DEC 22 1975

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

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

The Commission has issued the enclosed Amendment No. 16, Technical Specification Change No. 26 for License No. DPR-38; Amendment No. 16 Technical Specification Change No. 21 for License No. DPR-47; and Amendment No. 13, Technical Specification Change No. 13 for License No. DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments are in response to your request dated January 15, 1975.

The amendment incorporates into the Oconee Nuclear Station Technical Specifications changes to the reporting requirements. Changes to your proposal were necessary to meet our requirements. These have been discussed with your staff. The technical specifications are based on Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4.

We request that you use the formats presented in the Appendices to Regulatory Guide 1.16, Revision 4, for reporting operating information and that you report events of the type described under the section "Events of Potential Public Interest". Instructions for using these reporting formats are contained in Regulatory Guide 1.16 (a copy is enclosed for your use), and AEC report OOE-SS-001 titled "Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File" (a copy of which was provided you previously). This report is modified by updated instructions dated December 8, 1975, which are enclosed. Copy requirements are summarized in Regulatory Guide 10.1, "Compilation of Reporting Requirements for Persons Subject to NRC Regulations", a copy of which is also enclosed. This Guide will assist you in identifying reports that are required by the Commission's regulations set forth in Title 10 Code of Federal Regulations but are not contained in your technical specifications. Reports that are required by the regulations have not been repeated in your technical specifications.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

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

Copies of the related Safety Evaluation and the Federal Register Notice also are enclosed.

Sincerely,

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
1. Amendment No. 16
2. Amendment No. 16
3. Amendment No. 13
4. Regulatory Guide 1.16
5. Updated Instructions
6. Regulatory Guide 10.1
7. Safety Evaluation
8. Federal Register Notice

cc w/enclosures:
Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Mr. Troy B. Conner
Conner & Knotts
1747 Pennsylvania Avenue, NW
Washington, D.C. 20006

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

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

cc w/enclosures & incoming:
Mr. Elmer Whitten
State Clearinghouse
Office of the Governor
Division of Administration
1295 Pendleton Street
Fourth Floor
Columbia, South Carolina 29201

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SURNAME ▶	GZech:mer	(see note dtd	RAPurple			
DATE ▶	12/16/75	12/3/75)	12/22/75			

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated January 15, 1975, complies with the standards and requirements of the Atomic Energy Act of of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-38 is hereby amended to read as follows:

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

"B. Technical Specifications

The Technical Specifications contained in Appendixes A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 26."

3. This license amendment is effective January 1, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 26 to the
Technical Specifications

Date of Issuance: DEC 22 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated January 15, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-47 is hereby amended to read as follows:

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

"B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised, by issued changes thereto through Change No. 21."

3. This license amendment is effective January 1, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 21 to the
Technical Specifications

Date of Issuance: DEC 22 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated January 15, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-55 is hereby amended to read as follows:

"B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 13."

3. This license amendment is effective January 1, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Attachment:
Change No. 13 to the
Technical Specifications

Date of Issuance: DEC 22 1975

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SURNAME➤						
DATE➤						

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 16 TO FACILITY LICENSE NO. DPR-38
CHANGE NO. 23 TO TECHNICAL SPECIFICATIONS;

AMENDMENT NO. 16 TO FACILITY LICENSE NO. DPR-47
CHANGE NO. 21 TO TECHNICAL SPECIFICATIONS;

AMENDMENT NO. 13 TO FACILITY LICENSE NO. DPR-55
CHANGE NO. 13 TO TECHNICAL SPECIFICATIONS

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

Revise Appendix A as follows:

Remove Pages	Insert New Pages
i	i
ii	ii
iii	iii
iv	iv
v	v
vi	vi
1-5	1-5 (blank)
3.1-19	3.1-19
--	3.1-19a
3.1-20	3.1-20
4.2-1	4.2-1
4.2-2	4.2-2
4.2-3	4.2-3
4.4-1	4.4-1
4.4-2	4.4-2
4.4-3	4.4-3
4.4-4	4.4-4
4.4-7	4.4-7
4.4-8	4.4-8
4.4-9	4.4-9
4.4-10	4.4-10
4.13-1	4.13-1
6.1-2	6.1-2
6.1-4	6.1-4
6.2-1	6.2-1
6.6-1 thru 6.6-12	6.6-1 thru 6.6-9

26/21/13

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3.1.8 Single Loop Restrictions

Specification

The following special limitations are placed on single loop operation in addition to the limitations set forth in Specification 2.3.

3.1.8.1 Single loop operation is authorized for test purposes only.

3.1.8.2 At least 23 incore detectors meeting the requirements of Technical Specification 3.5.4.1 and 3.5.4.2 shall be available throughout this test to check gross core power distribution.

3.1.8.3 The pump monitor trip setpoint shall be set at no greater than 50 percent of rated power.

3.1.8.4 The outlet reactor coolant temperature trip setpoint shall be set at no greater than 610^{o}F.

3.1.8.5 At 15 percent of rated power and every 10 percent of rated power above 15 percent, measurements shall be taken of each operable incore neutron detector and each operable incore thermocouple, reactor coolant loop flow rates and vessel inlet and outlet temperature, and evaluation of this data determined to be acceptable before proceeding to higher power levels.

3.1.8.6 A report covering single loop operation, permitted by Specification 3.1.8, shall be submitted within 90 days after completion of testing. This report shall include the data obtained together with analyses and interpretations of these data which demonstrate:

(1) Coolant flows in the idle loop and operating loop are as 26
predicted. 21

(2) Relative incore flux and temperature profiles remain es- 13
sentially the same as for four pump operation at each power level taking into account the reduced flow in single loop operation.

