the location relative to the piping runs of safety related equipment including ventilation equipment, intakes, and ducts.

15. A discussion should be provided of the potential for flooding of safety related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.

16. A description should be provided of the quality control and inspection programs that will be required or have been utilized for piping systems outside containment.

17. If leak detection equipment is to be used in the proposed modifications, a discussion of its capabilities should be provided.

18. A summary should be provided of the emergency procedures that would be followed after a pipe break accident, including the automatic and manual operations required to place the reactor unit(s) in a cold shutdown condition. The estimated times following the accident for all equipment and personnel operational actions should be included in the procedure summary.

19. A description should be provided of the seismic and quality classification of the high energy fluid piping systems including the steam and feedwater piping that run near structures, systems, or components important to safety.

20. A description should be provided of the assumptions, methods, and results of analyses, including steam generator blowdown, used to calculate the pressure and temperature transients in compartments, pipe tunnels, intermediate buildings, and the turbine building following a pipe rupture in these areas. The equipment assumed to function in the analyses should be identified and the capability of systems required to function to meet a single active component failure should be described.

21. A description should be provided of the methods or analyses performed to demonstrate that there will be no adverse effects on the primary and/or secondary containment structures due to a pipe rupture outside these structures.

AEC

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

No. P-429
Contact: Frank Ingram
Tel. 301/973-7771

FOR IMMEDIATE RELEASE
(Wednesday, December 13, 1972)

AEC REGULATORY STAFF REQUESTS DATA ON PIPE BREAKS IN NUCLEAR PLANTS

The Atomic Energy Commission's Regulatory Staff is asking all utilities that operate nuclear power plants or have applied for operating licenses to assess the effects on essential auxiliary systems of a major break of the largest main steam or feedwater line. These lines carry steam from inside the reactor containment building to the main turbine in the turbine building, and hot feedwater back from the turbine condenser. The utility assessments will be evaluated by the AEC's Regulatory Staff.

The probability of a steam-line rupture is low. Nonetheless it will have to be considered in the AEC's safety evaluation.

The review of the pipe break problem has been under way for several weeks. It was started after the Advisory Committee on Reactor Safeguards received a letter raising questions about the location of pipes in the two-unit Prairie Island plant in Minnesota.

The Regulatory Staff has reviewed the Northern States Power Company application to operate Prairie Island, and on the basis of data available it has concluded that design changes will be required at Prairie Island.

Based on the new information--to be submitted by utilities as soon as possible--the Staff will determine what corrective action, if any, is necessary in each case. The changes could include such steps as relocating piping, providing venting of compartments, the addition of piping restraints, and, in some cases, structural strengthening.

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Docket Nos. 50-269
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and 50-287

DEC 6 1972

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:

We have made a preliminary review of the Duke Power Company Oconee Nuclear Station Security Plan, issued November 17, 1972, and find that we require additional information to complete our review. We also find the Plan to be deficient in the following areas:

1. the testing of alarms and communication links,

2. the maintenance of records (visitors log; results of tests, inspection and maintenance; list of false alarms and actions taken),

3. the reporting of threatened or actual attempts of sabotage,

4. the periodic review and update of the Plan,

5. the surveillance of the protected area at least twice per shift,

6. the surveillance of vital areas to ascertain equipment status,

7. the provision for drills, exercises, and tests, and

8. the security of the operating unit(s), during the period the remaining unit(s) are under construction.

Of those areas which are addressed, many are lacking the required detail. Specific deficiencies are detailed in the enclosure to this letter.

OFFICE ▶						
SURNAME ▶						0V
DATE ▶						

You should provide the required information, correct the noted deficiencies and submit the plan as proprietary information in two copies as an amendment to your application for operating licenses for Oconee Units 1, 2, and 3 by January 5, 1973.

If you have any questions regarding this matter please contact us.

Sincerely,

Original Signed by
Albert Schwencer

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

Enclosure:
Request for Additional Information

cc:
William Porter, Esquire
Duke Power Company
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UNITED STATES
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WASHINGTON, D.C. 20545

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and 50-287

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5. the surveillance of the protected area at least twice per shift,

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If you have any questions regarding this matter please contact us.

Sincerely,

Original Signed by
Albert Schwencer

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

Enclosure:
Request for Additional Information

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

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IAPeltier
RHouston

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SURNAME ▶	IAPeltier:emp	ASchwencer				
DATE ▶	12/6/72	12/6/72				

REQUEST FOR ADDITIONAL INFORMATION
OCONEE NUCLEAR STATION
SECURITY PLAN

Section	Deficiency/Additional Information Required
2.3	The definition of Vital Area is too narrow. What about the reactor itself, and the spent fuel pool?
3.2	What is the height and construction of the security fence? Will it be lighted? Will it be alarmed? Under what conditions and for how long, will the main gate and guardhouse not be manned? Under what conditions will visitors not require an escort while inside the protected area?
3.3	Where will the TV monitor for the intake structure be located? Where will the readout(s) be located? Are any additional TV monitors contemplated for surveillance of vital area? Will visitors be required to sign in and sign out?
4.1	State that the Plan will be implemented by written procedures.
4.2	Describe the actions taken in the event of an intrusion into the controlled area, and the protected area. State and justify the time required for deployment of the security guards. What is the average response time for the primary local law enforcement agency?
4.4	Will the security guard force be armed, or will weapons be readily available?
4.5	What is meant by "as required" in the third sentence?
Sketch	Provide separate sketches, to scale, as follows: the controlled area and fence, the protected area and fence, and the vital areas showing all entrances.

See Central Files for concurrence copy + distribution

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

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

November 20, 1972

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

RE: Oconee Nuclear Station, Units 1, 2, and 3

Gentlemen:

The Commission's Regulatory Staff has completed a review of fuel densification and its effect on reactor operation including transients and postulated loss-of-coolant accidents. The Staff's investigations and conclusions are reported in "Technical Report on Densification of Light Water Reactor Fuels" dated November 14, 1972, a copy of which is enclosed for your information and guidance. This report concludes that densification of fuel may occur and that the resulting formation of fuel column gaps should be anticipated in all light water reactor fuels. The report also provides the essential elements to be included in calculational models used to account for the effects of fuel densification.

The Regulatory Staff believes that the fuel for the subject facility(s) is susceptible to densification. Therefore, we request that you provide the necessary analyses and other relevant data for determining the consequences of densification and the effects on normal operation, anticipated transients, and accidents, including the postulated loss-of-coolant accident, using the guidance provided in the enclosed report. If the analyses indicate that changes in design or operating conditions are necessary to maintain required margins, you should submit proposed changes and operating limitations with the analyses.

DV

In order that the Regulatory Staff can conduct an expeditious and orderly review of these matters, we request that you submit the analyses and additional information within 45 days from the date of this letter.

It is requested that this information be provided with one signed original and thirty-nine additional copies. If your submittal is for more than one unit, a total of sixty copies is needed.

Sincerely,

A Giambusso

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

Enclosure:
Technical Report on Densification
(November 14, 1972)

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

Miss Louise Mancum, Librarian
Oconee County Library
201 S. Spring Street
Walhalla, South Carolina 29691

UNITED STATES
ATOMIC ENERGY COMMISSION
XXXXXXXXXXXXXXXXXXX
REGION II - SUITE 818
230 PEACHTREE STREET, NORTHWEST
ATLANTA, GEORGIA 30303
TELEPHONE: (404) 526-4503
DIRECTORATE OF REGULATORY OPERATIONS

DEC 4 1972

In Reply Refer To:
RO:II:RFW
50-269/72-9

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 refers to the inspection conducted by Mr. Warnick and others of this office on October 3-6, 1972, of activities authorized by AEC Construction Permit No. CPPR-33 for the Oconee Unit 1 facility, and to the discussion of our findings held by Mr. Warnick with Mr. Dick and other members of your staff at the conclusion of the inspection.

Areas examined during this inspection are as described in the enclosed inspection report No. 50-269/72-9. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with plant personnel, and observations by the inspector.

Within the scope of the inspection, no violations or safety items were observed.

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 (or your contractors) 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 we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

In regards to the item of noncompliance brought to your attention in our letter, dated August 30, 1972, please be advised that we have no further questions at this time regarding this item.

A reply to this letter is not necessary. Should you have any questions concerning the matters discussed in this letter, however, we will be glad to discuss them with you.

Very truly yours,

John G. Davis
Director

Enclosure:
Inspection Report No. 50-269/72-9

UNITED STATES
ATOMIC ENERGY COMMISSION
XXXXXXXXXXXXXXXXXXXXX
REGION II - SUITE 818
230 PEACHTREE STREET, NORTHWEST
ATLANTA, GEORGIA 30303
TELEPHONE: (404) 526-4503
DIRECTORATE OF REGULATORY OPERATIONS

In Reply Refer To:
RO:II:50-269,270,287

December 4, 1972

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:

The attached Directorate of Regulatory Operations Bulletin No. 72-3, "Limitorque Valve Operator Failures," is sent to you to provide you with information we received from the Northern States Power Company and the Commonwealth Edison Company concerning valve operator malfunctions experienced at their respective facilities. This information may relate to the performance of similar motor operated valves at your facilities.

Very truly yours,

John G. Davis
Director

Enclosure:
Bulletin 72-3

December 4, 1972
Directorate of Regulatory
Operations Bulletin 72-3

LIMITORQUE VALVE OPERATOR FAILURES

We recently received information relating to the malfunction of electric type valve operators at two reactors. The valve operators were identified as Limitorque Models SMB-00 and SMB-000 which are used extensively in safety related systems at a number of PWR and BWR reactor facilities. Subsequent investigation identified a specific production group of these models which were manufactured between 1969 and mid-1971. The specific deficiencies are described as follows:

Plant A

Testing of valves and valve operators used in safety related systems at this facility disclosed ten valves that failed to close following a "valve full open operation" test. The cause of failure was attributed to malfunction of the valve operator torque switch due to a lack of proper clearance between the moving parts of the torque switch unit and the inability of the "torque switch torsion spring" to return the electrical contacts to a closed position following operation of the valve. The weak torsion spring is considered a common mode of failure. Approximately 150 valves ranging up to eight inches in size were equipped with valve operators having the faulty switches.

Plant B

During a reactor startup, the inboard steam supply valve of the reactor core isolation coolant (RCIC) system failed in the open position. Several attempts were made unsuccessfully to close the valve. The failure was attributed to an internal torsion spring in the valve operator torque switch which normally resets the electrical contacts. The valve operator in question is similar to the units which failed at Plant A.

Two additional facilities have recently experienced similar failures since those reported at Plants A and B.

It is requested that you determine whether valve operators of the described make, model and vintage are installed or scheduled to be installed in your facility. If your findings show that valves installed or scheduled to be installed are equipped with the described valve operators, please inform this office within thirty days, in writing, of the number of valves equipped with the valve operators, the systems in which the subject valves are installed or scheduled to be installed, a description of corrective actions taken or planned, and the scheduled completion date of your corrective actions.

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

Docket No. 50-269

NOV 13 1972

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:

This refers to your letter of September 11, 1972, requesting a further review of our stated position regarding minimum crew size for multi-unit operation at the Oconee Station and technical comments on your basis for proposing fewer personnel than we believe necessary from a safety standpoint.

As stated in our August 15, 1972 letter to you, significant station operating experience is necessary to provide assurance that a shift crew of the size you have proposed has the capability to maintain the Oconee Station in a safe condition during normal and abnormal operation, and to protect the health and safety of the public during potential emergency situations. We appreciate the fact that you have researched and analyzed your normal and emergency procedures to determine which of these are expected to be most demanding on shift personnel and have proposed a minimum crew size based on these considerations. However, in the absence of any operating experience with a large reactor of the same design of the Oconee reactor, such analyses are not considered to be of sufficient technical validity to warrant relaxation of our normal practice, nor an adequate substitute for operating experience.

Until your operating and emergency procedures have been tested through use, and modified if considered necessary or desirable, and operating personnel have demonstrated their capability to suitably respond to

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Docket No. 50-269

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:

This refers to your letter of September 11, 1972, requesting a further review of our stated position regarding minimum crew size for multi-unit operation at the Oconee Station and technical comments on your basis for proposing fewer personnel than we believe necessary from a safety standpoint.

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Until your operating and emergency procedures have been tested through use, and modified if considered necessary or desirable, and operating personnel have demonstrated their capability to suitably respond to

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Docket No. 50-269

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:

This refers to your letter of September 11, 1972, requesting a further review of our stated position regarding minimum crew size for multi-unit operation at the Oconee Station and technical comments on your basis for proposing fewer personnel than we believe necessary from a safety standpoint.

As stated in our August 15, 1972 letter to you, significant station operating experience is necessary to provide assurance that a shift crew of the size you have proposed has the capability to maintain the Oconee Station in a safe condition during normal and abnormal operation, and to protect the health and safety of the public during potential emergency situations. We appreciate the fact that you have researched and analyzed your normal and emergency procedures to determine which of these are expected to be most demanding on shift personnel and have proposed a minimum crew size based on these considerations. However, in the absence of any operating experience with a large reactor of the same design of the Oconee reactor, such analyses are not considered to be of sufficient technical validity to warrant ~~the~~ relaxation of our ~~usual standards~~ nor an adequate substitute for operating experience.
normal practice

Until your operating and emergency procedures have been tested through use, and modified if considered necessary or desirable, and operating personnel have demonstrated their capability to suitably respond to

OFFICE ▶						
SURNAME ▶						
DATE ▶						

expected and unexpected occurrences during plant operation, we remain convinced that the crew size should, as a minimum, correspond to that specified in our letter to you of August 15.

