ATTACHMENT 2

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Proposed Technical Specification Revision and
Facility Operating License Condition Amendment
to Incorporate NUREG-0578 Category A Requirements

3.1.12 Reactor Coolant System Subcooling Margin Monitor

Specification

3.1.12.1

a. The Reactor Coolant System subcooling monitors shall be operable when the average RCS coolant temperature is above 300^{o}F.

b. If one monitor is inoperable, the monitor shall be restored to operable status within 7 days or the unit shall be in hot shutdown within the next 12 hours.

c. If no monitors are operable, then restore at least one monitor to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

Applicability

Applies to the secondary system requirements for removal of reactor decay heat.

Objective

To specify minimum conditions necessary to assure the capability to remove decay heat from the reactor core.

Specification

3.4.1 Emergency Feedwater System

The reactor shall not be heated above 250°F unless the following conditions are met:

a. Three emergency feedwater pumps (one steam-driven pump capable of being powered from an operable steam supply system and two-motor-dirven pumps), associated initiation circuitry, shall be operable.

b. Two 100% emergency feedwater flow paths shall be operable. Each flow path shall have at least one flow indicator operable.

c. If one emergency feedwater pump or emergency feedwater flow path is inoperable then, restore it to operable status within 60 hours. Otherwise, the unit shall be in a hot shutdown condition within an additional 12 hours and below 250°F in another 12 hours.

3.4.2 The 16 steam system safety valves shall be operable.

3.4.3 A minimum of 72,000 gallons of water per operating unit shall be available in the upper surge tank, condensate storage tank, and hotwell.

3.4.4 The emergency condenser circulating water system shall be operable.

3.4.5 The controls of the emergency feedwater system shall be independent of the Integrated Control System.

Bases

The Main Feedwater System and the Turbine Bypass System are normally used for decay heat removal and cooldown above 250°F. Feedwater makeup is supplied by operation of a hotwell pump condensate booster pump and a main feedwater pump.

The Emergency Feedwater (EFW) System assures sufficient feedwater supply to the steam generators of each unit, in the event of loss of the main Feedwater System, to remove energy stored in the core and primary coolant. The EFW System is designed to provide sufficient secondary side steam generator heat sink to enable cooldown from reactor trip power operation down to cold shutdown conditions.

A 100% emergency feedwater flowpath shall be considered to be either: 1) the steam-driven turbine pump, associated valves and piping capable of feeding either steam generator or 2) both motor-driven pumps, associated valves and piping each capable of feeding the associated steam generator.

One flow indicator or steam generator level indicator per steam generator is sufficient to provide indication of emergency feedwater flow to the steam generators and to confirm emergency feedwater system operation. In the event that at least one indicator per steam generator is not available, then the flowpath to this steam generator is considered to be inoperable.

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps or low main feedwater header pressure. The EFW System will supply sufficient feedwater for approximately five-hour cooldown at a flowrate of at least 720 gpm to enable the Reactor Coolant System to reach conditions at which the Decay Heat Removal System may be operated.

Two motor-driven emergency feedwater pumps are installed in each unit in addition to the steam-driven emergency feedwater pump. The motor-driven pumps are powered from diverse emergency power supplies.

All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

Normally, decay heat is removed by steam relief through the turbine bypass system to the condenser. Condenser cooling water flow is provided by a siphon effect from Lake Keowee through the condenser for final heat rejection to the Keowee Hydro Plant tailrace. Decay heat can also be removed from the steam generators by steam relief through the main steam relief valves.

The minimum amount of water in the upper surge tank, condensate storage tank and hotwell is the amount needed for 11 hours of operation per unit. This is based on the conservative estimate of normal makeup being 0.5% of throttle flow. Throttle flow at full load, 11,200,000 lbs/hr., was used to calculate the operation time. For decay heat removal the operation time with the volume of water specified would be considerably increased due to the reduced throttle flow.

The total relief capacity of the 16 steam system safety valves is 13,105,000 lbs/hr.

REFERENCE

FSAR, Section 10

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and two channels each of the following are operable: four reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels, four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and high reactor building pressure instrument channels. The engineered safety features actuation system must have two analog channels functioning correctly prior to a startup. Additional operability requirements are provided by Technical Specifications 3.1.12 and 3.4 for equipment which are not part of the RPS or ESFAS.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column B (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE-279 as described in FSAR Section 7.