(3) Operating loop temperatures and flows are obtained which justify the revised safety system setting prescribed for the temperature and flow instruments located in the operating loop (which must sense the combined core flow plus the cooler bypass flow of the idle loop).

Subsequent single loop operation shall be contingent upon Commission approval.

Bases

The purpose of single loop testing is to (1) supplement the 1/6 scale model test information, (2) verify predicted flow through the idle loop, (3) verify that changes in power level do not affect flow distribution or core power

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distribution, and (4) demonstrate that limiting safety system settings (pump monitor trip setpoint and reactor coolant outlet temperature trip setpoint) can be conservatively adjusted taking into account instrument errors.

Limiting the pump monitor trip setpoint to 50 percent of rated power and the reactor coolant outlet temperature trip setpoint to 610°F to perform this confirmatory testing assures operation well within the core protective safety limits shown in Figure 2.1-3, Curve 2.

Incore thermocouples will be installed and data will be taken to check outlet core temperature profiles. These data will be used in evaluating test results.

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3.1.9 Low Power Physics Testing Restrictions

Specification

The following special limitations are placed on low power physics testing.

3.1.9.1 Reactor Protective System Requirements

a. Below 1720 psig shutdown bypass trip setting limits shall apply in accordance with Table 2.3-1A - Unit 1.
2.3-1B - Unit 2.
2.3-1C - Unit 3.

b. Above 1800 psig nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1A - Unit 1.
2.3-1B - Unit 2.
2.3-1C - Unit 3.

3.1.9.2 Startup rate rod withdrawal hold shall be in effect at all times. This applies to both the source and intermediate ranges.

Bases

Technical Specification 3.1.9.2 will apply to both the source and intermediate ranges.

The above specification provides additional safety margins during low power physics testing.

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4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the Reactor Coolant System pressure boundary.

Objective

To assure the continued integrity of the Reactor Coolant System pressure boundary.

Specification

4.2.1 Prior to initial unit operation, an ultrasonic test survey shall be made of Reactor Coolant System pressure boundary welds as required to establish preoperational integrity and baseline data for future inspections.

4.2.2 Post-operational inspections of components shall be made in accordance with the methods and intervals indicated in IS-242 and IS-261 of Section XI of the ASME Boiler and Pressure Vessel Code, 1970, including 1970 winter addenda, except as follows:

IS-261 Item	Component	Exception
1.4	Primary Nozzle to Vessel Welds	1 RC outlet nozzle to be inspected after approximately 3 1/3 years operation. 2nd RC outlet nozzle to be inspected after approx. 6 2/3 yrs. operation. 4 RC inlet nozzles and 2 core flooding nozzles to be inspected at or near end of interval
3.3	Primary Nozzle to Safe End Welds	Not Applicable
4.3	Valve Pressure Retaining Bolting Larger than 2"	Not Applicable
6.1	Valve Body Welds	Not Applicable
6.3	Valve to Safe End Welds	Not Applicable
6.6	Integrally Welded Valve Supports	Not Applicable
6.7	Valve Supports & Hangers	Not Applicable

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[redacted]

4.2.3 The structural integrity of the Reactor Coolant System boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated, including evaluation of comparable areas of the Reactor Coolant System.

4.2.4 The results of the Inservice Inspections performed pursuant to Specifications 4.2.1, 4.2.2, and 4.2.3 shall be reported to the Commission within 90 days of completion. 2 3
2 1
1 3

4.2.5 To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.

4.2.6 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.

4.2.7 The inservice inspection program shall be reviewed at the end of five years to consider incorporation of new inspection techniques and equipment which have been proved practical and the conclusions of this review and evaluation shall be discussed with the NRC/ORI

4.2.8 At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an in-place, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed, if the interval measured from the previous such inspection is greater than 6 2/3 years.

4.2.9 For Unit 1 and Unit 2, a B Type vessel specimen capsule shall be withdrawn after one year of operation and an A Type capsule shall be withdrawn after 11, 17, and 22 years of operation. The withdrawal schedules may be modified to coincide with those refueling outages or unit shutdowns most closely approaching the withdrawal schedule. Specimens thus withdrawn shall be tested in accordance with ASTM-E-185-70. For Unit 3, a B Type vessel specimen capsule shall be withdrawn after one year of operation and an A Type capsule shall be withdrawn after 7, 14, and 17 years of operation. The withdrawal schedules may be modified to coincide with those refueling outages or unit shutdowns most closely approaching the withdrawal schedule. Specimens thus withdrawn shall be tested in accordance with ASTM-E-185-72. The results of these examinations shall be reported to the Commission within 90 days of completion of testing. 2 3
2 1
1 3

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4.2.10 During the first two refueling periods, two reactor coolant system piping elbows shall be ultrasonically inspected along their longitudinal welds (4 inches beyond each side) for clad bonding and for cracks in both the clad and base metal. The elbows to be inspected are identified in B&W Report 1364 dated December 1970.

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, 1970, including 1970 winter addenda, edition. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The reactor vessel specimen surveillance program for Unit 1 and Unit 2 is based on equivalent exposure times of 1.8, 19.8, 30.6 and 39.6 years. The contents of the different type of capsules are defined below.

A Type	B Type
Weld Material	HAZ Material
HAZ Material	Baseline Material
Baseline Material	

For Unit 3, the Reactor Vessel Surveillance Program is based on equivalent exposure times of 1.8, 13.3, 26.7, and 30.0 years. The specimens have been selected and fabricated as specified in ASTM-E-185-72.

