Duke Power Company
Oconee Nuclear Station

Attachment 1

Proposed Technical Specification Revision

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## 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.

Bases - Units 1, 2 and 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 $\Delta T$ 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 $0.16 \times 10^{-4}$ $(\Delta k/k)/°F$ 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%

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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.

## 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.

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 XS1, XS2, or XS3), and their loads) of each unit's 4160 volt Engineered Safety Features Power System may be inoperable for 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.

(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 | NA | MO | NA | |
| 2. Control Rod Drive Trip Breaker | NA | MO | NA | |
| 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 | |

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 | |

Table 4.1-1 (CONTINUED)

| Channel Description | Check | Test | Calibrate | Remarks |
| --- | --- | --- | --- | --- |
| 20. Reactor Building Spray System Logic | NA | MO | NA | |
| 21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure | NA | MO | RF | |
| 22. Pressurizer Temperature | ES | NA | RF | |
| 23. Control Rod Absolute Position | ES(1) | NA | RF(2) | (1) Check with Relative Position Indicator.<br>(2) Calibrate rod misalignment channel. |
| 24. Control Rod Relative Position | ES(1) | NA | RF(2) | (1) Check with Absolute Position Indicator.<br>(2) Calibrate rod misalignment channel. |
| 25. Core Flood Tanks | | | | |
| a. Pressure | ES | NA | RF | |
| b. Level | ES | NA | RF | |
| 26. Pressurizer Level | ES | NA | RF | |
| 27. Letdown Storage Tank Level | DA | NA | RF | |
| 28. Delete | | | | |
| 29. High and Low Pressure Injection Systems Flow Channels | NA | NA | RF | |

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 | |

Table 4.1-1 (CONTINUED)

| Channel Description | Check | Test | Calibrate | Remarks |
| --- | --- | --- | --- | --- |
| 41. Engineered Safeguards Channel 1 HP Injection Reactor Building Isolation Manual Trip | NA | RF | NA | Includes Reactor Building isolation of non-essential systems only |
| 42. Engineered Safeguards Channel 2 HP Injection & Reactor Building Isolation Manual Trip | NA | RF | NA | Includes Reactor Building isolation of non-essential systems only |
| 43. Engineered Safeguards Channel 3 LP Injection Manual Trip | NA | RF | NA | |
| 44. Engineered Safeguards Channel 4 LP Injection Manual Trip | NA | RF | NA | |
| 45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip | NA | RF | NA | Includes Reactor Building isolation of essential systems only |
| 46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip | NA | RF | NA | Includes Reactor Building isolation of essential systems only |
| 47. Engineered Safeguards Channel 7 Spray Manual Trip | NA | RF | NA | |
| 48. Engineered Safeguards Channel 8 Spray Manual Trip | NA | RF | NA | |

Table 4.1-1 (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

# 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 Itegrated 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.

The Operating Engineer 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

6.1.3.4 Audits

Audits of station activities shall be performed under the cognizance of the NSRB. These audits shall encompass:

a. The conformance of station operation to provisions contained within the Technical Specifications and applicable facility operating license conditions at least once per year.

b. The performance, training and qualifications of the station staff at least once per year.

c. The results of actions taken to correct deficiencies occurring in equipment, structures, systems or methods of operation that affect nuclear safety at least once per six months.

d. The performance of activities required by the quality assurance program the criteria of Appendix B to 10 CFR 50 at least once per two years.

e. The station emergency plan and implementing procedures at least once per 12 months.

f. The station security plan and implementing procedures at least once per 12 months.

g. Any other area of station operation considered appropriate by the NSRB or the Vice President, Nuclear Production Department.

h. The station fire protection program and implementing procedures at least once per 24 months.

