APPENDIX A TO OPER LIC DPR-38

TECHNICAL SPECIFICATIONS

50-269 Oconee Unit 1

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APPENDIX A

TO

OPERATING LICENSE NO. DPR-38

TECHNICAL SPECIFICATIONS

FOR THE

OCONEE NUCLEAR STATION UNIT 1

DUKE POWER COMPANY

DOCKET NO. 50-269

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

INTRODUCTION

These Technical Specifications apply to the Oconee Nuclear Station, Units 1 and 2 and are in accordance with the requirements of 10 CFR 50, Section 50.36. The bases, which provide technical support or reference the pertinent FSAR section for technical support of the individual specifications, are included for informational purposes and to clarify the intent of the specification. The bases are not part of the Technical Specifications, and they do not constitute limitations or requirements for the licensee. The technical specifications while applying to Units 1 and 2 are written on a single unit basis; exceptions to this are identified.

TECHNICAL SPECIFICATIONS

1 DEFINITIONS

The following terms are defined for uniform interpretation of these specifications.

1.1 RATED POWER

Rated power is defined as a steady state reactor core output of 2568 MWt.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 Cold Shutdown

The reactor is in the cold shutdown condition when it is subcritical by at least 1 percent $\Delta k/k$ and T_{avg} is no more than 200°F. Pressure is defined by Specification 3.1.2.

1.2.2 Hot Shutdown

The reactor is in the hot shutdown condition when it is subcritical by at least 1 percent $\Delta k/k$ and T_{avg} is at or greater than 525°F.

1.2.3 Reactor Critical

The reactor is critical when the neutron chain reaction is self-sustaining and $K_{eff} = 1.0$.

1.2.4 Hot Standby

The reactor is in the hot standby condition when all of the following conditions exist:

a. T_{avg} is greater than 525°F.

b. The reactor is critical.

c. Indicated neutron power on the power range channels is less than 2 percent of rated power.

1.2.5 Power Operation

The reactor is in a power operating condition when the indicated neutron power is above 2 percent of rated power as indicated on the power range channels.

1.2.6. Refueling Shutdown

The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least 1 percent $\Delta k/k$ and the coolant temperature at the low pressure injection pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.2.7 Refueling Operation

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

1.2.8 Refueling Period

Time between normal refuelings of the reactor, not to exceed 18 months without prior approval of the AEC.

1.2.9 Startup

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.

1.3 OPERABLE

A component or system is operable when it is capable of performing its intended function within the required range. The component or system shall be considered to have this capability when: (1) it satisfies the limiting conditions for operation defined in Specification 3, and (2) it has been tested periodically in accordance with Specification 4 and has met its performance requirements.

1.4 PROTECTIVE INSTRUMENTATION LOGIC

1.4.1 Instrument Channel

An instrument channel is the combination of sensor, wires, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control and/or protection. An instrument channel may be either analog or digital in nature.

1.4.2 Reactor Protective System

The reactor protective system is shown in Figures 7-1 and 7-6 of the FSAR. It is that combination of protective channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protective channels, their associated instrument channel inputs, manual trip switch, all rod drive protective trip breakers and activating relays or coils.

1.4.3 Protective Channel

A protective channel as shown in Figure 7-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply units, amplifiers and bistable modules provided for every reactor protective safety parameter) is a combination of instrument channels forming a single digital output to the protective system's coincidence logic. It includes a shutdown bypass circuit, a protective channel bypass circuit and reactor trip module and provision for insertion of a dummy bistable.

1.4.4 Reactor Protective System Logic

This system utilizes reactor trip module relays (coils and contacts) in all four of the protective channels as shown in Figure 7-1 of the FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one out of two times two logic. Each element of the one out of two times two logic is controlled by a separate set of two out of four logic contacts from the four reactor protective channels.

1.4.5 Engineered Safety Features System

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7-3 of the FSAR. The logic sub-system is wired to provide appropriate signals for the actuation of redundant Engineered Safety Features equipment on a two-of-three basis for any given parameter.

1.4.6 Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

1.5 INSTRUMENTATION SURVEILLANCE

1.5.1 Trip Test

A trip test is a test of logic elements in a protective channel to verify their associated trip action.

1.5.2 Channel Test

A channel test is the injection of an internal or external test signal into the channel to verify its proper output response; including alarm and/or trip initiating action where applicable.

1.5.3 Instrument Channel Check

An instrument channel check is a verification of acceptable instrument performance by observation of its behavior and/or state; this verification includes comparison of output and/or state of independent channels measuring the same variable.

1.5.4 Instrument Channel Calibration

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel test.

1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power.

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a heat balance on the secondary side of the steam generator considering all heat losses and additions.

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100 \left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right)$$

The power in any quadrant is determined from the power range channel displayed on the console for that quadrant. The average power is determined from an average of the outputs of the power range channels. If one of the power range channels is out of service, the incore detectors will be used. The quadrant power tilt limits as a function of power are stated in Specification 3.5.2.4.

1.6.2 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except as in b below.

b. At least one door of the personnel hatch and the emergency hatch is closed and sealed during refueling or during personnel passage through these hatches.

c. All non-automatic containment isolation valves and blind flanges are closed as required.

d. All automatic containment isolation valves are operable or locked closed.

e. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

1.8 ABNORMAL OCCURRENCE

An abnormal occurrence means the occurrence of any plant condition that:

a. Results in exceeding a safety limit, or

b. Results in a protective instrumentation setting less conservative than the Limiting Safety System Setting as established in Technical Specifications, or

c. Exceeds a Limiting Condition for Operation as established in the Technical Specifications, or

d. Causes any significant uncontrolled or unplanned release of radioactive material from the site, or

e. Results in the failure of one or more components of an Engineered Safety System or Reactor Protective System that causes or threatens to cause the system to be incapable of performing its intended function, or

f. Results in abnormal degradation of one of the several boundaries which are designed to contain the radioactive materials resulting from the fission process, or

g. Results in uncontrolled or unanticipated changes in reactivity greater than 1% $\Delta k/k$ except for trip.

1.9 UNUSUAL EVENTS

An unusual event is:

a. Discovery of any substantial errors in the transient or accident analyses, or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the Technical Specifications.

b. Any substantial variance from performance specifications contained in the Technical Specifications or the Safety Analysis Report.

c. Any observed inadequacy in the implementation of administrative or procedural controls during the operation of the facility which could significantly affect the safety of operations.

d. Any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

e. Any occurrence arising from natural or off site man-made events that affect or threaten to affect the safe operation of the plant.

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1A - Unit 1. If the actual pressure/temperature point is below
2.1-1B - Unit 2

and to the right of the line, the safety limit is exceeded.

The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2A - Unit 1. If the actual-reactor-thermal-power/
2.1-2B - Unit 2

power-imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the W-3 correlation.(1) The W-3 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.3. A DNBR of 1.3 corresponds to a 94.3% probability at a 99% confidence level that DNB will not occur; this is considered a conservative margin to DNB for all

operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip set points to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A represents the conditions at which a
2.1-1B

minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112%) when four reactor coolant pumps are operating(minimum reactor coolant flow is 131.3 x 10^6 lbs/hr. This curve is based on the following nuclear power peaking factors(2) with potential fuel densification effects;

$$F_q^N = 2.67; \; F_{\Delta H}^N = 1.78; \; F_z^N = 1.50$$

The design peaking combination results in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2A are based on the more restrictive of two thermal
2.1-2B

limits and include the effects of potential fuel densification:

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.67$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.3 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.1 kw/ft - Unit 1
19.8 kw/ft - Unit 2

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, 3, and 4 of Figure 2.1-2A correspond
2.1-2B

to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

The curve of Figure 2.1-1A is the most restrictive of all possible reactor
2.1-1B

coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.
2.1-3B

The curves of Figure 2.1-3A represent the conditions at which a minimum DNBR
2.1-3B

of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 15%,(3) whichever condition is more restrictive.

Using a local quality limit of 15% at the point of minimum DNBR as a basis for Curves 2 and 4 of Figure 2.1-3A is a conservative criterion even though the
2.1-3B

quality of the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the W-3 correlation continually increases from point of minimum DNBR, so that the exit DNBR is 1.7 or higher, depending on the pressure. Extrapolation of the W-3 correlation beyond its published quality range of +15% is justified on the basis of experimental data.(4)

The maximum thermal power for three pump operation is 87% - Unit 1
86% - Unit 2

due to a power level trip produced by the flux-flow ratio 75% flow x 1.08 = 81% power
1.07 = 80%

plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For each curve of Figure 2.1-3A, a pressure-temperature point above and to the
2.1-3B

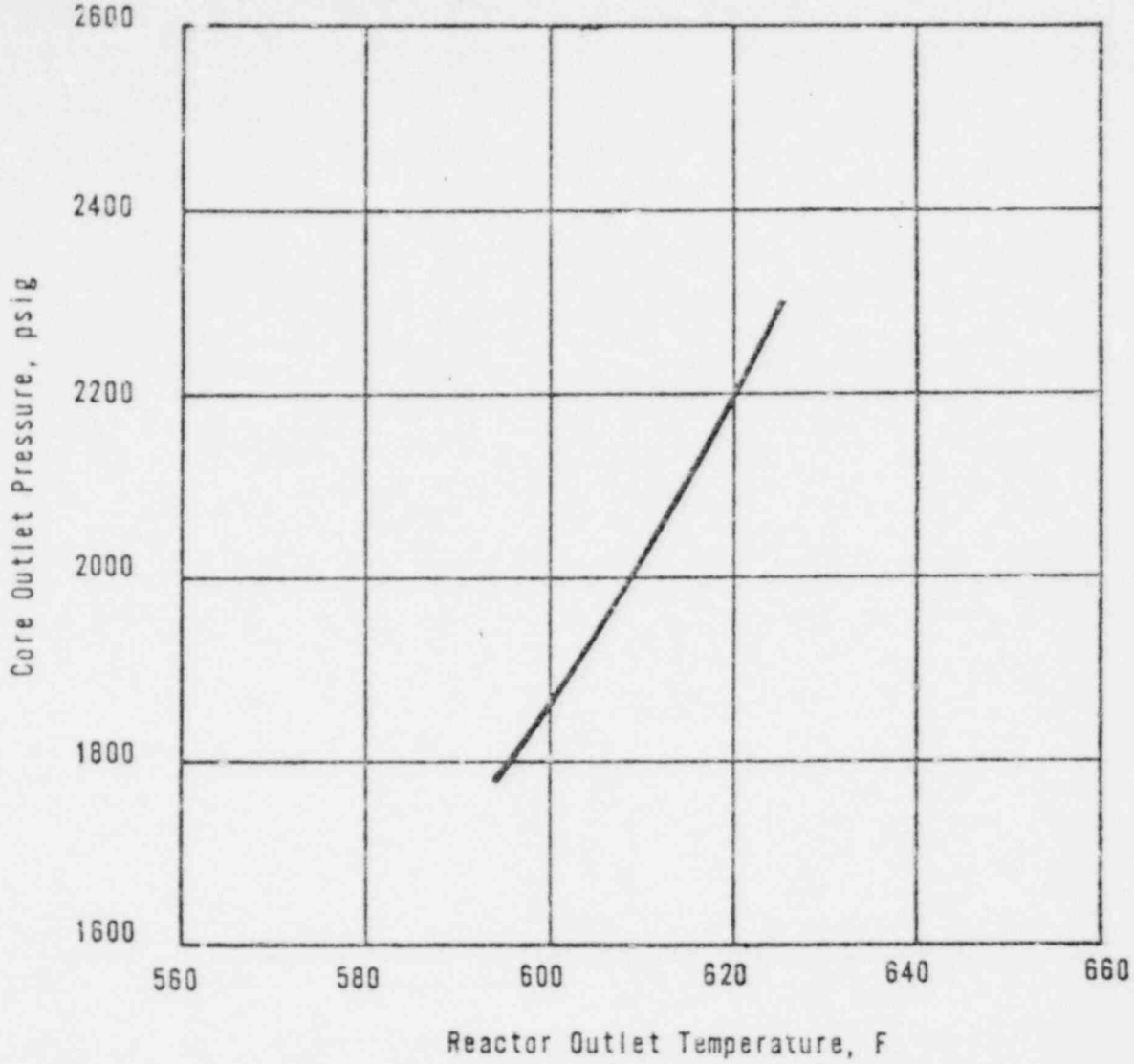
left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 15% for that particular reactor coolant pump situation. The 1.3 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

REFERENCES

(1) FSAR, Section 3.2.3.1.1
(2) FSAR, Section 3.2.3.1.1.c
(3) FSAR, Section 3.2.3.1.1.k
(4) The following papers which were presented at the Winter Annual Meeting, ASME, November 18, 1969, during the "Two-phase Flow and Heat Transfer in Rod Bundles Symposium":

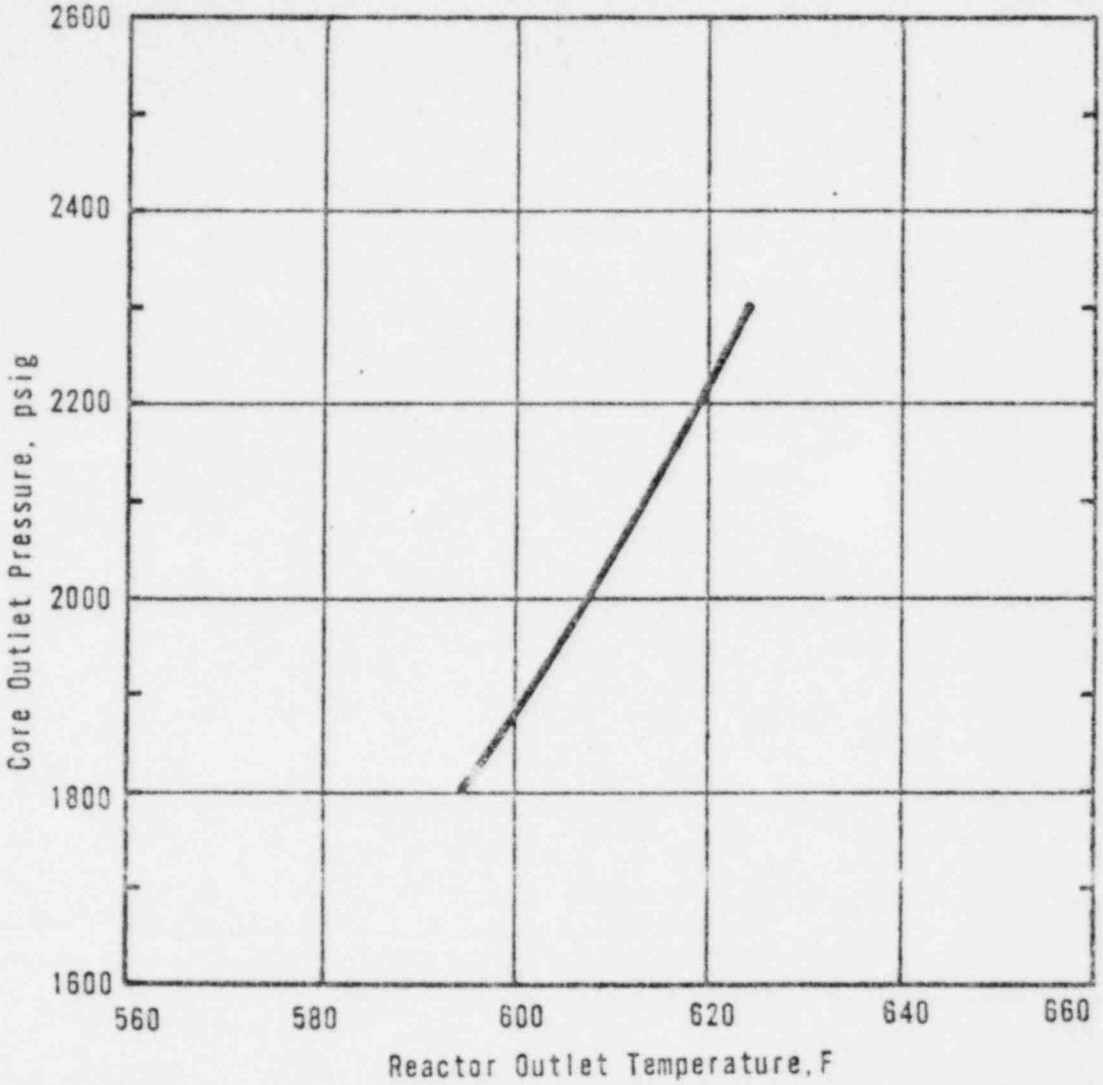
(a) Wilson, et. al.
"Critical Heat Flux in Non-Uniform Heater Rod Bundles."

(b) Gellerstedt, et. al.
"Correlation of a Critical Heat Flux in a Bundle Cooled by Pressurized Water."



