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

December 11, 1987

Posted
Amdt. 165
to DPR-47
① (See correction letter of 2-4-88)
② (See correction letter of 4-28-88)

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

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment Nos. 165, 165, and 162 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 - Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. 60132, 60133, 60134)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 165, 165, and 162 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments consist of changes to the Stations's common Technical Specifications (TSs) in response to your request dated August 15, 1985.

The amendments revise the TSs to correct typographical errors in several sections; correct a section title in the Table of Contents; address a change in nomenclature; update Final Safety Analysis Report (FSAR) references; delete out-of-date footnotes; delete an unnecessary section; change wording for clarification; and also, update organizational charts that appear in the TSs. Proposed revisions to TS page 6.1.5 are denied because you did not provide any technical justification or no significant hazards consideration determination. This type of submittal is unacceptable and suggests a lack of management oversight.

A copy of our Safety Evaluation is also enclosed. Notice of issuance of the enclosed amendments will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Helen N. Pastis

Helen N. Pastis, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II

Enclosures:
1. Amendment No. 165 to DPR-38
2. Amendment No. 165 to DPR-47
3. Amendment No. 162 to DPR-55
4. Safety Evaluation

cc w/enclosures: See next page

Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

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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. 165
License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated August 15, 1985, 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 set forth in 10 CFR Chapter 1;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 165, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtan N. Jabbour

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: December 11, 1987

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. 165
License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated August 15, 1985, 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 set forth in 10 CFR Chapter 1;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 165, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtan N. Jabbour

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: December 11, 1987

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. 162
License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated August 15, 1985, 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 set forth in 10 CFR Chapter 1;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 162, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kahtan N. Jabbour

Kahtan N. Jabbour, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: December 11, 1987

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 165 TO DPR-38

AMENDMENT NO. 165 TO DPR-47

AMENDMENT NO. 162 TO DPR-55

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

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

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iii	iii
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3.3-1	3.3-1
3.5-11	3.5-11
3.7-2	3.7-2
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Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system components which must be met to ensure safe reactor operation.

Specification

3.1.1 Operational Components

a. Reactor Coolant Pumps

1. Whenever the reactor is critical, single pump operation shall be prohibited, single-loop operation shall be restricted to testing, and other pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

2. Except for test purposes and limited by Specification 2.3, power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

3. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one low pressure injection pump is circulating reactor coolant.

b. Steam Generator

1. One steam generator shall be operable whenever the reactor coolant average temperature is above 250°F.

c. Pressurizer Safety Valves

1. All pressurizer code safety valves shall be operable whenever the reactor is critical.

2. At least one pressurizer code safety valve shall be operable whenever all reactor coolant system openings are closed, except for hydrostatic tests in accordance with the ASME Section III Boiler and Pressure Vessel Code.

Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, startup and shutdown operations, and inservice leak and hydrostatic tests. The various categories of load cycles used for design purposes are provided in Table 5.2-1 of the FSAR.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1699 and BAW-1697.

The Figures specified in 3.1.2.1, 3.1.2.2 and 3.1.2.3 present the pressure-temperature limit curves for normal heatup, normal cooldown and hydrostatic tests respectively. The limit curves are applicable up to the indicated effective full power years of operation. These curves are adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations.

The cooldown limit curves are not applicable to conditions of off-normal operation (e.g., small LOCA and extended loss of feedwater) where cooling is achieved for extended periods of time by circulating water from the HPI through the core. If core cooling is restricted to meet the cooldown limits under other than normal operation, core integrity could be jeopardized.

The pressure-temperature limit lines shown on the figures specified in 3.1.2.1 for reactor criticality and one the figures referred to in 3.1.2.3 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region, or in test reactors.

The limitations on steam generator pressure and temperature provides protection against nonductile failure of the secondary side of the steam generator. At metal temperatures lower than the RT_{NDT} of +60°F, the protection against nonductile failure is achieved by limiting the secondary coolant pressure to 20 percent of the preoperational system hydrostatic test pressure.

3.1.7 Moderator Temperature Coefficient of Reactivity

Specification

The moderator temperature coefficient shall not be positive at power levels above 95 percent of rated power.

Bases

A non-positive moderator coefficient at power levels above 95% or rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ $\Delta k/k/°F$ corrected to 95% rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ $\Delta k/k°F$. The moderator coefficient is expected to be zero or negative prior to completion of startup tests.

When the hot zero power value is corrected to obtain the hot full power value, the following corrections will be applied.