Sincerely,

(Signed) John F. O'Leary

John F. O'Leary
Director of Licensing

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

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SEE PREVIOUS YELLOW FOR CONCURRENCES

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Docket No. 50-269

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:

Please refer to your letter to me of September 11, 1972 requesting that we further review our position regarding minimum crew size for multi-unit operation at the Oconee Station and your basis for proposing fewer personnel than required by our position.

As stated in our August 15, 1972 letter to you, experience is necessary to prove safe operation with crew sizes specified and to verify the validity of operating, abnormal condition, and emergency procedures. In the absence of station operating experience with a large B&W reactor, there is no defensible basis for changing our position.

We appreciate your concern for safe operation and anticipate your thorough evaluation of this matter after significant station operating experience has been acquired at Oconee.

Sincerely,

John F. O'Leary
Director of Licensing

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

OFFICE ▶	L:PWR-4	L:OSB	L:~~AKXX~~DDTR	L:DDRP	L:DDL	DL
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Form AEC-318 (Rev. 9-53) AECM 0240   U. S. GOVERNMENT PRINTING OFFICE : 1970 O - 405-346

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

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. BOX 2178

September 11, 1972

Mr. John F. O'Leary
Director of Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Mr. O'Leary:

Please refer to Mr. R. C. DeYoung's letter of August 15, 1972 discussing the minimum shift crew size for Oconee Unit 1, Units 1 and 2, and Units 1, 2, and 3. We wish to state our position that we believe the shift size requirements identified in this letter are unnecessarily large.

We have researched our normal and emergency procedures to determine which of these would be the most demanding on our shift personnel for a particular situation, and it was determined that the loss of control room would require the maximum personnel. On March 12, 1970 in Bethesda, Maryland, on July 15, 1972 in Bethesda, Maryland, and on July 12, 1972 at Oconee Nuclear Station, our operating personnel presented to members of your staff the steps that would be taken by shift members to shut down Oconee Unit 1 and 2 from outside the control room. Our analysis showed that only two operators were required to safely shut down both units and maintain them in a hot shutdown condition from outside the control room. We have proposed five operators per shift for Units 1 and 2.

Our proposed staffing for the Oconee units was based on detailed analysis which was derived from years of fossil experience including our newest supercritical units at Marshall Station which are successfully operated with two men per shift per unit; experience in operating the reactor at Carolinas-Virginia Tube Reactor; and experience in reactor operations at Oak Ridge National Laboratory. Our design of the control boards at Oconee is backed up by 50 man-years of reactor operating experience.

We believe that our proposals of five men per shift for Units 1 and 2 and eight men per shift for Units 1, 2, and 3 represent the optimum shift size designed to employ all shift members in meaningful operations while on duty. Dilution of responsibility with additional manpower can only lead to decreased experience and effectiveness per man and lower morale. The shift size as stated represents a minimum which would be on duty at all times and allows for no relief personnel. For special operations during the life of the plant and for initial startup of each Oconee unit, we propose to increase the shift size appropriately. These initial startup shift sizes have been previously discussed with your staff and are identified in Section 15, Technical Specifications of the FSAR.

Even though Duke Power Company has presented sufficient justification for our proposed shift staffing and has received no technical objection from the AEC, we are proceeding to train an adequate number of operators for the shift staffing as required by your August 15 letter. Your further review of this matter will be appreciated since we believe that we have demonstrated that the numbers now required by the AEC are unnecessarily large.

Very truly yours,

A. C. Thies

ACT:vr

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

AUG 1 5 1972

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

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

Gentlemen:

The Director of Licensing, AEC, stated in a letter to you dated February 13, 1970, the AEC position regarding the minimum shift crew size for Oconee Unit 1. We also expressed our thinking at that time regarding the shift complement for two- and three-unit operation. At recent meetings with our staff, you have presented your proposals for the minimum staff requirements for one-, two-, and three-unit operation.

We have evaluated the information you presented, considering the experience of the proposed Oconee staff, the control room layout, the use of the computer as a data-logger, and the three visual displays of computer information. We have also taken into account the fact that no operating experience with a large B&W nuclear reactor has been accumulated to date.

Assurance must be provided that the minimum shift crew is of sufficient size to maintain the Oconee Nuclear Station in a safe condition during normal and abnormal operations, and to protect the health and safety of the general public during postulated emergency conditions. For example, the crew must be available to perform manual manipulation of failed automatic control systems, maintain surveillance of backup instrumentation if normal instrumentation becomes inoperable, record data manually if the automatic data logging system is out of service, and initiate any procedural actions in response to annunicated alarms. The crew must be also able to supplement automatic action as necessary to mitigate the consequences of an abnormal condition to prevent its degradation into an emergency. It is here that the shift manpower may be taxed to the greatest extent.

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 licensed Senior Reactor Operator and two persons with Reactor Operator licenses. For the initial operation of Oconee Units 1 and 2, which have a common control room, a minimum of seven men per shift will be required. This will include two licensed Senior Reactor Operators and three persons with Reactor Operator licenses. For three-unit operation, we have concluded that the minimum shift complement shall be eleven. This crew is composed of three licensed Senior Reactor Operators, four persons with Reactor Operator licenses, and four unlicensed operators.

Requirements in addition to those already listed in Section 6.1.1.7 of the Oconee Unit 1 Technical Specifications shall be as follows:

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

2. A licensed Reactor Operator, with no responsibilities for an operating reactor, will be present to monitor the status of any shutdown reactor.

After significant station operating experience has been obtained, at your request, we will consider a proposal for a smaller minimum shift crew size. Experience is necessary to prove safe operation with the crew sizes specified above, and to verify the validity of both operating and abnormal condition procedures. Your next revision to the Technical Specification for Oconee Units 2 and 3 should reflect the minimum crew sizes and additions specified.

Sincerely,

R C DeYoung

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

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

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Docket No. 50-269

OCT 20 1972

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:

It is our understanding that the B&W topical reports, all dated September 1972, listed below are referenced in the Final Safety Analysis Report for the Oconee Nuclear Power Station:

BAW-10037, "Reactor Model Flow Testing," Revision 1
BAW-10038, "Prototype Vibration Measurement Program for Reactor Internals"
BAW-10050, "Evaluation of Oconee Reactor Component Failures"
BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibration"

Enclosed for your information is a copy of our letter to B&W requesting additional information on these topical reports.

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

Sincerely,

Original Signed by
Albert Schwencer
A. Schwencer, Chief
Pressurized Water Reactors Branch No. 4
Directorate of Licensing

Enclosure
Letter to B&W

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

OFFICE ▶			L:PWR	L:PWR	
SURNAME ▶			IPeltier;jm	ASchwencer	
DATE ▶			10/18/72	10/19/72	

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

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

OCT 18 1972

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

Dear Mr. Mallay:

We have completed our initial review of your topical reports, listed below, and find that we need additional information to complete our evaluation.

BAW-10037, Revision 1, "Reactor Model Flow Testing"
BAW-10038, "Prototype Vibration Measurement Program for Reactor Internals"
BAW-10050, "Evaluation of Oconee Reactor Component Failure"
BAW-10051, "Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibration"

The specific information required is listed in the enclosures.

In order to maintain our licensing review schedules for facilities referencing these topical reports we will need a prompt and completely adequate response. Please inform us within seven (7) days after receipt of this letter of your schedule for submitting the complete response. If your reply is not prompt or fully responsive to our requests it is highly likely that the overall schedule for completing the licensing reviews for these facilities will have to be extended.

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

Sincerely,

R. C. DeYoung

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

Enclosure:
Requests for Additional Information

REQUEST FOR ADDITIONAL INFORMATION

"DESIGN OF REACTOR INTERNALS AND INCORE INSTRUMENT NOZZLES FOR FLOW-INDUCED VIBRATION"

B&W REPORT BAW-10051, SEPTEMBER 1972

1. Describe the loading combinations and the analytical methods used to confirm the structural integrity of the instrumentation guide tubes. Provide the basis for the criteria that redesign is not necessary if two guide tubes fail during hot functional testing.

2. As shown in Table 3-3 (page 3-26) the cantilever part of the guide tube and the flow distributor assembly (vertical) have approximately the same first mode frequencies. The configuration shown in Figure 3-3 indicates that the vertical motion of the flow distributor may produce rotation and therefore lateral motion at the lower tip of the guide tube. Provide a summary of the dynamic analyses used to account for possible dynamic coupling of the guide tube and the flow distributor assembly. Include the efforts of cross flow on the cantilever portion of the guide tube. The associated cyclic bending stresses at the incore instrument nozzle should also be provided.

3. The shedding frequency used for computing the β value of the drag force acting on the incore instrument nozzles was actually based upon a two (2) inch diameter (page 3-5) of the lower portion. Since the upper portion is 1 1/8 inch diameter (β=1), provide a summary of the analysis to show that excessive response amplitudes of the instrument nozzles will not occur.

4. Provide the basis for assuming that the lowest mode deflection of the thermal shield is 0.06 inches.

5. Provide the basis for assuming that the amplitude of other predominate modes of the thermal shield are a function of the ratio of the frequencies squared to the first mode (page 3-14).

6. Provide the basis for neglecting the combined modal contribution effects in predicting the maximum radial deflection of the thermal shield under the hot functional testing and normal operational loadings (Table 3-5).

REQUEST FOR ADDITIONAL INFORMATION

"EVALUATION OF OCONEE REACTOR COMPONENT FAILURE"

B&W REPORT BAW-10050, SEPTEMBER 1972

1. As stated in page 4-12, the first mode frequency of the instrumentation guide tube is 250 Hz while the vortex shedding frequency is approximately 385 Hz, therefore, the first mode response may be excluded as a failure mode. However, higher modes may be in the range of the vortex shedding frequency or other forcing frequencies.

   (a) Provide a comparison of the higher mode guide tube frequencies with the shedding frequency.

   (b) Provide the criteria that was used for redesign of the instrumentation guide tubes.

   (c) Provide a discussion of other possible causes of failure, such as the mentioned radnom excitation of turbulence and the reactor coolant pump excitation. Include the effect of the pump shaft frequency of 20 Hz (page 4-9).

2. Provide a discussion on the following possible failure mode on the incore instrument nozzles: The core structure vibratory motion and the cross flow loading may produce a rotational vibration model in the guide tubes and associated lateral deformation of the lower tips. The lateral motion may produce vibratory contact with the inserted tip of the incore instrument nozzle and result in cyclic bending stresses at the bottom of the nozzle to failure.

REQUEST FOR ADDITIONAL INFORMATION

"PROTOTYPE VIBRATION MEASUREMENT PROGRAM FOR REACTOR INTERNALS"

B&W REPORT BAW-10038, SEPTEMBER 1972

1. Since flow-induced forcing functions have not been identified or postulated provide a description of the method that was employed to determine the predicted responses.

2. Supplement Table 6-1 by providing predicted readings or estimated stress levels at all sensor locations.

REQUEST FOR ADDITIONAL INFORMATION

"REACTOR MODEL FLOW TESTING"

B&W REPORT BAW-10037, REVISION 1, SEPTEMBER 1972

1. Verify the possible omission of the flow area term in the equation (1-1).

2. The flow frequency content and the related energy distribution was not determined by the measurements during the 1/6 scale model testing. Identify the contribution of this model testing to the postulation of forcing functions for response prediction analysis. Provide the basis for the use of the simple equation setforth on Page 3-4 of BAW-10051 to compute the shedding frequency since this model is valid only for a simple flow condition.

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DDavis

OCT 8 1972

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:

The Final Safety Analysis Report for Oconee Nuclear Station references Babcock & Wilcox (B&W) Topical Report entitled, "Study of Intergranular Separations in Low Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding," BAW-10013, dated December 1972 with revised pages 2-1, 2-2, 2-3, 5-3, and 5-4 dated February 15, 1972. A copy of our letter to B&W concerning the regulatory staff's evaluation of this report is enclosed for your information.

Sincerely,

Original Signed by
Albert Schwencer

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

Enclosure:
As stated

cc: William L. Porter, Esquire
Duke Power Company
P.O. Box 2173
422 S. Church Street
Charlotte, North Carolina 28201

CRESS T-6103 R-6
& T-7036 R-4-10
10/18/72

OFFICE ▶	L:PWR:4	L:PWR:4				
SURNAME ▶	DDavis:ttb	ASchwencer				
DATE ▶	10/18/72	10/ /72				

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

OCT 11 1972

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

Dear Mr. Mallay:

The regulatory staff has completed its review of Babcock & Wilcox Topical Report BAW-10013, dated December 1971, and entitled "Study of Intergranular Separations in Low Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding" with revised pages 2-1, 2-2, 2-3, 5-3 and 5-4 dated February 15, 1972. A summary of our review is enclosed for your information.

As a result of our review we have concluded that Topical Report BAW-10013 as revised February 15, 1972, will be acceptable by reference in applications for construction permits and operating licenses.

Sincerely,

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

Enclosure:
Topical Report Evaluation

TOPICAL REPORT EVALUATION

Report Identification: BAW-10013

Report Title: Study of Intergranular Separations in Low Alloy Steel Heat-Affected Zones Under Austenitic Stainless Steel Weld Cladding

Report Date: December 1971 with revised pages 2-1, 2-2, 2-3, 5-3 and 5-4 dated February 15, 1972

Originating Organization: Babcock and Wilcox

Reviewed By: Materials Engineering Branch, AEC Directorate of Licensing, September 1972

SUMMARY OF TOPICAL REPORT

Intergranular separations in low alloy steel heat-affected zones under austenitic stainless steel weld claddings have been detected in reactor vessels constructed by various manufacturers which were clad by high-heat-input weld cladding processes.