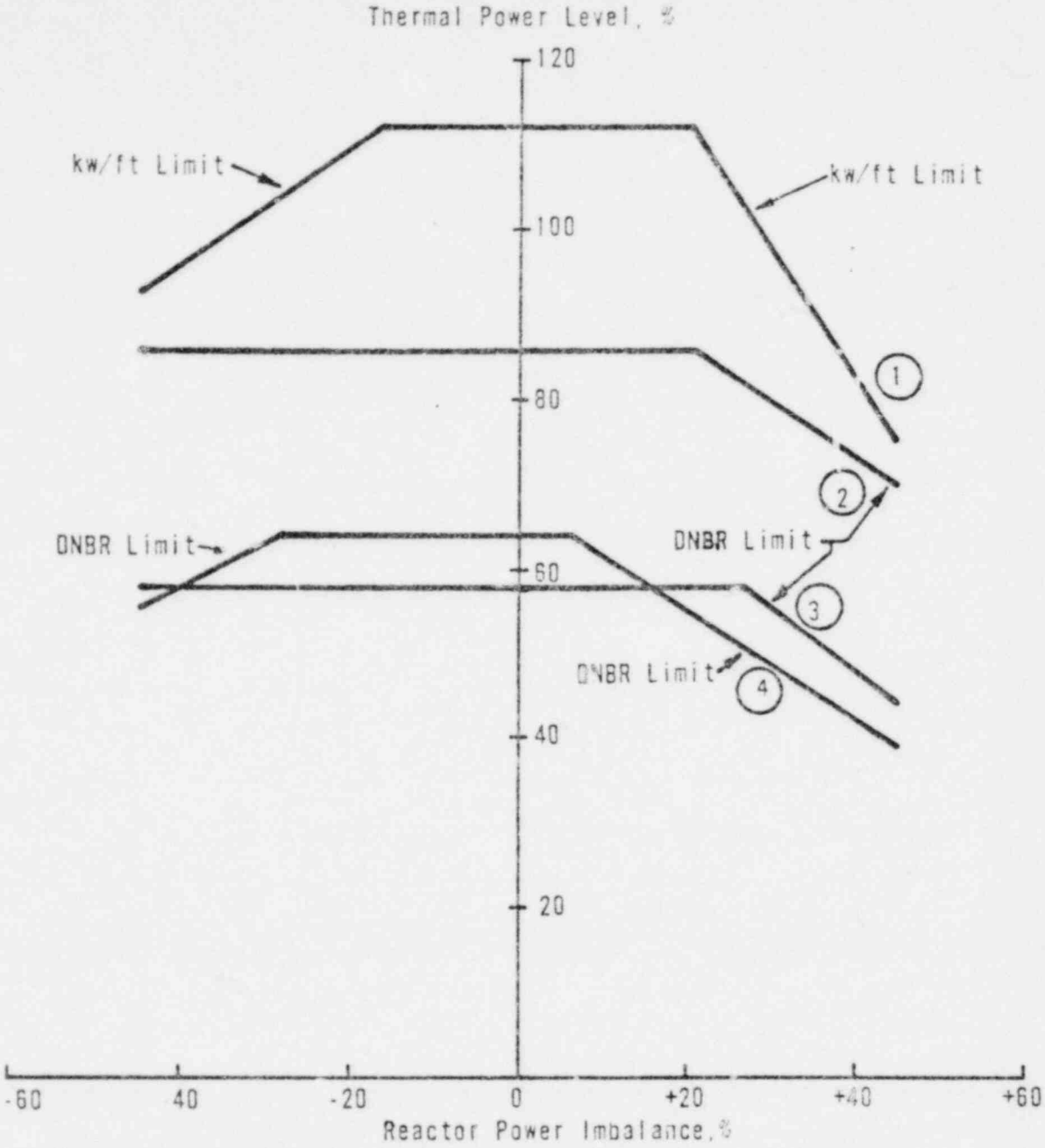
CORE PROTECTION SAFETY LIMIT

UNIT 1
OCONEE NUCLEAR STATION
Figure 2.1 - 1A



CORE PROTECTION SAFETY LIMIT

UNIT 2
OCONEE NUCLEAR STATION
Figure 2.1 - 1B



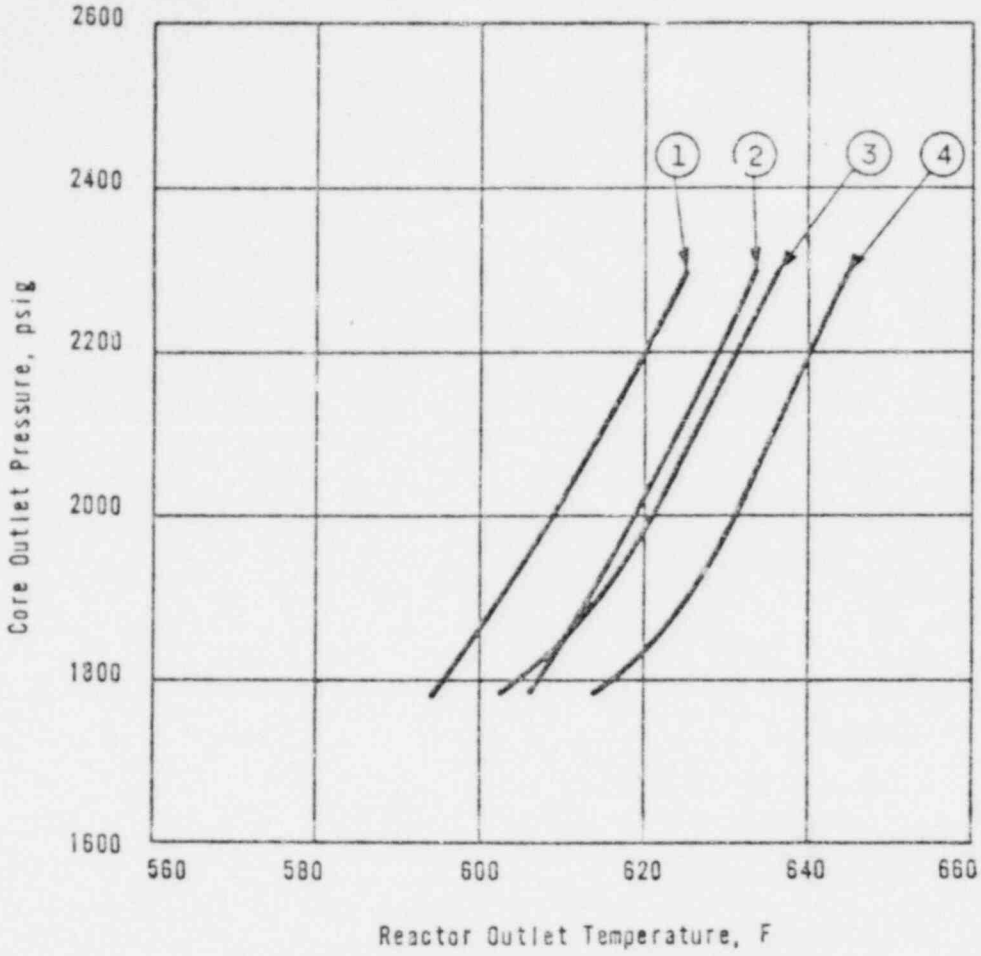
CURVE	REACTOR COOLANT FLOW (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6
4	60.1×10^6

CORE PROTECTION SAFETY LIMITS

UNIT 2

OCONEE NUCLEAR STATION

Figure 2.1 - 2B



CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	131.3×10^6 (100%)	112%	FOUR PUMPS (DNBR LIMIT)
2	60.1×10^6 (45.8%)	65%	TWO PUMPS IN ONE LOOP (QUALITY LIMIT)
3	98.1×10^6 (74.7%)	87%	THREE PUMPS (DNBR LIMIT)
4	64.4×10^6 (49.0%)	59%	ONE PUMP IN EACH LOOP (QUALITY LIMIT)

CORE PROTECTION SAFETY LIMITS

UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1 - 3A

2.2 SAFETY LIMITS - REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.

Specification

2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1967.

Bases

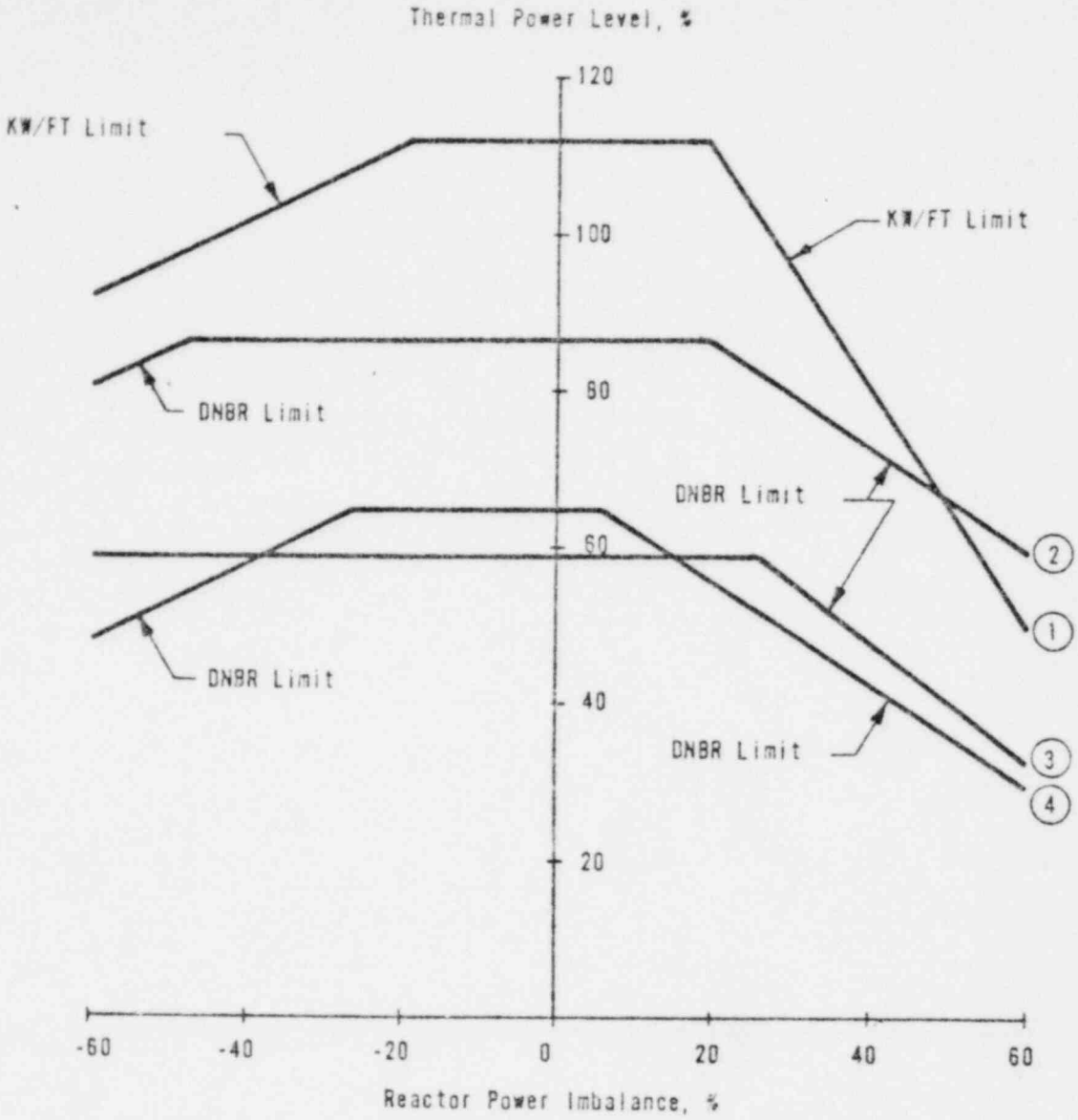
The reactor coolant system[1] serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110% of design pressure.[2] The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under USAS Section B31.7 is 110% of design pressure. Thus, the safety limit of 2750 psig (110% of the 2500 psig design pressure) has been established.[2] The settings, the reactor high pressure trip (2355 psig) and the pressurizer safety valves (2500 psig)[3] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psig (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the Reactor Coolant pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2255 psig.

REFERENCES

(1) FSAR, Section 4

(2) FSAR, Section 4.3.10.1

(3) FSAR, Section 4.2.4



Curve	Reactor Coolant Flow (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6
4	60.1×10^6

CORE PROTECTION SAFETY LIMITS

UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1 - 2A

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protective action to prevent any combination of process variables from exceeding a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1A - Unit 1 and
2.3-1B - Unit 2

Figure 2.3-2A - Unit 1.
2.3-2B - Unit 2.

The pump monitors shall produce a reactor trip for the following conditions:

a. Loss of two pumps and reactor power level is greater than 55% of rated power.

b. Loss of two pumps in one reactor coolant loop and reactor power level is greater than 0.0% of rated power. (Reactor power level trip setpoint is reset to 55% of rated power for single loop operation.)

c. Loss of one or two pumps during two-pump operation.

Bases

The reactor protective system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protective system instrumentation are listed in Table 2.3-1A - Unit 1. The safety analysis has been based upon these protective
2.3-1B - Unit 2

system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis.(4)

Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108% and reactor flow rate is 100%, or flow rate is 93% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 81.0% and reactor flow rate is 74.7% or flow rate is 69% and power level is 75%.

3. Trip would occur when two reactor coolant pumps are operating in a single loop if power is 59% and the operating loop flow rate is 54.5% or flow rate is 43% and power level is 46%.

4. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 53% and reactor flow rate is 49.0% or flow rate is 45% and the power level is 49%.

For safety calculations the maximum calibration and instrumentation errors for the power level trip were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2

level trip and associated reactor power/reactor power-imbalance boundaries by
1.08% - Unit 1 for a 1% flow reduction.
1.07% - Unit 2

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure set point is reached before the nuclear overpower trip set point. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2

for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(1)

The low pressure (1800 psig) and variable low pressure (16.25 T_{out}-7769) trip
(16.25 T_{out}-7756)

setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B

ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction.(2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of (16.25 T_{out} -7809)
(16.25 T_{out} -7796)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B

temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620 F.

Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1A. Two conditions are imposed when

2.3-1B

the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0% of rated power during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The over power trip set point of $\leq$ 5.0% prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0% of rated power if none of the reactor coolant pumps were operating.

Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped. After the pump contact monitor trip has occurred the following actions will permit single loop operation:

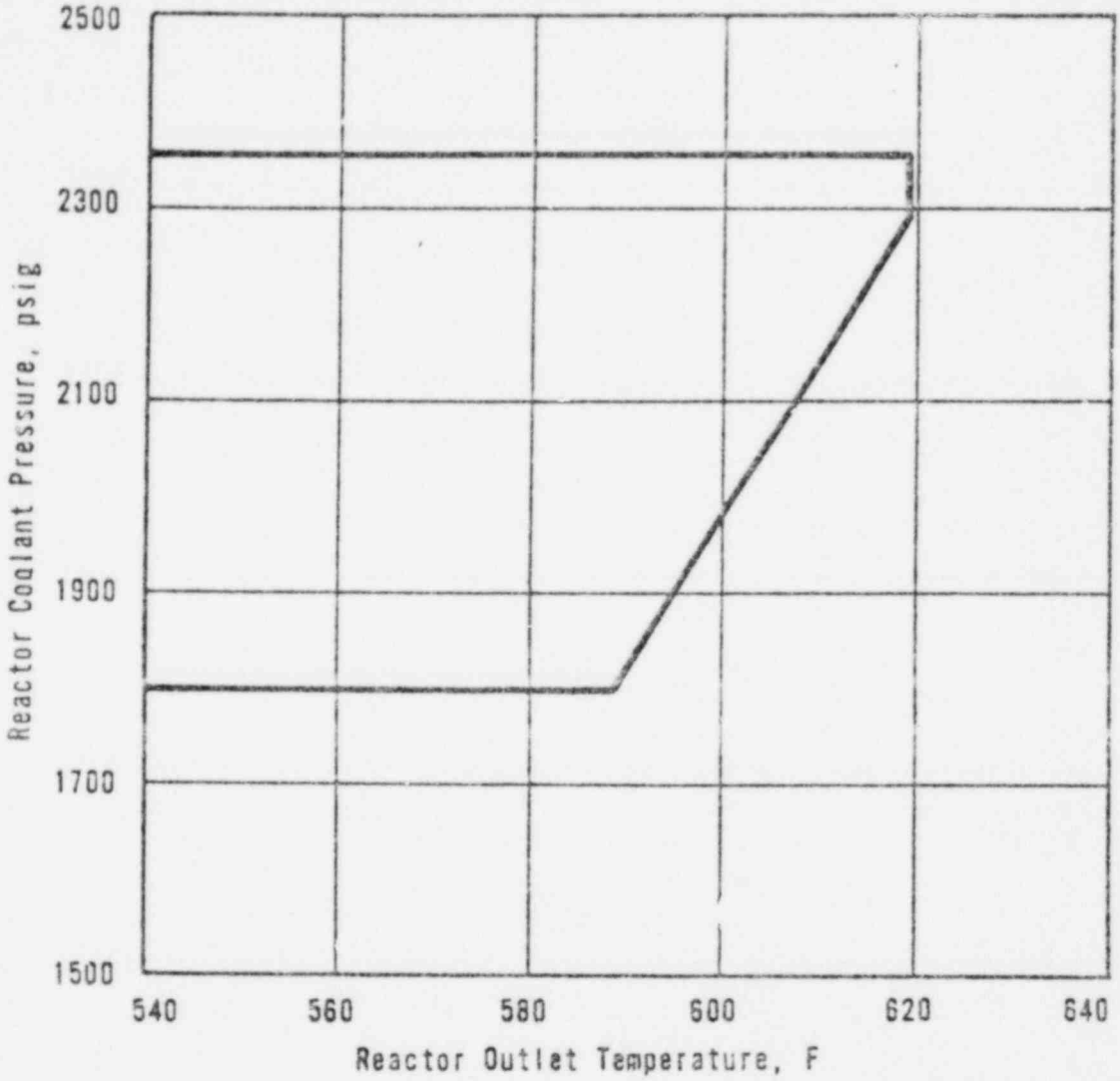
1. Reset the pump contact monitor power level trip set point to 55.0%.

2. Trip one of the two protective channels receiving outlet temperature information from sensors in the idle loop.

Tripping one of the two protection channels receiving outlet temperature information from the idle loop assures a protective system trip logic of one out of two.

REFERENCES

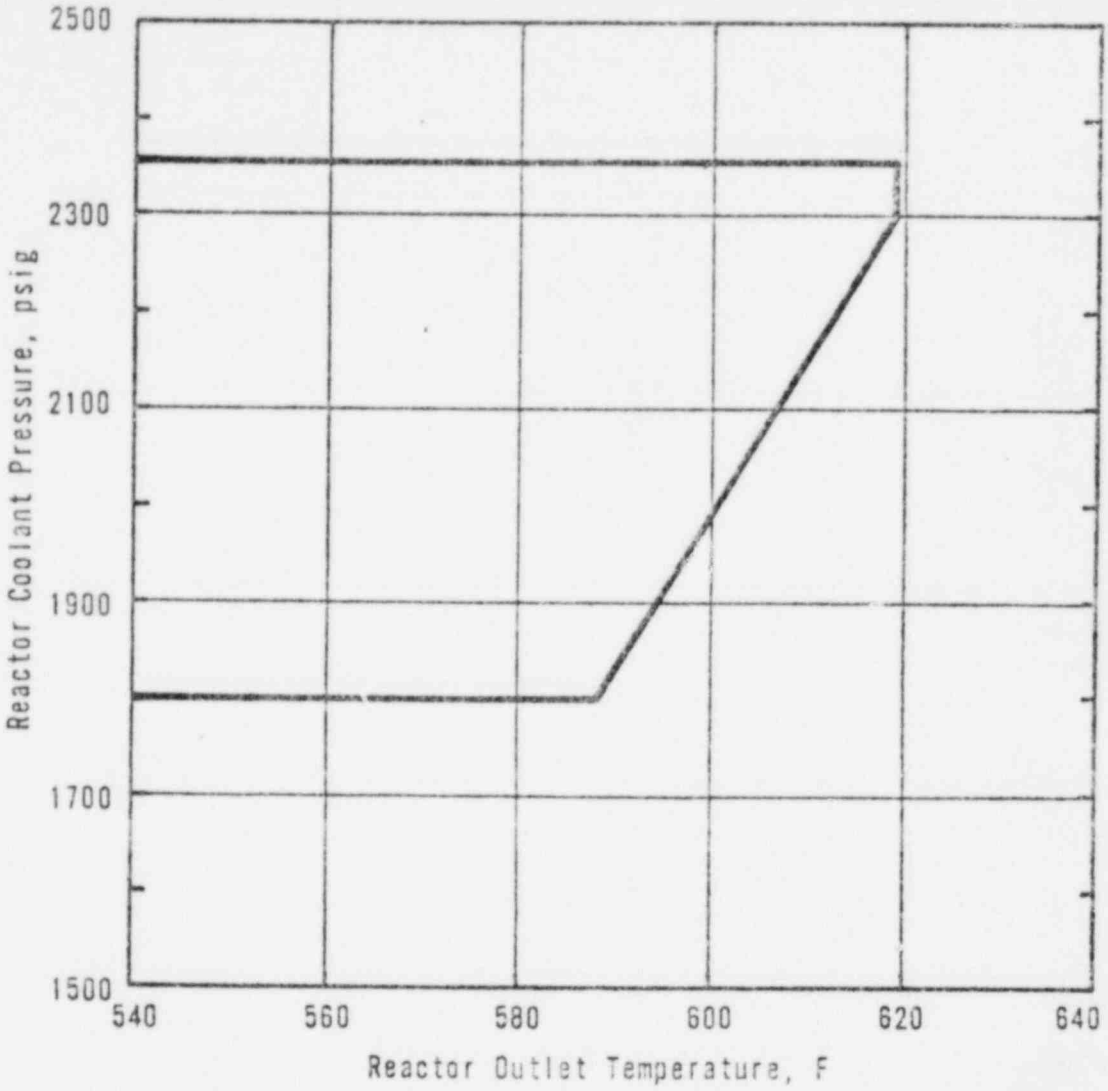
(1) FSAR, Section 14.1.2.2
(2) FSAR, Section 14.1.2.7
(3) FSAR, Section 14.1.2.8
(4) FSAR, Section 14.1.2.3
(5) FSAR, Section 14.1.2.6