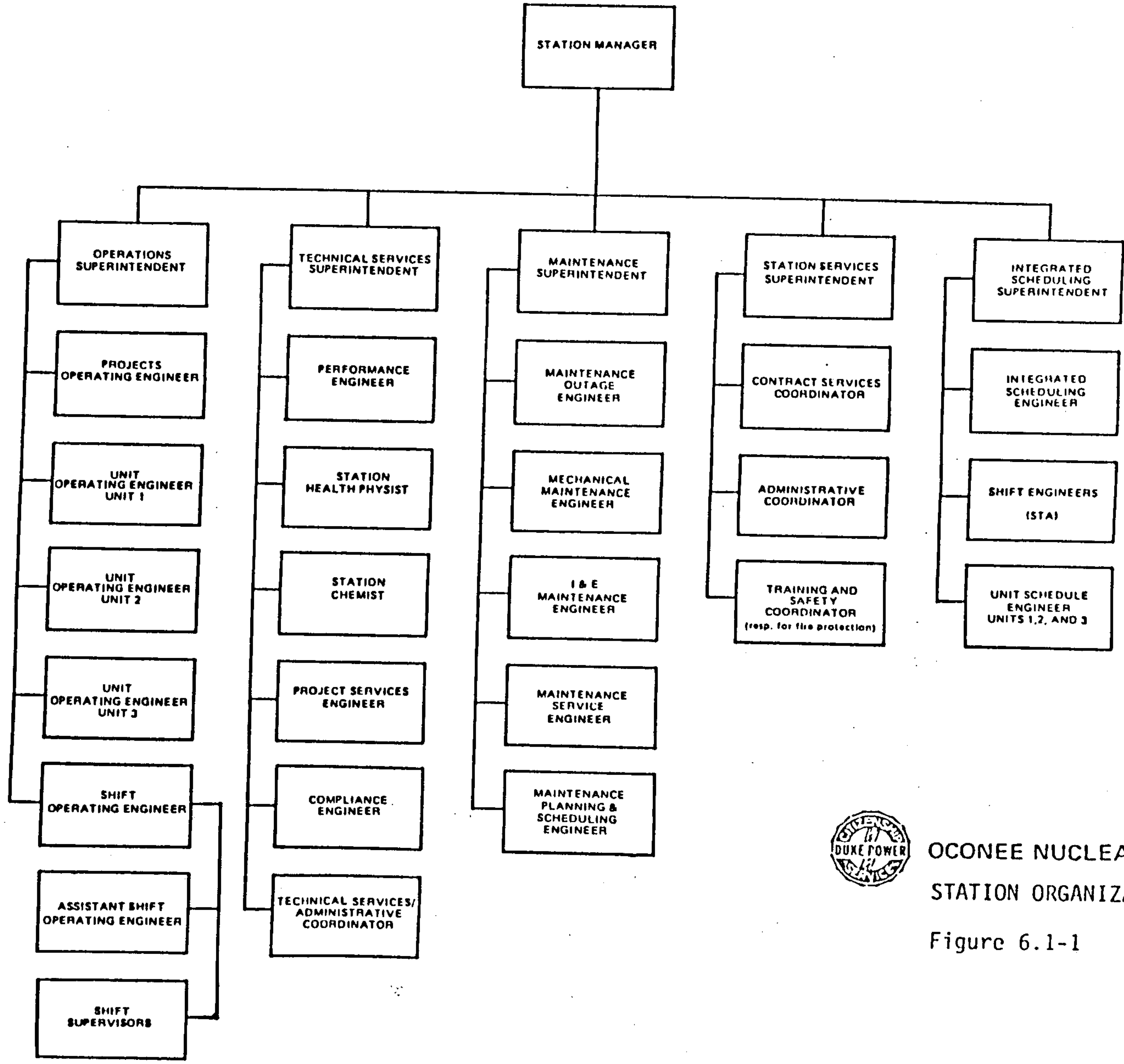
i. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

j. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months.

k. The Process Control Program and implementing procedures for solidification of radioactive wastes at least once per 24 months.