B&W investigations revealed that the subject flaws are present only in SA-508, Class 2 forgings manufactured to a coarse grain practice, and clad by high-heat-input submerged arc processes such as the 6 wire, strip, and the 2-wire. No anomalies were observed in SA-533 Grade B, Class 1 plate materials clad by any of the high-heat-input processes. Their fracture mechanics studies revealed that a critical crack size, on the order of 4 inches, is required to initiate fast fracture. This is several orders of magnitude greater than the maximum flaw size (i.e., 0.156 inch in depth and 0.500 inch in length) plus a predicted growth of 0.058 inch over a 40 year period due to design fatigue cycles, and it is considered by B&W that the subject flaws would have no detrimental effect on the integrity of B&W vessels under all operating conditions during the design life of the vessel.

SUMMARY OF REGULATORY EVALUATIONS

We consider the findings of B&W that the flaws are present only in SA-508 Class 2 coarse grain forgings, but not in SA-533 Grade B, Class 1 plates when clad by high-heat-input processes valid, since they have been confirmed by other investigators. We have reviewed B&W's fracture mechanics analysis and agree with their statement regarding critical crack size. However, their calculations on crack growth are based on controlled short term experiments, which do not necessarily reflect actual reactor operating conditions. However, even if the crack growth were several times greater than calculated, the initial maximum flaw size plus such a value would still be relatively insignificant when compared to the critical crack size, which was determined to be of approximately 4 inches.

SUMMARY OF REGULATORY POSITION

We concur with B&W's finding that the integrity of a vessel having flaws such as described in the subject report would not be compromised during the life of the plant. This report is acceptable and may be referenced in future case applications where similar underclad grain boundary separations have been detected. However, such flaws should be avoided and we recommend that future applicants state in their PSARs what steps they plan to take in this regard.

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Regulatory Operations (3)
Project Manager
Licensing Assistant
DDavis

Docket No. 50-269

OCT 12 1972

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:

Enclosed for your information is a copy of our letter to the Babcock & Wilcox Company (B&W) concerning the regulatory staff's evaluation of their topical report entitled "A Study of Discontinuities in Control Rod Drive Motor Tube Extensions," BAW-10047, Revision 1, dated August 1972. B&W has indicated that the information in this report is applicable for Oconee Nuclear Station Unit 2.

Sincerely,

Original Signed by
Albert Schwencer

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

Enclosure:
As stated

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

Appl RW

T-6003-R-03
T-7051 R-1-6
10/3/72

OFFICE ▶	L:PWR	L:PWR				
SURNAME ▶	DKD DDavis:ttb	AS ASchwencer				
DATE ▶	10/12/72	10/12/72				

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

OCT 11 1972

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

Dear Mr. Mallay:

The regulatory staff has completed its review of Babcock & Wilcox Topical Report BAW-10047, Revision 1, dated August 1972 and entitled "A Study of Discontinuities in Control Rod Drive Motor Tube Extensions." You have indicated that this revision replaces the original Topical Report BAW-10047, June 1972. A summary of our review is enclosed for your information.

As a result of our review, we have concluded that Topical Report BAW-10047, Revision 1, will be acceptable by reference in applications for construction permits and operating licenses provided the results of the testing and surveillance programs are consistent with the information supplied in this report. We understand that Oconee Nuclear Station Unit 2 has been selected for these programs. The results of the testing program should be reported to us within sixty (60) days upon completion of the tests and the results of the surveillance program should be reported within sixty (60) days upon the end of the first of the fuel cycles. The Topical Report should be revised or supplemented to include the results of these programs.

Sincerely,

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

Enclosure:
Topical Report Evaluation

TOPICAL REPORT EVALUATION

Report Identification: BAW-10047, Revision 1

Report Title: A Study of Discontinuities in Control Rod Drive Motor Tube Extensions

Report Date: August 1972

Originating Organization: Babcock & Wilcox

Reviewed By: Materials Engineering Branch, Mechanical Engineering Branch and Reactor Systems Branch, AEC Directorate of Licensing

SUMMARY OF TOPICAL REPORT

Some motor extension tubes exhibited localized variations in wall thickness and discontinuities on the inner surface, not detected by ultrasonic testing techniques certified to be in accordance with ASME Code, Section III. The investigation included metallographic and chemical examinations, specialized ultrasonic inspection techniques and fracture mechanics analysis.

Tubes meeting the Code-required minimum wall thickness based on the original design temperature of 650°F and having no indicated discontinuities deeper than 5% of the nominal-finish wall thickness will be accepted and returned to the field for service.

Where wall thickness permits, discontinuity indications greater than 5% of the nominal-finish wall thickness will be reduced by honing the I.D. surface of the tubes while maintaining the ASME Code, Section III required wall thickness based on the original design temperature. Tubes will then be reexamined ultrasonically for discontinuity depth and for proper wall thickness.

The remaining tubes withheld, because of wall thickness, will be re-evaluated based on a revised design temperature of 450°F, which is a more realistic but still conservative value of the maximum temperature at which the upper extension tubes will operate. The basis for this revised temperature has been established by extensive mockup testing at the Alliance Research Center and measurements at Oconee 1. A testing and surveillance program will also be implemented on one of the first

facilities utilizing the affected motor tube extensions in order to verify that the mockup tests accurately simulated the operating conditions of a typical facility.

SUMMARY OF REGULATORY EVALUATION

Repair of tubes which showed excessive discontinuities has been accomplished by honing the I.D., which is allowable in accordance with ASME Code, Section III, Class 1 Components, paragraph NB-2558.

Some motor tube extensions have localized variations and some repaired tubes also had a wall thickness below the Code-allowable thickness for a design condition of 650°F and 2500 psig. These tubes were re-evaluted on the basis of 450°F and 2500 psig, which is allowable by ASME Code, Section III, Class 1 Components, paragraph NB-3112.2. The actual maximum metal temperature which exists under the specified normal operating condition was determined by mockup testing and measurement at Oconee 1.

Specialized ultrasonic test inspection techniques have been developed to accurately define the depth of discontinuities. Indicated discontinuity depths no greater than 5% of the nominal wall thickness are allowed. This is in accordance with ASME Code, Section III, Class 1 Components, paragraph NB-2552.1.

A fracture mechanics analysis by Southwest Research Institute has concluded that a 90 mil notch in the worst case longitudinal direction is acceptable for the design life of the tube. We concur with the SRI findings.

REGULATORY POSITION

We conclude that the motor tube extensions accepted under the criteria proposed by B&W have an adequate margin of safety and that the subject report is acceptable provided the results of the testing and surveillance program are consistent with the mockup tests. These results should be documented by a revision or supplement to this Topical Report.

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

AUG 1 5 1972

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

Gentlemen:

The Director of Licensing, AEC, stated in a letter to you dated February 13, 1970, the AEC position regarding the minimum shift crew size for Oconee Unit 1. We also expressed our thinking at that time regarding the shift complement for two- and three-unit operation. At recent meetings with our staff, you have presented your proposals for the minimum staff requirements for one-, two-, and three-unit operation.

We have evaluated the information you presented, considering the experience of the proposed Oconee staff, the control room layout, the use of the computer as a data-logger, and the three visual displays of computer information. We have also taken into account the fact that no operating experience with a large B&W nuclear reactor has been accumulated to date.

Assurance must be provided that the minimum shift crew is of sufficient size to maintain the Oconee Nuclear Station in a safe condition during normal and abnormal operations, and to protect the health and safety of the general public during postulated emergency conditions. For example, the crew must be available to perform manual manipulation of failed automatic control systems, maintain surveillance of backup instrumentation if normal instrumentation becomes inoperable, record data manually if the automatic data logging system is out of service, and initiate any procedural actions in response to annunciated alarms. The crew must be also able to supplement automatic action as necessary to mitigate the consequences of an abnormal condition to prevent its degradation into an emergency. It is here that the shift manpower may be taxed to the greatest extent.

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SURNAME ▶						
DATE ▶						

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 licensed Senior Reactor Operator and two persons with Reactor Operator licenses. For the initial operation of Oconee Units 1 and 2, which have a common control room, a minimum of seven men per shift will be required. This will include two licensed Senior Reactor Operators and three persons with Reactor Operator licenses. For three-unit operation, we have concluded that the minimum shift complement shall be eleven. This crew is composed of three licensed Senior Reactor Operators, four persons with Reactor Operator licenses, and four unlicensed operators.

Requirements in addition to those already listed in Section 6.1.1.7 of the Oconee Unit 1 Technical Specifications shall be as follows:

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

2. A licensed Reactor Operator, with no responsibilities for an operating reactor, will be present to monitor the status of any shutdown reactor.

After significant station operating experience has been obtained, at your request, we will consider a proposal for a smaller minimum shift crew size. Experience is necessary to prove safe operation with the crew sizes specified above, and to verify the validity of both operating and abnormal condition procedures. Your next revision to the Technical Specification for Oconee Units 2 and 3 should reflect the minimum crew sizes and additions specified.

Sincerely,

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

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

SEE DOCKET NO. 50-269 FOR CONCURRENCES

OFFICE ▶	L:PWR-4	L:OS	L:PWR-4	L:AD/PWRs	L:OS	
SURNAME ▶	IAPeltier:emp	CRVanNiel	ASchwencer	RCDeYoung	DThompson	
DATE ▶	8/14/72	8/15/72	8/15/72	8/15/72	8/15/72	

JUL 26 1972

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, North Carolina 28201

Gentlemen:

Requirements for peaking factor reductions as a result of recent evaluations of emergency core cooling system performance indicate that present quadrant power tilt limits in the Technical Specifications for the Oconee 1 plant may be too large, permitting design peaking factors or safety limits to be exceeded before detection by the ex-core instrumentation. These limits on the quadrant power tilt are specified in the Control Rod and Power Distribution Limits section of your Technical Specifications. Two tilt limits, as determined by ex-core detectors, are specified: 1) a lower limit that provides a warning of potential violation of design peaking factors, and 2) an upper limit that provides a warning of potential violation of safety limits. Action appropriate to each situation is also specified.

In this regard, please submit by August 25, 1972, a reevaluation of the ability of the ex-core detectors to provide a warning of potential violation of design peaking factors and safety limits from the x-y plane power tilts for the Oconee 1 plant assuming the most adverse permissible axial peaking factor. If the results of your analysis show that lower quadrant tilt limits are needed to provide the above warnings, we will require appropriate changes to the Technical Specifications. In this event, your response should include your proposed changes to the Technical Specifications. Please inform us within seven (7) days after receipt of this letter of your confirmation of the above submittal date or the date you will be able to meet.

If you desire any discussion or clarification of the material requested, please contact us.

Sincerely,

Original Signed by
R. C. DeYoung

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

OFFICE ▶						
SURNAME ▶	See next page					
DATE ▶						

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

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R. W. Klecker
OGC
Rg Operations (3)
Project Mangger (I. Peltier)
Valeria Wilson (2)

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SURNAME ▶	IAPeltier	ASchwencer	VMoore	RCDeYoung		
DATE ▶	7/ /72	7/ /72	7/ /72	7/ /72		

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PWR Branch Chiefs
RWKlecker
OGC
Regulatory Operations (3)
Project Leader
Licensing Assistant
DDavis

JUL 17 1972

Docket Nos. 50-269
and 50-270

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:

Enclosed for your information is a copy of our letter to the Babcock & Wilcox Company (B&W) requesting additional information on their topical report entitled "Study of Discontinuities in Control Rod Drive Motor Tube Extensions," BAW-10047 dated June 1972. B&W has indicated that the information in this report is applicable for Oconee Nuclear Station Units 1 and 2.

Please contact us if you desire any discussion or clarification of our needs as specified in the enclosed letter.

Sincerely,

Original signed by
R. C. DeYoung

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

Enclosure:
B&W letter requesting
additional information

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

OFFICE ▶	L:PWR-4	L:PWR-4	L:PWR:AD			
SURNAME ▶	DDavis;ttb	Schwencer	RCDeYoung			
DATE ▶	7/14/72	7/14/72	7/15/72			

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

JUL 17 1972

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

Dear Mr. Mallay:

We have completed our initial review of your topical report BAW-10047, "Study of Discontinuities in Control Rod Drive Motor Tube Extensions," and find that we need additional information to complete our evaluation. The specific information required is listed in the enclosure. Much of this information was discussed with your representatives on July 7, 1972 at a meeting in Bethesda, Maryland.

In general, we find your investigation into the cause of the discontinuities and your subsequent inspection program adequate. However, before our final acceptance, we will require a completely adequate response to the information requested, revision or supplementation to your topical report to reflect this additional information, and incorporation of this topical report with revisions or supplements in all appropriate applications. In addition, we will also require a testing program and surveillance program. The testing program will verify the predictions of the temperature response of the CRDM tube extensions on a statistically significant number of assemblies during the preoperational and startup tests for the first facility utilizing these redesigned units. The surveillance program will assure that the maximum temperature of the CRDM tube extension does not approach the revised design temperature during plant operation.

The additional information and your response to our requirements should be provided in accordance with the review schedules we have established for those applications affected by this matter. We understand that you are aware of these schedules through your normal communication channels with the respective applicants and will inform them of your schedules for responding to our needs.

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

Sincerely,

R. C. DeYoung

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

Enclosure:
Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION

BABCOCK & WILCOX TOPICAL REPORT, BAW-10047

1. Provide more information describing the conditions of each test in Table 7-1, such as air flow conditions, insulation, thermal barrier, and drive activity.