Early inspection of Reactor Coolant System piping elbows is considered desirable in order to reconfirm the integrity of the carbon steel base metal when explosively clad with sensitized stainless steel. If no degradation is observed during the two annual inspections, surveillance requirements will revert to Section XI of the ASME Boiler and Pressure Vessel Code.

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4.4 REACTOR BUILDING

4.4.1 Containment Leakage Tests

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leak Rate Tests

4.4.1.1.1 Design Pressure Leak Rate

The maximum allowable integrated leak rate, La, from the Reactor Building at the 59 psig design pressure, P_p, shall not exceed 0.25 weight percent of the building atmosphere at that pressure per 24 hours.

4.4.1.1.2 Testing at Reduced Pressure

The periodic integrated leak rate test may be performed at a test pressure, P_t, of not less than 29.5 psig provided the resultant leakage rate, Lt, does not exceed a pre-established fraction of La determined as follows:

a. Prior to reactor operation the initial value of the integrated leak rate of the Reactor Building shall be measured at design pressure and at the reduced pressure to be used in the periodic integrated leak rate tests. The leak rates thus measured shall be identified as Lpm and Ltm respectively.

b. Lt shall not exceed La(Ltm/Lpm) for values of (Ltm/Lpm) not greater than 0.7.

c. Lt shall not exceed $La(Pt/Pp)^{\frac{1}{2}}$ for values of (Ltm/Lpm) above 0.7.

d. If Ltm/Lpm is less than 0.3, the initial integrated test results shall be subject to review by the NRC to establish an acceptable value of Lt.

4.4.1.1.3 Conduct of Tests

a. The test duration shall be at least 24 hours, except that if both the following conditions are met, the test duration shall be at least 10 hours:

   (1) All test conditions, including the test procedure, shall be similar to the initial integrated leak rate tests.

   (2) When the test is terminated, building pressure shall have stabilized and shall not be increasing.

[redacted]

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b. Test accuracy shall be verified by supplementary means, such as measuring the quantity of air required to return to the starting point or by imposing a known leak rate to demonstrate the validity of measurements.

c. Closure of containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves without preliminary exercises or adjustment.

4.4.1.1.4 Frequency of Test

After the initial preoperational leak rate test, two integrated leak rate tests shall be performed at approximately equal intervals between each major shutdown for inservice inspection, to be performed at 10 year intervals. In addition, an integrated leak rate test shall be performed at each 10 year interval, coinciding with the inservice inspection shutdown.

4.4.1.1.5 Conditions for Return to Criticality

a. If Lt is not greater than 50 percent of the value permitted in 4.4.1.1.2, local leak rate testing need not be completed prior to return to criticality following a periodic integrated leak rate test.

b. If Lt is greater than 50 percent and not greater than 100 percent of the value permitted in 4.4.1.1.2, return to criticality will be performed conditioned upon demonstration that local leakage into the penetration room, measured at full design pressure, accounts for all leakage above 50 percent of that permitted by 4.4.1.1.2. If this cannot be demonstrated within 30 days of returning to criticality, the reactor shall be shut down.

c. If Lt is greater than 100 percent of the value permitted by 4.4.1.1.2, the unit shall not be made critical.

4.4.1.1.6 Corrective Action and Retest

If repairs are necessary to meet the criteria of 4.4.1.1.1 or 4.4.1.1.2, the integrated leak rate test need not be repeated, provided local leak rate measurements are made before and after repair to demonstrate that the leak rate reduction achieved by repairs reduces the overall measured integrated leak rate to an acceptable value.

4.4.1.1.7 Report of Test Results

The results of the initial Containment integrated leak rate test and subsequent 26
periodic tests shall be the subject of a summary technical report which shall 21
be submitted to the Commission within 90 days of completion of the test. 13

4.4.1.2 Local Leak Rate Tests

4.4.1.2.1 Scope of Testing

The local leak rate shall be measured for each of the following components:

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a. Personnel hatch
b. Emergency hatch
c. Equipment hatch seals
d. Fuel transfer tube seals
e. Reactor Building normal sump drain line
f. Reactor coolant pump seal outlet line
g. Reactor coolant pump seal inlet line
h. Quench tank drain line
i. Quench tank return line
j. Quench tank vent line
k. Normal makeup to Reactor Coolant System
l. High pressure injection line
m. Electrical penetrations
n. Reactor Building purge inlet line
o. Reactor Building purge outlet line
p. Reactor Building sample lines
q. Reactor coolant letdown line

4.4.1.2.2 Conduct of Tests

a. Local leak rate tests shall be performed at a pressure of not less than 59 psig.

b. Acceptable methods of testing are halogen gas detection, soap bubbles, pressure decay, hydrostatic flow or equivalent.

4.4.1.2.3 Acceptance Criteria

The total leakage from all penetrations and isolation valves shall not exceed 0.125 weight percent of the Reactor Building atmosphere per 24 hours.

4.4.1.2.4 Corrective Action and Retest

a. If at any time it is determined that the criterion of 4.4.1.2.3 above is exceeded, repairs shall be initiated immediately.

b. If conformance to the criterion of 4.4.1.2.3 is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed annually, except that:

a. The equipment hatch and fuel transfer tube seals shall be additionally tested after each opening.

b. The personnel hatch and emergency hatch outer door seals shall be tested at four-month intervals, except when the hatches are not opened during that interval. In no case shall the test interval be longer than 12 months.