A. Uncertainty in isothermal measurement

The measured moderator temperature coefficient will contain uncertainty on the account of the following:

1. ±0.2°F in the ΔT of the base and perturbed conditions.

2. Uncertainty in the reactivity measurement of $\pm 0.1 \times 10^{-4}$ $\Delta k/k$.

Proper corrections will be added for the above conditions to result in a conservative moderator coefficient.

B. Doppler coefficient at hot zero power

During the isothermal moderator coefficient measurement at hot zero power, the fuel temperature will increase by the same amount as the moderator. The measured temperature coefficient must be increased by 0.16×10^{-4} $(\Delta k/k)/°F$ to obtain pure moderator temperature coefficient.

C. Moderator temperature change

The hot zero power measurement must be reduced by $.09 \times 10^{-4}$ $(\Delta k/k)/°F$. This corrects for the difference in water temperature at zero power (532°F) and 15% power (580°F) and for the increased fuel temperature effects at 15% power. Above this power, the average moderator temperature remains 580°F. However, the coefficient, $_m$, must also be adjusted for the interaction of an average moderator temperature with increased fuel temperatures. This correction is $-.001 \times 10^{-4}$ $\Delta_m/\Delta T$ power. It adjusts the 15%

Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

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3.3 EMERGENCY CORE COOLING, REACTOR BUILDING COOLING, REACTOR BUILDING SPRAY, AND LOW PRESSURE SERVICE WATER SYSTEMS

Applicability

Applies to the emergency core cooling, reactor building cooling, reactor building spray, and low pressure service water systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building cooling, reactor building spray and low pressure service water systems.

Specification

3.3.1 High Pressure Injection (HPI) System

a. Prior to initiating maintenance on any component of the HPI system, the redundant component shall be tested to assure operability.

b. When the reactor coolant system (RCS), with fuel in the core, is in a condition with temperature above 350°F and reactor power less than 60% FP:

(1) Two independent trains, each comprised of an HPI pump and a flow path capable of taking suction from the borated water storage tank and discharging into the reactor coolant system automatically upon Engineered Safeguards Protective System (ESPS) actuation (HPI segment) shall be operable.

(2) Test or maintenance shall be allowed on any component of the HPI system provided one train of the HPI system is operable. If the HPI system is not restored to meet the requirements of Specification 3.3.1.b(1) above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.1.b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS temperature below 350°F within an additional 24 hours.

c. For all Units, when reactor power is greater than 60% FP:

(1) In addition to the requirements of Specification 3.3.1.b(1) above, the remaining HPI pump and valves HP-409 and HP-410 shall be operable and valves HP-99 and HP-100 shall be open.

(2) Tests or maintenance shall be allowed on any component of the HPI system, provided two trains of HPI system are operable. If the inoperable component is not restored to operable status within 72 hours, reactor power shall be reduced below 60% FP within an additional 12 hours.

Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

Bases

Operation at power with an inoperable control rod is permitted within the limits provided. These limits assure that an acceptable power distribution is maintained and that the potential effects of rod misalignment on associated accident analyses are minimized. For a rod declared inoperable due to misalignment, the rod with the greatest misalignment shall be evaluated first. Additionally, the position of the rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments. When a control rod is declared inoperable, boration may be initiated to achieve the existence of 1% $\Delta k/k$ hot shutdown margin.

The power-imbalance envelope defined in Figures 3.5.2-10 (Unit 1)
3.5.2-11 (Unit 2)
3.5.2-12 (Unit 3)
is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-16) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification power spike factors (Units 1 and 2 only)
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping rods)

** Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

1. Both 125 VDC instrumentation and control distribution centers (DCA and DCB);

2. All four 125 VDC instrumentation and control panelboards (DIA, DIB, DIC, and DID), including the associated isolating transfer diodes and diode monitors (ADA 1 & 2, ADB 1 & 2, ADC 1 & 2, ADD 1 & 2);

3. All four 120 VAC vital instrumentation power panelboards (KVIA, KVIB, KVIC, and KVID), including the associated static inverters;

4. The 240/120 VAC regulated power panelboard (KRA).

Additionally, the 125 VDC instrumentation and control batteries with an associated charger shall be operable as follows:

1. For operation of Unit 1 only, 1CA or 1CB, and 2CA or 2CB
   Unit 2 only, 2CA or 2CB, and 3CA or 3CB
   Unit 3 only, 3CA or 3CB, and 1CA or 1CB

2. For operation of any two units, 1CA or 1CB, 2CA or 2CB, and 3CA or 3CB.

3. For operation of all three units, five of the six batteries with their associated chargers.

(g) Both of the 125 VDC 230KV switching station batteries (SY-1, SY-2), with associated chargers, distribution centers, and panelboards shall be operable.