2. Supply all data used to conclude that 450°F is the maximum temperature which could occur. Supply data used to justify statements 3, 4, and 5 on page 7-2.

3. Define the drive activities, cycling and intermittent tripping, to include the speed, frequency and length of the rod movement for each activity.

4. Was the test facility used for these tests shown in B&W topical report, BAW-10029? If so, provide the flow velocity in the vicinity of the CRD Motor Tube flange and compare this to the Oconee 1 hot functional test results and updated velocities from your additional model tests.

5. Supply all test results from Oconee 1 that would be appropriate for this subject.

6. Provide the maximum temperature increase expected for one full length trip.

7. For the situation where stator cooling water is lost, but electric power to the stator is still supplied, (1) show that the integrity of the reactor coolant pressure boundary and the functionality of the control rods are maintained, or (2) show that sufficient time and suitable instrumentation are available to prevent the above from happening. In either case provide a conservative "worst case" analysis including the effect of single failures.

8. Provide the "worst operating conditions as defined by the design specifications" referred to on page 7-2.

9. Revise your predicted maximum motor tube extension to include the maximum outlet temperature of 619°F.

10. State the validity of these tests and analyses for all types (A,B,C) of B&W CRD.

11. Discuss the effect of venting the CRD on the motor tube extension temperature.

12. Provide data to show the transient temperature behavior of the extension for the various test and operating conditions.

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IAPeltier

Docket No. 50-269

JUN 27 1972

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

Gentlemen:

The Atomic Energy Commission has issued an Order extending the latest completion date for Duke Power Company's Oconee Nuclear Station, Unit 1. In lieu of the latest completion date of June 30, 1972, as specified previously in Provisional Construction Permit No. CPPR-33, the latest completion date has been extended to February 28, 1973.

A copy of the Order which has been transmitted to the Office of the Federal Register for publication, is enclosed for your information.

Sincerely,

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

Enclosure:
Order Extending Construction
Completion Date

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

bcc: H. J. Mueller, GMR/H
H. McAlduff, ORO
J. A. Harris, IS
A. A. Wells, ASLBP
J. R. Buchanan, ORNL
T. W. Laughlin, DTIE
St. George T. Arnold, ORNL
N. H. Goodrich, ASLB
F. W. Karas, SECY

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SURNAME ▶	EGoulbourne:emp	IAPeltier	ASchwencer	RCDeYoung		
DATE ▶	6/22/72	6/22/72	6/22/72	6/22/72	6/24/72	

ATOMIC ENERGY COMMISSION

DOCKET NO. 50-269

DUKE POWER COMPANY
(Oconee Nuclear Station, Unit 1)

ORDER EXTENDING PROVISIONAL CONSTRUCTION PERMIT COMPLETION DATE

By application dated June 2, 1972, Duke Power Company requested an extension of the latest completion date specified in Provisional Construction Permit No. CPPR-33. The permit authorizes the construction of a pressurized water nuclear reactor, designated as the Oconee Nuclear Station, Unit 1, on the applicant's site in Oconee County, South Carolina, approximately eight miles northeast of Seneca, South Carolina.

Good cause having been shown for this extension pursuant to Section 185 of the Atomic Energy Act of 1954, as amended, and Section 50.55(b) of 10 CFR Part 50 of the Commission's regulations, IT IS HEREBY ORDERED THAT the latest completion date specified in Provisional Construction Permit No. CPPR-33 is extended from June 30, 1972 to February 28, 1973.

Dated at Bethesda, Maryland this 27th day of June 1972.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
A. Giambusso

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

50-269

APR 11 1972

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:

The regulatory staff has prepared the attached draft criteria regarding industrial security. This draft material reflects preliminary thinking by the staff on this subject. It is furnished to applicants for the purposes of illustrating the scope of security planning that is considered appropriate and identifying specific aspects of security planning that should be addressed in security plans. Applicants are encouraged to use the draft criteria as a "checklist" in the preparation of security plans.

These criteria are not and should not be regarded as firm requirements of the regulatory staff; conformance with every criterion is not essential.

Sincerely,
Original signed by
R. C. DeYoung

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

Enclosure:
Draft Criteria on
Industrial Security

cc: See next page

CRESS T-N4010 R-1 4/7/72						
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SURNAME ▶	IPeltier-hem	ASchwencer	RCDeYoung			LB
DATE ▶	4/11/72	4/11/72	4/11/72			

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1971-446-154

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

Mr. William S. Lee
Senior Vice President
Engineering & Construction
Duke Power Company
422 S. Church Street
P. O. Box 2178
Charlotte, N. C. 28201

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R. C. DeYoung, DRL
Licensing Assistant (2)
Project Leader

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SURNAME ▷						
DATE ▷						

DRAFT CRITERIA ON
INDUSTRIAL SECURITY

The Draft criteria reflects preliminary thinking by the staff on this subject. These criteria are not and should not be regarded as firm requirements of the regulatory staff. Conformance with every criteria is not essential.

I - APPROACH:

Criterion 1 - General Guidelines

Physical protection shall be based on controlling access to the facility and obtaining assistance from local alw enforcement authorities. As a minimum the plan shall combine at least the following elements: (i) em-employee investigation; (ii) a security force; (iii) a lighted fence or other lighted physical barrier which surrounds the facility; (iv) a lock and key system; (v) a system of intrusion alarms to protect each door or other opening in vital buildings; (vi) liaison with local law enforcement authorities; (vii) redundant tamper-resistant communication links with local law enforcement authorities.

Criterion 2 - Employee Intent on Sabotage

This class of opponent is considered to be neutralized by pre-employment investigation, by restrictions on package and vehicle access to the facility, and by procedures which minimize access to vital rooms, buildings, and structures.

Criterion 3 - Non-employee Intent on Sabotage

The fence surrounding the facility shall be considered as preventing irresolute opponents from gaining access to the outside of facility buildings. To the resolute oppoenet the fence shall be considered as offering only a brief time delay. Upon breaching the fence the opponent is considered to proceed to a vital building and attempt to gain access. The triggering of any portal or interior alarm shall be considered as announcing a sabotage threat until proven otherwise. Triggering of an alarm shall be sufficient cause to immediately alert local law enforcement authorities, who will respond with a force unless the alert is subsequently cancelled in accordance with a preagreed arrangement.

Criterion 4 - Physical Protection Program

A physical protection program shall be established at the earliest practicable time. Physical protection requirements shall be considered during each new facility design. The program shall be documented by written policies, procedures, and instructions.

Criterion 5 - Separate Document

The physical protection plan shall be an individual document physically separate and distanct from other elements of the applications.

Criterion 6 - Compatibility With Emergency Plan

The physical protection plan and related procedures shall be compatible with the emergency plan and related procedures.

II. PERSONNEL SCREENING:

Criterion 7 - Investigation

A background investigation shall be conducted to support the belief that each employee who has access to the protected area is trustworthy.

III. SECURITY FORCE:

Criterion 8 - Requirement For The Force

A force of guards and watchmen shall be established, organized, and trained to carry out actions specified in this Appendix and to act as a deterrent to those intent on industrial sabotage.

Criterion 9 - Response Time

Deployment of the security force shall permit one or more watchmen to arrive at any portal protected by an intrusion alarm within four (4) mintues.

IV. PHYSICAL BARRIERS:

Criterion 10 - Requirement For the Barrier

Vital buildings and structures shall be encompassed by a physical barrier which forms a protected area.

Criterion 11 - Location

A design aim shall be to install the physical barrier at least 50 feet from vital buildings and structures.

Criterion 12 - Buildings and Structures as Part of the Physical Barrier

Buildings or structures which are not vital and which offer intrusion protection equal to or better than the main segment of the physical barrier may be incorporated into and form a segment of the physical barrier. In such buildings or structures, each exterior door, window, or other portal shall be (i) protected by a watchman or (ii) locked and protected by an intrusion alarm.

Criterion 13 - Entrances

Each gate, door, or other intended entrance to the protected area shall be (i) under the control of a watchman, or (ii) locked and protected by an intrusion alarm.

Criterion 14 - Clear Area

The area from at least 25 feet inside to at least 25 outside of the protected area shall at all times be clear and free of objects that would aid in concealing a person.

Criterion 15 - Lighting

The physical barrier shall be lighted between sunset and sunrise so that the illumination (i) at any point 3 feet above the ground and 5 feet from the physical barrier exceed 0.2 foot candles, and (ii) at the ground level from at least 10 feet inside to at least 25 feet outside the barrier permits ready detection of persons.

V. ACCESS CONTROL:

Criterion 16 - Personnel Access to the Protected Area

Personnel access through gates, doors, and other entrances to the protected area shall be controlled by one or more watchmen.

Criterion 17 - Personnal Access to Vital Buildings, Rooms and Structures

Personnel requirements for each vital building, vital room, or vital structure shall be considered individually. Written procedures shall be prepared and implemented to minimize the number of persons who may enter or otherwise have access to each vital building, vital room, or vital structure.

Criterion 18 - Personal Vehicles

No personal vehicle shall be permitted inside the protected area.

Criterion 19 - Essential Vehicles

Vehicles and vehicular equipment essential to the operation, maintenance, and safety of the facility together with their cargoes shall be subject to search before entering the protected area. The physical protection plan shall define essential vehicles.

Criterion 20 - Package Search

Packages shall be subject to search before being permitted into the protected area.

VI. VITAL BUILDINGS AND STRUCTURES:

Criterion 21 - Portal Protection

In vital buildings and structures each door, window, or other portal which is accessible from the ground or any part of which is within 15 feet of the ground shall be (i) under the control of a watchman, or (ii) locked and protected by an intrusion alarm.

VII. INTRUSION ALARMS:

Criterion 22 - Interior Alarms

Interior intrusion alarms and associated components shall satisfy the requirements of Interim Federal Specification W-A-0054A (GSA-FSS) Alarm Systems, Interior, Security, Components for.

Criterion 23 - Exterior Alarms

The selection and installation of exterior intrusion alarms shall be guided by considerations similar to those set forth in Interim-Federal Specification W-A-00450A (GSA-FSS) Alarm Systems, Interior, Security, Components for. Additional considerations shall include (i) the local environmental conditions and (ii) the resistance offered by candidate alarms to tampering.

Criterion 24 - Alarm Termination

Each intrusion alarm shall terminate in central panels located in the control room and in at least one other place within the protected area. Each central panel shall provide for (i) an indication of which individual alarm is triggered, and (ii) a single master alarm which is triggered whenever one or more of the individual alarms is triggered.

VIII. LIAISON WITH LOCAL LAW ENFORCEMENT AUTHORITIES

Criterion 25 - Planning

Liaison with local law enforcement authorities (LLEA) shall be established with the aim of devising a comprehensive cooperative protection plan for the facility. This plan shall incorporate and develop the following features as a minimum: (i) facility personnel shall without delay or investigation alert the LLEA when any intrusion alarm activates; (ii) a prescribed delay period shall follow during which facility personnel shall investigate the cause for the alarm; (iii) a prearranged cancel arrangement for use only during the delay period shall be devised to avoid full LLEA response to postulated false alarms; and (iv) in the event the prearranged cancel is not received within the prescribed delay period, the LLEA shall respond by dispatching an armed force to the facility to investigate.

Criterion 26 - Communication Links

At least two independent communication links with the LLEA shall be established. At least one of these links shall utilize electromagnetic waves (e.g. radio, microwave link, or LASER) in a design which does not depend primarily on wire or cable transmission lines outside the protected area; this link is referred to as the "primary link." The links shall be accessible from each intrusion alarm central panel.

Criterion 27 - Protection of Communication Links

At least one control panel together with all other onsite components essential to the proper functioning of the primary link shall be protected as a vital structure.

Criterion 28 - Training

The security force and other appropriate employees shall be trained to operate communication links and to implement the plan required in Criterion 25 - Planning.

IX. TESTING AND MAINTENANCE:

Criterion 29 - Testing and Maintenance

Intrusion alarms, protected areas, and communications links utilized pursuant to the requirements of this part shall be tested and maintained as follows:

(a) Intrusion alarms, physical barriers, and communications links shall be maintained in operable and effective condition.

(b) Intrusion alarms and communications links shall be inspected and tested for operability and required functional performance daily.

(c) Physical barriers shall be inspected at intervals not exceeding four (4) hours.

X. OVERT THREATS:

Criterion 30 - Overt Threats

The plan shall develop measures for dealing with potential dangers such as bomb threats and civil disturbances.

XI. WRITTEN PROCEDURES:

Criterion 31 - Written Procedures

Written procedures shall be prepared and kept readily available for the use of guards, watchmen, and other employees in implementing the following criteria set forth in this Appendix:

(a) Criterion 9 - Response time

(b) Criterion 12 - Buildings and structures as part of the physical barrier

(c) Criterion 13 - Entrances

(d) Criterion 14 - Clear area

(e) All criteria under Section V - ACCESS CONTROL

(f) Criterion 21 - Portal protection

(g) Criterion 27 - Protection of communication links

(h) Criterion 28 - Training

(i) All criteria under Section IX - TESTING AND MAINTENANCE

(j) Criterion 30 - Overt threats

XII. RECORDS AND REPORTS:

Criterion 32 - Records

Pursuant to the requirements of this Appendix the following records shall be maintained at each facility:

(a) Except for regular employees, the name, address, and purpose of visit for each individual who enters the protected area.

(b) Results of all tests, inspections, and maintenance which have been performed on physical barriers, intrusion alarms, and communication links,

(c) Chronological list of false alarms from intrusion detectors identifying the circuit, area or portal protected, and action taken.