[redacted]

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4.4.1.3 Isolation Valve Functional Tests

Quarterly, remotely-operated Reactor Building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during unit operation. The latter valves shall be tested during each refueling shutdown.

4.4.1.4 Annual Inspection

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any integrated leak rate test, to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak rate test. Results of the inspection shall be reported to the Commission within 90 days of completion. 26 21 13

4.4.1.5 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a local leak rate test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.4 and 4.4.1.2.3, respectively.

Bases

The Reactor Building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 286°F. Prior to initial operation, the containment is strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment is also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests verify that the leak rate from Reactor Building pressurization satisfies the relationships given in the specification.

The performance of a periodic integrated leak rate test during unit life provides a current assessment of potential leakage from the containment, in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner. The test pressure of 29.5 psig for the periodic integrated leak rate test is sufficiently high to provide an accurate measurement of the leak rate and it duplicates the preoperational leak rate test at 29.5 psig. The specification provides a relationship for relating the measured leakage of air at 29.5 psig to the potential leakage at 59 psig. The frequency of the periodic integrated leak rate test is normally keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leak rate tests is based on three major considerations. First is the low probability of leaks in the

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its significance to the load-carrying capability of the structure. The sheathing filler will be sampled and inspected for changes in physical appearance.

Wire samples shall be selected in such a manner that with the third inspection, wires from all nine surveillance tendons shall have been inspected and tested.

4.4.2.2 Inspection Intervals and Reports

For Unit 1, the initial inspection shall be within 18 months of the initial Reactor Building Structural Integrity Test. The inspection intervals, measured from the date of the initial inspection, shall be two years, four years and every five years thereafter or as modified based on experience. For Units 2 and 3 the inspection intervals measured from the date of the initial structural test shall be one year, three years and every five years thereafter or as modified based on experience. Tendon surveillance may be conducted during reactor operation provided design conditions regarding loss of adjacent tendons are satisfied at all times.

A quantitative analytical report covering results of each inspection shall be 26
submitted to the Commission within 90 days of completion, and shall especially 21
address the following conditions, should they develop: 13

a. Broken wires.

b. The force-time trend line for any tendon, when extrapolated, that extends beyond either the upper or lower bounds of the predicted design band.

c. Unexpected changes in corrosion conditions or sheathing filler properties.

4.4.2.3 End Anchorage Concrete Surveillance

a. The end anchorages and adjacent concrete surfaces of the surveillance tendons will be inspected. In addition, other locations for surveillance will be determined by information obtained from design calculations, prestressing records, observations, and deformation measurements made during prestressing.

b. The inspection interval will be approximately one-half year and one year after the operation of the unit and will occur during the warmest and coldest part of the year.

c. The inspections made shall include:

   (1) Visual inspection of the end anchorage concrete exterior surfaces.

   (2) A determination of the temperatures of the liner plate area or containment interior surface in locations near the end anchorage concrete under surveillance.

   (3) Measurement of concrete temperatures at specific end anchorage concrete surfaces being inspected.

(4) The mapping of the predominant visible concrete crack patterns.

(5) The measurement of the crack widths, by use of optical comparators or wire feeler gauges.

(6) The measurement of movements, if any, by use of demountable mechanical extensometers.

d. The measurements and observations shall be compared with those to which prestressed structures have been subjected in normal and abnormal load conditions and with those of preceding measurements and observations at the same location on the reactor containment.

e. The acceptance criteria shall be as follows:

If the inspections determine that the conditions are favorable in comparison with experience and predictions, the close inspections will be terminated by the last of the inspections stated in the schedule. If the inspections detect symptoms of greater than normal cracking or movements, an immediate investigation will be made to determine the cause.

f. Results of the inspection shall be reported to the Commission within 90 days of completion. 26 21 13

4.4.2.4 Liner Plate Surveillance

4.4.2.4.1 The liner plate will be examined prior to the initial pressure test in accessible areas to determine the following:

a. Location of areas which have inward deformations. The magnitude of the inward deformations shall be measured and recorded. These areas shall be permanently marked for future reference and the inward deformations shall be measured between the angle stiffeners which are on 15-inch centers. The measurements shall be accurate to $\pm$ 0.01 inch. Temperature readings shall be obtained on both the liner plate and outside containment wall at the locations where inward deformations occur.

b. Locations of areas having strain concentrations by visual examination with emphasis on the condition of the liner surface. The location of these areas shall be recorded.

4.4.2.4.2 Shortly after the initial pressure test and approximately one year after initial startup, a re-examination of the areas located in Section 4.4.2.4.1 shall be made. Measurements of the inward deformations and observations of any strain concentrations shall be made.

4.4.2.4.3 If the difference in the measured inward deformations exceeds 0.25 inch (for a particular location) and/or changes in strain concentration exist, an investigation shall be made. The investigation will determine any necessary corrective action.

4.4.2.4.4 The surveillance program shall be discontinued after the one year after initial startup inspection if no corrective action was needed. If corrective action is required, the frequency of inspection for a continued surveillance program shall be determined.

4.4.2.4.5 Results of the surveillance shall be reported to the Commission within 90 days of completion.

Bases

Provisions have been made for an in-service surveillance program, covering the first several years of the life of the unit, intended to provide sufficient evidence to maintain confidence that the integrity of the Reactor Building is being preserved. This program consists of tendon, tendon anchorage and liner plate surveillance.

To accomplish these programs, the following representative tendon groups have been selected for surveillance:

Horizontal - Three 120° tendons comprising one complete hoop system below grade.