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS

UNIT 1

OCONEE NUCLEAR STATION
Figure 2.3 - 1A

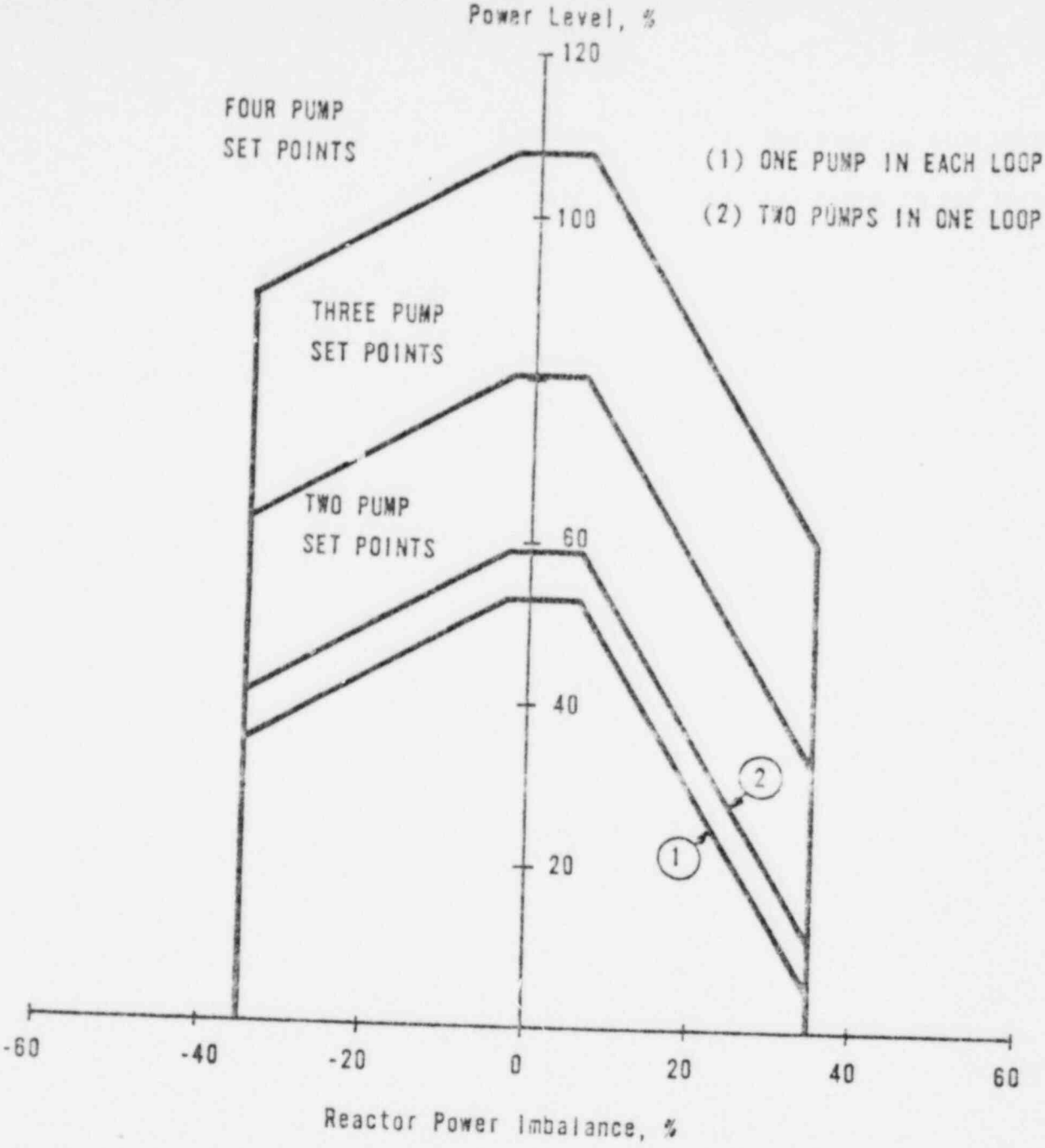


PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS

UNIT 2

OCONEE NUCLEAR STATION

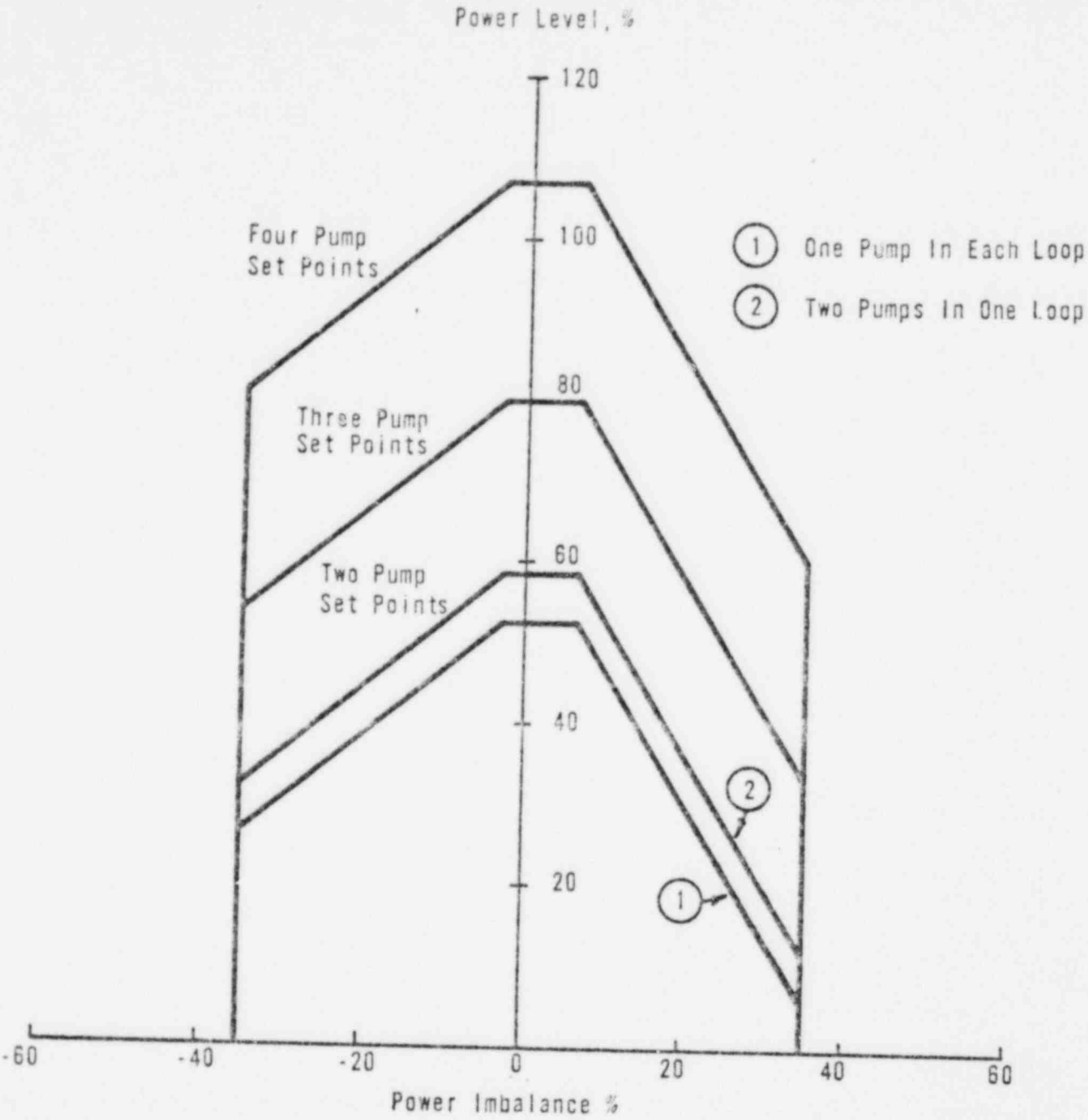
Figure 2.3 - 1B



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS

UNIT 1

OCONEE NUCLEAR STATION
Figure 2.3 - 2A



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SET POINTS

UNIT 2

OCONEE NUCLEAR STATION

Figure 2.3 - 2B

Table 2.3-1A

Reactor Protective System Trip Setting Limits

RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1. Nuclear Power Max. (% Rated)	105.5	105.5	105.5	105.5	5.0(3)
2. Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3. Nuclear Power Max. Based on Pump Monitors, (%, Rated)	NA	NA	55% (5)(6)	55%	Bypassed
4. High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	2355	1720(4)
5. Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	1800	Bypassed
6. Variable Low Reactor Coolant System Pressure psig, Min.	$(16.25\ T_{out} - 7769)$(1)	$(16.25\ T_{out} - 7769)$(1)	$(16.25\ T_{out} - 7769)$(1)	$(16.25\ T_{out} - 7769)$(1)	Bypassed
7. Reactor Coolant Temp. F., Max.	619	619	619 (6)	619	619
8. High Reactor Building Pressure, psig, Max.	4	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (°F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

(5) Reactor power level trip set point produced by pump contact monitor reset to 55.0%.

(6) Specification 3.1.8 applies. Trip one of the two protection channels receiving outlet temperature information from sensors in the idle loop.

Table 2.3-18

Reactor Protective System Trip Setting Limits

RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	Two Reactor Coolant Pumps Operating in A Single Loop (Operating Power -46% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1. Nuclear Power Max. (% Rated)	105.9	105.5	105.5	105.5	5.0(3)
2. Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	Bypassed
3. Nuclear Power Max. Based on Pump Monitors, (%, Rated)	NA	NA	55% (5)(6)	55%	Bypassed
4. High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	2355	1720(4)
5. Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	1800	Bypassed
6. Variable Low Reactor Coolant System Pressure psig, Min.	$(16.25\ T_{out}-7756)$(1)	$(16.25\ T_{out}-7756)$(1)	$(16.25\ T_{out}-7756)$(1)	$(16.25\ T_{out}-7756)$(1)	Bypassed
7. Reactor Coolant Temp. F., Max.	619	619	619 (6)	619	619
8. High Reactor Building Pressure, psig, Max.	4	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (^{o}F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

(4) Automatically set when other segments of the RPS are bypassed.

(5) Reactor power level trip set point produced by pump contact monitor reset to 55.0%.

(6) Specification 3.1.8 applies. Trip one of the two protection channels receiving outlet temperature information from sensors in the idle loop.

3 LIMITING CONDITIONS FOR OPERATION

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. 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

A reactor coolant pump or low pressure injection pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One low pressure injection pump will circulate the equivalent of the reactor coolant system volume in one half hour or less. (1)

The low pressure injection system suction piping is designed for 300°F and 370 psig; thus the system with its redundant components can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident at hot shutdown. (5) The pressurizer code safety valve lift set point shall be set at 2500 psig ± 1% allowance for error and each valve shall be capable of relieving 300,000 lb/hr of saturated steam at a pressure no greater than 3% above the set pressure.

REFERENCES

(1) FSAR Tables 9-11 and 4-3 through 4-7.

(2) FSAR Sections 4.2.5.1 and 9.5.2.3.

(3) FSAR Section 4.2.5.4.

(4) FSAR Sections 4.3.10.4 and 4.2.4.

(5) FSAR Sections 4.3.7 and 14.1.2.2.3.

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 Hydro Tests

For thermal steady state system hydro test the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core and to ASME Code Section III limits when no fuel assemblies are present provided:

a. Prior to initial criticality the reactor coolant system temperature is 118°F or greater or

b. After initial criticality and during the first two years of operation the reactor coolant system temperature is 215°F or greater.

3.1.2.2 Leak Tests

a. Leak tests may be conducted under the provisions of 3.1.2.1 a and b above or

b. After initial criticality and during the first two years of operation the system may be tested to a pressure of 1150 psig provided that the system temperature is 175°F or greater.

3.1.2.3 For the first two years of power operation (1.7×10^6 thermal megawatt days) the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-1 and Figure 3.1.2-2, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-1. The heatup rates shall not exceed those shown on Figure 3.1.2-1.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the left of and below the limit line in Figure 3.1.2-2. Cooldown rates shall not exceed those shown on Figure 3.1.2-2.

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the vessel shell is below 100°F.

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 410°F.

3.1.2.6 Within two years of power operation, Figures 3.1.2-1 and 3.1.2-2 shall be updated in accordance with appropriate criteria accepted by the AEC.

Bases

All reactor coolant system compor s are designed to withstand the effects of cyclic loads due to system te rature and pressure changes. (1) These cyclic loads are int.duced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown of 100°F per hour satisfies stress limits for cyclic operation. (2) The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT. (3) The reactor vessel plate material and welds have been tested to verify conformity to specified requirements and a maximum NDTT value of 20°F has been determined based on Charpy V-Notch tests. The maximum NDTT value obtained for the steam generator shell material and welds was 40°F.

Figures 3.1.2-1 and 3.1.2-2 contain the limiting reactor coolant system pressure-temperature relationship for operation at DTT(4) and below to assure that stress levels are low enough to preclude brittle fracture. These stress levels and their bases are defined in Section 4.3.3 of the FSAR.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with accumulated nuclear operation. The predicted maximum NDTT increase for the 40-year exposure is shown on Figure 4.10:(4) The actual shift in NDTT will be determined periodically during plant operation by testing of irradiated vessel material samples located in this reactor vessel.(5) The results of the irradiated sample testing will be evaluated and compared to the design curve (Figure 4-11 of FSAR) being used to predict the increase in transition temperature.

The design value for fast neutron (E > 1 MeV) exposure of the reactor vessel is 3.0×10^{10} n/cm^2--s at 2,568 MWt rated power and an integrated exposure of 3.0×10^{19} n/cm^2 for 40 years operation. (6) The calculated maximum values are 2.2×10^{10} n/cm^2--s and 2.2×10^{19} n/cm^2 integrated exposure for 40 years operation at 80 percent load. (4) Figure 3.1.2-1 is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1.2-1 is based on the projected NDTT at the end of the first two years of operation. During these two years, the energy output has been conservatively estimated to be 1.7×10^6 thermal megawatt days, which is equivalent to 655 days at 2,568 MWt core power. The projected fast neutron exposure of the reactor vessel for the two years is 1.7×10^{18} n/cm^2 which is based on the 1.7×10^6 thermal megawatt days and the design value for fast neutron exposure.

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for the increases in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are peri ally changed to stay within the established stress li its duri tup and cooldown.

The NDTT shift and the magnitudes of the thermal and pressure stresses are sensitive to integrated reactor power and not to instantaneous power level. Figure 3.1.2-1 and 3.1.2-2 are applicable to reactor core thermal ratings up to 2,568 MWt.

The pressure limit line on Figure 3.1.3-1 has been selected such that the reactor vessel stress resulting from internal pressure will not exceed 15 percent yield strength considering the following:

1. A 25 psi error in measured pressure.

2. System pressure is measured in either loop.

3. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

For adequate conservatism in fracture toughness including size (thickness) effect, a maximum pressure of 550 psig below 275°F with a maximum heatup and cooldown rate of 50°F/hr has been imposed for the initial two year period as shown on Figure 3.1.2-1. During this two year period, a fracture toughness criterion applicable to Oconee Unit 1 beyond this period will be developed by the AEC. It will be based on the evaluation of the fracture toughness properties of heavy section (thickness) steels, both irradiated and unirradiated, for the AEC-HSST program and the PVRC program, and with considerations of test results of the Oconee Unit 1 reactor surveillance program.

The spray temperature difference restriction is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

REFERENCES

(1) FSAR Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

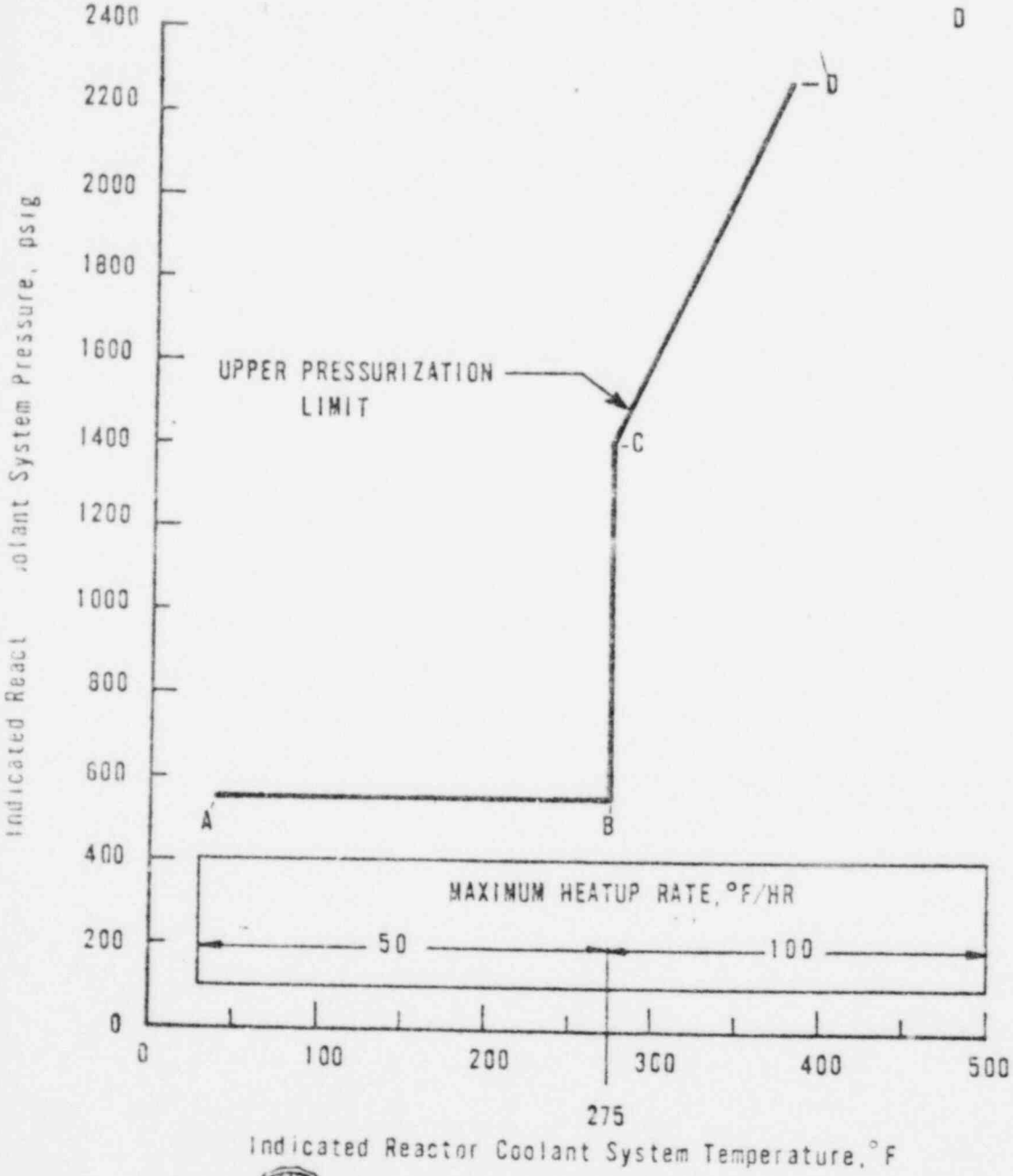
(3) FSAR Section 4.3.10.5

(4) FSAR Section 4.3.3

(5) FSAR Section 4.4.6

(6) FSAR Sections 4.1.2.8 and 4.3.3

POINT	TEMP.	PRESS.
A	40	550
B	275	550
C	275	1400
D	380	2275



OCONEE NUCLEAR STATION

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
(APPLICABLE UP TO AN INTEGRATED EXPOSURE OF 1.7×10^{18} n/cm^2 OR DTT 144 °F)