XIII. QUALITY ASSURANCE:

Criterion 33 - Quality Assurance

A separate section of the physical protection plan shall describe tests and inspections designed to demonstrate and confirm the intended performance of each commitment made in the physical protection plan.

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Docket No. 50-269

MAR 27 1972

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

Gentlemen:

A copy of Supplement No. 1 to the Safety Evaluation prepared by the Division of Reactor Licensing relating to your Oconee Nuclear Station Unit 1 is enclosed for your information.

Sincerely,

Original signed by
R. C. DeYoung

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

Enclosure:
Supplement No. 1 to the
Safety Evaluation

cc w/encl:
William L. Porter, Esq.
Duke Power Company
P. O. Box 2178
Charlotte, North Carolina 22801

bcc w/encls:
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SURNAME ▶	FWKaras:emp	ASchwencer	RCDeYoung	JGallo	PAMorris	
DATE ▶	3/23/72	3/27/72	3/27/72	3/ /72	3/ /72	

SUPPLEMENT NO. 1

TO

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

OCONEE NUCLEAR STATION UNIT 1

DOCKET NO. 50-269

MAR 2 4 1972

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 General

The Safety Evaluation by the Division of Reactor Licensing dated December 29, 1970, included a description of the Oconee Nuclear Station Units 1, 2, and 3 emergency core cooling system (ECCS) and our evaluation of the performance analysis of this system for the spectrum of break sizes up to and including the double-ended severance of the largest pipe of the reactor coolant pressure boundary. This evaluation was based upon ECCS analyses performed by the applicant and reported in the Oconee Nuclear Station operating license application. These analyses were performed using computer codes developed by B&W for analysis of large PWR reactors having safety injection systems.

Subsequently, the Atomic Energy Commission has reevaluated the theoretical and experimental bases for predicting the performance of emergency core cooling systems, including new information obtained from industry and AEC research programs in this field. As a result of this reevaluation, the Commission has developed interim acceptance criteria for emergency core cooling systems for light-water power reactors. These criteria are described in an Interim Policy Statement issued on June 25, 1971, and published in the Federal

Register on June 29, 1971, (36 F.R. 12247). By letter dated July 9, 1971, the Division of Reactor Licensing informed the applicant of the additional information that would be required for our evaluation of the performance of the Oconee Unit No. 1 ECCS in accordance with the Interim Policy Statement. The applicant provided a revised analysis of the Oconee Nuclear Station Unit 1 performance in a report titled "Multinode Analysis of B&W's 2568-MWt Nuclear Plants During a Loss-of-Coolant Accident" dated October 1971. The applicant also provided a supplement to this report, identified as Supplement 10 to the Oconee FSAR and dated December 17, 1971, that discusses the analysis of ECCS using unpressurized fuel pins in Oconee Units 1 and 3. The analysis was performed using the B&W Evaluation Model in conformance with the Interim Policy Statement, Appendix A, Part 4. The analysis was performed assuming the occurrence of a loss-of-coolant accident during operation at 102% of the requested power level of 2568 MW thermal.

1.2 Recent Experimental Information

Small-scale experiments have been conducted by the Aerojet Nuclear Corporation (formerly Idaho Nuclear Corporation) under contract to the U. S. Atomic Energy Commission as part of the reactor safety research and development work being carried out at the National Reactor Testing Station, principally to assist in the development of analysis methods to be used in the design and execution of the LOFT Project. During the past several years tests under this program have been performed to investigate the phenomena of blowdown of heated high pressure water from:

(1) a simulated reactor vessel with and without internals,

(2) a simulated reactor primary system with a vessel and single operating loop,

(3) a single loop system with an electrically-heated simulated reactor core, and

(4) a single loop, electrically-heated core system with accumulator ECC injection.

The results of some of these tests (LOFT Semiscale series 845-851) conducted in late 1970 and early 1971 showed that the analytical technique (RELAP-3 code) used by ANC at that time for blowdown analysis did not accurately predict the phenomena that occurred during blowdown after the cold ECCS water was introduced. The analysis had assumed that uniform and instantaneous mixing of the cold injection water and the hot residual fluid took place in the appropriate zones of the Semiscale system. The test showed that mixing is incomplete. In addition, the analysis did not predict that the cold ECCS water would be ejected from the vessel after injection. This phenomenon was observed in several cold leg Semiscale tests; the performance of the ECCS was satisfactory for the hot leg tests.

Although the LOFT Semiscale tests in this series have provided information for evaluation of the adequacy of analytical models, the results of these tests cannot be applied directly to describe the performance of pressurized water reactors following a loss-of-coolant accident because the test loop used was not designed so as to properly scale parameters affecting system performance. These include (1) the elevation head of the inlet annulus water, (2) the ratio of steam bubble diameters to the width of the vessel inlet annulus, (3) multiple flow loops, (4) relative loop and core resistances, (5) containment back pressure, (6) surface to volume ratios, (7) pump flow resistance, (8) steam generator model, (9) core heat rate, and (10) core internals.

Although the results of the small LOFT Semiscale experiments would not be expected to describe the performance of large power reactors, we have taken into account the results of these tests in establishing the acceptability of PWR interim evaluation models listed in Appendix A of the Commission's policy statement by including the conservative assumption that all of the water injected by the accumulators during blowdown is lost. Another consideration that led to this conservative assumption was the inadequacy of the currently used calculational techniques to predict accumulator water behavior during blowdown. As further experimental information or improved calculational techniques become available, this conservative assumption will be reevaluated.

2.0 DESCRIPTION OF EMERGENCY CORE COOLING SYSTEM

The Oconee Unit 1 emergency core cooling system (ECCS) consists of a high pressure injection system, an injection system employing core flooding tanks, and a low pressure injection system with external (to the containment) recirculation capability. Various combinations of these systems are employed to assure core cooling for the complete range of break sizes.

The high pressure injection system includes three pumps, each capable of delivering 450 gpm at 585 psig reactor vessel pressure and discharges to the reactor coolant inlet lines. One pump will provide the required minimum flow. The high pressure injection pumps are located in the auxiliary building adjacent to the containment. A concentrated boric acid solution from the boric acid water storage tank is provided to the suction side of the high pressure pumps during ECCS operation.

During normal reactor operation, the high pressure injection system recirculates reactor coolant for purification and for supply of seal water to the reactor coolant circulation pumps. The high pressure injection system is initiated at a low reactor coolant system pressure of 1500 psig or a reactor building pressure of 4 psig. Automatic actuation switches the system from normal to emergency operating mode. One of the three high pressure pumps is normally in operation. The system is designed to withstand a single failure of an active component without a loss of function.

The two core flooding tanks are located in the containment outside of the secondary shield. Each accumulator has a total volume of 1410 ft^3 with a minimum stored borated water volume of 1040 ft^3 pressurized with nitrogen to 600 psig. Each accumulator is connected to a separate reactor vessel core flooding nozzle by a flooding line incorporating two check valves and a motor operated normally open stop valve adjacent to the tank. The core flooding tanks will therefore inject water automatically whenever the pressure in the primary system is reduced below the core flooding tank pressure of 600 psig.

The low pressure injection system includes two pumps plus a spare pump each capable of delivering 3000 gpm at 100 psig reactor vessel pressure arranged to deliver water to the reactor vessel through two separate injection lines. One low pressure injection pump is capable of removing the heat energy generated after a loss-of-coolant accident.

The low pressure injection system pumps take their suction from the borated water storage tank (initially) and the reactor building emergency sump. The recirculation system components are redundant so as to withstand a single failure of an active or passive component without loss of function at the required flow.

The low pressure injection system is actuated on a low reactor coolant system pressure of 500 psig or a high reactor building pressure of 4 psig.

All of the ECCS subsystems can accomplish their function when operating on emergency (onsite) power as well as offsite power. If there is a loss of normal power sources the engineered safeguards power line is connected to the Keowee hydro unit which will start up and accelerate to full speed in 23 seconds or less. The pumps and valves of the injection system will be energized at less than 100% voltage and frequency to achieve the design injection flow rate within 25 seconds.

3.0 PERFORMANCE ANALYSIS OF EMERGENCY CORE COOLING SYSTEM

3.1 General

We have developed a set of conservative assumptions and procedures to be used in conjunction with the Babcock and Wilcox developed codes to analyze the ECCS functions. The assumptions and procedures used by B&W in analyzing the performance of the Oconee Unit No. 1 ECCS are described in Appendix A, Part 4 of the Interim Policy Statement published in the Federal Register on December 18, 1971 (F.R. Vol. 36, No. 244). Report BAW-10034 "Multinode Analysis of B&W's 2568 MWt Nuclear Plants During a Loss-of-Coolant Accident," October 1971, covers the performance of cores for which all fuel pins are pressurized. In addition, Supplement 10 of the FSAR presents the B&W LOCA analysis for cores having unpressurized pins as will be the case for Oconee Units 1 and 3. Unit 1 will have unpressurized and pressurized (a mixture) pins for the first two cycles and Unit 3 will have a mixture for the first cycle only. The analysis for the core with a mixture of pressurized and unpressurized pins resulted in a heat rate limit of 17.4 KW/ft. for the 102% power case to meet the 2300°F maximum cladding temperature criteria. The applicant submitted an analysis in Supplement 10 to support his claim that the 17.4 KW/ft. limitation would not result in power penalty and that there would be adequate margin below this limit through core life. For comparison, the analysis reported in BAW-10034 is based upon an 18.15 KW/ft peak linear heat rate for cores with pressurized pins only. The 8.55 ft^2 cold leg split is the limiting case accident with a peak temperature of 2284°F in the case of the mixed core and 2177°F in the case of the pressurized pin only case.

3.2 Analysis of the Blowdown Period

The applicant used the CRAFT and THETA 1-B computer codes for the analysis of the blowdown phase of the transient. Using these codes, and the evaluation model specified in Appendix A, Part 4, of the Interim Policy Statement, the applicant provided the reevaluation of the ECCS performance in compliance with the Commission's Interim Policy Statement.

For the blowdown portion of the accident, we have concluded that the applicant's analyses as reported in BAW-10034 and Supplement 10 of the FSAR, conform to the requirements specified in the Commission's Interim Policy Statement, Appendix A, Part 4.

3.3 Analysis of the Refill and Reflood Period

The applicant has considered the thermal behavior of the core during the refill and reflood portion of the loss-of-coolant accident, which is explained as follows:

(1) The vessel refill is provided initially by the core flooding tanks, and later by the pumping systems, and is assumed to start at the end of the blowdown period. The reactor vessel is assumed to be essentially dry at the end of the blowdown period, as a result of the conservative assumption in Appendix A, Part 4, of the Interim Policy Statement that water injected by the core flooding tanks prior to end-of-blowdown is ejected from the primary system.

(2) No heat transfer in the core is assumed until the level of water reaches the bottom of the core, at which time refill is considered complete and the core reflood starts.

The end of blowdown is 14.6 seconds after rupture for the 8.55 ft^2 cold leg double ended break and reflood (to the bottom of the core) is complete about 23 seconds after rupture. The end of blowdown is 18.7 seconds after rupture for the 8.55 ft^2 cold leg split and reflood is complete about 26 seconds after rupture.

(3) The reflood of the core is characterized initially by a rapid liquid level rise both in the core and in the vessel annulus until enough of the core is covered to generate substantial amounts of steam. The re-flood rate

increases and peaks in about 8.5 seconds after the end of blowdown at about 11 to 12 inches per second, then decreases rapidly leveling off at about 5.5 inches per second about 10 seconds after the end of blowdown. At 10 seconds after the end of blowdown, the water covers about 12 inches of the core for the case of a double ended cold leg break and 20 inches of the core for the case of a 8.55 ft^2 cold leg split.

(4) The amount of steam generated in the core together with the steam flow path resistance governs the rate of steam flow. The steam flow path is assumed to be only through the vent valves within the reactor vessel and no credit is taken for steam flow around the loop. The steam flow resistance also limits the rate of liquid rise in the core, but the annulus water level continues to increase until the liquid level reaches the inlet nozzle. Core flood tanks and low pressure injection system water is piped directly to the reactor vessel with no intervening reactor coolant system piping.

(5) The peak temperature reached in the transient for the limiting 8.55 ft^2 cold leg split occurs about 30 seconds after the break.

Based on our review of "Multinode Analysis of B&W's 2568 MWt Nuclear Plants During a Loss-of-Coolant Accident" BAW-10034, October 1971, and Supplement 10 to the FSAR we have concluded that the applicant has evaluated the refill and reflood events in an acceptable manner.

3.4 Results

The applicant has calculated the following temperatures for Oconee Unit No. 1 at 102% of a nominal power level of 2568 MWt:

Cold Leg Pipe Breaks		Peak Clad Temperatures (°F)	
(Area)	(Type Break)	Pressurized Pins	Unpressurized Pins
8.55 ft^2	(Double Ended)	2052	2072
8.55 ft^2	(Split)	2177*	2284*
3.0 ft^2	(Split)	1652	1662
0.5 ft^2	(Split)	1614	1561
Hot leg			
14.1 ft^2	(Split)	1621	1605

*Limiting case.

The total core metal-water reaction is less than 1% for each of the assumed pipe breaks.

4.0 CONCLUSIONS

On the basis of our evaluation of the additional B&W analyses, described in 3.1 above, we conclude that our acceptance criteria, as described in the Commission's Interim Policy Statement have been met:

(1) The maximum calculated fuel element cladding temperature does not exceed 2300°F.

(2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor.

(3) The calculated clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is so embrittled as to fail during or after quenching.