(h) Both of the 125 VDC Keowee batteries (Bank 1 & 2) with associated chargers and distribution centers (1DA & 2DA) shall be operable.

(i) The level of Keowee Reservoir shall be at least 775 feet above sea level.

3.7.2 With the reactor heated above 200°F, provisions of 3.7.1 may be modified to allow the following conditions to exist:

(a) One of the two independent on-site emergency power paths, as defined in 3.7.1(b), may be inoperable for periods not exceeding 72 hours for test or maintenance, provided the alternate power path is verified operable within one hour of the loss and every eight hours thereafter.

(b) The circuits or channels of any single functional unit of the EPSL may be inoperable for test or maintenance for periods not exceeding 24 hours, provided that:

1. The conditions of Table 3.7-1 for degraded operation are satisfied for that specific functional unit; and

2. The conditions of Table 3.7-1 for normal operation are satisfied for all other functional units.

The circuits or channels of more than one functional unit of the EPSL may be inoperable only if:

1. The inoperability results from a loss of power due to the inoperability of a 125 VDC instrumentation and control panelboard (see 3.7.2(e) below); and

2. The conditions of Table 3.7-1 for degraded operation are satisifed for the affected functional units.

If any event, if the reactor is subcritical, the inoperable circuit(s) or channel(s) shall be restored to operability and the conditions of Table 3.7-1 for normal operation shall be satisifed for all functional units before the reactor is returned to criticality.

(c) One 4160 volt main feeder bus may be inoperable for 24 hours.

(d) One complete single string (i.e., 4160 volt switchgear (TC, TD, or TE), 600 volt load center, (X8, X9, or X10), 600-208 volt MCC(XS1, XS2, or XS3), and their loads) of each unit's 4160 volt Engineered Safety Features Power System may be inoperable for 24 hours.

(e) One or more of the following DC distribution components may be inoperable for periods not exceeding 24 hours (except as noted in 3.7.2(g) below):

1. One complete single string or single component (i.e., 125VDC battery, charger, distribution center, and panelboards) of the 125VDC 230KV Switching Station Power System.

*2. One complete single string or single component (i.e., 125VDC battery, charger, and distribution center) of the Keowee 125VDC Power System may be inoperable provided the remaining string of Keowee is operable and electrically connected to an operable Keowee hydro unit.

3. One complete single string or single component (i.e., 125VDC battery, charger, distribution center, and associated isolating and transfer diodes) of any units 125VDC Instrumentation and Control Power System. Only one battery more than the number allowed to be inoperable per 3.7.1 (f) for the Station may be removed from service under this paragraph.

4. One 125 VDC instrumentation and control panelboard and its associated loads, per unit, provided that no additional AC buses are made inoperable beyond the provisions of 3.7.2(a), (c), and (d), and provided that the conditions of Table 3.7-1 for normal operation are satisfied for all functional units of the EPSL before the 125 VDC instrumentation and control panelboard becomes inoperable. Additionally, the provisions of 3.7.2.(h) must be observed for the 120 VAC vital instrumentation power panelboard which is powered by the affected 125 VDC panelboard.

*A one-time extension of inoperability for a period of 10 days per battery is granted to allow for installation of new Keowee batteries and battery racks.

Correction letter of 2-4-88

(f) For periods not to exceed 24 hours each unit's 125 VDC system may be separated from its backup unit via the isolating and transfer diodes.

(g) One battery each, from one or more of the following 125VDC systems may be simultaneously inoperable for 72 hours in order to perform an equalizer charge after the surveillance requirements of Specification 4.6.10.

1. 230 KV Switching Station 125VDC Power System

2. Keowee Hydro Station 125VDC Power System

3. Each unit's 125VDC Instrumentation and Control Power System, provided that the unit's remaining battery is operable. However, for operation of 1 or 2 units, no more batteries than those allowed to be inoperable per 3.7.1 (f) may be removed from service. For operation of 3 units, at least 4 or the 6 station IC& batteries shall be operable.