Figure 3.1.2-1

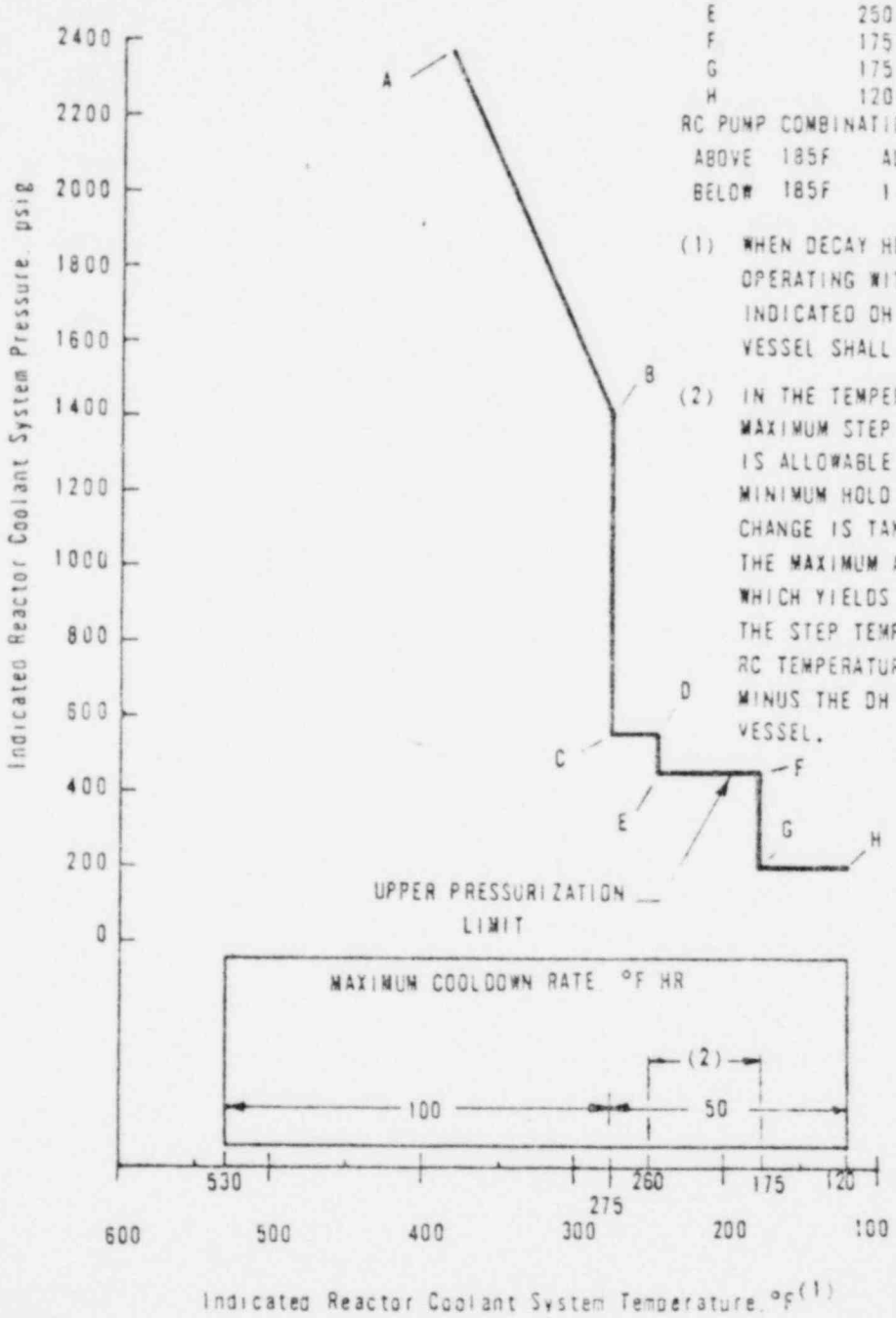
POINT	TEMP	PRESS
A	380	2275
B	275	1400
C	275	550
D	250	550
E	250	450
F	175	450
G	175	200
H	120	200

RC PUMP COMBINATIONS ALLOWABLE

ABOVE	185F	ALL
BELOW	185F	1-A,1-B;0-A,2-B;1-A,0-B;0-A,1-B

(1) WHEN DECAY HEAT REMOVAL SYSTEM (DH) IS OPERATING WITHOUT ANY RC PUMPS OPERATING, INDICATED DH RETURN TEMP TO THE REACTOR VESSEL SHALL BE USED.

(2) IN THE TEMPERATURE RANGE 260F TO 175F, A MAXIMUM STEP TEMPERATURE CHANGE OF 75F IS ALLOWABLE FOLLOWED BY A ONE HOUR MINIMUM HOLD ON TEMPERATURE. IF THE STEP CHANGE IS TAKEN BELOW 250F RC TEMPERATURE, THE MAXIMUM ALLOWABLE STEP SHALL BE THAT WHICH YIELDS A FINAL TEMPERATURE OF 175F. THE STEP TEMPERATURE CHANGE IS DEFINED AS RC TEMPERATURE(BEFORE STOPPING ALL RC PUMPS) MINUS THE DH RETURN TEMPERATURE TO THE REACTOR VESSEL.



REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
(APPLICABLE UP TO DTT = 185°F)

OCONEE NUCLEAR STATION
Figure 3.1.2 - 2

3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above DTT + 10°F.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1%Δk/k until a steam bubble is formed and a water level between 80 and 396 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality and before the reactor is critical.

Bases

At the beginning of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods.(1) Calculations show that above 525°F, the consequences are acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature,(2) startup and operation of the reactor when reactor coolant temperature is less than 525°F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient(2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1%Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(1) and the small integrated Δk/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below DTT + 10°F provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1% subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident of a start-up accident.[3] ,

The requirement that the safety rod groups be fully withdrawn before criticality provides an increased shutdown margin during startup.

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.4

(3) FSAR, Supplement 3, Answer 14.4.1

3.1.4 Reactor Coolant System Activity

Specification

The total activity of the reactor coolant due to nuclides with half lives longer than 30 minutes shall not exceed 224/$\bar{E}$ microcuries per ml whenever the reactor is critical. $\bar{E}$ is the average (mean) beta and gamma energies per disintegration, in MeV, weighted in proportion to the measured activity of the radionuclides in reactor coolant samples.

Bases

The above specification is based on limiting the consequences of a postulated accident involving the double-ended rupture of a steam generator tube. The rupture of a steam generator tube enables reactor coolant and its associated activity to enter the secondary system where volatile isotopes could be discharged to the atmosphere through condenser air-ejectors and through steam safety valves (which may lift momentarily). Since the major portion of the activity entering the secondary system is due to noble gases, the bulk of the activity would be discharged to the atmosphere. The activity release continues until the operator stops the leakage by reducing the reactor coolant system pressure below the set point of the steam safety valves and isolates the faulty steam generator. The operator can identify the faulty steam generator by using the N^{16} detectors on the steam lines in conjunction with the off-gas monitors on the condenser air ejector lines; thus he can isolate the faulty steam generator within 34 minutes after the tube break occurred. During that 34 minute period, a maximum of 2760 ft^3 of 580°F reactor coolant leaked into the secondary system. (This is equivalent to a cold makeup volume of 1980 ft^3).

The activity discharged to the atmosphere as the result of a steam generator tube rupture will not be increased by the loss of station power since condenser cooling water flow can be maintained by gravity flow from Lake Keowee through the emergency condenser cooling water discharge to the Keowee Hydro tailrace.

The controlling dose for the steam generator tube rupture accident is the whole-body dose resulting from immersion in the cloud of released activity. To insure that the public is adequately protected, the specific activity of the reactor coolant will be limited to a value which will insure that the whole-body dose at the site boundary will not exceed 0.5 Rem should a steam generator tube rupture accident occur.

Although only volatile isotopes will be released from the secondary system, the following whole-body dose calculation conservatively assumes that all of the radioactivity which enters the secondary system with the reactor coolant is released to the atmosphere. Both the beta and gama radiation from these isotopes contribute to the whole-body dose. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employs the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. It is further assumed that meteorological conditions during the course of the accident correspond to Pasquill Type F and 1 meter per second wind speed, resulting in a X/Q value of 1.16×10^{-4} sec/m^3, which includes a correction factor of 2.2 to the dilution calculated by the Pasquill method. This correction factor was shown

appropriate by on site diffusion measurements using SF_6 (sulfur hexafluoride) as a gas tracer.

The combined gamma and beta whole body dose from a semi-infinite cloud is given by:

$$\text{Dose (Rem)} = 1/2[\overline{E}\cdot A\cdot V\cdot X/Q\cdot(3.7 \times 10^{10}\ \text{dps/Ci})\cdot(1.33 \times 10^{-11}\ \text{Rem/MeV/m}^3)]$$

$$\text{Dose (Rem)} = 0.246\cdot\overline{E}\cdot A\cdot V\cdot X/Q$$

$$A_{max}(\mu c/cc) = \frac{(\text{Dose})_{max}}{0.246\cdot\overline{E}\cdot V\cdot X/Q} = \frac{0.5}{0.246 \times \overline{E} \times 78.25 \times 1.16 \times 10^{-4}}$$

$$A_{max}(\mu c/cc) = 224/\overline{E}$$

Where

A = Reactor coolant activity (μCi/ml = Ci/m^3)

V = Reactor coolant volume at 580°F leaked into secondary system (2763 ft^3 = 78.25 m^3)

X/Q = Atmospheric dispersion coeffieient at site boundary for a two hour period (1.16×10^{-4} sec/m^3)

$\overline{E}$ = Average beta and gamma energies per disintegration (MeV) corrected to operating temperature and pressure.

Calculations required to determine $\overline{E}$ will consist of the following:

1. Quantitative measurement of the specific activity (in units of μCi/cc) of radionuclides with half lives longer than 30 minutes, which make up at least 95% of the total activity in reactor coolant samples.

2. A determination of the average beta and gamma decay energies per disintegration for each nuclide, measured in (1) above, by utilizing known decay energies and decay schemes (e.g., Table of Isotopes, Sixth Edition, March 1968).

3. A calculation of $\overline{E}$ by the average beta and gamma energy for each radionuclide in proportion to its specific activity, as measured in (1) above.

REFERENCES

FSAR, Section 14.1.2.10

3.1.5 Chemistry

Specification

3.1.5.1 If the concentration of oxygen in the primary coolant exceeds 0.1 ppm during power operation, corrective action shall be initiated within eight hours to return oxygen levels to $\leq$ 0.1 ppm.

3.1.5.2 If the concentration of chloride in the primary coolant exceeds 0.10 ppm during power operation, corrective action shall be initiated within eight hours to return chloride levels to $\leq$ 0.10 ppm.

3.1.5.3 If the concentration of fluorides in the primary coolant exceeds 0.10 ppm following modifications or repair to the primary system involving welding, corrective action shall be initiated within eight hours to return fluoride levels to $\leq$ 0.10 ppm.

3.1.5.4 If the concentration limits of oxygen, chloride or fluoride in 3.1.5.1, 3.1.5.2 and 3.1.5.3 above are not restored within 24 hours the reactor shall be placed in a hot shutdown condition within 12 hours thereafter. If the normal operational limits are not restored within an additional 24-hour period, the reactor shall be placed in a cold shutdown condition within 24-hours thereafter.

3.1.5.5 If the oxygen concentration and the chloride or fluoride concentration of the primary coolant system individually exceed 1.0 ppm. the reactor shall be immediately brought to the hot shutdown condition using normal shutdown procedure and action is to be taken immediately to return the system to within normal operation specifications. If normal operating specifications have not been reached in 12 hours, the reactor shall be brought to a cold shutdown condition using normal procedure.

Bases

By maintaining the chloride, fluoride and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack. (1,2)

The oxygen concentration in the reactor coolant system is normally expected to be below detectable limits since dissolved hydrogen is used when the reactor is critical and a residual of hydrazine is used when the reactor is subcritical to control the oxygen. The requirement that the oxygen concentration not exceed 0.1 ppm during power operation is added assurance that stress corrosion cracks will not occur. (4)

If the oxygen, chloride, or fluoride limits are exceeded, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchange resin, increase the hydrogen concentration in the makeup tank, etc.) and further because of the time dependent nature of any adverse effects arising from chlorides or oxygen concentrations in excess of the limits, it is unnecessary to shutdown immediately, since the condition can be corrected.

The oxygen and halogen limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time. (4) Thus, the period of eight hours to initiate corrective action and the period of 24 hours to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analyses are correct and to locate the source of contamination. If corrective action has not been effective at the end of 24 hours, then the reactor coolant system will be brought to the hot shutdown condition within 12 hours and corrective action will continue. If the operational limits are not restored within an additional 24 hour period, the reactor shall be placed in a cold shutdown condition within 24 hours thereafter.

The maximum limit of 1 ppm for the oxygen and halogen concentration that will not be exceeded was selected as the hot shutdown limit because these values have been shown to be safe at 500°F. (3)

REFERENCES

(1) FSAR, Section 4.1.2.7

(2) FSAR, Section 9.2.2

(3) Stress Corrosion of Metals, Logan

(4) Corrosion and Wear Handbook, O. J. DePaul, Editor

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

3.1.6.3 If any reactor coolant leakage exists through a non-isolable fault in a RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2 or 3.1.6.3, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable.

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20.

3.1.6.6 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2 or 3.1.6.3 the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.7 When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different operating principles shall be operable, with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for 48 hours provided two other means to detect leakage are operable.

3.1.6.8 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1, 3.1.6.2, 3.1.6.3, 3.1.6.4, 3.1.6.5, or 3.1.6.6 except that such losses when added to leakage shall not exceed 30 gpm.

Bases

Every reasonable effort will be made to reduce reactor coolant leakage including evaporative losses (which may be on the order of .5 gpm) to the lowest possible rate and at least below 1 gpm in order to prevent a large leak from masking the presence of a smaller leak. Water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the

reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small breaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Operating Staff and will be documented in writing and approved by the Superintendent. Under these conditions, an allowable reactor coolant system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one high pressure injection pump and makeup would be available even under the loss of off-site power condition.

If leakage is to the reactor building it may be identified by one or more of the following methods:

a. The reactor building air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument is sensitive are .10 gpm to greater than 30 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in coolant leakage of 1 gpm is detectable within 10 minutes after it occurs.

b. The iodine monitor, gaseous monitor and area monitor are not as sensitive to corrosion product activity.(1) It is calculated that the iodine monitor is sensitive to an 8 gpm leak and the gaseous monitor is sensitive to a 230 gpm leak based on the presence of tramp uranium (no fission products from tramp uranium are assumed to be present). However, any fission products in the coolant will make these monitors more sensitive to coolant leakage.

c. In addition to the radiation monitors, leakage is also monitored by a level indicator in the reactor building normal sump. Changes in normal sump level may be indicative of leakage from any of the systems located inside the reactor building such as reactor coolant system, low pressure service water system, component cooling system and steam and feedwater lines or condensation of humidity within the reactor building atmosphere. The sump capacity is 15 gallons per inch of height and each graduation on the level indicates 1/2 inch of sump height. This indicator is capable of detecting changes on the order of 7.5 gallons of leakage into the sump. A 1 gpm leak would therefore be detectable within less than 10 minutes.

d. Total reactor coolant system leakage rate is periodically determined by comparing indications of reactor power, coolant temperature, pressurizer water level and letdown storage tank level over a time interval. All of these indications are recorded. Since the pressurizer level is maintained essentially constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the letdown storage tank resulting in a tank level decrease. The letdown storage tank capacity is 31 gallons per inch of height and each graduation on the level recorder represents 1 inch of tank height. This inventory monitoring method is capable of detecting changes on the order of 31 gallons. A 1 gpm leak would therefore be detectable within approximately one half hour.

As described above, in addition to direct observation, the means of detecting reactor coolant leakage are based on 2 different principles, i.e., activity, sump level and reactor constant inventory measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the reactor building.

The upper limit of 30 gpm is based on the contingency of a complete loss of station power. A 30 gpm loss of water in conjunction with a complete loss of station power and subsequent cooldown of the reactor coolant system by the turbine bypass system (set at 1,040 psia) and steam driven emergency feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore electrical power to the station and makeup flow to the reactor coolant system.

REFERENCES

FSAR Section 11.1.2.4.1

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% of rated power is specified such that the maximum clad temperatures will not exceed the Interim Acceptance Criteria based on LOCA analyses. Below 95% of rated power the Interim Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ $\Delta k/k/^{o}F$. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ $\Delta k/k/^{o}F$. 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. $\pm 0.2°F$ 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 $0.16 \times 10^{-4}(\Delta k/k)/°F$ to obtain a 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, α_m, must also be adjusted for the interaction of an average moderator temperature with increased fuel temperatures. This correction is $-.001 \times 10^{-4}$ $\Delta\alpha_m/\Delta\%$ power. It adjusts the 15% power α_m to the moderator coefficient at any power level above 15% power. For example, to correct to 100% power, α_m is adjusted by $(-.001 \times 10^{-4})$ (85%), which is $-.085 \times 10^{-4}\Delta\alpha_m$.

D. Dissolved boron concentration

This correction is for any difference in boron concentration, if required, between zero and full power. Since the moderator coefficient is more positive for greater dissolved boron concentrations, the sign of the correction depends on whether boron is added or removed. The correction is 0.16×10^{-6} $\Delta\alpha_m/\Delta$PPM. (The magnitude of the correction varies slightly with boron concentration; the value presented above, however, is valid for a range in boron concentrations from 1000 to 1400 ppm.)

E. Control rod insertion

This correction is for the difference in control rod worth (% $\Delta k/k$) in the core between zero and full power. The correction is 0.17×10^{-4} $\Delta\alpha_m/\%\Delta k/k$, where the sign for rod worth change is negative for rod insertion, because the moderator coefficient is more negative for a larger rod worth in the core.

F. Isothermal to distributed temperature

The correction for spatially distributed moderator temperature has been found to be less than or equal to zero. Therefore, zero is a conservative correction value for distributed effects.