(4) The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long lived radioactivity remaining in the core.

These are the same acceptance criteria that we stated on pages 42 and 43 of our Safety Evaluation on Oconee Unit 1.

The results of the applicant's analyses for a loss-of-coolant accident initiated at a core power level of 2568 MWt show that the acceptance criteria are met on the basis of analyses performed in accordance with an acceptable evaluation model given in the Interim Policy Statement.

On the basis of our evaluation of the additional B&W analyses described in 3.1 above, we have determined that the conclusion that the emergency core cooling system is acceptable and will provide adequate protection for any loss-of-coolant accident, as set forth on page 43 of our Safety Evaluation dated December 29, 1970, remains applicable for the Oconee Nuclear Station reactors for core powers up to 2568 MWt.

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Docket No. 50-269

MAR 3 1972

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

Gentlemen:

Your letter of November 15, 1971 requested a deviation from Technical Specification 3.1.2.1 for zero power physics testing in Oconee Unit No. 1. Specifically you indicated a desire to perform the initial physics tests at 250°F and 900 psig and stated that the tests would be conducted at isothermal conditions in an unirradiated reactor vessel.

We have reviewed your request and have concluded that the proposed test conditions satisfy both our current AEC criteria and the recently adopted fracture toughness rules of the ASME Boiler and Pressure Vessel Code, Section III. On this basis you are authorized to perform the initial physics tests, as proposed at 250°F and 900 psig.

Please contact us if you desire further discussion of this matter.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

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

bcc: H. Mueller, GMR/H
JRBuchana, ORNL
JWLaughlin, DTIE
STRobinson, SECY

SEE PREVIOUS CONCURRENCES

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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:

Your letter of November 15, 1971 to Dr. Morris requested a deviation from Technical Specification 3.1.2.1 for zero power physics testing in Oconee Unit No. 1. Specifically you wish to perform the initial physics tests at 250°F and 900 psig.

This deviation is requested for the initial physics tests only and will be at isothermal conditions in an unirradiated reactor vessel. We concur in your request to perform the initial physics test at 250°F and 900 psig.

Please contact us if you desire further discussion of this matter.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

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

bcc: H. Mueller, GMR/H
J. R. Buchana, ORNL
J. W. Laughlin, DTIE
S. Robinson, SECY

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FWKaras (2)

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:

Your letter of November 15, 1971 to Dr. Morris requested a deviation from Technical Specification 3.1.2.1 for zero power physics testing in Oconee Unit No. 1. Specifically you wish to perform the initial physics tests at 250°F and 900 psig which would violate TS 3.1.2.1.

Since the deviation is requested for the initial test only which will be at isothermal conditions and in an unirradiated reactor vessel we suggest that you proceed on the basis that the deviation will be granted or that the technical specifications will be changed to provide for the above condition based on conservative technical justification.

Please contact us if you desire further discussion of this matter.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

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

bcc: H. Mueller, GMR/H
J. R. Buchana, ORNL
J. W. Laughlin, DTIE
S. Robinson, SECY

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Form AEC-318 (Rev. 9-53) AECM 0240 ☆U. S. GOVERNMENT PRINTING OFFICE: 1970—407-758 surfaces wetted by reactor coolant leakage to detect evidence of corrosion.

The following corrective measures shall be applied:

(a) An evaluation of the effect of any corroded area upon the structural integrity of the component shall be performed in accordance with the provisions of Article IS-311 of Section XI Code.

(b) Repairs of corroded areas, if necessary, shall be performed in accordance with the procedures of Article IS-400 of Section XI Code.

(3) The visual examinations of (1) and (2) above shall be conducted in conformance with the procedures of Article IS-211 of Section XI of the ASME Boiler and Pressure Vessel Code.

B. Corrective Measures

(1) The source of any reactor coolant leakage detected by the examinations of A(1) above shall be located by the removal of insulation where necessary and the following corrective measures applied:

(a) Normally expected leakage from component parts (e.g., valve stems) shall be minimized by appropriate repairs and maintenance procedures. Where such leakage may reach the surface of ferritic components of the reactor coolant pressure boundary, the leakage shall be suitably channeled for collection and disposal.

(b) Leakage from through-wall flaws in the pressure-retaining membrane of a component shall be eliminated, either by corrective repairs or by component replacement. Such repairs shall conform with the requirements of Article IS-400 of Section XI of the ASME Boiler and Pressure Vessel Code.

(2) In the event boric acid residues are detected by the examinations of A(2) above, insulation from ferritic steel components shall be removed to the extent necessary for examination of the component

Recommended PWR Inservice Inspection Program for Detection of Effects of Reactor Coolant Leakage

A. Inspection Requirements

(1) Prior to reactor startup following each refueling outage, all pressure-retaining components of the reactor coolant pressure boundary shall be visually examined for evidence of reactor coolant leakage while the system is under a test pressure not less than the nominal system operating pressure at rated power.

This examination (which need not require removal of insulation) shall be performed by inspecting (a) the exposed surfaces and joints of insulations, and (b) the floor areas (or equipment) directly underneath these components.

At locations where reactor coolant leakage is normally expected and collected (e.g., valve stems, etc.), the examination shall verify that the leakage collection system is operative and leaktight.

(2) During the conduct of the examinations of (1) above, particular attention shall be given to the insulated areas of components constructed of ferritic steels to detect evidence of boric acid residues resulting from reactor coolant leakage which might have accumulated during the service period preceding the refueling outage.

50-269

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S. H. Hanauer, DR
D. J. Skovholt, DRL
P. F. Collins, DRL
CO (2)
R. C. DeYoung, DRL
F. Schroeder, DRL
T. R. Wilson, DRL
PWR Branch Chiefs
Project Leader
Licensing Assistant (2)
H. Denton, DRL
R. Klecker, DRL
R. Maccary, DRS

FEB 10 1972

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:

As you know, an event occurred at a foreign pressurized water power reactor in which an unusual corrosion mechanism occurred when prolonged leakage of borated reactor coolant onto the reactor vessel head was undetected. Subsequent tests have indicated that this corrosion potential might exist under certain conditions when borated fluid has prolonged contact with carbon steel.

To preclude additional experiences of this type, an appropriate program of inservice inspection should be implemented to detect such effects at an early stage. The ASME Code Committee for Inservice Inspection is considering revision of the ASME Code for Inservice Inspection of Nuclear Reactors. However, as an interim measure, we believe that the inspection program described in the enclosure should be incorporated into your inservice inspection program.

Please advise us within thirty days concerning your adoption of the provisions of the enclosure.

Sincerely,

Original signed by
R. C. DeYoung

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

Enclosure:
PWR Inservice Inspection Program

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

CRESS
4105 22
4106 1&2
2/9/72

OFFICE ▶	DRL:PWR-4	DRL:AD:PWRs		
SURNAME ▶	AS ASchwencer:mlm	RCDeYoung		
DATE ▶	2/9/72	2/10/72		

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DJSkovholt
PFCollins
FWKaras (2)
A Schwencer
LOCAL - PDR

Docket No. 50-269

JAN 2 8 1972

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

Gentlemen:

The Atomic Energy Commission has issued an Order extending the latest completion date for Duke Power Company's Oconee Nuclear Station Unit 1. In lieu of the latest completion date of January 31, 1972 as specified previously in Provisional Construction Permit No. CPPR-33, the latest completion date has been extended to June 30, 1972.

A copy of the Order which has been transmitted to the Office of the Federal Register for publication, is enclosed for your information.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Order Extending Construction
Completion Date

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

bcc: H. J. McAlduff, ORO
H. Mueller, GMR/H
J. A. Harris, PI
R. L. Leith, OC
J. R. Buchanan, ORNL
T. W. Laughlin, DTIE
E. B. Tremmel, IP
J. Saltzman, SLR
N. H. Goodrich, ASLB
A. Nussbaumger, DML
S. T. Robinson, SECY

CRESS #02 OFFICE ▶	DRL:AD/PWRs	DRL:PWR-4	DRL:AD/PWRs	OGC	DRL	
M/C#218-194,etc. SURNAME ▶	FWKaras:ccls	ASchwencer	RCDeYoung		PAMorris	LB
1/26/72 DATE ▶	1/26/72	1/27/72	1/28/72	1/ /72	1/28/72	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1971 - 446 - 154

ATOMIC ENERGY COMMISSION

DOCKET NO. 50-269

DUKE POWER COMPANY

Order Extending Provisional Construction Permit Completion Date

By application dated December 20, 1971, Duke Power Company requested an extension of the latest completion date specified in Provisional Construction Permit No. CPPR-33. The permit authorizes the construction of a pressurized water nuclear reactor designated as the Oconee Nuclear Station Unit 1 at the applicant's site in Oconee County, South Carolina, approximately eight miles northeast of Seneca, South Carolina.

Good cause having been shown for this extension pursuant to Section 185 of the Atomic Energy Act of 1954, as amended, and Section 50.55(b) of 10 CFR Part 50 of the Commission's regulations, IT IS HEREBY ORDERED THAT the latest completion date specified in Provisional Construction Permit No. CPPR-33 is extended from January 31, 1972 to June 30, 1972.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Date of Issuance: JAN 28 1972

50-269

Docket No. 50-269

JAN 28 1972

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SHHanauer, DR	OGC
FSchroeder, DRL	CO (2)
TRWilson, DRL	FWKaras (2)
RSBoyd, DRL	IAPeltier
RCDeYoung, DRL	ACRS (16)
DJSkovholt, DRL	
HRDenton, DRL	
RWKlecker, DRL	

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

Gentlemen:

In our review of your "Structural Integrity Test Report of the Reactor Containment Building" report, dated October 29, 1971, which reports the structural integrity tests performed on Oconee No. 1 containment, we find the report weak in the following areas. We request that you provide more discussion of your experience with embedded gages and more description of the surface conditions (state of cracking, spalling, etc.) of the dome. These items should be addressed in detail taking into account previous experience with containments at other plants.

Sincerely,

Original signed by
R. C. DeYoung

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

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

OFFICE ▶	DRL:PWR-4	DRL:SEB	DRL:PWR-4	DRL:AD/PWRs	DRS/AD/ENG	
SURNAME ▶	IAPeltier:emp	FSchauer	ASchwencer	RCDeYoung	RR MACCARY	LB
DATE ▶	1/27/72	1/27/72	1/27/72	1/28/72	1/27/72	

Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE: 1970—407-758

JAN 26 1972

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

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production and Operation
P. O. Box 2187
Charlotte, North Carolina 28201

Gentlemen:

A problem regarding adequate separation of redundant instrumentation and control cables in the Oconee Nuclear Station was brought to your attention by our Division of Compliance. We observed the extent of this problem during a site visit on November 30, 1971. At a January 19, 1972 meeting with us in Bethesda, you indicated your intent to resolve this problem by taking the following actions:

1. Install "Glastic" fire resistant barriers to the bottom of Oconee Unit 1 cable trays in all areas where the minimum spacing between the cables in the bottom of one tray is less than three inches from the cables in the top of the tray immediately below it.

2. Institute a cable temperature checking program in the critical areas of cable tray overfill in Oconee Unit 1. This program will be carried out for a reasonable but limited period of time and will include temperature checks during initial startup, normal and adverse operating conditions.

3. Revise the FSAR to incorporate the above Unit 1 cable tray modifications and the cable temperature checking program and to show that for Oconee Units 2 and 3 the original cable separation criteria will be met including no cable tray overloading and a minimum of five inches rail-to-rail space between all vertical trays.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆U. S. GOVERNMENT PRINTING OFFICE: 1970—407-758

Based on our review of this matter, we conclude that your proposal as noted above is acceptable.

Sincerely,

Original signed by
R. C. DeYoung

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

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

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DRL Reading
PWR-4 Reading
RCDeYoung
TRWilson
FWKaras
IAPeltier
Compliance (2)
RWKlecker

OFFICE ▶	DRL:PWR-4	DRL:PWR-4	DRL:AD/PWRs	DRS		
SURNAME ▶	IAPeltier:emp	ASchwencer	RCDeYoung	VAMoore		
DATE ▶	1/24/72	1/24/72	1/26/72	1/26/72		

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

DEC 8 1971

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

Gentlemen:

During our continuing review of your application for an operating license for the Oconee Nuclear Station Units 1, 2, and 3, we submitted copies of your Pre-Operational Environmental Radioactivity Monitoring Program Report for the Oconee Nuclear Station to the U. S. Department of the Interior, Fish and Wildlife Service for their use. A copy of the Fish and Wildlife Service's comments on your report is enclosed for your information. Copies of the comments are also being sent to the appropriate State and local officials.

We concur with the Department of the Interior's comments and request that you inform us in detail how you will correct the deficiencies cited by the Department of the Interior.

Twenty copies of your reply to this letter should be submitted to us for our review and distribution to the Fish and Wildlife Service.

Sincerely,

Original signed by
R. C. DeYoung

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

Enclosure:
FWS ltr dtd 10/4/71

cc: See Attached List

OFFICE ▶	DRL:AD/PWRs	DRL:PWR-4	DREP	DRL:AD/PWRs		
SURNAME ▶	FWKaras:emp	ASchwencer	GABlanc	RCDeYoung		
DATE ▶	12/8/71	12/8/71	12/8/71	12/9/71		

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

cc: w/encl.

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

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

cc: w/o encl.