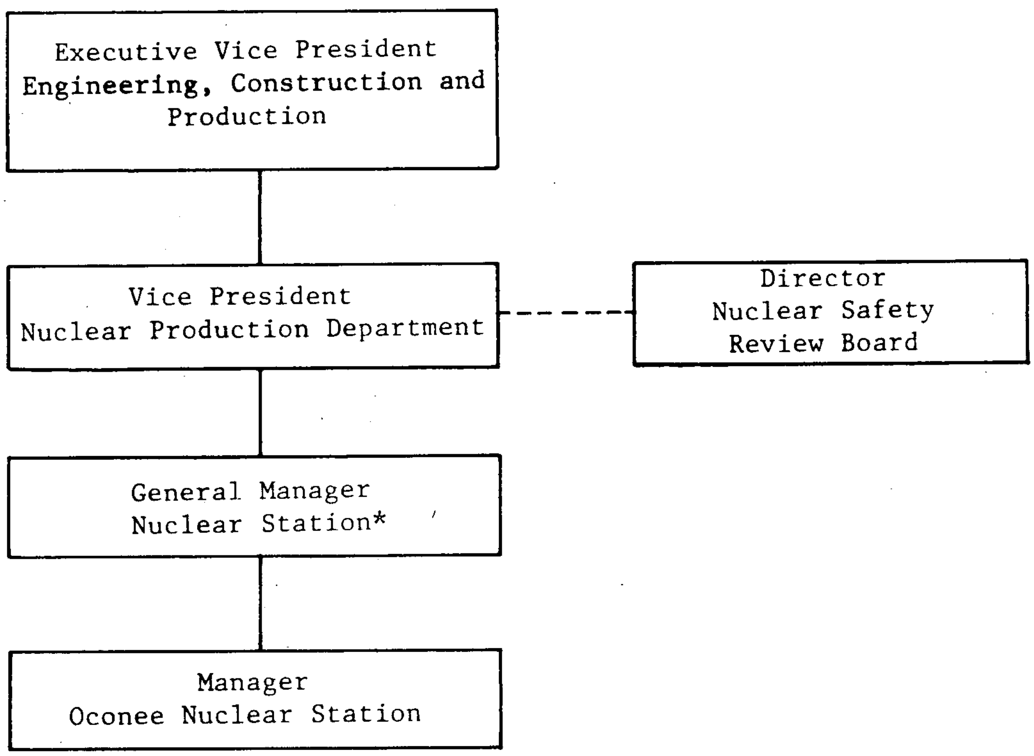
l. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months.

NUCLEAR STATION ORGANIZATION
FOR
OCONEE




DUKE POWER

OCONEE NUCLEAR STATION
STATION ORGANIZATION CHART
Figure 6.1-1




*Responsible for Fire Protection Program

OCONEE NUCLEAR STATION

MANAGEMENT ORGANIZATION CHART

Figure 6.1-2

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 the reporting period.*

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.

*In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining the summary of required meteorological data onsite in a file that shall be provided to the NRC upon request.

### 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

Duke Power Company
Oconee Nuclear Station

Attachment 2

No Significant Hazards Consideration Evaluation

# No Significant Hazards Consideration Evaluation

Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the Commission's regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of these standards by providing certain examples (48 FR 14870). Example (i) of the types of amendments considered not likely to involve significant hazards considerations is applicable to this amendemnt request. This specific example involves amendment requests that are considered to be purely administrative changes to the Technical Specifications--for example,

(a) a change to achieve consistency throughout the technical specification; or

(b) correction of an error; or

(c) a change in nomenclature.

The proposed Technical Specifications amendments addressed in this submittal has been determined by Duke to contain administrative changes only. The requested changes are required so that the Technical Specifications will be consistent throughout and past omissions will be corrected.

Briefly, the proposed amendment corrects typographical errors in several places; revises the table of contents to provide for consistency; changes nomenclature in one place; updates two organizational charts; updates FSAR figures and tables being referenced; deletes out-of-date footnotes; and, changes wording for clarification. A more detailed discussion of each revision is provided in the Technical Justification (Attachment 3).