Vertical - Three tendons spaced approximately 120° apart.

Dome - Three tendons spaced approximately 120° apart.

The inspection during this initial period of at least one wire from each of the nine surveillance tendons (one wire per group per inspection) is considered sufficient representation to detect the presence of any wide spread tendon corrosion or pitting conditions in the structure. This program will be subject to review and revision as warranted based on studies and on results obtained for this and other prestressed concrete reactor buildings during this period of time.

REFERENCES

(1) FSAR Section 5.6.2.2

4.4.3 Hydrogen Purge System

Applicability

Applies to testing Reactor Building Purge System.

Objective

To verify that this system and components are operable.

Specification

4.4.3.1 Operating Tests

An in-place system test shall be performed annually. This test shall consist of a visual inspection, hook-up of the system to one of the three reactor buildings, a flow measurement using flow instruments in the portable purging station and pressure drop measurements across the filter banks. Flow shall be design flow or higher, and pressure drops across the filter bank shall not exceed two times the pressure drop when new. Fan motors shall be operated continuously for at least one hour, and valves shall be proven operable. This test shall demonstrate that under simulated emergency conditions the system can be taken from storage and placed into operation within 48 hours.

4.4.3.2 Filter Tests

Annually, leakage tests using DOP on HEPA units and Freon-112 (or equivalent) on charcoal units shall be performed at design flow on the filter. Removal of 99.5% DOP by each entire HEPA filter unit and removal of 99.0% Freon-112 (or equivalent) by each entire charcoal absorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing.

4.4.3.3 H_2 Detector Test

Hydrogen concentration instruments shall be calibrated annually with proper consideration to moisture effect.

Bases

The purge system is composed of a portable purging station and a portion of the Penetration Room Ventilation System. The purge system is operated as necessary to maintain the hydrogen concentration below the control limit. The purge discharge from the Reactor Building is taken from one of the Penetration Room Ventilation System penetrations and discharged to the unit vent. A suction may be taken on the Reactor Building via isolation valve PR-7 (Figure 6-5 of the FSAR) using the existing vent and pressurization connections.

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4.13 FUEL SURVEILLANCE

Applicability

Applies to the fuel surveillance program for fuel rods of Unit 1.

Objective

To specify the fuel surveillance program for fuel rods.

Specification

4.13.1 Visual Inspection

Two (2) Oconee Unit 1 fuel assemblies will be designated for visual inspection. These same assemblies will be inspected during each of the first three refuelings of Unit 1. Underwater viewing devices will be used to determine that the fuel rods have maintained their structural integrity.

4.13.2 Dimensional Examination

Measurements of the length and outside diameter will be made on selected peripheral rods of the following fuel assemblies of the first core of Unit 1 both prior to operation and at the times specified:

a. One assembly after the first cycle.

b. Four assemblies after the second cycle.

c. Two assemblies after the third cycle.

4.13.3 Results of the fuel surveillance program shall be submitted to the Commission within 90 days of completion of the program. 26 13
21

Bases

This fuel surveillance program provides substantiating information for the first core in the present generation of B&W reactors. It provides for examination of fuel rods at the end of the first, second, and third cycles of Unit 1 to determine if fuel rods have maintained their integrity and to determine the extent, if any, of dimensional changes in diameter and length.

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c. Quorum

The chairman plus two members shall constitute a quorum.

d. Responsibilities

The committee shall have the following responsibilities:

1. Review all new procedures or changes to existing procedures determined by the station Manager or his designate to affect operational safety.

2. Review station operation and safety considerations.

3. Review reportable occurrences and violations of Technical Specifications and make recommendations to prevent recurrence. | 26/21/13

4. Review all proposed tests that affect nuclear safety or radiation safety.

5. Review proposed changes to Technical Specifications and safety-related changes or modifications to the station design.

e. Authority

The Station Review Committee shall make recommendations to the station Manager regarding Specification 6.1.2.1-d.

f. Records

Minutes of all meetings of the committee shall be kept at the station, and copies shall be sent to the station Manager, Vice President, Steam Production, and the chairman of the Nuclear Safety Review Committee. | 26/21/13

6.1.2.2 Nuclear Safety Review Committee

a. The Executive Vice President and General Manager shall appoint a Nuclear Safety Review Committee having responsibility to verify that operation of the station is consistent with company policy and rules, approved operating procedures, and license provisions; to review important proposed station changes, and tests; to verify that abnormal occurrences and unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and to detect trends which may not be apparent to a day-to-day observer.

b. The activities of the Nuclear Safety Review Committee shall be guided by a written charter that contains the following:

Subjects within the purview of the committee
Responsibility and authority
Mechanisms for convening meetings
Provisions for use of specialists or subgroups

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f. Meeting Frequency:

The committee shall meet at least three times per year at intervals not to exceed five months and as required on call by the chairman. During the period of initial operation, this committee shall meet at least once per calendar quarter.

g. Quorum:

The chairman or vice-chairman plus three members, or appointed alternates, shall constitute a quorum. No more than a minority of the quorum shall have direct line responsibility for station operation.

h. Meeting Minutes:

Minutes of all scheduled meetings of the committee shall be prepared and shall identify all documentary materials reviewed. These minutes shall be formally approved, retained, and also promptly distributed to the Executive Vice President and General Manager; Senior Vice President, Engineering and Construction; Senior Vice President, Production and Transmission; Vice President, Design Engineering; Vice President, 26/21/13
Steam Production; and station Manager. A copy of these minutes shall be kept on file at the station.

i. As a safety review to the normal operating organization, the committee shall review the following:

1. Proposed tests and experiments, and results thereof, when these constitute an unreviewed safety question defined in 10CFR50.59.