There are four reactor protective channels. A fifth channel that is isolated from the reactor protective system is provided as a part of the reactor control system. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. Minimum trip logic on other channels is one out of two.

The four reactor protective channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed. There will be one reactor protective system bypass switch key permitted in the control room. That key will be under the administrative control of the Shift Supervisor. Spare keys will be maintained in a locked storage accessible only to the station Manager.

Each reactor protective channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. There are four shutdown bypass keys in the control room under the administrative control of the Shift Supervisor. The use of a key operated shutdown bypass switch for on-line testing or maintenance during reactor power operation has no significance when used in conjunction with a key operated channel bypass switch since the channel trip relay is locked in the untripped state. The use of a key operated shutdown bypass switch alone during power operation will cause the channel to trip. When the shutdown bypass switch is operated for on-line testing or maintenance during reactor power operation, reactor power and RCS pressure limits as specified in Table 2.3-1A, B, or C are not applicable.

The source range and intermediate range nuclear instrumentation overlap by one decade of neutron flux. This decade overlap will be achieved at 10^{-10} amps on the intermediate range instrument.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 600 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state on-line repairs to the failed device, when practical, will be made, and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

Containment isolation valves on non-essential systems are isolated by diverse signals from high containment pressure and low reactor coolant system pressure devices. The systems considered to be non-essential include:

1. Letdown line
2. RC Pump seal return line
3. Quench Tank sample line
4. Quench Tank gaseous vent
5. Reactor Building purge lines
6. Reactor Building sump drain line
7. Reactor Building atmosphere sample line
8. Pressurizer sample line
9. OTSG sample line
10. OTSG drain line

Containment isolation valves on essential systems are isolated by high containment pressure only. The systems considered to be essential include:

1. Component cooling to RC pumps
2. Low pressure service water cooling to RC pump motor

REFERENCE

FSAR, Section 7.1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
11. ESF High Pressure Injection System & Reactor Building Isolation (Non-essential Systems)			
a. Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
b. Reactor Building 4 PSIG Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
c. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
12. ESF Low Pressure Injection System			
a. Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
b. Reactor Building 4 PSIG Instrument Channels	2	1	Bring to hot shutdown within 12 hours (e)
c. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
13. ESF Reactor Building Isolation (Essential Systems) & Reactor Building Cooling System			
a. Reactor Building 4 PSIG Instrument Channel	2	1	Bring to hot shutdown within 12 hours (e)
b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
14. ESF Reactor Building Spray System			
a. Reactor Building High Pressure Instrument Channel	2	1	Bring to hot shutdown within 12 hours (e)

3.5.3 Engineered Safety Features Protective System Actuation Setpoints

Applicability

This specification applies to the engineered safety features protective system actuation setpoints.

Objective

To provide for automatic initiation of the engineered safety features protective system in the event of a breach of RCS integrity.

Specification

The engineered safety features protective actuation setpoints and permissible bypasses shall be as follows:

Functional Unit	Action	Setpoint
High Reactor Building Pressure	Reactor Building Spray	$\leqq$30 psig
	High-Pressure Injection Reactor Building Isolation (Non-essential Systems)	$\leqq$4 psig
	Low-Pressure Injection	$\leqq$4 psig
	Start Reactor Building Cooling & Reactor Building Isolation (Essential Systems)	$\leqq$4 psig
	Penetration Room Ventilation	$\leqq$4 psig
Lower Reactor Coolant System Pressure	High Pressure Injection(1) & Reactor Building Isolation (Non-essential systems)	$\geqq$1500 psig
	Low Pressure Injection(2)	$\geqq$500 psig

(1) May be bypassed below 1750 psig and is automatically reinstated aboved 1750 psig.

(2) May be bypassed below 900 psig and is automatically reinstated above 900 psig.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached immediately in the event of a DBA, cover the entire spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1500 psig low reactor coolant pressure setpoint for high pressure injection initiation and 500 psig for low pressure injection is to

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days.