G. Azimuthal xenon stability

Before commercial operation a test will be performed to verify that divergent azimuthal xenon oscillations do not occur.

REFERENCES

(1) FSAR, Section 14

(2) FSAR, Section 3

(3) FSAR, Section 14.2.2.3.4

3.1.8 Single Loop Restrictions

Specification

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

3.1.8.1 Single loop operation is authorized for test purposes only.

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

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

3.1.8.4 The outlet reactor coolant temperature trip set point shall be set at no greater than 610°F.

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

3.1.8.6 DOL shall be notified of the scheduled date of single loop testing. Upon completion of test, results shall be reported to AEC/DOL. Subsequent single loop operation shall be contingent upon written approval by DOL.

Bases

The purpose of single loop testing is to (1) supplement the 1/6 scale model test information, (2) verify predicted flow through the idle loop, (3) verify that changes in power level do not affect flow distribution or core power distribution, and (4) demonstrate that limiting safety system settings (pump monitor trip set point and reactor coolant outlet temperature trip set point) can be conservatively adjusted taking into account instrument errors.

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

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

3.1.9 Low Power Physics Testing Restrictions

Specification

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

3.1.9.1 Reactor Protective System Requirements

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

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

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

Bases

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

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

3.1.10 Control Rod Operation

Specification

3.1.10.1 Allowable combinations of pressure and temperature for control rod operation shall be to the left of and above the limiting pressure versus temperature curve as shown in Figure 3.1.10-1.

3.1.10.2 The dissolved gas concentration shall not exceed 100 standard cc/liter.

3.1.10.3 If either the limits of 3.1.10.1 or 3.1.10.2 are exceeded, the center control rod drive mechanism shall be checked for accumulation of undissolved gases.

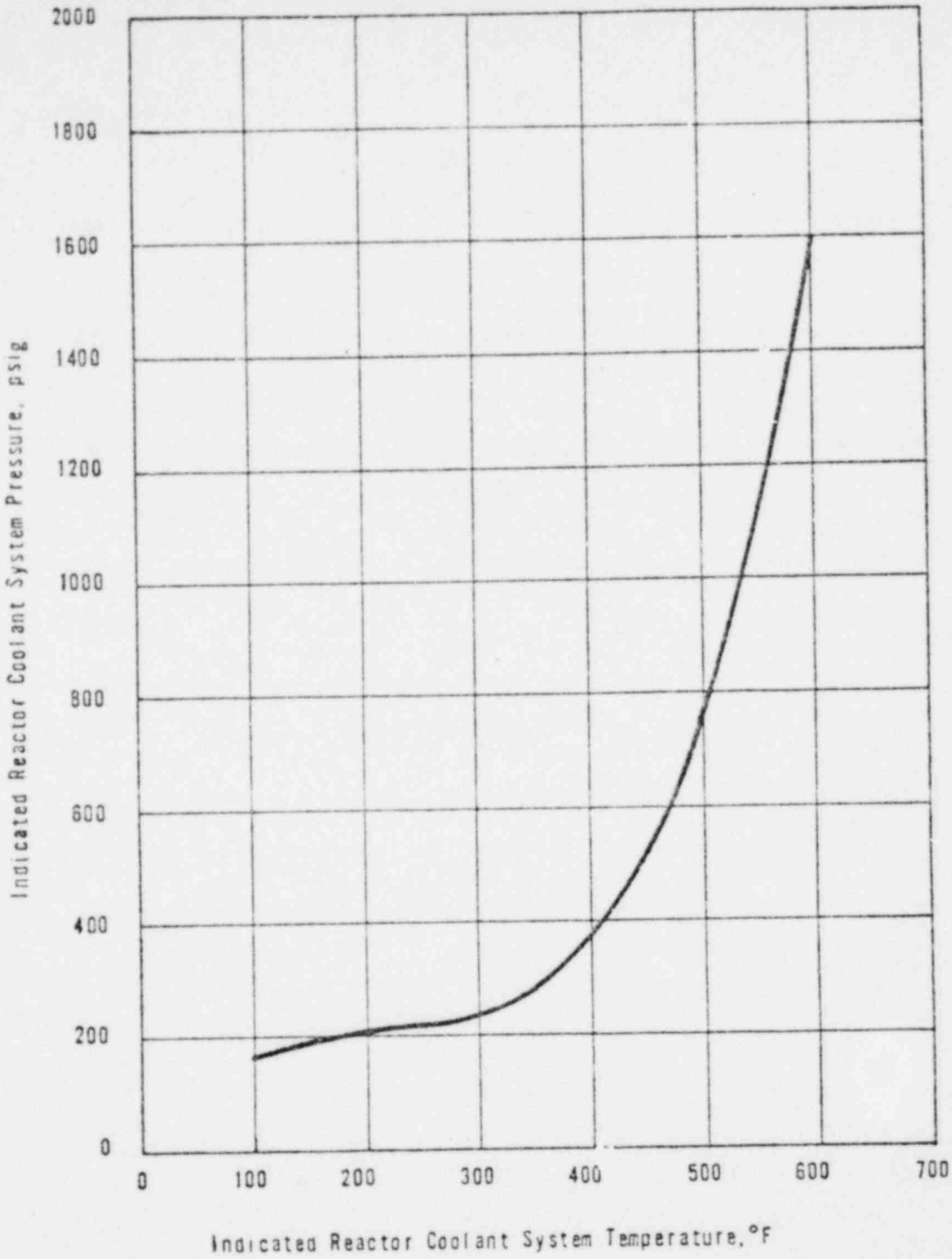
Bases

The limiting pressure versus temperature curve for dissolved gases is determined by the equilibrium pressure versus temperature curve for the dissolved gas concentration of 100 std. cc/liter of water. This equilibrium total pressure is the sum of the partial pressure of the dissolved gases plus the partial pressure of water at a given temperature.

By maintaining the reactor coolant temperature and pressure as specified above, any dissolved gases in the reactor coolant system are maintained in solution.

Although the dissolved gas concentration is expected to be approximately 20-40 std. cc/liter of water, the dissolved gas concentration is conservatively assumed to be 100 std. cc/liter of water at the reactor vessel outlet temperature.

If either the maximum dissolved gas concentration (100 std. cc/liter of water) is exceeded or the operating pressure falls below or to the right of the limiting pressure versus temperature curve, the center CRDM should be checked for accumulation of undissolved gases.



LIMITING PRESSURE VS TEMPERATURE
CURVE FOR 100 STD CC/LITER H_2O

OCONEE NUCLEAR STATION
Figure 3.1.10 - 1

3.2 HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the high pressure injection and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification

The reactor shall not be critical unless the following conditions are met:

3.2.1 Two high pressure injection pumps per unit are operable except as specified in 3.3.

3.2.2 One source per unit of concentrated soluble boric acid in addition to the borated water storage tank is available and operable. This can be either:

a. The boric acid mix tank containing at least 74 inches (450 ft^3) of 10,600 ppm boron as boric acid solution at a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall also be operable and shall have at least the same temperature requirement as the boric acid mix tank. One associated boric acid pump shall be operable. If the daily average air temperature in the vicinity of this tank and associated flow path piping is less than 85°F, at least one channel of heat tracing shall be in operation for this tank and piping.

b. The concentrated boric acid storage tank containing at least 26 inches (550 ft^3) of 8700 ppm boron as boric acid solution with a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall be operable and shall have the same temperature requirement as the concentrated boric acid storage tank. One associated boric acid pump shall be operable. If the daily average air temperature in the vicinity of this tank is less than 70°F, at least one channel of heat tracing shall be in operation for this tank and associated piping.

Bases

The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the concentrated boric acid storage tank. An alternate method of

boration will be use of the high pressure injection pumps taking suction directly from the borated water storage tank (2)

The quantity of boric acid in storage from any of the 3 above mentioned sources is sufficient to borate the reactor coolant system to a 1% subcritical margin in the cold condition at the end of core life. The maximum required is the equivalent of 396 ft^3 of 10,600 ppm boron as boric acid solution. A minimum of 450 ft^3 of 10,600 ppm boron as boric acid solution in the boric acid mix tank, a minimum of 550 ft^3 of 8,700 ppm boron as boric acid solution in the concentrated boric acid storage tank or a minimum of 350,000 gallons of 1800 ppm boron as boric acid solution in the borated water storage tank(3) will satisfy the requirements. The specification assures that at least two of these supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. The quickest method allows for the necessary boron addition in less than one hour. The slowest method (using one 10 gpm pump taking suction from the boric acid storage tank) would require approximately 3 hours to inject enough boron to keep the reactor 1% subcritical with xenon in the core. As xenon decays out, more boron would have to be added. Therefore, in order to account for xenon decay, the 10 gpm pump would pump for something less than 5 hours. At this time, the reactor coolant system would be at a temperature of approximately 175°F and the core would be more than 1% subcritical.

The concentration of boron in the boric acid mix tank and concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept 10°F above the crystallization temperature for the concentration present. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

The boric acid mix tank concentration of 10,600 ppm boron corresponds to a precipitation temperature of 80°F, and the concentrated boric acid storage tank concentration of 8700 ppm corresponds to a precipitation temperature of 68°F. It is expected that the surface temperatures of these tanks and associated piping will be 10°F above the precipitation temperatures. If the air temperature should approach a precipitation temperature, at least one channel of heat tracing in service assures that heat losses to the atmosphere will be made up to maintain this 10°F margin.

REFERENCES

(1) FSAR, Section 9.1; 9.2

(2) FSAR Figure 6.2

(3) Technical Specification 3.3

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING COOLING, REACTOR BUILDING SPRAY, AND PENETRATION ROOM VENTILATION SYSTEMS

Applicability

Applies to the emergency core cooling, reactor building cooling, reactor building spray, and reactor building penetration room ventilation systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building cooling, reactor building spray, and reactor building penetration room ventilation systems.

Specification

3.3.1 The following equipment shall be operable whenever there is fuel in the reactor vessel and reactor coolant pressure is 350 psig or greater or reactor coolant temperature is 250°F or greater:

(a) One reactor building spray pump and its associated spray nozzle header.

(b) Two low pressure service water pumps. The valve in the discharge from the reactor building cooler (LPSW 108 and 2LPSW 108) shall be locked open.

(c) Reactor building penetration room ventilation system consisting of both penetration room fans and their associated filters. Manual operated valves PR-12, PR-14, PR-16, and PR-18 shall be locked open.

(d) A and B Engineered Safety Feature low pressure injection pumps shall be operable.

(e) Two low pressure injection coolers shall be operable.

(f) Two BWST level instrument channels shall be operable.

(g) The borated water storage tank shall contain a minimum level of 46 feet of water having a minimum concentration of 1,800 ppm boron at a temperature not less than 40°F. The manual valve, LP-28, on the discharge line from the borated water storage tank shall be locked open.

(h) The two reactor building emergency sump isolation valves shall be either manually or remote-manually operable.

(i) Two reactor building cooling fans and associated cooling units.

(j) The Engineered Safety Features valves and interlocks associated with each of the above systems shall be operable.

3.3.2 In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350°F and irradiated fuel is in the core:

(a) Two high pressure injection pumps shall be maintained operable to provide redundant and independent flow paths.

(b) Engineered Safety Feature valves and interlocks associated with 3.3.2a above shall be operable.

3.3.3 In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:

(a) The two core flooding tanks shall each contain a minimum of 13 ± .44 ft. (1040 ± 30 ft^3) of borated water at 600 ± 25 psig.

(b) Core flooding tank boron concentration shall not be less than 1,800 ppm boron.

(c) The electrically-operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.

(d) One pressure instrument channel and one level instrument channel per core flood tank shall be operable.

3.3.4 The reactor shall not be made critical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 is operable.

(a) The other reactor building spray pump and its associated spray nozzle header.

(b) The remaining reactor building cooling fan and associated cooling unit.

(c) Engineered Safety Feature valves and interlocks associated with 3.3.4a and 3.3.4b shall be operable.

3.3.5 Except as noted in 3.3.6 below, maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, low pressure service water, reactor building spray, reactor building cooling or penetration room ventilation systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1, 3.3.2, 3.3.3, or 3.3.4, within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.1, 3.3.2, 3.3.3, or 3.3.4 are not met within an additional 48 hours, the reactor shall be placed in a condition below that reactor coolant system condition required in Specification 3.3.1, 3.3.2, 3.3.3, or 3.3.4 for the component degraded.

3.3.6 Exceptions to 3.3.5 shall be as follows:

(a) Both core flooding tanks shall be operational above 800 psig.

(b) Both motor-operated valves associated with the core flooding tanks shall be fully open above 800 psig.

(c) One pressure instrument channel and one level instrument channel per core flood tank shall be operable above 800 psig.

(d) One reactor building cooling fan and associated cooling unit shall be permitted to be out of service for seven days provided both reactor building spray pumps and associated spray nozzle headers are in service at the same time.

3.3.7 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to assure operability.

Bases

The requirements of Specification 3.3 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two high pressure injection pumps and two low pressure injection pumps are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.(1)

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(2)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent freezing. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70°F without any control rods in the core. This concentration is 1,338 ppm boron while the minimum value specified in the tanks is 1,800 ppm boron.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.5 and 3.3.6 provided requirements in Specification 3.3.7 are met which assure operability of the duplicate components. Operability of the specified com-

ponents shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated immediately prior to removal. The basis of acceptability is a likelihood of failure within 24 hours following such demonstration.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the reactor building design pressure will not be exceeded with one spray and two coolers operable. Therefore, a maintenance period of seven days is acceptable for one reactor building cooling fan and its associated cooling unit.(3)

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severence, limit the peak clad temperature to less than 2,300°F and the metal-water reaction to that representing less tha 1 percent of the clad.

Three low pressure service water pumps serve Oconee Units 1 and 2. One low pressure service water pump per unit is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

A single train of reactor building penetration room ventilation equipment retains full capacity to control and minimize the release of radioactive materials from the reactor building to the environment in post-accident conditions.

REFERENCES

(1) FSAR, Section 14.2.2.3
(2) FSAR, Section 9.5.2
(3) FSAR, Section 14.2.2.3.5
(4) FSAR, Section 6.4

3.4 STEAM & POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat.

Objective

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

Specification

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

3.4.1 Capability to remove a decay heat load of 5 percent full reactor power from at least one of the following means:

a. A hotwell pump, condensate booster pump, and a main feedwater pump.
b. The emergency feedwater pump.
c. A hotwell pump and a condensate booster pump.

3.4.2 The sixteen steam system safety valves are operable.

3.4.3 The turbine bypass system shall have four valves operable, except that one valve may be removed from service for maintenance for a period not to exceed 24 hours.

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

3.4.5 The emergency condenser circulating water system shall be operable as per Specification 4.1.

Bases

The 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 feedwater flow required to remove decay heat corresponding to 5 percent full power with saturated steam in the pressure range from 30 psia (saturation pressure at 250°F) to 1065 psia (lowest setting of steam safety valve) as a function of feedwater temperature is:

°F	Flow, GPM
60	750
90	770
120	790
180	840

One hotwell pump plus one condensate booster pump will supply at least 3000 GPM at 550 psia, and one hotwell pump plus one booster pump plus one main

feed pump will supply at least 3000 gpm at 1065 psia. The emergency feed pump will supply 1080 gpm at 1065 psia.

In the event of complete loss of electrical power, feedwater is supplied by a turbine driven emergency feedwater pump which takes suction from the upper surge tanks and hotwell. Decay heat is removed from steam generator 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.

The minimum amount of water in the upper surge tank and condensate storage tank is the amount needed for 11 hours of operating 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 relief capacity of the sixteen steam system safety valves is 13,105,000 lbs/hr. The capacity of the four turbine bypass valves is 2,817,000 lbs/hr.

REFERENCE

FSAR, Section 10

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Columns A and B are met.

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Columns A and B; operation shall be limited as specified in Column C.

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 The key-operated shutdown bypass switch associated with each reactor protective channel shall not be used during reactor power operation.

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sources supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes. The condition will be corrected and the remaining trip devices shall be tested within eight hours. If the condition is not corrected and the remaining trip devices tested within the eight hour period, the reactor shall be placed in the hot shutdown condition within an additional four hours.

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.

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 Superintendent.

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. These keys will not be used during reactor power operation.

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.

REFERENCE

FSAR, Section 7.1

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS

	Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree Of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
1.	Nuclear Instrumentation Intermediate Range Channels	1	0	Bring to hot shutdown within 12 hours (b)
2.	Nuclear Instrumentation Source Range Channels	1	0	Bring to hot shutdown within 12 hours (b)(c)
3.	RPS Manual Pushbutton	1	0	Bring to hot shutdown within 12 hours
4.	RPS Power Range Instrument Channels	3(a)	1(a)	Bring to hot shutdown within 12 hours
5.	RPS Reactor Coolant Temperature Instrument Channels	2(d)	1	Bring to hot shutdown within 12 hours
6.	RPS Pressure-Temperature Instruments Channels	2(d)	1	Bring to hot shutdown within 12 hours
7.	RPS Flux Imbalance Flow Instrument Channels	2	1	Bring to hot shutdown within 12 hours
8.	RPS Reactor Coolant Pressure			
	a. High Reactor Coolant Pressure Instrument Channels	2	1	Bring to hot shutdown within 12 hours
	b. Low Reactor Coolant Pressure Channels	2	1	Bring to hot shutdown within 12 hours
9.	RPS Power-Number of Pumps Instrument Channels	2	1	Bring to hot shutdown within 12 hours
10.	RPS High Reactor Building Pressure Channels	2	1	Bring to hot shutdown within 12 hours

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (Cont'd)

Functional Unit	(A) Minimum Operable Analog 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			
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 & 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)

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (Cont'd)

Functional Unit	(A) Minimum Operable Analog Channels	(B) Minimum Degree Of Redundancy	(C) Operator Action If Conditions Of Column A and B Cannot Be Met
b. Manual Pushbutton	2	1	Bring to hot shutdown within 12 hours (e)
15. Turbine Stop Valves Closure	2	1	Bring to hot shutdown within 12 hours (f)

(a) For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

(d) Single loop operation at power (after testing and approval by the AEC/DOL) is not permitted unless the operating channels are the two receiving Reactor Coolant Temperature from operating loop.

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in the cold shutdown condition within 24 hours.

(f) One operable channel with zero minimum degree of redundancy is allowed for 24 hours before going to the hot shutdown condition.

3.5.2 Control Rod Group and Power Distribution Limits

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall be not less than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

a. Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted.

b. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existance of 1% $\Delta k/k$ hot shutdown margin. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

c. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that a 1% $\Delta k/k$ hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established.

d. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the thermal power allowable for the reactor coolant pump combination.

f. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2, operation may continue provided the rods in the group are positioned such that the rod that was declared inoperable is maintained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5c.

3.5.2.3 The worth of a single inserted control rod shall not exceed 0.5% $\Delta k/k$ at rated power or 1.0% $\Delta k/k$ at hot zero power except for physics testing when the requirements of Specification 3.1.9 shall apply.