2. Proposed changes in equipment or systems which constitute an unreviewed safety question defined in 10CFR50.59, or which are referred by the operating organization.

3. All requests to the NRC/DRL for changes in Technical Specifications or license that involve unreviewed safety questions as defined in 10CFR50.59. 26/21/13

4. Violations of statutes, regulations, orders, Technical Specifications, license requirements, or internal procedures, or instructions having safety significance as determined by the NSRC.

5. Reportable Occurrences as defined in 6.6.2.1 of these specifications. 26/21/13

6. Special reviews or investigations as required by the Vice President President, Steam Production, or the station Manager. 26/21/13

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6.2 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE

6.2.1 Any reportable occurrence shall be investigated promptly by the station Manager.

6.2.2 The station Manager shall promptly notify the Vice President, Steam Production, of any reportable occurrence. The Station Review Committee shall review a written report which shall describe the circumstances leading up to and resulting from the occurrence and shall recommend appropriate action to prevent or minimize the probability of a recurrence.

6.2.3 The Station Review Committee report shall be submitted to the Nuclear Safety Review Committee for review of any recommendations. Copies shall also be sent to the station Manager and the Vice President, Steam Production.

6.6 STATION REPORTING REQUIREMENTS

6.6.1 Routine Reports

The following reports shall be submitted to the Director, Office of Inspection and Enforcement Region II, Atlanta, Georgia.

6.6.1.1 Startup Report

A summary report of unit startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the facility license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the unit. Startup reports shall be submitted (1) within 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) nine months following initial criticality, whichever occurs first. If a startup report does not cover all three events, i.e., initial criticality, completion of the startup test program and resumption or commencement of commercial power operation, supplementary reports shall be submitted at least every three months until all three events are completed.

6.6.1.2 Annual Operating Report

Routine operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to April 1 of each year. The initial report shall be submitted prior to April 1 of the year following initial criticality.

Each annual operating report shall provide the following:

a. Operations Summary

(1) A narrative summary of operating experience during the report period relating to safe operation of the facility, including safety-related maintenance not covered in 6.6.1.2.a(2e)

(2) For each outage or forced reduction in power[1/] of over 20 percent of design power level where the reduction extends for greater than four hours.

1/The term "forced reduction in power" is defined as the occurrence of a component failure or other condition which requires that the load on the unit be reduced for corrective action immediately or up to and including the very next weekend. Note that routine preventive maintenance, surveillance and calibration activities requiring power reductions are not covered by this section.

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(a) the proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction);

(b) a brief discussion of (or reference to reports of) any reportable occurrences pertaining to the outage or power reduction;

(c) corrective action taken to reduce the probability of recurrence, if appropriate;

(d) operating time lost as a result of the outage or power reduction (for scheduled or forced outages,2/ use the generator off-line hours; for forced reductions in power, use the approximate duration of operation at reduced power);

(e) a description of major safety-related corrective maintenance performed during the outage or power reduction, including the system and component involved and identification of the critical path activity dictating the length of the outage or power reduction; and

(f) a report of any single release of radioactivity or unusual radiation exposure specifically associated with the outage which accounts for more than 10 percent of the allowable annual values.

b. Changes, Tests and Experiments

A brief description and the summary of the safety evaluation for those changes, tests, and experiments carried out without prior Commission approval pursuant to the provisions of 10CFR50.59.

c. Reporting of Radioactive Effluent Releases 3/

Data shall be reported to the Commission in a form similar to that shown in Table 6.6-1 and shall include the following:

(1) Gaseous Releases

(a) Total radioactivity (in curies) releases of noble and activation gases.

(b) Maximum noble gas release rate during any one-hour period.

(c) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

2/The term "forced outage" is defined as the occurrence of a component failure or other condition which requires that the unit be removed from service for corrective action immediately or up to and including the very next weekend.

3/ Shall be reported on a semi-annual basis.

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(d) Percentage applicable limits released.

(2) Iodine Releases

(a) Total I-131, I-133, I-135 radioactivity (in curies) released.

(b) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

(c) Percentage of limit.

(3) Particulate Releases

(a) Gross radioactivity (β-γ) released (in curies) excluding background radioactivity.

(b) Gross alpha radioactivity released (in curies) excluding background radioactivity.

(c) Total radioactivity released (in curies) of nuclides with half-lives greater than eight days.

(d) Percentage of limit.

(4) Liquid Releases

(a) Gross radioactivity (β-γ) released (in curies) excluding tritium and average concentration released to the unrestricted area at the Keowee Hydro unit.

(b) The maximum concentration of gross radioactivity (β-γ) released to the unrestricted area (averaged over the period of release).

(c) Total tritium and alpha radioactivity (in curies) released and average concentration released to the unrestricted area at the Keowee Hydro unit.

(d) Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area at the Keowee Hydro unit.

(e) Total volume (in liters) of Keowee Hydro liquid waste released.

(f) Total volume (in liters) of dilution water used prior to release from the restricted area.

(g) Total radioactivity (in curies) released, by nuclide, based on representative isotopic analyses performed.

(h) Percentage of limit for total activity released.

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(5) Solid Waste

(a) The total amount of solid waste packaged (in cubic feet).

(b) Estimated total radioactivity (in curies):

(c) Disposition including date and destination if shipped off site.