The reason for the deletion of Technical Specification 3.1.8, single loop restriction, is because it is obsolete and no longer applicable to Oconee. Further, there are currently no plans to ever use this specification at Oconee. This specification was included to provide, during special tests being conducted, additional restrictions for single loop operation. During routine operations, single loop operation restriction is provided by Specification 2.3. In addition, prior to invoking Specification 3.1.8, specific commission approval was required. Thus, the deletion of Specification 3.1.8 would not result in the removal or reduction in any limitation, restriction, control or margin of safety. Furthermore, this deletion would not significantly increase the probability or consequences of a previously evaluated accident nor would it create the possibility of a new or different kind of accident.

Duke has determined, based on the above consideration that the requested amendments are administrative in nature, that the revisions do not involve a significant increase in the probability or consequences of accidents previously considered, nor create the possibility of a new or different kind of accident, and will not involve a significant decrease in a safety margin. Therefore, Duke concludes that there is a No Significant Hazards Consideration involved in this amendment request.

Duke Power Company
Oconee Nuclear Station

Attachment 3

Technical Justification

## Technical Justification

The proposed Technical Specification revisions addressed in this submittal are adminstrative in nature and, as such, are of no public health or safety significance. Briefly, the proposed amendment corrects typographical errors in several sections; corrects a section title in the Table of Contents; addresses a change in nomenclature; updates FSAR references; deletes out-of-date footnotes; deletes an unnecessary section; changes wording for clarification; and also, updates organizational charts that appear in the Technical Specifications.

There are several areas where typographical errors were found in the Oconee Technical Specification. High pressure valves are designated as HP, but were mistakenly referred to as 3HP in two places. The work "and" was used instead of "or" in section 3.7.1. and thirdly, a "f" is shown instead of a "g" in a reference in section 3.8.2(e). The last two errors occurred as part of Technical Specification Revision 127/127/124 on March 3, 1984. In section 3, the work present was misspelled, and an underline was usde instead of a minus sign to denote "±". Finally, in section 3.5.2 the words that relate to the acronym APSR were incorrect and are now being corrected. The changes included in this proposed amendment correct these errors. These changes are considered to be purely administrative in that they only correct errors.

An inconsistency was found between the Table of Contents and the title for section 1.2.3. The Table of Contents refers to "Reactor Control" when it should be "Reactor Critical". This change will provide for uniformity throughout the Technical Specifications, and thus, assure a consistent application of the term.

The initial Oconee FSAR update was provided as required by 10 CFR 50, §50.71 by a Duke letter dated July 19, 1982. The updated FSAR was reformatted to be consistent with present FSAR format criteria. This resulted in the FSAR references within the Technical Specifications being out of date. These references were corrected by Duke's Technical Specification revision submitted on February 13, 1984, but it has since been noted that several references were mistakenly omitted. The updating of the reference to the FSAR within the Technical Specifications assures that the appropriate figure of the FSAR is being identified. The updating of the Technical Specifications is an administrative change to achieve consistency with other documents.

In section 6.1.1.4 of the Technical Specifications, a change in nomenclature is requested. The Health Physicists at the Oconee Nuclear Station are referred to as Station Health Physicists, not Site Health Physicists.

Two footnoted special exemptions should be deleted as they are no longer applicable. In both cases, the dates of which the footnotes are valid have passed, therefore they can be deleted.

In sections 6.1.3 and 6.6.2, some wording has been changed in order to achieve clarity and consistency throughout the Technical Specification. "Individuals" was changed to "members of the public" to clarify which individuals and "during the reporting period" is being used instead of "each quarter" and "each calender quarter" to be consistent with other Technical Specifications. "Container volume" was 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" was changed to "numbers of shipments". Finally, a footnote was added concerning Radioactive Effluent Release Reports to achieve consistency with other Technical Specifications.

Technical Specification 3.1.8, single loop restriction, is being deleted 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 performed, 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. In addition, the reference to Single Loop Restrictions in section 6.6.3 is being deleted.

The final revisions are updates to the Station Organizational Chart and Management Organization Chart, for Oconee Nuclear Station, to achieve consistency with Duke Power's current organization.