3.5.2.4 Quadrant tilt:

a. If the quadrant power tilt exceeds 5%, except for physics tests, power shall be limited to 90% of the thermal power allowable for the reactor coolant pump combination.

b. If the quadrant power tilt exceeds 10%, except for physics tests, power shall be limited to 80% of the thermal power allowable for the reactor coolant pump combination.

c. If the quadrant power tilt exceeds 20%, except for physics tests, power shall be limited to 60% of the thermal power allowable for the reactor coolant pump combination.

d. Within a period of four hours, the quadrant tilt shall be reduced to less than 5%, except for physics tests, or the reactor power/imbalance envelope trip setpoints will be reduced 2% in power for each 1% tilt.

e. If quadrant tilt is in excess of 25%, except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable for the reactor coolant pump combination is restricted by a reduction of 2% in power for each 1% tilt.

f. Quadrant tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

3.5.2.5 Control rod positions:

a. Technical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

b. Operating rod group overlap shall not exceed 30% between two sequential groups, except for physics tests.

c. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1-1 (for up to 435 full power days of operation) and 3.5.2-1-2 (for after 435 full power days of operation) for four pump operation and on Figure 3.5.2-2 for three or two pump operation.

d. Above 80 percent of rated power, the boron concentration shall not be changed to compensate for transient xenon. Below 80 percent of rated power, the boron concentration may be changed to compensate for transient xenon, but power may not be increased above 80 percent of rated power for a period of 12 hours after the last boron concentration change for transient xenon. Changes in boron concentration to compensate for reactivity effects other than transient xenon may be made at any time.

e. Reactor Power Imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Imbalance shall be maintained within the envelope defined by Figure 3.5.2-3. If the imbalance is not within the envelope defined by Figure 3.5.2-3, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.6 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent.

Bases

The power-imbalance envelope defined in Figure 3.5.2-3, is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-4) such that the maximum clad temperature will not exceed the Interim Acceptance Criteria. Operation outside of the power imbalance envelope alone does not constitute a situation that would cause the Interim Acceptance Criteria to be exceeded should a LOCA occur.* The power-imbalance envelope represents the boundary of operation limited by the Interim Acceptance Criteria only if the control rods are at the withdrawal limits as defined in Figures 3.5.2-1 and 3.5.2-2 and if a 5 percent quadrant power tilt exists. Additional conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration uncertainty
c. Fuel densification effects
d. Hot rod manufacturing tolerance factors

The 30 percent 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:

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

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

The minimum available rod worth provides for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains in the full out position.(1)

Inserted rod groups during power operation will not contain single rod worths greater than 0.5% Δk/k. This value has been shown to be safe by the safety analysis of the hypothetical rod ejection accident.(2) A single inserted control rod worth of 1.0% Δk/k at beginning of life, hot, zero power would result in the same transient peak thermal power and therefore the same environmental consequences as a 0.5% Δk/k ejected rod worth at rated power.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Groups 6 and 7 to be partially inserted.

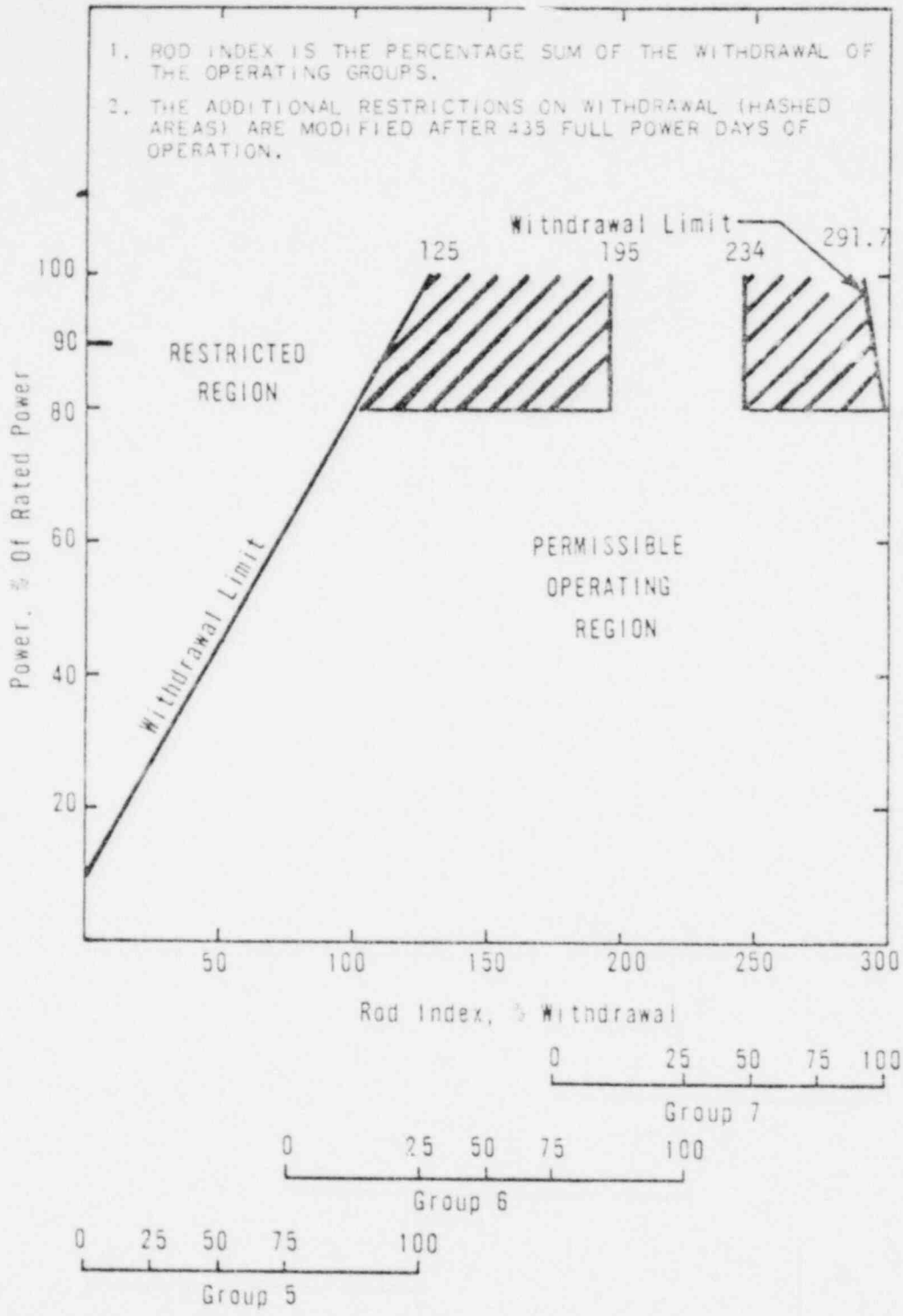
The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position limits in Specification 3.5.2.5c ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

The quadrant tilt and axial imbalance monitoring in Specifications 3.5.2.4f and 3.5.2.5e respectively normally will be performed in the process computer. The two hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