(6) Environmental Monitoring

(a) For each medium sampled during the reporting period, the following information shall be provided.

1. Number of sampling locations.

2. Total number of samples.

3. Number of locations at which levels are found to be significantly greater than local backgrounds.

4. Highest, lowest, and the average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(b) If levels of station-contributed radioactive materials in environmental media indicate the likelihood of public intakes in excess of 3 percent of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided. (These values are comparable to the top of Range I, as defined in FRC Report No. 2.)

(c) If statistically significant variations in off-site environmental concentrations with time are observed and are attributed to station releases, correlation of these results with effluent releases shall be provided.

d. Personnel Exposure and Monitoring

A tabulation (supplementing the requirements of 10 CFR 20.407) of the number of personnel receiving exposures greater than 100 mrem in the reporting period and their associated man-rem exposure, according to duty function, e.g., routine plant surveillance and inspection (regular duty), routine plant maintenance, special plant maintenance (describe maintenance), routine fueling operation, special refueling operation (describe operation), and other job-related exposures.

e. Fuel Examinations

Indication of failed fuel resulting from irradiated fuel examinations, including results of eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

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6.6.2 Non-Routine Reports

6.6.2.1 Reportable Occurrences

a. Prompt Notification with Written Followup

The types of events listed below shall be reported within 24 hours of discovery (by telephone, telegraph, mailgram, or facsimile transmission to the Director, Office of Inspection and Enforcement, Region II, or his designate) with a written followup report within two weeks to the Director, Office of Inspection and Enforcement, Region II (copy to the Director, Office of Management Information and Program Control, USNRC).

(1) Failure of the Reactor Protective System to trip, as required, when a monitored parameter reaches the setpoint specified as the limiting safety system setting in the Technical Specifications.

(2) Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the Technical Specifications.

(3) Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary or primary containment.

(4) Reactivity anomalies involving disagreement with predicted value of reactivity balance under steady-state conditions greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds, or if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or any unplanned criticality.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the Safety Analysis Report.

(6) Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the Safety Analysis Report.

(7) Conditions arising from natural or man-made events that, as a direct result of the event, require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

(8) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

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b. <u>Thirty-Day Written Reports</u>

The types of events listed below shall be the subject of written reports to the Director, Office of Inspection and Enforcement, Region II, within 30 days of discovery of the event. (Copy to the Director, Office of Management Information and Program Control, USNRC).

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or shutdown required by a limiting condition for operation.

(3) Observed inadequacies in the implementation of administrative or procedural controls during operation of a unit which could cause reduction of degree of redundancy provided in the Reactor Protective System or Engineered Safety Feature Systems.

6.6.2.2 Environmental Monitoring

a. If individual milk samples show I-131 concentrations of 10 picocuries per liter or greater, a plan shall be submitted within one week advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any individual.

b. If milk samples collected over a calendar quarter show average concentrations of 4.8 picocuries per liter or greater, a plan shall be submitted within 30 days advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any individual.

c. If, during any annual report period, a measured level of radioactivity in any environmental medium other than those associated with gaseous radioiodine releases exceeds ten times the control station value, a written notification will be submitted within one week advising the NRC of this condition. This notification should include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

d. If, during any annual report period, a measured level of radioactivity in any environmental medium other than those associated with gaseous radioiodine releases exceeds four times the control station value, a written notification will be submitted within 30 days advising the NRC of this condition. This notification should include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

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6.6.3 Special Reports

Special reports shall be submitted to the Director, Office of Inspection and Enforcement, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Electrical System Degradation, Specification 3.7.
b. Excessive Liquid Waste Releases, Specification 3.9.
c. Excessive Gaseous Waste Releases, Specification 3.10.
d. Inservice Inspection, Specification 4.2.4.
e. Reactor Vessel Specimen Surveillance, Specification 4.2.8.
f. Containment Integrated Leak Rate Test, Specification 4.4.1.1.7.
g. Reactor Building Annual Inspection Report, Specification 4.4.1.4.
h. Tendon Stress Surveillance, Specification 4.4.2.2.
i. End Anchorage Concrete Surveillance, Specification 4.4.2.3.
j. Liner Plate Surveillance, Specification 4.4.2.4.
k. Single Loop Operation, Specification 3.1.8.
l. Fuel Surveillance Program, Specification 4.13.

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DUKE POWER COMPANY
OCONEE NUCLEAR STATION
ONS-S/A-07

TABLE 6.6-1

REPORT OF RADIOACTIVE EFFLUENTS

Year ____________________

I. Liquid Releases

	Units	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	TOTAL
1. Gross Radioactivity (β,γ)														
a) Total release	Curies													
b) Average concentration released	μCi/ml													
c) Maximum concentration released	μCi/ml													
2. Tritium														
a) Total release	Curies													
b) Average concentration released	μCi/ml													
3. Dissolved noble gases														
a) Total release	Curies													
b) Average concentration released	μCi/ml													
4. Gross Alpha Radioactivity														
a) Total release	Curies													
b) Average concentration released	μCi/ml													
5. Volume of liquid waste to discharge canal	liters													
6. Volume of dilution water	liters													
7. Isotopes Released	Curies													
Ba+La-140														
Sr-89														
I-131														
Xe-133														
Xe-135														
Cs-137														
Cs-134														
Co-60														
Co-58														
Cr-51														
Mn-54														
Zn-65														
Sr-90														
8. Percent of technical specification limit for total activity released	%													

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TABLE 6.6-1 (CONTINUED)