(h) One 120 VAC vital instrumentation power panelboard per unit and/or its associated static inverter may be inoperable for periods as specified below:

Panelboard	Maximum Allowed Period of Inoperability
KVIA	4 hours
KVIB	4 hours
KVIC	24 hours
KVID	24 hours

A single vital bus static inverter per unit may continue to be inoperable beyond the specified period, but no longer than 7 days total, provided that its associated 120 VAC vital instrumentation power panelboard is connected to the 240/120 VAC Regulated Power System and verified to be operable once every 24 hours.

(i) 1. A startup transformer may be inoperable for periods not exceeding 72 hours for test or maintenance, provided the underground feeder path, through transformer CT4; and to one 4160V standby bus is verified operable within one hour of loss and every eight hours thereafter. The remaining operable startup transformers can be shared between units within the same 72 hours of the above startup transformer being determined inoperable. Prior to exceeding 72 hours, they shall be aligned and connected such that each one is providing a path for power to one and only one unit.

2. In the event that a startup transformer becomes inoperable for unplanned reasons, then one unit shall be in cold shutdown within 72 hours with its loads powered from the standby buses. The remaining operable startup transformers can be shared between units within the same 72 hours of the above startup transformer being determined inoperable. Prior to exceeding 72 hours, they shall be aligned and connected such that each one is providing a path for power to one and only one unit.

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA	
2. Control Rod Drive Trip Breakers, SCR Control Relays E and F	NA	MO(1)	NA	(1) This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	MO	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Intermediate Range	ES(1)	PS	NA	(1) When in service.
6. Source Range	ES(1)	PS	NA	(1) When in service.
7. Reactor Coolant Temperature	ES	MO	RF	
8. High Reactor Coolant Pressure	ES	MO	RF	
9. Low Reactor Coolant Pressure	ES	MO	RF	
10. Flux-Reactor Coolant Flow Comparator	ES	MO	RF	
11. Reactor Coolant Pressure Temperature Comparator	ES	MO	RF	

Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

Correction letter of 4-28-89

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
12. Pump-Flux Comparator	ES	MO	RF	
13. High Reactor Building Pressure	DA	MO	RF	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	NA	MO	NA	
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

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Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
30. Borated Water Storage Tank Level Indicator	WE	NA	RF	
31. Boric Acid Mix Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
32. Concentrated Boric Acid Storage Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
33. Containment Temperature	NA	NA	RF	
34. Incore Neutron Detectors	MO(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout.
35. Emergency Plant Radiation Instruments	MO(1)	NA	RF	(1) Battery check.
36. Environmental Monitors	MO(1)	NA	RF	(1) Check functioning.
37. Reactor Manual Trip	NA	PS	NA	
38. Reactor Building Emergency Sump Level	NA	NA	RF	
39. Steam Generator Water Level	WE	NA	RF	
40. Turbine Overspeed Trip	NA	NA	RF	

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Amendment No. 162 (Unit 3)

Table -1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
49. Emergency Feedwater Flow Indicators	MO	NA	RF	
50. PORV and Safety Valve Position Indicators	MO	NA	RF	
51. RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	MO	RF	
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
a) Control Oil Pressure Switches	NA	MO	RF	
b) Discharge Pressure Switches	NA	MO	RF	
53. Emergency Feedwater Initiation Circuits				
a) Control Oil Pressure Switches	NA	MO	RF	
b) Discharge Pressure Switches	NA	MO	RF	

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage

Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

5 DESIGN FEATURES

5.1 SITE

5.1.1 The Oconee Nuclear Station is approximately eight miles northeast of Seneca, South Carolina. Figure 2.1-4 of the Oconee FSAR shows the plan of the site. The minimum distance from the reactor center line to the boundary of the exclusion area and to the outer boundary of the low population zone as defined in 10 CFR 100.3, shall be one mile and six miles respectively.

5.1.2 For the purposes of satisfying 10 CFR Part 20, the "Restricted Area," for gaseous release purposes only, is the same as the exclusion area as defined above.

REFERENCE

(1) FSAR, Chapter 2

(2) Technical Specification 3.10

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, REVIEW, AND AUDIT

6.1.1 Organization

6.1.1.1 The Station Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.1.2 In all matters pertaining to actual operation and maintenance and to these Technical Specifications, the Station Manager shall report to and be directly responsible to the Vice President, Nuclear Production Department, through the General Manager, Nuclear Stations. The organization is shown in Figure 6.1-2.

6.1.1.3 The station organization for Operations, Technical Services, Maintenance, Station Services, and Integrated Scheduling shall be functionally as shown in Figure 6.1-1. Minimum operating shift requirements are specified in Table 6.1-1.