REFERENCES

[1] FSAR, Section 3.2.2.1.2

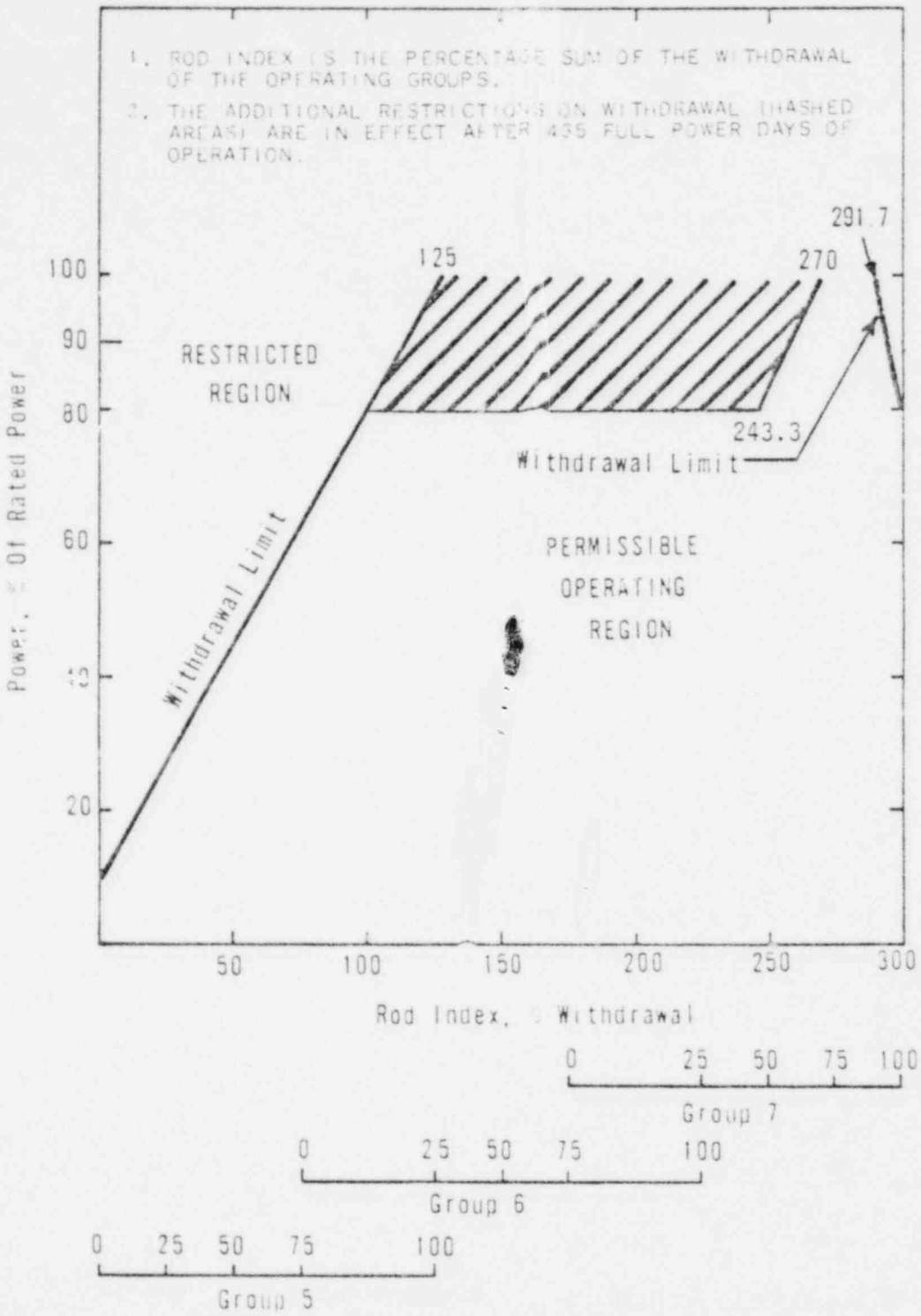
[2] FSAR, Section 14.2.2.2



CONTROL ROD GROUP WITHDRAWAL LIMITS
FOR 4 PUMP OPERATION

OCONEE NUCLEAR STATION
Figure 3.5.2 1-1

Rev. 30. 9/4/73

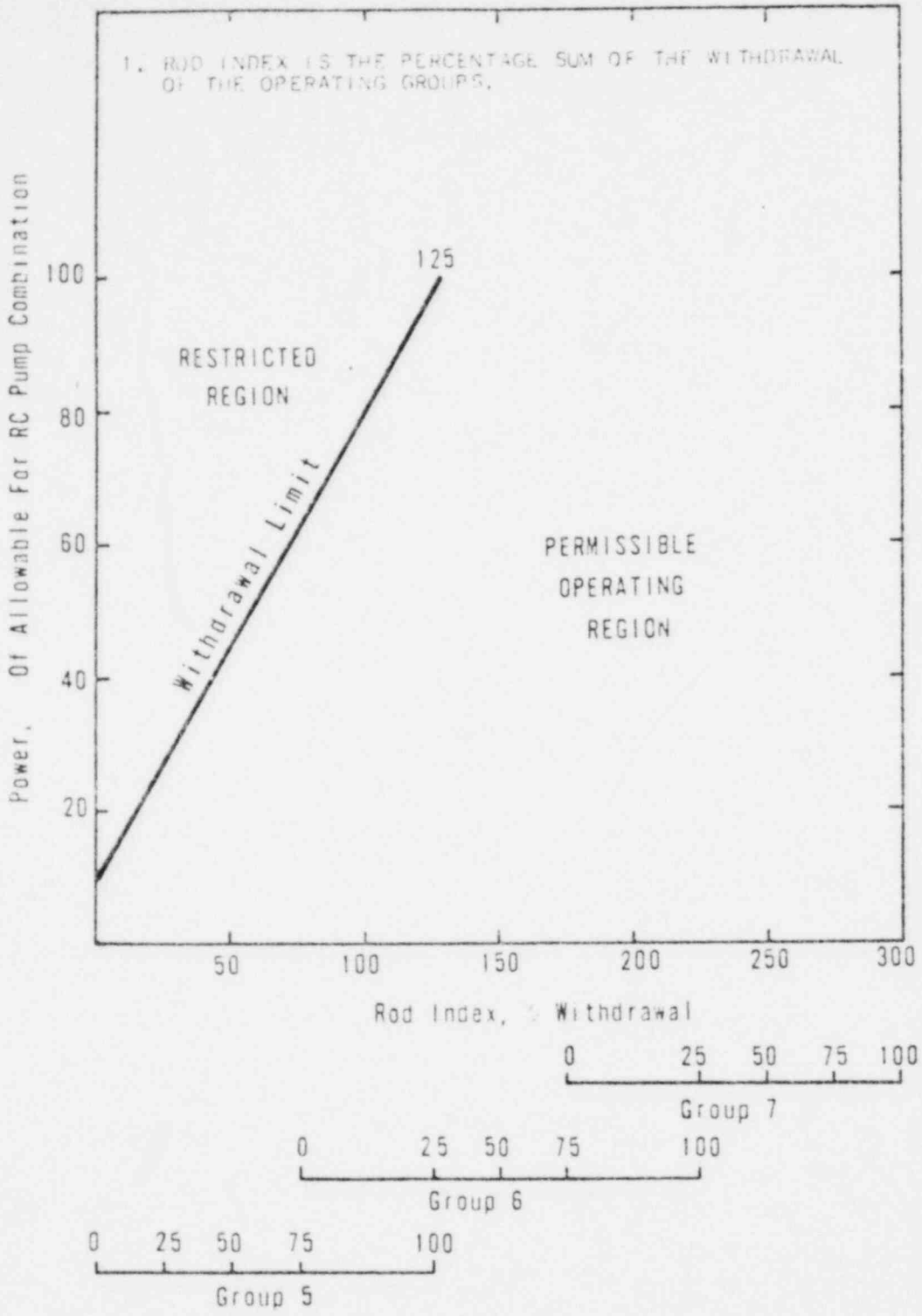


CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION

OCONEE NUCLEAR STATION

Figure 3.5.2-1-2

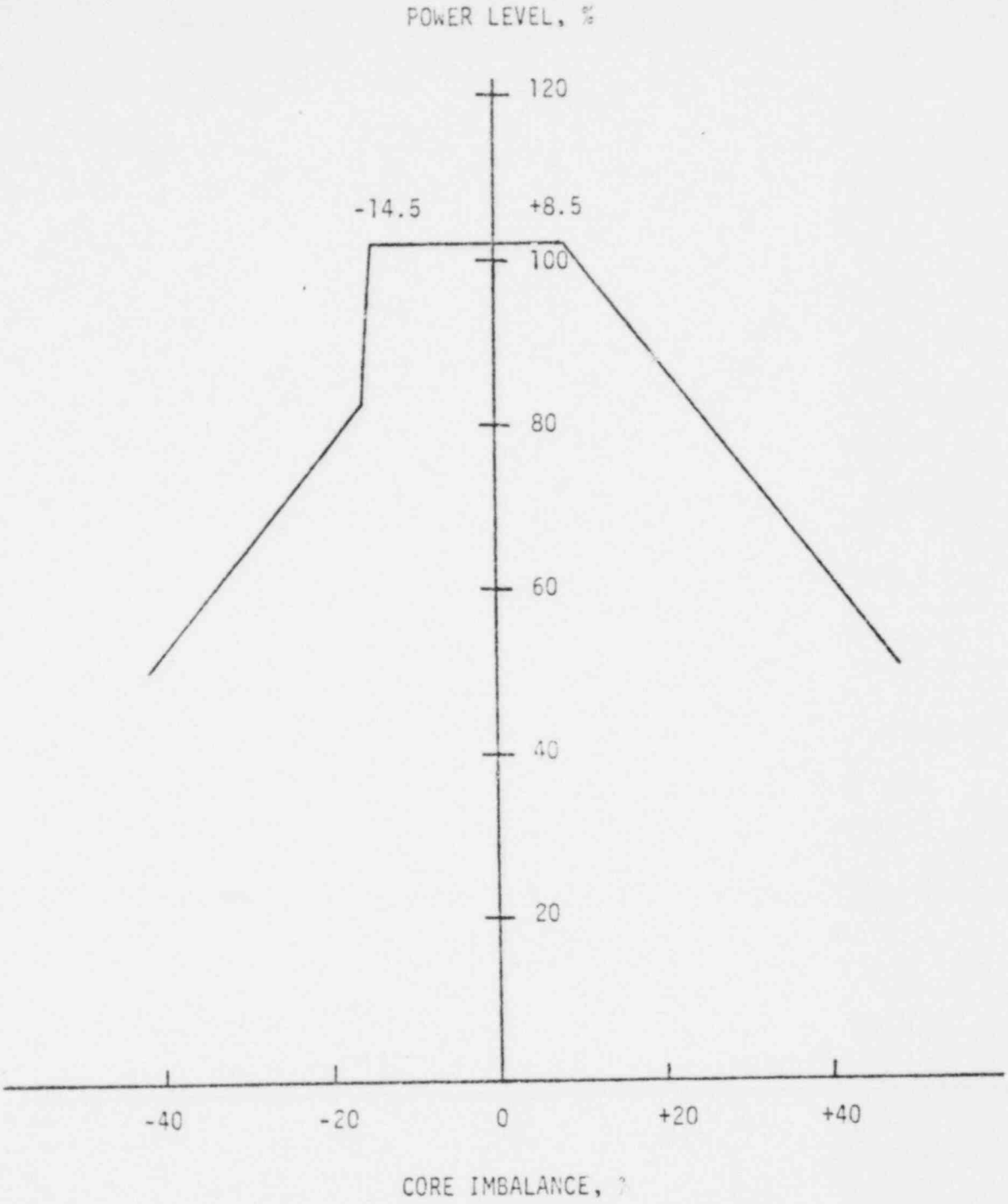
Rev. 30. 9/4/73



CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 3 AND 2 PUMP OPERATION

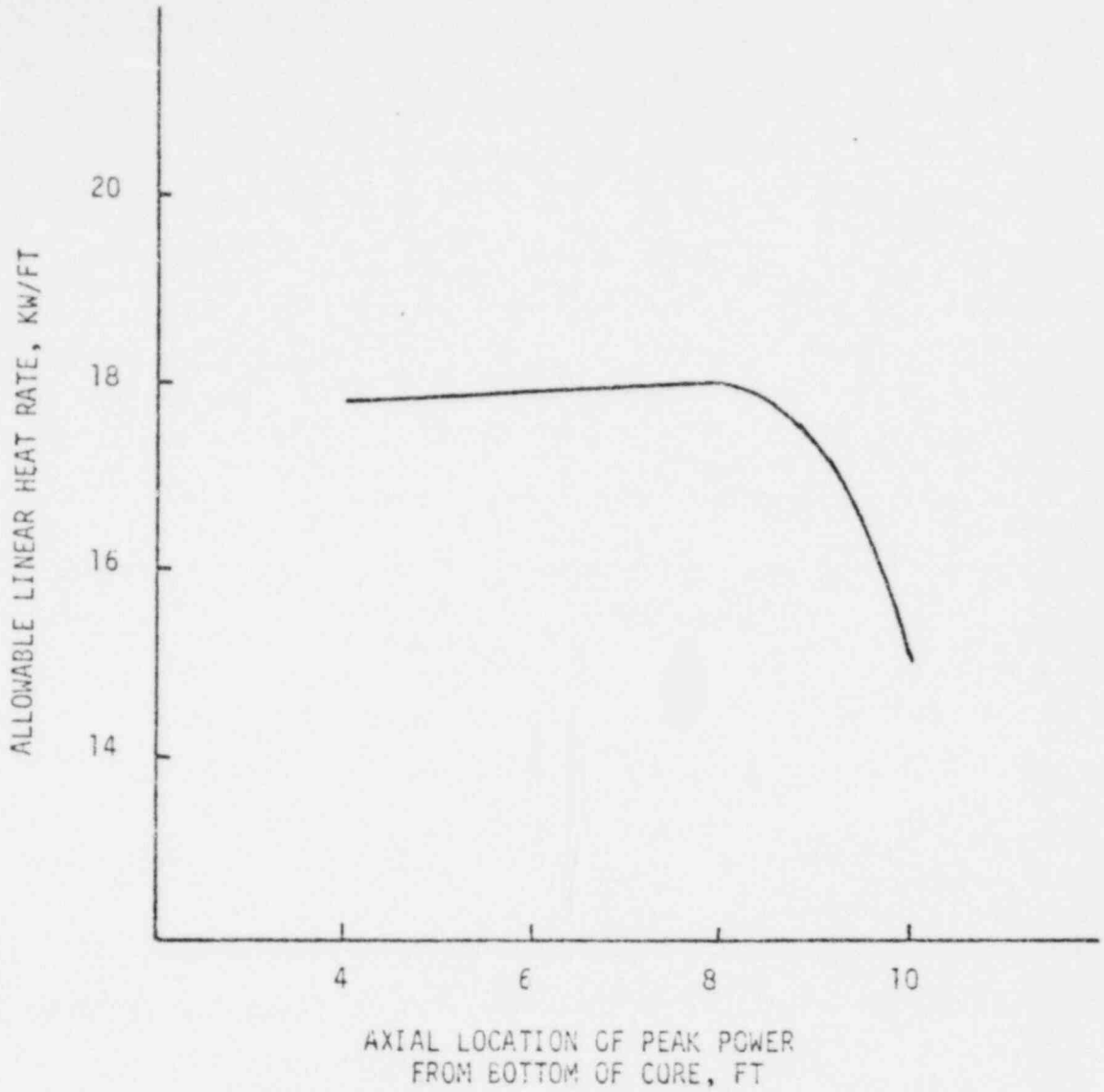
OCONEE NUCLEAR STATION

Figure 3.5.2 - 2



POWER-IMBALANCE ENVELOPE

OCONEE NUCLEAR STATION
Figure 3.5.2 - 3



MAXIMUM ALLOWABLE LINEAR HEAT RATE
PER INTERIM ACCEPTANCE CRITERIA

OCONEE NUCLEAR STATION
Figure 3.5.2-4

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	$\leq$30 psig
	High-Pressure Injection	$\leq$4 psig
	Low-Pressure Injection	$\leq$4 psig
	Start Reactor Building Cooling & Reactor Building Isolation	$\leq$4 psig
	Penetration Room Ventilation	$\leq$4 psig
Low Reactor Coolant System Pressure	High Pressure Injection(1)	$\geq$1500 psig
	Low Pressure Injection(2)	$\geq$500 psig

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

establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation.(1)

REFERENCES

(1) FSAR, Section 14.2.2.3.

3.5.4 Incore Instrumentation

Applicability

Applies to the operability of the incore instrumentation system.

Objective

To specify the functional and operational requirements of the incore instrumentation system.

Specification

Above 80% of operating power determined by the reactor coolant pump combination, reference table 2.3.1, at least 23 individual incore detectors shall be operable to check gross core power distribution and to assist in the periodic calibration of the out-of-core detectors in regard to the core imbalance trip limits. The detectors shall be arranged as follows and may be a part of both basic arrangements.

3.5.4.1 Axial Imbalance

a. Three detectors in each of 3 strings shall lie in the same axial plane with 1 plane in each axial core half.

b. The axial planes in each core half shall be symmetrical about the core mid-plane.

c. The detector strings shall not have radial symmetry.

3.5.4.2 Radial Tilt

a. Two sets of 4 detectors shall lie in each core half. Each set of 4 shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

Bases

A system of 52 incore flux detector assemblies with 7 detectors per assembly has been provided. The system includes data display and record functions and is used primarily for out-of-core nuclear instrumentation calibration and for core power distribution verification.

A. The out-of-core nuclear instrumentation calibration includes:

1. Calibration of the split detectors at initial reactor startup, during the power escalation program, and periodically thereafter.

2. A comparison check with the incore instrumentation in the event one of the four out-of-core detector assemblies gives abnormal readings during operation.

3. Confirmation that the out-of-core axial power splits are as expected.

B. Core power distribution verification includes:

1. Measurement at low power initial reactor startup to check that power distribution is consistent with calculations.

2. Subsequent checks during operation to insure that power distribution is consistent with calculations.

3. Indication of power distribution in the event that abnormal situations occur during reactor operation.

C. The safety of unit operation at or below 80% of operating power(1) for the reactor coolant pump combinations without the core imbalance trip system has been determined by extensive 3-D calculations. This will be verified during the physics startup testing program.

D. The minimum requirement for 23 individual incore detectors is based on the following:

1. An adequate axial imbalance indication can be obtained with 9 individual detectors. Figure 3.5.4-1 shows three detector strings with 3 detectors per string that will indicate an axial imbalance that is within 8% (calculated) of the real core imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings. Both steady state and design transient data from the Oconee 1 maneuvering analysis were used for this comparison.

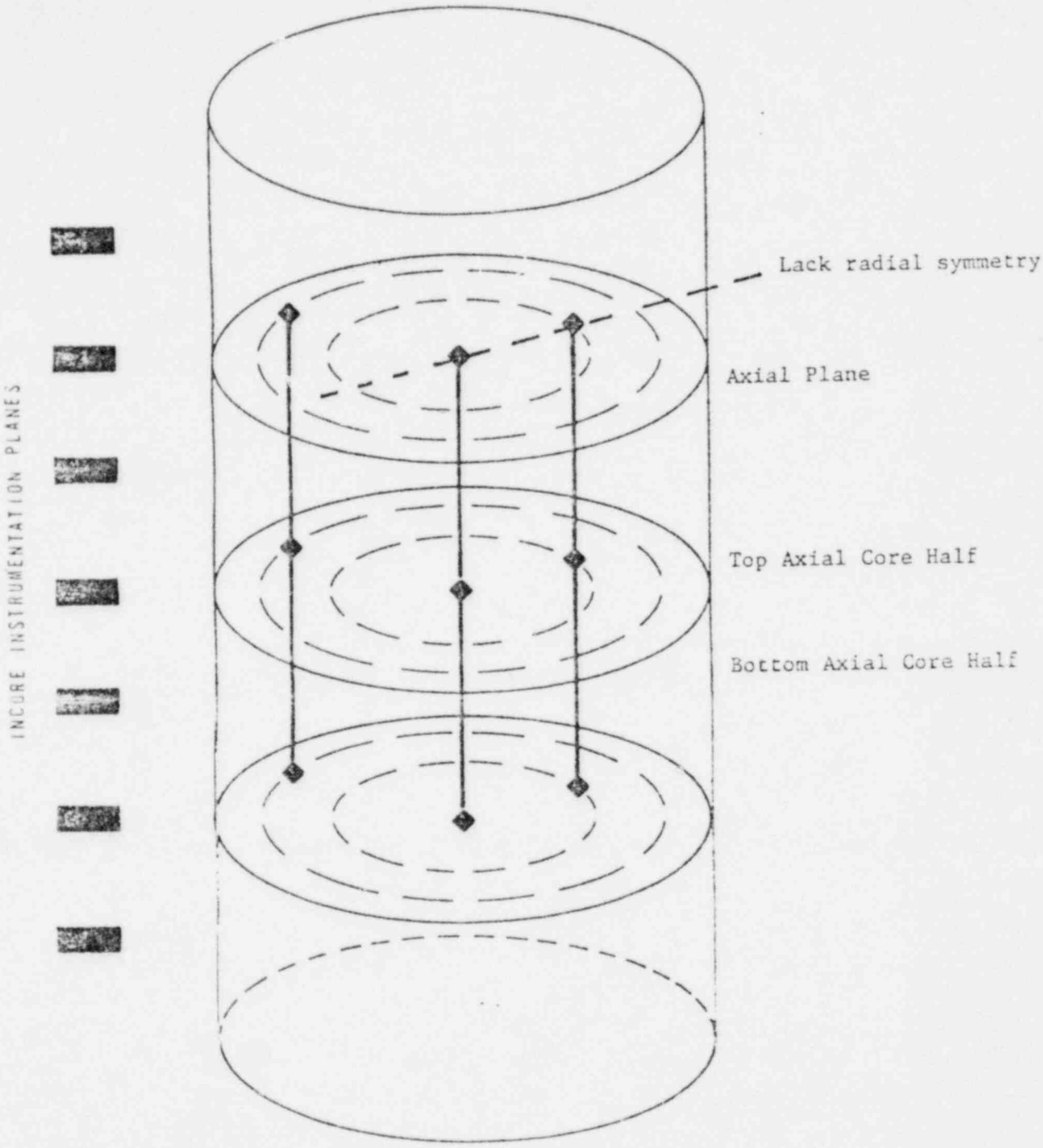
2. Figure 3.5.4-2 shows a detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from 2 detectors in a radial quandrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

3. Figure 3.5.4-3 combines Figures 3.5.4-1 and 3.5.4-2 to illustrate a set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions. Startup testing will verify the adequacy of this set of detectors for the above functions.

E. At least 23 specified incore detectors will be operable to check power distribution above 80% power determined by reactor coolant pump combination. These incore detectors will be read out either on the computer or on a recorder. If 23 detectors in the specified locations are not operable, power will be decreased to or below 80% for the operating reactor coolant pump combination.

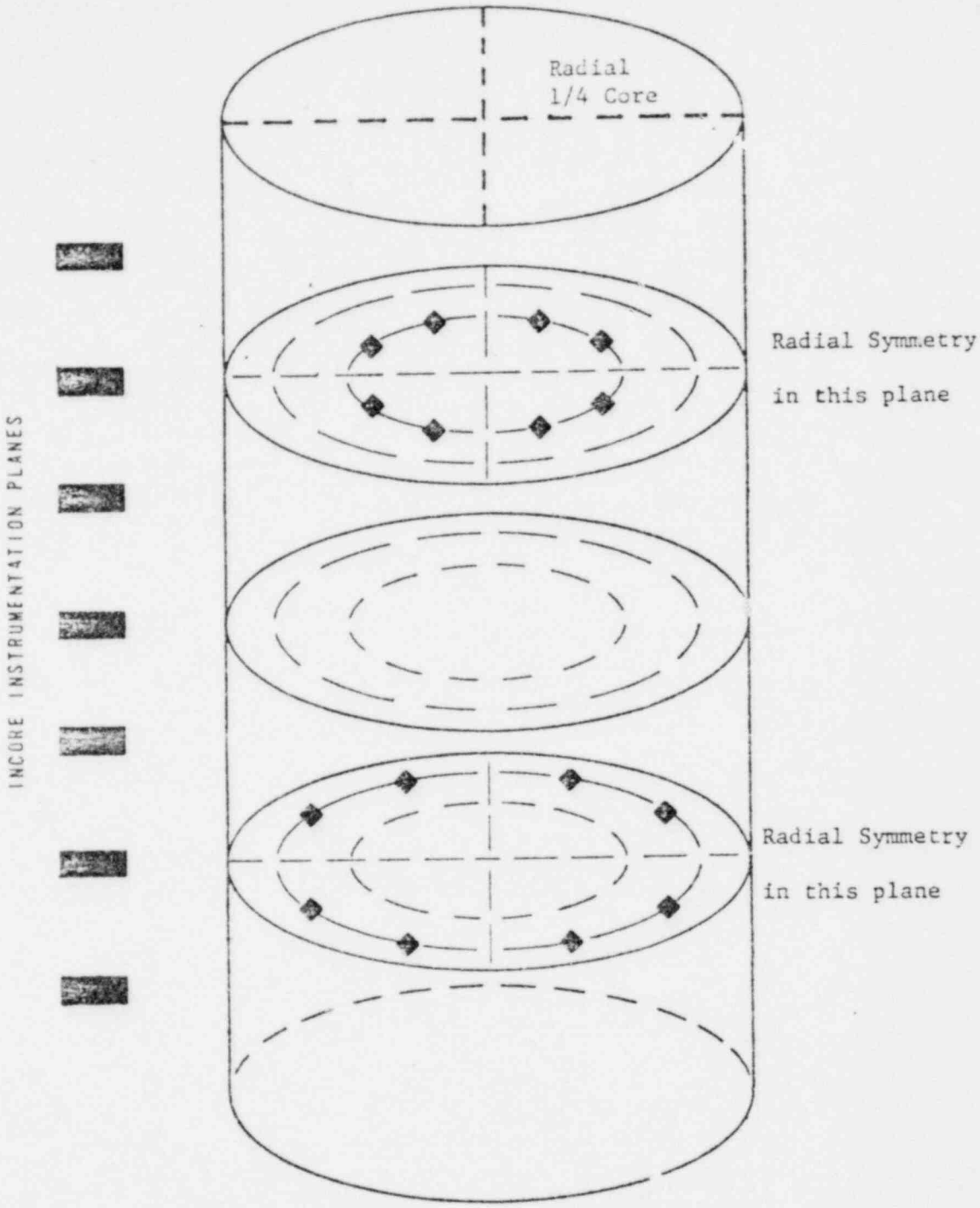
REFERENCE

(1) FSAR, Section 4.1.1.3



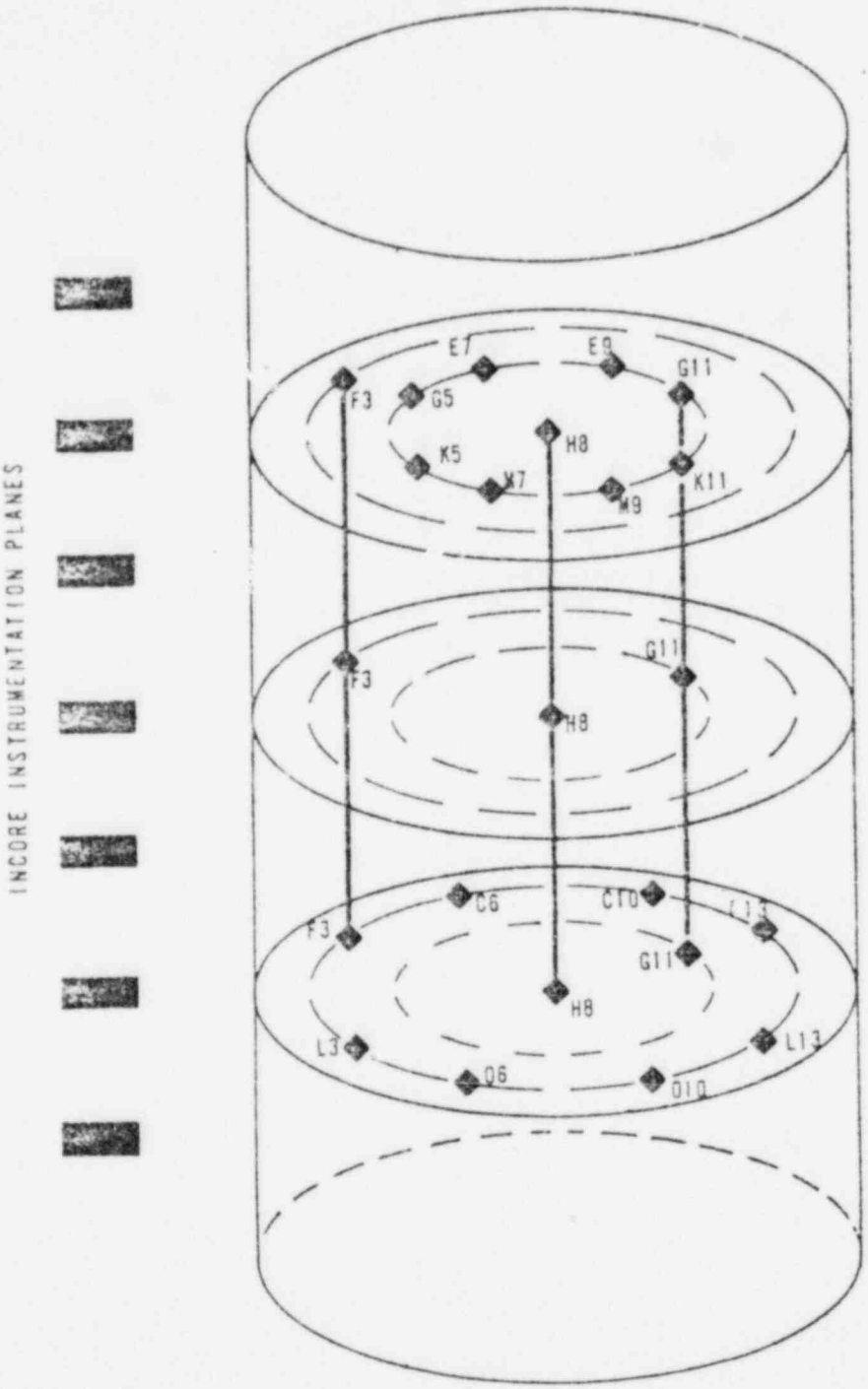
INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION

OCONEE NUCLEAR STATION
Figure 3.5.4 1



INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX TILT INDICATION

OCONEE NUCLEAR STATION
Figure 3.5.4 - 2



INCORE INSTRUMENTATION SPECIFICATION

OCONEE NUCLEAR STATION
Figure 3.5.4 - 3

3.6 REACTOR BUILDING

Applicability

Applies to the containment when the reactor is subcritical by less than 1% $\Delta k/k$.

Objective

To assure containment integrity during startup and operation.

Specification

3.6.1 Containment integrity shall be maintained whenever all three (3) of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200°F or greater.

c. Nuclear fuel is in the core.

3.6.2 Containment integrity shall be maintained when the reactor coolant system is open to the containment atmosphere and the requirements for a refueling shutdown are not met.

3.6.3 The containment integrity shall be intact whenever positive reactivity insertions which would result in the reactor being subcritical by less than 1% $\Delta k/k$ are made by control rod motion or boron dilution.

3.6.4 The reactor building internal pressure shall not exceed 1.5 psig or five inches of Hg if the reactor is critical.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and tagged.