Mr. Daniel W. Slater, Chief
Division of River Basin Studies
Bureau of Sport Fisheries and Wildlife
U. S. Department of the Interior
Washington, D. C. 20240

OFFICE ▶						
SURNAME ▶						
DATE ▶						

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

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

Gentlemen:

During our continuing review of your application for an operating license for the Oconee Nuclear Station Units 1, 2, and 3, we submitted copies of your Pre-Operational Environmental Radioactivity Monitoring Program Report for the Oconee Nuclear Station to the U. S. Department of the Interior, Fish and Wildlife Service for their use. A copy of the Fish and Wildlife Service's comments on your report is enclosed for your information. Copies of the comments are also being sent to the appropriate State and local officials.

We concur with the Department of the Interior's comments and recommendations and therefore request that you implement them, including continuing to cooperate with appropriate Federal and State agencies in developing the necessary program for the preoperational and post-operational environmental monitoring surveys. Please inform us in detail how you have responded to the Department of the Interior's comments.

Twenty copies of your reply to this letter should be submitted to us for our review and distribution to the Fish and Wildlife Service.

Sincerely,

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

Enclosure

cc: See attached list

OFFICE ▶			DRL:AD/PWRs	DRL:PWR-4	DREP	DRL:AD/PWRs
SURNAME ▶			FWK FWKaras:emp	Schwencer	GABlanc	RCDeYoung
DATE ▶			12/7/71	12/ /71	12/ /71	12/ /71

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

ADDRESS ONLY THE DIRECTOR
BUREAU OF SPORT FISHERIES
AND WILDLIFE

50-269
50-270
50-287

United States Department of the Interior
FISH AND WILDLIFE SERVICE
BUREAU OF SPORT FISHERIES AND WILDLIFE
WASHINGTON, D.C. 20240

RECEIVED
OCT 6 1971
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

OCT 4 1971

Mr. Harold L. Price
Director of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Price:

This is in response to Mr. DeYoung's letter of July 12, 1971, transmitting copies of the Pre-Operational Environmental Radioactivity Monitoring Program Report for the Duke Power Company's Oconee Nuclear Station, South Carolina, AEC Docket Nos. 50-269, 270, and 287.

The environmental monitoring program for this station for the most part is adequate, but data for benthic organisms are omitted in some tables. The list of criteria for the selection of sampling locations as mentioned on pages 6 and 7 of the report appears adequate. However, in subsequent tables, listing samples and locations (summary, 2-1 and 2-1a) no data are shown for benthic organisms. Also, the distance from the radioactive waste discharge point to the nearest downstream sampling station is not clearly indicated.

Sincerely yours,

Willis King

Assistant
Director

4313

Docket File

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

DEC 6 1971

Docket No. 50-269

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

Gentlemen:

We have reviewed the evaluation of twenty-one damaged control rod drive mechanisms (CRDM's) as presented in your September 29, 1971 report, "Description of Cause and Correction of Damage to Control Rod Drive Mechanisms During Preoperational Testing of Oconee No. 1".

In general, we conclude that significant damage occurred to at least eleven CRDM's as a direct result of the presence of nitrogen gas in the region of the buffer piston at the bottom of the torque tube. Although not all drives within the probable gas bubble area were removed from the head (nozzle positions 10, 12, 13, 14, 16, 17, 18, 23, 24, and 25 were not removed), we believe that the autoclave tests provided a reasonable basis for the in-place inspection of all CRDM's not removed from the head. While it is not entirely clear, we presume that the autoclave trips used to establish threshold of damage as a function of feet of water displaced by gas at 2000, 400 and 0 psig were 100% withdrawal trips.

With the addition of some remarks, we concur in your planned corrective action as described in Section V of your report. In paragraph V-A, we understand this to mean that the CRDM's identified on pages A-9 through A-33 will not be used in Oconee 1. The limits described in paragraphs V-B.4 and V-B.5 should be made specific, on justifiable bases, and added to the Technical Specifications for Oconee Unit 1. Also, because of the consequences of a "dry scram", minimum trip times should be added, with supporting bases, to the Technical Specifications. The design changes described in paragraphs V-C.1 through V-C.4 should be added to the FSAR with an adequate description of how venting will be accomplished under all plant operating conditions.

LB

With regard to the potential for recurrence of a dry trip, we believe that, until further notice Duke Power Company should continue to monitor for gas build up on the center CRDM. Because of the unique, high location of the buffer piston on the Oconee CRDM's we believe significant operating experience should be obtained before concluding that no form of operational venting is required.

Please contact us if you desire any discussion or clarification of this matter.

Sincerely,

Original signed by
R. C. DeYoung

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

cc:
William L. Porter, Esq.
P. O. Box 2178
Charlotte, North Carolina 28201

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Docket
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PWR #4 Reading
R. C. DeYoung, DRL
D. Skovholt, DRL
E. G. Case, DRS
F. Karas, DRL
J. Knotts, OGC
Compliance (Engelken)
DTIE (Laughlin)
NSIC (Buchanan)
A. Schwencer, DRL
AEC PDR

OFFICE ▶	DRL:PWR-4	DRL:AD/PWRs	CO			
SURNAME ▶	Schwencer [redacted]	DeYoung	Engelken			
DATE ▶	11/30/71	12/4/71	12/2/71			

JUL 9 1971

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

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production & Operation
422 South Church Street
Charlotte, North Carolina 28201

Gentlemen:

On June 19, 1971 the AEC adopted interim acceptance criteria for the performance of emergency core cooling systems (ECCS) in light-water nuclear power plants. A copy of the Commission's interim policy statement on this matter is enclosed for your information. In accordance with Section IV.B. of the interim policy statement, before we can complete our evaluation of the ECCS for the Oconee Nuclear Station Units No. 1, 2 and 3, we need information to show that the system meets the general criteria of Section IV.A. using a suitable evaluation model. We are continuing discussions with representatives of the Babcock & Wilcox Company (B&W) that are directed toward establishing a suitable evaluation model using B&W computer codes for evaluation of plants incorporating a nuclear steam supply system designed by B&W.

The information that we need regarding analyses performed with a suitable evaluation model is:

(1) For the break size range, location and type mentioned in Appendix A, Part 1 of the interim policy statement, provide information pertaining to (a) the system pressure, (b) the hot-spot clad temperature, local mass velocity, fluid temperature, and heat transfer coefficient, (c) the core pressure drop, quality, and mass velocity, (d) the heat flux distribution in the hot channel, (e) the flow rates in the upper and lower plenums, (f) the flow rates in the broken and intact cold-leg and hot-leg piping, (g) the flow rate out of the break, and (h) percent clad metal-water reaction.

(2) Provide a detailed discussion of the calculation used to predict heat transfer during the reflood portion of the transient.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

(3) Discuss in detail any deviations in the evaluation model used in the foregoing studies from that described in Appendix A, Part 1 of the interim policy statement, if that model is used.

In addition, you should submit for our review any changes to the Technical Specifications for the plant that may be required on the basis of the results of your analyses.

When you have obtained the required information, please submit it as an amendment to your application.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
AEC Interim Policy Statement

cc: w/enclosure

Mr. Roy B. Snapp
1725 K Street, N.W.
Washington, D. C. 20006

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JBKnotts, OGC
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WNyer (2)
Seismic Design Consultant
ASchwencer

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SURNAME ▶	FWKaras:tls	CGLong	RCDeYoung	FSchroeder		PAMorris
DATE ▶	7/9/71	7/9/71	7/9/71	7/ /71		7/9/71

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

June 14, 1971

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

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

Gentlemen:

In our review of your Oconee Nuclear Station Units 1, 2 and 3, the degree of availability of the various sources of auxiliary electrical power has been the subject of considerable discussion. At present, we have focused on those sources which will be available at the time Unit 1 becomes operational. Taking into account the availability of backup auxiliary power from a Lee Station gas turbine, we conclude that there is reasonable assurance that no single failure in the offsite or onsite auxiliary power systems could result in a condition where one more failure would deprive Unit 1 of all auxiliary ac power.

To obtain this reasonable assurance, we conclude that until an additional viable source of onsite auxiliary power is shown to be available, every effort should be made to keep the Keowee hydro-station generators operational when Unit 1 is operating. Minor maintenance and routine inspections requiring removal of both hydro-generators from service should be performed when the Unit 1 reactor is subcritical.

With a view towards multiple nuclear unit operation, we understand that you intend to demonstrate that the Oconee Nuclear Units can, without reliance on other ac power sources, continue to supply all safety related station auxiliary electrical loads following a loss of the external grid when operating at or near rated power. Following our review of the results of such a demonstration, it is expected that when there are at least two Oconee nuclear units in operation, the complete Keowee hydro-station can be permitted to be shutdown for

OFFICE ▶						
SURNAME ▶						
DATE ▶						

limited periods to perform minor maintenance and routine scheduled inspections without requiring the nuclear units to be shutdown also.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

cc: Mr. Roy B. Snapp
1725 K Street, N.W.
Washington, D. C.

Distribution:
Docket File (3)
AEC PDR
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DR Reading
DRL Reading
PWR-2 Reading
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S. Hanauer
F. Schroeder
R. S. Boyd
R. C. DeYoung
D. Skovholt
T. R. Wilson
E. G. Case
R. R. Maccary
R. W. Klecker
DRS/DRL Branch Chiefs
F. W. Karas (3)
Attorney, OGC
ACRS (16)
W. Nyer (2)
A. W. Schwencer

OFFICE ▶	DRL:PWR-2	DRL:PWR-2	DRS	DRL:AD/PWRs	DRL	
SURNAME ▶	AWSchwencer:eag	GGLong	VAMoore	RCDeYoung	PAMorris	
DATE ▶	6/11/71	6/11/71	6/ /71	6/12/71	6/14/71	

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

Docket No. 50-269

JAN 11 1971

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production & Operation
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Gentlemen:

This supplements my letter to you dated December 31, 1970.

I am forwarding for your information a copy of a Safety Evaluation by the Division of Reactor Licensing dated December 29, 1970. The document relates to the operation of your Oconee Nuclear Station Unit 1 on your site in Oconee County, South Carolina.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
DRL Safety Evaluation
dated December 29, 1970

bcc: H. J. McAlduff, ORO
E. E. Hall, GMR/H
E. B. Tremmel
R. Leith, OC
J. R. Buchanan, ORNL
T. W. Laughlin, DTIE
A. A. Wells, ASLB
J. J. DiNunno, SAGMEA
J. Verme, SMM
D. A. Nussbaumer, DML
S. T. Robinson, SECY
J. D. Saltzman, SLR
G. I. Ertter
J. A. Harris, PI

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C. G. Long
N. L. Dube (w/8 encl.)
D. J. Skovholt
OGC
OC (2)
P. F. Collins
A. Schwencer
F. W. Karas (2)

CRESS OFFICE ▶	DRL:AD/PWRs	DRL:PWR-2	DRL:AD/PWRs	DRL	
T54, R11 SURNAME ▶	FWK FW Karas	CGL CG Long	RCDeY RC DeYoung	PAM PA Morris	
ebk DATE ▶	1/8/71	1/8/71	1/9/71	1/11/71	

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

Docket No. 50-269

JAN 11 1971

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production & Operation
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Gentlemen:

This supplements my letter to you dated December 31, 1970.

I am forwarding for your information a copy of a Safety Evaluation by the Division of Reactor Licensing dated December 29, 1970. The document relates to the operation of your Oconee Nuclear Station Unit 1 on your site in Oconee County, South Carolina.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
DRL Safety Evaluation
dated December 29, 1970

bcc: H. J. McAlduff, ORO
E. E. Hall, GMR/H
E. B. Tremmel
R. Leith, OC
J. R. Buchanan, ORNL
T. W. Laughlin, DTIE
A. A. Wells, ASLB
J. J. DiNunno, SAGMEA
J. Verme, SMM
D. A. Nussbaumer, DML
S. T. Robinson, SECY
J. D. Saltzman, SLR
G. I. Ertter
J. A. Harris, PI

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DRL Reading
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R. C. DeYoung
C. G. Long
N. L. Dube (w/8 encl.)
D. J. Skovholt
OGC
OC (2)
P. F. Collins
A. Schwencer
F. W. Karas (2)

d

CRESS OFFICE ▶	DRL:AD/PWRs	DRL:PWR-2	DRL:AD/PWRs	DRL	
T54, R11 SURNAME ▶	FWK FW Karas	CGL CG Long	RCD RC DeYoung	PAM PA Morris	
ebk DATE ▶	1/8/71	1/9/71	1/9/71	1/11/71	

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

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

DEC 2 3 1970

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production & Operation
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Dear Mr. Thies:

This is in response to your letters dated October 16, 1969, February 9, 1970, June 26, 1970, and July 9, 1970 requesting that certain information which you submitted to the Atomic Energy Commission be withheld from public disclosure.

We have determined that disclosure of the information contained in the documents listed below is not required in the public interest nor by 10 CFR Part 9, and would adversely affect the interests of The Babcock & Wilcox Company or Duke Power Company. Accordingly, we are withholding the information contained in the following documents from public inspection pursuant to Section 2.790 of 10 CFR Part 2.

BAW-10002 "Once-Through Steam Generator Research and Development Report" which summarizes development and testing completed to provide confidence the once-through steam generator will perform as expected and will give satisfactory service throughout its design life.

BAW-10002, Suppl. 1 - revised pages to BAW-10002, above.

BAW-10005, Rev. 1 "Internals Vent Valve Evaluation" which describes the internals vent valve evaluation program and the results of tests conducted on a prototype internals vent valve for use on PWR plants designed by B&W.

BAW-10008 Part 2, Rev. 1 "Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation" which presents information on the reactor vessel and fuel assembly dynamic models and on the tests performed to establish fuel assembly frequency and damping values used in the analyses.