Bases

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
12. Pump-Flux Comparator	ES	MO	AN	
13. High Reactor Building Pressure	DA	MO	AN	
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	AN	
b. Reactor Building Pressure (4 psig)	ES	MO	AN	
16. Low Pressure Injection Logic	NA	MO	NA	
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	AN	
b. Reactor Building Pressure (4 psig)	ES	MO	AN	
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	AN	

Table 4.1-1 (CONTINUED)

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

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
49. Emergency Feedwater Flow Indicators	MO	NA	AN	
50. PORV and Safety Valve Position Indicators	MO	NA	AN	

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rod Movement (1)	Movement of Each Rod	Monthly
2.	Pressurizer Safety Valves	Setpoint	50% Annually
3.	Main Steam Safety Valves	Setpoint	25% Annually
4.	Refueling System Interlocks	Functional	Prior to Refueling
5.	Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly
6.	Reactor Coolant System Leakage (2)	Evaluate	Daily
7.	Condenser Cooling Water System Gravity Flow Test	Functional	Annually
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9.	Spent Fuel Cooling System	Functional	Prior to Refueling
10.	High Pressure and Low Pressure Injection System (3)	Vent Pump Casings	Monthly and Prior to Testing
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature.	Functional	Each Refueling
12.	RCS Subcooling Monitor	Functional	Each Refueling

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

4.9 EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine-driven and motor-driven emergency feedwater pumps and associated valves.

Objective

To verify that the emergency feedwater pumps and associated valves are operable.

Specification

4.9.1 Pump Test

Monthly, the turbine-driven and motor-driven feedwater pumps shall be operated on recirculation to the upper surge tank for a minimum of one hour.

4.9.2 Valve Test

Quarterly, automatic valves in the emergency feedwater flow path will be determined to be operable in accordance with the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

4.9.3 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly.

Bases

The monthly testing frequency is sufficient to verify that the emergency feedwater pumps are operable. Verification of correct operation is made both from the control room instrumentation and direct visual observation of the pumps. The parameters which are observed are detailed in the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

REFERENCES

(1) FSAR, Section 10.2.2

(2) FSAR, Section 14.1.2.8.3

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, Steam Production, through the Manager, Nuclear Production. The organization is shown in Figure 6.1-2.

6.1.1.3 The station organization for Operations, Technical Services and Maintenance 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 Site Health Physicist.

The Site 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 Site Health Physicist by the station Manager, based on the recommendations of the System Health Physicist and as approved by the Manager, Nuclear Production.

6.1.1.5 Retraining and replacement of station personnel shall be in accordance with Section 5.5 of the ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel."

6.1.1.6 A training program for the fire brigade shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except that training sessions may be held quarterly.

6.1.1.7 The two functions of the Shift Technical Advisor, namely accident assessment and operating experience assessment, are fulfilled in the following manner:

a. An experienced SRO, who has been instructed in additional academic subjects, will be assigned on-shift to provide the accident assessment capability.

b. Several engineers, familiar with plant operations and representing diverse technical backgrounds will be assigned to provide the operating experience assessment.

6.1.2 Technical Review and Control

6.1.2.1 Activities

a. Procedures required by Technical Specification 6.4 and other procedures which affect station nuclear safety, and changes (other than editorial or typographical changes) thereto, shall be prepared by a qualified individual/organization. Each such procedure, or procedure change, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or procedure change, but who may be from the same organization as the individual/group which prepared the procedure, or procedure change. Such procedures and procedure changes may be approved for temporary use by two members of the station staff, at least one of whom holds a Senior Reactor Operator's License on the unit(s) affected. Procedures and procedure changes shall be approved prior to use or within seven days of receiving temporary approval for use by the station Manager; or by the Operating Superintendent, the Technical Services Superintendent or the Maintenance Superintendent, as previously designated by the station Manager.

b. Proposed changes to the Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the station Manager.

c. Proposed modifications to station nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to station nuclear safety-related structures, systems and components shall be approved prior to implementation by the station Manager; or by the Operating Superintendent, the Technical Services Superintendent, or the Maintenance Superintendent, as previously designated by the station Manager.

d. Individuals responsible for reviews performed in accordance with 6.1.2.1.a, 6.1.2.1.b, and 6.1.2.1.c shall be members of the station supervisory staff, previously designated by the station Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated station review personnel.

e. Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the station Manager; or by the Operating Superintendent, the Technical Services Superintendent or the Maintenance Superintendent, as previously designated by the station Manager.

f. Incidents reportable pursuant to Technical Specification 6.6.2.1 and violations of Technical Specifications shall be investigated and a report prepared which evaluates the occurrence and which provides recommendations to prevent recurrence. Such reports shall be approved by the station Manager and transmitted to the Vice President, Steam Production, or his designee; and to the Director of the Nuclear Safety Review Board.

g. The station Manager shall assure the performance of special reviews and investigations, and the preparation and submittal of reports thereon, as requested by the Vice President, Steam Production.