6.1.1.4 Incorporated in the staff of the station shall be personnel meeting the minimum requirements encompassing the training and experience described in Section 4 of ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel" except for the Station Health Physicist, the Superintendent of Operations and the Operating Engineer.

The Station Health Physicist shall have a bachelor's degree in a science or engineering subject or the equivalent in experience, including some formal training in radiation protection, and shall have at least five years of professional experience in applied radiation protection of which three years shall be in applied radiation protection work in one of Duke Power Company's nuclear stations.

A qualified individual who does not meet the above requirements, but who has demonstrated the required radiation protection management capabilities and professional experience in applied radiation protection work at one of Duke Power Company's multi-unit nuclear stations, may be appointed to the position of Station Health Physicist by the Station Manager, based on the recommendations of the System Health Physicist and as approved by the General Manager, Nuclear Stations.

The Superintendent of Operations shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training, on a one-for-one time basis. The Superintendent of Operations shall hold or have held a Senior Reactor Operator license.

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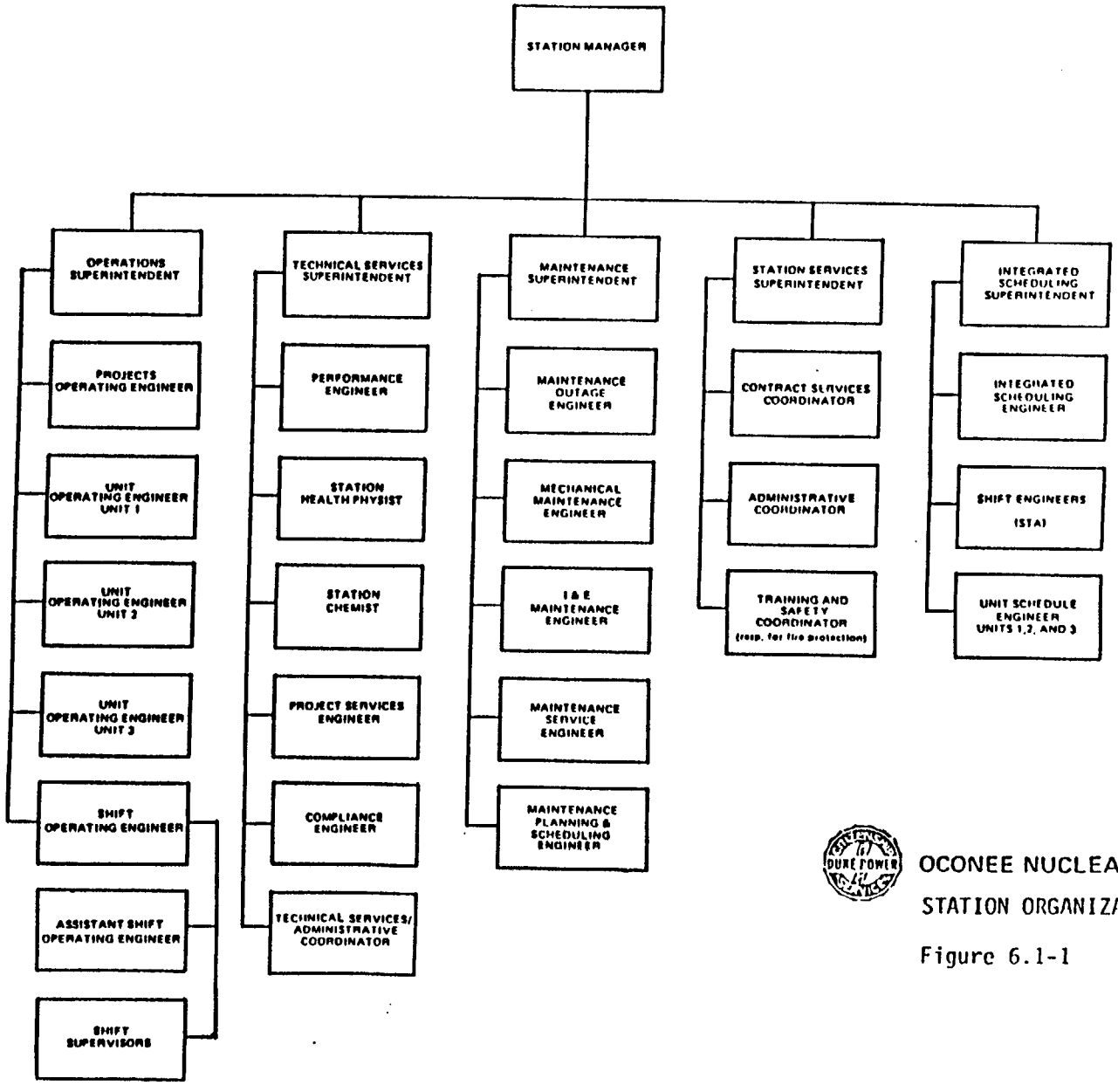
NUCLEAR STATION ORGANIZATION
FOR
OCONEE