BAW-10009 "Effect of Fuel Rod Failure on Emergency Core Cooling Effectiveness" which describes tests run to evaluate potential fuel

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rod cladding failure mechanisms: cladding swelling and perforation, brittle failure of cladding, and eutectic formation between zirconium in the cladding and iron and nickel in spacer grids. An analysis is performed to evaluate the effect of clad swelling on the ability to cool the core following a loss-of-coolant accident.

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Duke Power Company Application Amendment No. 14 dated July 9, 1970 which contains proprietary answers to AEC questions posed in letters dated February 13, 1970, and March 3, 1970.

Withholding of this information from public inspection, shall not, however, affect the right of persons properly and directly concerned to inspect it.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

DISTRIBUTION:
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DR Reading
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A. Schwencer

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T47, R06, 07

OFFICE ▶	AD/PWRs	DRL:PWR-2	OGC	DRL:AD/PWR	DRL	
SURNAME ▶	FWK FWKaras:blv	CGLong		RCDeYoung	PAMorris	
DATE ▶	12/16/70	12/ /70	12/2/70	12/17/70	12/ /70	

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969—O-364-598

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

DEC 2 2 1970

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production & Operation
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Dear Mr. Thies:

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BAW-10009 "Effect of Fuel Rod Failure on Emergency Core Cooling Effectiveness" which describes tests run to evaluate potential fuel

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Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

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T47, R06, 07

OFFICE ▶	AD/PWRs	DRL:PWR-2	OGC	DRL:AD/PWR	DRL	
SURNAME ▶	FWKaras:blv	CGLong		RCDeYoung	PAMorris	
DATE ▶	12/16/70	12/ /70	12/ /70	12/17/70	12/17/70	

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

JAN 19 1971

Mr. A. C. Thies
Vice President
Duke Power Company
Box 2178
Charlotte, North Carolina 28201

Dear Mr. Thies:

With regard to your letter of July 17, 1970, requesting an exemption for the use of respiratory protective equipment, we need additional information to complete our evaluation of your request. Most of the information required is set forth in 10 CFR 20.103 C(3) subparagraphs (i) through (iv) and in the Notice of Proposed Rule Making by the Atomic Energy Commission 10 CFR 20, Standards for Protection Against Radiation, "Exposure of Individuals to Concentrations of Radioactive Material in Restricted Areas," published in the Federal Register November 4, 1967 (Volume 32, Number 215, pages 15432 to 15434) (enclosed). Subparagraph 20.103 C(4) of the Notice of Proposed Rule Making further itemizes the information we need. In addition, we are currently requiring licensees to provide operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

The Notice of Proposed Rule Making, as published in 1967, includes an Appendix E which itemizes the protection factors that may be allowed for various types of equipment (see enclosure). On Appendix E of the enclosure we have noted three changes in the protection factors to reflect our current thinking. These protection factors are judged appropriate for respiratory protective equipment approved by the U. S. Bureau of Mines. Equipment not approved under U.S. Bureau of Mines Approval Schedules may be used only if the licensee demonstrates to the Commission by testing, or on the basis of reliable test information, that the

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material and performance characteristics of the equipment are at least equal to those afforded by U.S. Bureau of Mines approved equipment of the same type.

When the exemption request is granted, the Oconee Technical Specifications should be amended to provide information and requirements similar to those included in the Technical Specifications for Consumers Power Company's Palisades Plant (Docket No. 50-255) Section 6.4.5.

We currently anticipate that our action on your exemption request can be completed by the time of the issuance of the Oconee Unit 1 Operating License. Timely response to the information requested will help expedite our review of your request so that we can meet this schedule.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Notice of Proposed Rule
Making, AEC (10 CFR Part 20)

DISTRIBUTION:
Docket file
AEC PDR
DR READING
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PWR-2 Reading
CKBeck
MMMann
SHanauer
FSchroeder
RSBoyd
RCDeYoung
DSkovholt
TRWilson
EGCase
RRMaccary
RWKlecker
DRS, DRL Branch Chiefs
FWKaras (2)
Attorney, OGC
NThommasson
ASchwencer

See Previous Yellow for Additional Concurrences

OFFICE ▶					DRL	DRL
SURNAME ▶					FSchroeder	PAMorris
DATE ▶					1/7/71	1/10/71

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

Mr. A. C. Thies
Vice President
Duke Power Company
Box 2178
Charlotte, North Carolina 28201

Dear Mr. Thies:

With regard to your letter of July 17, 1970, concerning an exemption for use of respiratory protective equipment, we will need additional information in order to complete the processing of your request. Most of the details required are set forth in 10 CFR 20.103 C(3) subparagraphs (i) through (iv) and in the Proposed Rule Making by the Atomic Energy Commission 10 CFR 20, Standards for Protection Against Radiation, "Exposure of Individuals to Concentrations of Radioactive Material in Restricted Areas," published in the Federal Register November 4, 1967 (Volume 32, Number 215 pages 15432 to 15434) (enclosed). Subparagraph 20.103 C(4) of the Proposed Rule Making further itemizes the information which we need. In addition, we are currently requiring the licensee to provide operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

The Proposed Rule Making, as published in 1967, included an appendix which itemized the respiratory protective credit which may be allowed for various types of equipment (see enclosure). As noted in Appendix E of the enclosure three of the protection factors have been changed in order to provide you information which is consistent with our current thinking. These respiratory protection factors are judged appropriate for respiratory protective equipment approved by the U.S. Bureau of Mines. Equipment not approved under

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U.S. Bureau of Mines Approval Schedules may be used only if the licensee demonstrates to the Commission by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U.S. Bureau of Mines approved equipment of the same type.

When the exemption request is granted, the Oconee Technical Specifications should be amended to provide information and requirements similar to those included in the Technical Specifications for Consumers Power Company's Palisades Plant (Docket No. 50-255) Section 6.4.5.

We are currently anticipating the completion of processing your exemption request concurrent with the issuance of Oconee Unit 1 Operating License. Timely response to the information requested will help expedite our review of your request so that we can meet this schedule.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Proposed Rule Making,
AEC (10 CFR Part 20)

DISTRIBUTION:

Docket file	DSkovholt
AEC PDR	TRWilson
DR Reading	EGCase
DRL Reading	RRMaccary
PWR-2 Reading	RWKlecker
CKBeck	DRS, DRL Branch Chiefs
MMMann	FWKaras (2)
SHanauer	Attorney, OGC
FSchroeder	XXXXXXXX
RSBoyd	NThommasson
RCDeYoung	ASchwencer

OFFICE ▶	SE&SR/DRL	PWR-2/DRL	AD/PWRs	DRL	DRL	
SURNAME ▶	WNT NThommasson PWHowe:mlg	CGLong	RCDeYoung	FSchroeder	PAMorris	
DATE ▶	12/31/70	1/6/71	1/6/71			

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

December 23, 1970

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production & Operation
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Dear Mr. Thies:

This is in response to your letters dated October 16, 1969, February 9, 1970, June 26, 1970, and July 9, 1970 requesting that certain information which you submitted to the Atomic Energy Commission be withheld from public disclosure.

We have determined that disclosure of the information contained in the documents listed below is not required in the public interest nor by 10 CFR Part 9, and would adversely affect the interests of The Babcock & Wilcox Company or Duke Power Company. Accordingly, we are withholding the information contained in the following documents from public inspection pursuant to Section 2.790 of 10 CFR Part 2.

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DATE ▶						

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Withholding of this information from public inspection, shall not, however, affect the right of persons properly and directly concerned to inspect it.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

DISTRIBUTION:
AEC PDR (3)
Docket Files (3)
DR Reading
DRL Reading
PWR-2 Reading
R. C. DeYoung
F. W. Karas (2)
A. Schwencer

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T47, R06, 07

OFFICE ▶	AD/PWRs	DRL:PWR-2	OGC	DRL:AD/PWR	DRL
SURNAME ▶	FWKaras:blv	CGLong		RCDeYoung	PAMorris
DATE ▶	12/16/70	12/17/70	12/2/70	12/17/70	12/23/70

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

DEC 2 1970

Duke Power Company
Attn: Mr. Austin C. Thies
Vice President
Production & Operation
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Gentlemen:

During our review of your application for an operating license for the Oconee Nuclear Station, we requested the comments and recommendations of the U.S. Department of the Interior, Fish & Wildlife Service, concerning the radiological impact of the proposed facility on the areas of responsibility of the Fish & Wildlife Service.

A copy of the Fish & Wildlife Service Report, dated November 24, 1970, is enclosed for your information. Copies of this report are also being sent to the appropriate State and local officials. The radiological safety aspects of the material in the report will be considered in the evaluation of the safety of your project by the regulatory staff and the Advisory Committee on Reactor Safeguards. Our conclusions will be included in the safety evaluation which we will prepare prior to the issuance of an operating license for your nuclear power unit.

We call your attention to the recommendations of the Fish & Wildlife Service as they relate to the Oconee Nuclear Station and request that you cooperate with the Service in modifying and developing your detailed plans for radiological surveys.

The reports which you will submit on the preoperational surveys will continue to be evaluated by the Commission and by the Fish & Wildlife Service prior to the issuance of an operating license in this proceeding. The reports of surveys made after operations have begun will be similarly reviewed. You are requested to furnish these reports directly to the Commission in twenty copies, and we will forward copies of the reports to the Fish & Wildlife Service and make other appropriate distribution.

Matters discussed in the Fish & Wildlife Service Report properly relating to our responsibilities under the National Environmental Policy Act will be handled in accordance with the Interim Guidelines of the Council on appl'd

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DATE ▶						

Environmental Quality and our proposed revision of Appendix D to 10 CFR 50. To this end, our May 6, 1970 letter requested information from you on the environmental impact of the Oconee Nuclear Station plant. Matters relating to water quality will be handled in accordance with the sections of the Federal Water Pollution Control Act. Regarding these Acts, an applicant for a construction permit and operating license for any nuclear power plant which will discharge effluents into the navigable waters of the United States is required to provide the AEC with a certification from the State or interstate pollution control agency, or the Secretary of the Interior, as appropriate, that there is reasonable assurance that the plant will not violate applicable water quality standards. In the case of the Oconee Nuclear Station, where actual construction had lawfully commenced prior to the date of enactment of the Water Quality Improvement Act of 1970, the Act provides that a certification shall be required under Section 21 (b)(7) of this Act before April 3, 1973, or the permit to construct the facility or license to operate the completed facility will terminate on April 3, 1973. Thereafter, the AEC may not issue a permit or license for the plant until a water quality certification first has been received.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
F&WS ltr dtd 11/24/70

cc: See page 3

OFFICE ▶						
SURNAME ▶						
DATE ▶						

cc w/encl:

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

Mr. Reese A. Hubbard
County Supervisor of Oconee
County
Anderson, South Carolina 29622

cc w/o encl:

Mr. William M. White, Chief
Division of River Basin Studies
Bureau of Sport Fisheries and Wildlife
U.S. Department of the Interior
Washington, D.C. 20240

Distribution:
Docket File (3)
AEC PDR
DR Reading
DRL Reading
PWR-2 Reading
OGC
RSBoyd, DRL
RCDeYoung, DRL
DJSkovholt, DRL
JRTotter, DBM
PWHowe, DRL (2)
FWKaras, DRL (2)
ODParr, DRL

OFFICE ▶	DRL:AD/PWRs x7407	DRL:PWR-2	DRL:AD/PWRs	DRL:DIR		
SURNAME ▶	FWKaras:pt	CGLong	RCDeYoung	PAMorris		
DATE ▶	11/30/70	12/1/70	12/1/70	/ /70		

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

OCT 5 - 1970

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production & Operation
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Dear Mr. Thies:

A copy of a report dated September 23, 1970 from the Advisory Committee on Reactor Safeguards to Chairman Seaborg is enclosed for your information.

The report relates to the Committee's review of the Duke Power Company's application for a license to operate the Oconee Nuclear Plant Unit 1 in Oconee County, South Carolina.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
ACRS letter dated 9/23/70

Distribution:
AEC PDR
Docket Files (2)
DR Reading
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FWKaras (2)
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SURNAME ▶		FWK FWKaras	CGLong	RCDeYoung	PAMorris
DATE ▶		10/1/70	10/1/70	10/5/70	10/5/70

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ASchwencer

OCT 1 1970

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

Duke Power Company
ATTN: Mr. Austin C. Thies
Vice President
Production & Operation
Power Building
422 South Church Street
Charlotte, North Carolina 28201

Gentlemen:

In order to complete our review of your application for a facility license to operate the Oconee Nuclear Station Unit No. 1, it will be necessary that you provide financial data in accordance with Section 50.33(f) of 10 CFR Part 50, a copy of which is enclosed. This information should include:

1. Estimated annual costs of operating the nuclear facility for a five-year period.

2. Estimated costs of permanently shutting down the facility and maintaining it in a safe condition, if and when it may occur.

The above data should be filed as an amendment to your application with three copies signed under oath or affirmation and nineteen additional conformed copies. The information contained in your 1969 Annual Report which you recently submitted with your transmittal letter dated September 24, 1970, should be incorporated in your amendment by reference, and no further copies of your 1969 Annual Report need be submitted.

Sincerely,
Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
10 CFR Part 50

CRESS OFFICE ▶	DRL/AD/PWRs	DRL:PWR-2	OGC	DRL:AD/PWRs	OC Concurred by telephone FWK	DRL
T46 SURNAME ▶	FWK FWKaras:mlm	CGLong		RCDeYoung	CALovejoy	PAMorris
R3 DATE ▶	9/30/70	9/30/70	10/1/70	10/1/70	9/30/70	10/1/70

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