h. The station security program, and implementing procedures, shall be reviewed at least annually. Changes determined to be necessary as a result of such review shall be approved by the station Manager and transmitted to the Vice President, Steam Production, or his designee; and to the Director of the Nuclear Safety Review Board.

i. The station emergency plan, and implementing procedures, shall be reviewed at least annually. Changes determined to be necessary as a result of such review shall be approved by the station Manager and transmitted to the Vice President, Steam Production, or his designee; and to the Director of the Nuclear Safety Review Board.

j. The station manager shall assure that an independent fire protection and loss prevention inspection and audit shall be performed annually utilizing qualified off-site personnel and that an inspection and audit by a qualified fire consultant shall be performed at intervals no greater than three years.

6.1.2.2 Records

Records of the above activities shall be maintained.

6.1.3 Nuclear Safety Review Board

6.1.3.1 Function

The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations
b. Nuclear Engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Administrative control and quality assurance practices

6.1.3.2 Organization

a. The Director, members and alternate members of the NSRB shall be formally appointed by the Vice President, Steam Production, and shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years technical experience, of which a minimum of three years shall be in one or more areas given in 6.1.3.1.

b. The NSRB shall be composed of at least five members, including the Director, Members of the NSRB may be from the Steam Production Department, from other departments within the Company or from external to the Company. A maximum of one member of the NSRB may be from the Oconee Nuclear Station staff.

c. Consultants may be utilized by the NSRB to provide expert advice to the NSRB, as determined necessary by the Director of the NSRB.

d. Staff assistance may be provided to the NSRB in order to promote the proper, timely and expeditious performance of its functions.

e. The NSRB shall meet at least once per six months. The period between such meetings shall not exceed eight months.

f. A quorum of the NSRB shall consist of the Director, or his designated alternate, and at least two other NSRB members or alternate members. No more than a minority of the quorum shall have line responsibility for operation of Oconee Nuclear Station.

6.1.3.3 Subjects Requiring Review

The following subjects shall be reported to and reviewed by the NSRB:

a. The safety evaluations for (1) changes to procedures, equipment or systems, and (2) tests or experiments completed under the provisions of 10 CFR 50.59(a)(1) to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Proposed changes in Technical Specifications or the Facility Operating Licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of station equipment that affect nuclear safety.

g. Indicents that are the subject of non-routine reports submitted to the Commission.

h. Quality Assurance Department audits relating to station operations and actions taken in response to these audits.

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 to meet 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 two years.

f. The station security plan and implementing procedures at least once per two years.

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

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

6.1.3.5 Responsibilities and Authorities

a. The NSRB shall report to and advise the Vice President, Steam Production on those areas of responsibility specified in Specifications 6.1.3.3 and 6.1.3.4.

b. Minutes shall be prepared and forwarded to the Vice President, Steam Production, and to the Senior Vice President, Production and Transmission, within 14 days following each formal meeting of the NSRB.

c. Records of activities performed in accordance with Specifications 6.1.3.3 and 6.1.3.4 shall be maintained.

d. Audit reports encompassed by Section 6.1.3.4 shall be forwarded to the Vice President, Steam Production, and to the Senior Vice President, Production and Transmission and to the management position responsible for the areas audited within 30 days of completion of each audit.

ADDITIONAL REQUIREMENTS

1. One licensed operator per unit shall be in the Control Room at all times when there is fuel in the reactor vessel.

2. Two licensed operators shall be in the Control Room during startup and scheduled shutdown of a reactor.

3. At least one licensed operator shall be in the reactor building when fuel handling operations in the reactor building are in progress.

4. An operator holding a Senior Reactor Operator license and assigned no other operational duties shall be in direct charge of refueling operations.

5. At least one person per shift shall have sufficient training to perform routine health physics requirements.

6. If the computer for a reactor is inoperable for more than eight hours, an operator, in addition to those required above, shall supplement the shift crew.