OCONEE NUCLEAR STATION
STATION ORGANIZATION CHART

Figure 6.1-1

Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
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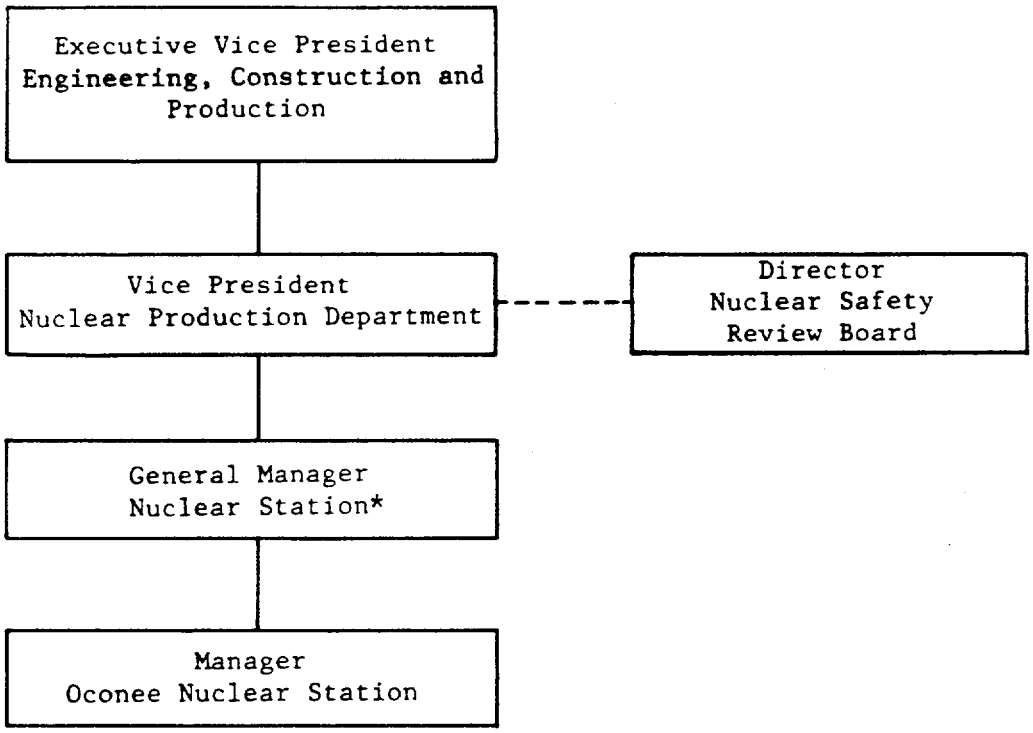

*Responsible for Fire Protection Program

OCONEE NUCLEAR STATION

MANAGEMENT ORGANIZATION CHART

Figure 6.1-2

Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the station during the reporting period.

The Radioactive Effluent Release Reports shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter.

The Radioactive Effluent Release Reports shall include an assessment of the radiation doses from radioactive effluents to members of the public due to their activities inside the unrestricted area boundary during the reporting period. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports.

The Radioactive Effluent Release Reports shall include the following information for all unplanned releases to unrestricted areas of radioactive materials in gaseous and liquid effluents:

a. A description of the event and equipment involved.

b. Cause(s) for the unplanned release.

c. Actions taken to prevent recurrence.

d. Consequences of the unplanned release.

The Radioactive Effluent Release Reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the station during each calender quarter. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The annual average meteorological conditions shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods are acceptable. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual.