DUKE POWER COMPANY
OCONEE NUCLEAR STATION
ONS-S/A-08

REPORT OF RADIOACTIVE EFFLUENTS

Year ____________________

II. Airborne Releases

	Units	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	TOTAL
1. Total noble gases	Curies													
2. Total halogens	Curies													
3. Total particulate gross radioactivity (β γ)	Curies													
4. Total tritium	Curies													
5. Total particulate gross alpha radioactivity	Curies													
6. Maximum noble gas release rate	μCi/sec													
7. Percent of applicable limit for:														
a. noble gases	%													
b. halogens	%													
c. particulates	%													
8. Isotope released:	Curies													
Particulates														
Cs-137														
Ba-La-140														
Sr-90														
Cs-134														
Sr-89														
Halogens														
I-131														
I-133														
I-135														
Gases														
Kr-85														
Xe-133														
Kr-88														
Kr-87														
Kr-85m														
Xe-138														
Xe-135m														
Xe-135														
Ar-41														

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FACILITY LICENSE NO. DPR-38
CHANGE NO. 26 TO TECHNICAL SPECIFICATIONS;

AMENDMENT NO. 16 TO FACILITY LICENSE NO. DPR-47
CHANGE NO. 21 TO TECHNICAL SPECIFICATIONS;

AMENDMENT TO 13 TO FACILITY LICENSE NO. DPR-55
CHANGE NO. 13 TO TECHNICAL SPECIFICATIONS

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

Introduction

By letter dated January 15, 1975, Duke Power Company (the licensee) requested a change in the Technical Specifications of Licenses No. DPR-38, DPR-47, and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. The proposed amendments would modify the station reporting requirements and delete the definition of an abnormal occurrence.

Discussion

The proposed changes would be administrative in nature and are intended to provide uniform license requirements. In Section 208 of the Energy Reorganization Act of 1974 "abnormal occurrences" is defined as an unscheduled incident or event which the Commission determines is significant from the standpoint of public health and safety. The term "abnormal occurrence" is reserved for usage by NRC. Regulatory Guide 1.16, "Reporting of Operating Information Appendix A Technical Specifications", Revision 4, enumerates required reports consistent with Section 208. The proposed change to required reports identifies the reports required of all licensees not already identified by the regulations and those unique to this facility. The proposal would formalize present reporting and would delete any reports no longer needed for assessment of safety related activities.

Evaluation

The new guidance for reporting operating information does not identify any event as an "abnormal occurrence." The proposed reporting requirements also delete reporting of information no longer required and duplication of reported information. The standardization of required reports and desired format for the information will permit more rapid recognition of potential problems.

During our review of the proposed changes, we found that certain modifications to the proposal were necessary to have conformance with the desired regulatory position. These changes were discussed with the licensee and have been incorporated into the proposal.

We have concluded that the proposal as modified improves the licensee's program for evaluating plant performance and the reporting of the operating information needed by the Commission to assess safety related activities and is acceptable. The modified reporting program is consistent with the guidance provided by Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications", Revision 4.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: DEC 22 1975

ROUTING AND TRANSMITTAL SLIP

TO (Name, office symbol or location)	INITIALS / DATE	ACTION
1 OELD - f/concurrences	INITIALS / DATE	CIRCULATE / COORDINATION
2 DLZiemann - f/signatures	INITIALS / DATE	FILE / INFORMATION
3 Reba - for final checks	INITIALS / DATE	NOTE AND RETURN / PER CON-VERSATION
4	INITIALS / DATE	SEE ME / SIGNATURE

REMARKS

Attached for your concurrence are five packages (Dresden Station, Quad Cities Station, Cooper, Pilgrim and Calvert Cliffs) of nine from ORB 2 which incorporate standard reporting requirement sections into their Appendix A Technical Specifications. One package, Pilgrim also revises the entire administrative controls section.

It is requested that, in the interest of review consistency, these packages (and the 4 future reporting requirements packages) be assigned to one OELD reviewer.

Questions may be directed to the PM for the particular case or to Mike Fletcher, coordinator for reporting (Exts. 7403, 7450)

11/3/75 No need for OELD concurrence this [illegible] this subject

Do NOT use this form as a RECORD of approvals, concurrences, disapprovals, clearances, and similar actions

FROM (Name, office symbol or location)	DATE / PHONE
DLZiemann DLZ	11-3-75 / 7380

OPTIONAL FORM 41
AUGUST 1967
GSA FPMR (41CFR) 100-11.206

5041-101

Should [illegible] only [illegible] one package on this subject.

11/3

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 16, 16, and 13 to Facility Operating Licenses No. DPR-38, DPR-47, and DPR-55, respectively, issued to Duke Power Company which revised Technical Specifications for operation of the Oconee Nuclear Station, Units 1, 2, and 3, located in Oconee County, South Carolina. The amendments are effective January 1, 1966.

These amendments revise the provisions in the Technical Specifications relating to Reporting Requirements.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments is not required since the amendments do not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendments dated January 15, 1975, (2) Amendments No. 16, 16,

OFFICE ➤						
SURNAME ➤						
DATE ➤						

and 13 to Licenses No. DPR-38, DPR-47, and DPR-55, with Changes No. 26, 21, and 13, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Oconee County Library, 201 South Spring Street, Walhalla, South Carolina 29691.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this DEC 22 1975

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
R. A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

OFFICE ➤	DRL:ORB#1	OELD	DRL:ORB#1			
SURNAME ➤	GGZech:dc	See note dtd 11/3/75	RAPurple			
DATE ➤	12/ /75	12/ /75	12/22/75			