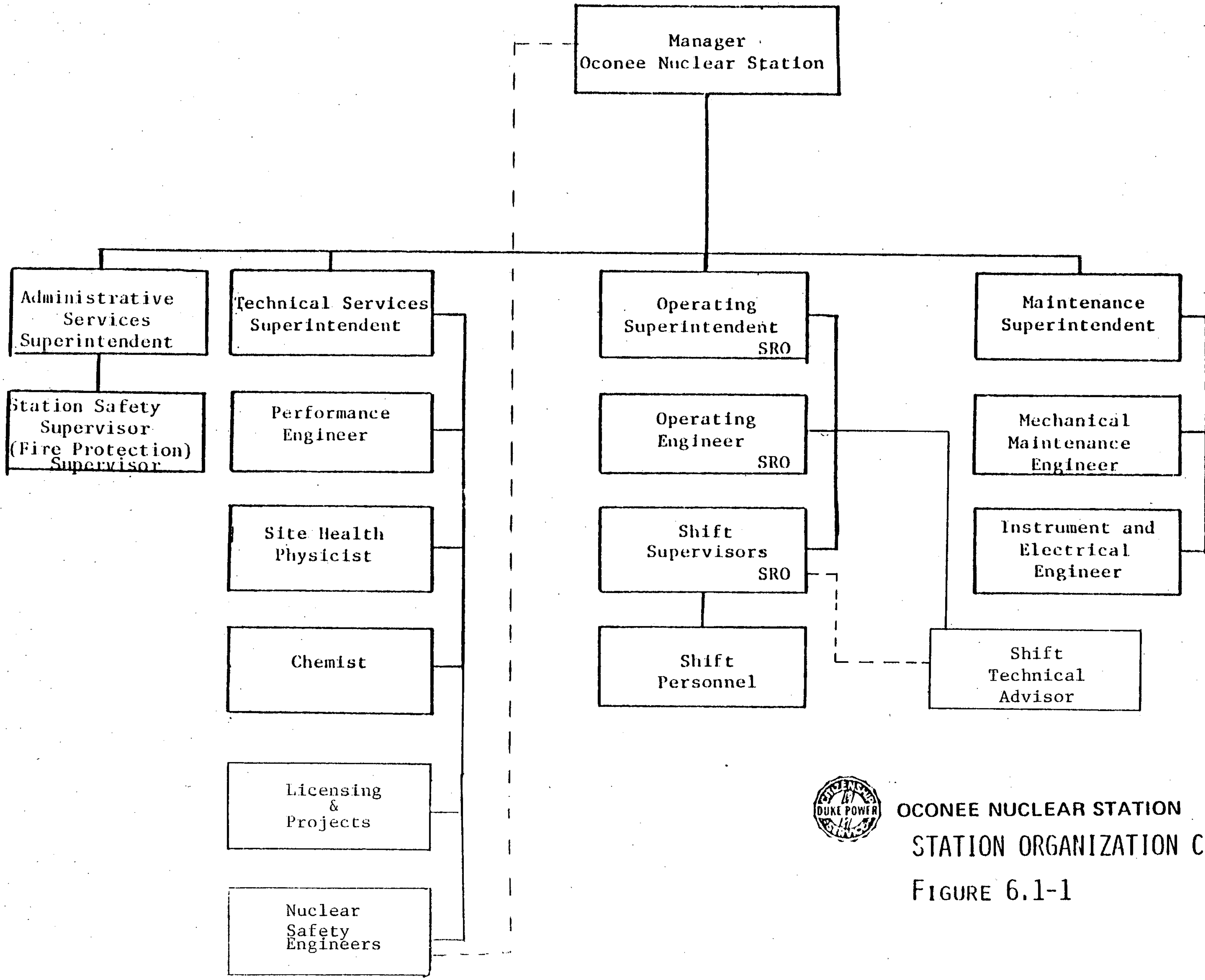
7. A fire brigade of 5 members shall be maintained on site at all times. This excludes 3 members of the minimum operating shift requirements that are required to be present in the control rooms.

TABLE 6.1-1
MINIMUM OPERATING SHIFT REQUIREMENTS
(With Fuel in the Three Reactor Vessels)

	One Unit Operating*	Two Units Operating*	All Units Operating*	All Units Shutdown
Shift Supervisor (SRO)	1	1	1	1
Additional SRO	1	2**	2	1
Shift Technical Advisor (SRO)	1	1	1	None
Reactor Operator	4	4	5	3
Nuclear Equipment Operator	2	5	4	3

* Above cold shutdown

** Only one SRO required if both units are operated from one Control Room.



OCONEE NUCLEAR STATION
STATION ORGANIZATION CHART
FIGURE 6.1-1

OCONEE NUCLEAR STATION
Proposed License Conditions

3H. Systems Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

3I. Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

3J. Backup Method for Determining Subcooling Margin

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.