The Radioactive Effluent Release Reports shall include the following information for each type of solid waste shipped offsite during the report period:

a. total container volume (cubic meters),

b. total curie quantity (determined by measurement or estimate),

c. principal radionuclides (determined by measurement or estimate),

d. type of waste, (e.g., spent resin, compacted dry waste evaporator bottoms),

e. number of shipments, and

f. solidification agent (e.g., cement, or other approved agents (media)).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to Unrestricted Areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

6.6.3 Special Reports

Special reports shall be submitted to the Regional Administrator, 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. Auxiliary Electrical Systems, Specification 3.7

b. Radioactive Liquid Effluents,
   Dose, Specification 3.9.2
   Liquid Waste Treatment, Specification 3.9.3
   Chemical Treatment Ponds, Specification 3.9.4

c. Radioactive Gaseous Effluents,
   Dose, Specification 3.10.2
   Gaseous Radwaste Treatment, Specification 3.10.3

d. Fire Protection and Detection Systems, Specification 3.17

e. Reactor Coolant System Surveillance,
   Inservice Inspection, Specification 4.2.1
   Reactor Vessel Specimen, Specification 4.2.4

f. Reactor Building Surveillance,
   Containment Leakage Tests, Specification 4.4.1

g. Structural Integrity Surveillance,
   Tendon Surveillance, Specification 4.4.2.2

h. Radiological Environmental Monitoring
   Program, Specification 4.11.1
   Land Use Census, Specification 4.11.2

i. Dose Calculations (40 CFR 190), Specification 4.21

Amendment No. 165 (Unit 1)
Amendment No. 163 (Unit 2)
Amendment No. 162 (Unit 3)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, and 3

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

I. INTRODUCTION

By letter dated August 15, 1985, Duke Power Company (the licensee) proposed changes to the Technical Specifications (TSs) of Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments would consist of changes to the Station's common TSs.

The proposed amendments would revise the TSs to correct typographical errors in several sections; correct a section title in the Table of Contents; address a change in nomenclature; update Final Safety Analysis Report (FSAR) references; delete out-of-date footnotes; delete an unnecessary section; change wording for clarification; and update organizationl charts that appear in the TSs.

II. EVALUATION

2.1 Typographical Errors (TS pages 3.3-1, 3.7-2, 3.7-3, 3.7-4, 3.1-17, and 3.5-11)

Page 3.3-1: In the present TSs, high pressure injection valves HP-409 and HP-410 are mistakenly referred to as 3HP and the proposed changes would designate the valves as HP. This is a typographical error and does not change the physical characteristics of the valves.

Page 3.7-2: The proposed changes would use the word "or" instead of "and" in TS 3.7.1(f)1. The TS designates which 125 VDC instrumentation and control batteries with an associated charger shall be operable. For the operation on only one unit, only two batteries need be operable and the present TSs require more through the use of the word "and." The word to be used should be "or."

Page 3.7-3: In TS 3.7.2(e) reference is made to section "f." The proposed changes would refer to Section "g"; the appropriate section being referred to for DC systems.

Page 3.7-4: TS 3.7.2(f) on the bottom of page 3.7-3 would be relocated to page 3.7-4. The actual text would not be changed.

Page 3.1-17: In the Bases section on moderator temperature coefficient of reactivity, section A.2. refers to the uncertainty, in the reactivity measurement of 0.1 X 10^{-4} $\Delta K/K$. The proposed changes would denote a plus and minus sign (±) instead of an underline before the uncertainty factor.

Page 3.5-11: In the present TSs, the Group 8 APSR are described as axial power shaping bank. The licensee proposes to revise the word bank to rods. Therefore, it would read "axial power shaping rods."

2.2 Table of Contents (TS page i)

Page i: In the Table of Contents, the proposed revision would change the entry for Section 1.2.3 from "Reactor Control" to "Reactor Critical". This revision would be consistent with the section title. The proposed revision is purely administrative in nature.

2.3 FSAR Updates (TS pages 3.1-4 and 5.1-1)

Pages 3.1-4 and 5.1-1: The proposed revisions would correct the Final Safety Analysis Report (FSAR) references which were changed during Duke's July 1982 update. These references were mistakenly omitted in their February 13, 1984 application for amendment.

2.4 Station Health Physicist (TS page 6.1-1)

Page 6.1-1: The proposed revision would change the job title of the Site Health Physicists to Station Health Physicists. This revision is a change in nomenclature.

2.5 Special Exemptions Deleted (TS pages 3.1-1, 4.1-3, 4.1-4, 4.1-5, 4.1-6, 4.1-7, and 4.1-8)

Page 3.1-1: The present TS 3.1.1.c.1 has a footnote which states that, "the requirements of Specification 3.1.1.c.1 are waived for Unit 1 until 11:59 p.m., October 29, 1982". Since this waiver has expired, the footnote may be deleted from the TSs.

Pages 4.1-3, 4.1-4, 4.1-5, 4.1-6, 4.1-7 and 4.1-8: Table 4.1-1 on instrument surveillance requirements is found on these pages. A double asterisk footnote states that, "a one-time extension is granted for the instrument calibration such that it be performed during the 1983 Unit 1 refueling outage, provided that such outage begins no later than July 16, 1983. Since this extension has expired, the footnote may be deleted from the TSs.

2.6 Single Loop Operation (TS pages iii, 3.1-19 and 6.6-5)

Page iii: The proposed revision would delete any reference to single loop restrictions from the Table of Contents.

Page 3.1-19: The proposed revision would delete TS 3.1.8 because it is obsolete and no longer applicable to Oconee. There are currently no plans to ever use this specification at Oconee. The original purpose for this section was 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 did not affect flow distribution or core power distribution and (4) demonstrate that the limiting safety system settings (pump monitor trip setpoint and reactor outlet temperature trip setpoint) could be conservatively adjusted taking into account instrument errors. In addition, this specification required prior Commission approval before it could be used.

In summary, this specification was included in Oconee Technical Specification to provide additional restrictions for single loop operation solely for the purpose of performing tests. During routine operations, single loop operation restriction is provided by Specification 2.3. Specification 3.1.8 is limited to when special tests are preformed, and in addition required prior Commission approval. Thus, the deletion of Specification 3.1.8 would not result in the removal or decrease in any limitation, restriction or control.

Page 6.6-5: The proposed revision would delete any reporting requirements associated with single loop restrictions.

2.7 Organization, Review and Audit (TS pages 6.1-1, 6.1-7 and 6.1-8)

Pages 6.1-1, 6.1-7 and 6.1-8: The proposed revisions would update the Station Organization Chart and Management Organization Chart for Oconee Nuclear Station to achieve consistency with Duke Power's current organization. The proposed revisions would expand the current organization shown in the TS, but would not change its basic structure. However, the Shift Engineer's (STA) reporting requirements changed from the Operations Organization to the Integrated Scheduling Superintendent. This revision is consistent with NUREG-0737, Appendix C.

2.8 Audits (TS page 6.1-5)

Page 6.1-5: The licensee proposed a revision to the TSs which was not addressed in the No Significant Hazards Consideration Evaluation or the Technical Justification. This revision would have deleted from the TSs Duke's commitments to Regulatory Guide 1.21, Revision 1, June 1974 and to Regulatory Guide 4.1, Revision 1, April 1975. This proposed revision is being denied because there was no technical justification given by the licensee.

2.9 Radioactive Effluent Release Report (TS page 6.6-2)

Page 6.6-2: The proposed revisions would change several words to TS 6.6.1.4.

In Section 6.6.2, some wording would be changed. "Individuals" would be changed to "members of the public" to clarify which individuals. This change would be consistent with the current Standard Technical Specifications. "During the reporting period" would be used instead of "each quarter" and "each calendar quarter" would be consistent with, other Technical Specifications. In the first paragraph of page 6.6.2 would add clarification. However, in the second paragraph of the same page it would remove the requirement for a quarterly breakdown of gaseous effluents that is required in the Standard Technical Specifications and in Regulatory Guide 1.21, Revision 1, June 1984. "Container volume" would be changed to "total container volume, in cubic meters," for clarification purposes. And, since 10 CFR Part 61 currently does not address types of containers "type of container" would be changed to "numbers of shipments." This change would be consistent with the guidance of Regulatory Guide 1.21. Finally, a footnote would be added concerning Radioactive Effluent Release Reports to achieve consistency with other Technical Specifications. Although the Standard Technical Specifications allow for relaxing the requirements of Regulatory Guide 1.21 to allow a one-year report for meterorological data onsite, the wording of this footnote is not clear that a year's worth of data will be supplied in the second half year Radioactive Effluent Release Report. These changes do not involve a significant hazards consideration. Changing "each quarter" to "during the reporting period" in paragraph 2 of page 6.6-2 is unacceptable as submitted and is not incorporated into the Technical Specifications. Proposed addition to the footnote on page 6.6-2 is unacceptable as submitted and is not incorporated into the Technical Specifications. The remaining proposed changes are accepted as submitted and are incorporated into the Technical Specifications.

III. ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

IV. CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (50 FR 43025) on October 23, 1985, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: P. Moore
H. Pastis

Dated: December 11, 1987