Bases

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence no pressure buildup in the containment if the Recator Coolant System ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here are conservative since an evaluation of

National Weather Service records for this area indicates that from 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg. and the highest is 30.85 inches of Hg.

When containment integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.

REFERENCES

FSAR, Section 5

3.7 AUXILIARY ELECTRICAL SYSTEMS

Applicability

Applies to the availability of off-site and on-site electrical power for station operation and for operation of station auxiliaries.

Objective

To define those conditions of electrical power availability necessary to provide for safe reactor operation and to provide for continuing availability of engineered safety features systems in an unrestricted manner and to prescribe safety evaluation and reporting requirements to be followed in the event that the auxiliary electric power systems become degraded.

Specification

3.7.1 Except as permitted by 3.7.2, 3.7.3, 3.7.4, 3.7.5, 3.7.6, and 3.7.7, the reactor shall not be heated above 200°F unless the following conditions are met.

(a) At least two 230 kV transmission lines, on separate towers, shall be in service.

(b) Two startup transformers shall be operable and available to the unit's 4160 volt Main Feeder Buses No. 1 and No. 2

(c) One operable Keowee hydro unit shall be available to supply power through the Underground Feeder Bus, Transformer CT4 and the 4160 volt Standby Buses No. 1 and No. 2 to the units 4160 volt Main Feeder Buses No. 1 and 2. The second Keowee hydro unit shall be available to supply power automatically through a startup transformer to the units 4160 volt Main Feeder Buses No. 1 and 2.

(d) The two 4160 volt main feeder buses shall be energized.

(e) The three 4160 volt Engineered Safety Features switchgear buses shall be energized.

(f) Three 600 volt load centers plus the three 600 volt-208V Engineered Safety Features MCC Buses shall be energized.

(g) For each unit, all 125 VDC instrumentation and control batteries with their respective chargers, buses, diode monitors, and diodes supplying the unit's vital instrumentation and the four instrumentation and control panel boards shall be operable.

(h) The 125 VDC switching station batteries with their respective chargers, buses, and isolating diodes shall be operable.

(i) The Keowee batteries with their respective chargers, buses, and isolating diodes shall be operable.

(j) The level of the Keowee Reservoir shall be at least 775 feet above seal level.

3.7.2 During hot standby or power operation, provisions of 3.7.1 may be modified to allow any one of the following conditions to exist:

(a) One of the two required startup transformers may be removed from service for 48 hours provided it is expected to be restored to service within 48 hours and the other required startup transformer is available for automatic connection to the unit's main feeder bus.

(b) One Keowee hydro unit may be inoperable for periods not exceeding 24 hours for test or maintenance provided the operable Keowee hydro unit is connected to the underground feeder circuit which is operable.

(c) The underground feeder circuit may be inoperable for 24 hours for test and maintenance.

(d) In each unit, the following items may be inoperable for periods not exceeding 24 hours:

1. One 4160 volt main feeder bus.

2. One complete single string of any unit's Engineered Safety Features 4160 volt switchgear bus, 600 volt load center - 600V-208V MCC and their loads.

3. One complete single string of any unit's 125 VDC instrumentation and control batteries, chargers, buses, and all associated isolating and transfer diodes.

4. One 125 VDC instrumentation and control panel board and/or its associated loads.

(e) One complete single string of the 125 VDC switching station batteries, buses, chargers, and the related diode assemblies may be de-energized for test or maintenance for periods not exceeding 24 hours.

(f) One complete single string of the Keowee batteries, chargers, buses, and isolating diodes may be de-energized for test or maintenance for periods not exceeding 24 hours.

(g) One 4160 volt standby bus may be inoperable for test or maintenance for periods not exceeding 24 hours.

3.7.3 In the event that the conditions of Specification 3.7.1 are not met within the time specified in Specification 3.7.2, except as noted below in Specification 3.7.4, 3.7.5, 3.7.6, 3.7.7, and 3.7.8, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of 3.7.1 are not met within an additional 4[illegible] hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

3.7.4 In the event that all conditions in Specification 3.7.1 are met except that one of the two Keowee hydro units is expected to be made unavailable for longer than the test or maintenance period of 24 hours, the reactor shall be permitted to remain critical or be restarted provided the following restrictions are observed:

(a) Prior to the restart of a shutdown reactor or within 30 minutes after the loss of one Keowee hydro unit, the remaining Keowee hydro unit shall be connected to the underground feeder, and the 4160 volt standby buses shall be energized by one of the three Lee gas turbines through the 100 kV circuit. The Lee gas turbine and the 100 kV transmission circuit shall be electrically separate from the system grid and non-safety-related loads.

(b) After loss of a hydro unit, this information shall be reported within 24 hours to the Directorate of Regulatory Operations, Region II. If the outage is expected to exceed 24 hours, a written report shall be submitted detailing the circumstances of the outage and the estimated time to return the unit to operating condition.

(c) The reactor coolant Tavg shall be above 525°F. Reactor coolant pump power may be used to elevate the temperature from 500°F to 525°F in the case of a restart. If Tavg decreases below 500°F, restart is not permitted by this specification.

3.7.5 In the event that all conditions in Specification 3.7.1 are met except that the underground feeder circuit to the standby buses is unavailable for longer than the test or maintenance period of 24 hours, the reactor shall be permitted to remain critical or be restarted provided the following restrictions are observed.

(a) Prior to the restart of a shutdown reactor or within 30 minutes of the feeder loss for an operating reactor, the 4160 standby buses shall be energized by one of the three Lee gas turbines through the 100 kV transmission circuit. The Lee gas turbine and the 100 kV transmission circuit shall be electrically separate from the system grid and non-safety-related loads.

(b) The reactor coolant Tavg shall be above 525°F. Reactor coolant pump power may be used to elevate the temperature from 500°F to 525°F in the case of a restart. If Tavg decreases to below 500°F, restart is not permitted by this specification.

(c) Afte of the underground feeder, this information shall be r within 24 hours to the Directorate of Regulatory Oper Region II. If the outage is expected to exceed 24 h written report shall be submitted detailing the circes of the outage and the estimated time to return the ound feeder to operating condition.

3.7.6 In the at all conditions of Specification 3.7.1 are met except t 230 kV transmission lines are lost, the reactor shall be ted to remain critical or be restarted provided the followin ictions are observed:

(a) Prio e restart of a shutdown reactor or within 30 minutes of 1 1 230 kV transmission lines for an operating reactor, the lt standby buses shall be energized by one of the thre as turbines through the 100 kV transmission circuit. The turbine and the 100 kV transmission circuit shall be ely separate from the system grid and non-safety-relads.

(b) Afte of all 230 kV transmission lines, this information shal ported within 24 hours to the Directorate of Regulat ations, Region II. If the outage is expected to exce ours, a written report shall be submitted detailing the stances of the outage and the estimated time to retu 230 kV transmission lines to operating condition.

(c) The coolant Tavg shall be above 525°F. Reactor coolant pump may be used to elevate the temperature from 500°F to 525°F in t of a restart. If Tavg decreases below 500°F, rest not permitted by this specification.

3.7.7 In the hat all conditions of Specification 3.7.1 are met except owee hydro units are unavailable, the reactor shall be permi remain critical or be restarted provided the followir rictions re met:

(a) Pri he restart of a shutdown reactor or within 30 minutes afte of both Keowee hydro units for an operating reactor, the lt standby buses shall be energized by one of the thre as turbines through the 100 kV transmission circuits. The turbine and the 100 kV transmission circuit shall be ely separate from the system grid and non-safety-relads.

(b) Afte of both Keowee hydro units, this information shall be d within 24 hours to the Directorate of Regulatory Oper, Region II. If the outage is expected to exceed 24 h written report shall be submitted detailing the circes of the outage and the estimated time to return the to operating condition.

(c) Prior to hot restart of a reactor from a tripped condition, the causes and the effects of the shutdown shall be established and analyzed. A restart will be permitted if the cause of such trips are the result of error or of minor equipment malfunctions. A restart will not be permitted if the trip is a result of system transients or valid protection system action.

(d) The reactor coolant Tavg shall be above 525°F. Reactor coolant pump power may be used to elevate the temperature from 500°F to 525°F in the case of a restart. If Tavg decreases below 500°F restart is not permitted by this specification.

3.7.8 Any degradation beyond Specification 3.7.2, 3.7.4, 3.7.5, 3.7.6, or 3.7.7 above shall be reported to the Directorate of Regulatory Operations, Region II, within 24 hours. A safety evaluation shall be performed by Duke Power Company for the specific situation involved which justifies the safest course of action to be taken. The results of this evaluation together with plans for expediting the return to the unrestricted operating conditions of Specification 3.7.1 above shall be submitted in a written report to the Directorate of Licensing, with a copy to the Directorate of Regulatory Operations, Region II, within five days.

Bases

The auxiliary electrical power systems are designed to supply the required Engineered Safeguards loads in one unit and safe shutdown loads of the other two units and are so arranged that no single contingency can inactivate enough engineered safety features to jeopardize plant safety. These systems were designed to meet the following criteria:

> "Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity and testability to permit the functions required of the engineered safety features of each unit."

The auxiliary power system meets the above criteria and the intent of AEC Criterion 17. The adequacies of the AC and DC systems are discussed below as are the bases for permitting degraded conditions for AC power.

Capacity of AC Systems

The auxiliaries of two units in hot shutdown (6.0 MVa each) plus the auxiliaries activated by ESG signal in the other unit (4.8 MVa) require a total AC power capacity of 16.8 MVa. The continuous AC power capacity available from the on-site power systems (Keowee Hydro Units) is 20 MVa (limited by transformer CT4) if furnished by the underground circuit or 30 MVa (limited by CT1 or CT2) if furnished through the 230 kV off-site transmission lines. Capacity available from the backup 100 kV off-site transmission line (Lee Station Gas Turbine Generator) is 20 MVa (limited by CT5).

Thus, the minimum available capacity from any one of the multiple sources of AC power, 20 MVa, is adequate.

Capacity of DC Systems

Normally, for each unit AC power is rectified and supplies the DC system buses as well as keeping the storage batteries on these buses in a charged state. Upon loss of this normal AC source of power, each unit's DC auxiliary systems important to reactor safety have adequate stored capacity (ampere-hours) to independently supply their required emergency loads for at least one hour. One hour is considered to be conservative since there are redundant sources of AC power providing energy to these DC auxiliary systems. The loss of all AC power to any DC system is expected to occur very infrequently, and for very short periods of time. The following tabulation demonstrates the margin of installed battery charger rating and battery capacity when compared to one hour of operation (a) with AC power (in amps) and (b) without AC power (in ampere hours) for each of the three safety-related DC systems installed at Oconee:

A. 125 VDC Instrumentation and Control Power System

Charger XCA, XCB, or XCS	a. 600 amps each
Battery 1CA and 1CB Combined Capacity (X = 1, 2, or 3)	b. 698 ampere-hours
Actual active loads on both 125 VDC I & C buses XDCA and XDCB during 1st hour of LOCA (X = 1, 2, or 3)	a. First min. - 1371 amps next 59 min. - 568.5 amps b. 581.9 ampere-hours

B. 125 VDC Switching Station Power System

Charger SY-1, SY-2, or SY-S Rating	a. 50 amps each
Battery SY-1 or SY-2 Capacity	b. 14.4 ampere-hours

Actual active load per battery during 1st hour of LOCA	a. First min. - 130 amps next 59 min. - 10 amps b. 12 ampere-hours

C. 125 VDC Keowee Station Power System

Charger No. 1, No. 2 or Standby Rating	a. 200 amps each
Battery No. 1 or No. 2 Capacity	b. 233 ampere-hours
Actual active load per battery during 1st hour of LOCA	a. First min. - 1031 amps next 59 min. - 179.4 amps b. 193.6 ampere-hours

Redundancy of AC Systems

There are three 4160 engineered safety feature switchgear buses per unit. Each bus can receive power from either of the two 4160 main feeder buses per unit. Each feeder bus in turn can receive power from the 230 kV switchyard through the startup transformers, through the unit auxiliary transformer by backfeeding through the main step-up transformer, or from the 4160V standby bus. Another unit's startup transformer serving as an alternate supply can be placed in service in one hour. The standby bus can receive power from the Hydro Station through the underground feeder circuit or from a combustion turbine generator at the Lee Steam Station over an isolated 100 kV transmission line. The 230 kV switchyard can receive power from the on-site Keowee Hydro station or from several off-site sources via transmission lines which connect the Oconee Station with the Duke Power system power distribution network.

Redundancy of DC Systems

A. 125 VDC Instrument and Control Power System

All reactor protection and engineered safety features loads on this system can be powered from either the Unit 1 and Unit 2 or Unit 2 and Unit 3 or Unit 3 and Unit 1 125 VDC Instrument and Control Power Buses. The units' 125 VDC Instrument and Control Power Buses can be powered from two battery banks and three battery chargers. As shown above, one battery (e.g., 1CA) can supply all loads for one hour. Also, one battery charger can supply all connected ESF and reactor protection loads.

B. 125 VDC Switching Station Power System

There are two essentially independent sybsystems each complete with an AC/DC power supply (battery charger), a battery bank, a battery charger bus, motor control center (distribution panel). All safety-related equipment and the relay house in which it is located are Class I (seismic) design. Each subsystem provides the necessary DC power to:

a. Continuously monitor operations of the protective relaying,

b. Isolate Oconee (including Keowee) from all external 230 kV grid faults,

c. Connect on-site power to Oconee from a Keowee hydro unit or,

d. Restore off-site power to Oconee from non-faulted portions of the external 230 kV grid.

Provisions are included to manually connect a standby battery charger to either battery/charger bus.

C. 125 VDC Keowee Station Power System

There are essentially two independent physically separated Class I (seismic) subsystems, each complete with an AC/DC power supply (charger) a battery bank, a battery/charger bus and a DC distribution center. Each subsystem provides the necessary power to automatically or manually start, control and protect one of the hydro units.

An open or short in any one battery, charger or DC distribution center cannot cause loss of both hydro units.

The 230 kV sources, while expected to have excellent availability, are not under the direct control of the Oconee Station and, based on past experience, cannot be assumed to be available at all times.

The operation of the on-site hydro-station is under the direct control of the Oconee Station and requires no off-site power to startup. Therefore, an on-site backup source of auxiliary power not subject to failure from the same cause as off-site power is provided in the form of twin hydro-electric turbine generators powered through a common penstock by water taken from Lake Keowee. The use of a common penstock is justified on the basis of past hydro plant experience of the Duke Power Company (since 1919) which indicates that the cumulative need to dewater the penstock can be expected to be limited to about one day a year, principally for inspection, plus perhaps four days every tenth year. In all other cases, it is expected that when one hydro unit is out for maintenance, the other unit will be available for service.

In the event that only one hydro unit is available to backup the off-site power sources, and it is considered important for an Oconee unit reactor to remain critical or return to criticality from a hot shutdown condition, the Lee Station combustion turbine is again available to assure a continued supply of shutdown power in the event that an external event should cause loss of all off-site power.

In a similar manner, in the event that none of the sources of off-site power are available and it is considered important to continue to maintain an Oconee unit reactor critical or return it to criticality from a hot shut-down condition, a Lee Station gas turbine can be made available as an additional backup source of power, thus assuring continued availability of auxiliary power to perform an orderly shutdown of a unit should a problem develop requiring shutdown of both hydro units.

There may be a rare occasion where both hydro units are unexpectedly lost and there are compelling reasons to maintain the Oconee reactors critical or return them to criticality from hot shutdown conditions for a specific period of time

rather than require it to remain subcritical or be shutdown. A scheduled shutdown for inspection or a shutdown to perform minor maintenance would not constitute a compelling reason. Factors to consider in justification of such a rare, limited period of criticality without the hydro station available would include number of off-site 230 kV power sources available, availability of the other units' startup transformer, availability of the Lee gas turbine, weather conditions and all other factors which could bear on potential for loss of these power sources. Also, the evaluation should show that reactor safety will not be compromised if during operation under such further degradation, an additional loss of AC power should be suffered.

3.8 FUEL LOADING AND REFUELING

Applicability

Applies to fuel loading and refueling operations.

Objective

To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by RIA-48 and RIA-49. Radiation levels in the spent fuel storage area shall be monitored by RIA-41. If any of these instruments becomes inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one low pressure injection pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required to shutdown the core to a $k_{eff} \leq 0.99$ if all control rods were removed.

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the reactor building at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Both isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.8 When two irradiated fuel assemblies are being handled simultaneously within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.
Irradiated fuel assemblies may be handled with the Auxiliary Hoist provided no other irradiated fuel assembly is being handled in the fuel transfer canal.

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

3.8.10 The reactor building purge system, including the radiation monitor, RIA 45, which initiates purge isolation, shall be tested and verified to be operable immediately prior to refueling operations.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The low pressure injection pump is used to maintain a uniform boron concentration. (1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core. (2) The boron concentration will be maintained above 1,800 ppm. Although this concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing of the reactor building purge isolation is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Specification 3.8.11 is required, as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.(3)

REFERENCES

(1) FSAR, Section 9.7

(2) FSAR, Section 14.2.2.1

(3) FSAR, Section 14.2.2.1.2

3.9 RELEASE OF LIQUID RADIOACTIVE WASTE

Applicability

Applies to the controlled release of all liquid waste discharged from the station which may contain radioactive materials.

Objective

To establish conditions for the release of liquid waste containing radioactive materials and to assure that all such releases are within the concentration limits specified in 10 CFR Part 20. In addition, to assure that the releases of radioactive material in liquid wastes (above background) to unrestricted areas meet the low as practicable concept, the following liquid release objectives shall apply:

a. The annual total quantity of radioactive materials in liquid waste, excluding tritium and dissolved gases, shall be less than 5 curies per unit;

b. The annual average concentration of radioactive materials in liquid waste, upon release from the Restricted Area, excluding tritium and dissolved noble gases, shall not exceed 2×10^{-8} μCi/ml;

c. The annual average concentration of tritium in liquid waste, upon release from the Restricted Area, shall not exceed 5×10^{-6} μCi/ml;

d. The annual average concentration of dissolved gases in liquid waste, upon release from the Restricted Area, shall not exceed 2×10^{-6} μCi/ml;

Specifications

3.9.1 If the experienced release of radioactive materials in liquid wastes, when averaged over a calendar quarter, is such that these quantities if continued at the same release rate for a year would exceed twice the annual objectives the licensee will:

a. Make an investigation to identify the causes for such release rates;

b. Define and initiate a program of action to reduce such release rates to the design levels, and;

c. Describe these actions in a report to AEC/DOL within 30 days after incurring the reporting obligation.

3.9.2 The release rate of radioactive liquid effluents, excluding tritium and dissolved noble gases, shall not exceed 10 curies during any calendar quarter without specific approval of the Commission. Similarly, the quarterly average concentration of tritium released from the Restricted Area shall not exceed 1×10^{-5} μCi/ml.

3.9.3 The rate of release of radioactive materials in liquid waste from the station shall be controlled such that the instantaneous concentrations of radioactivity in liquid waste upon release from the Restricted Area, does not exceed the values listed in 10CFR20, Appendix B, Table II, Column 2.

3.9.4 The equipment installed in the liquid radioactive waste system shall be maintained and operated for the purposes of keeping releases within the objectives of these specifications and shall process all liquids prior to their discharge in order to limit the activity, excluding tritium and dissolved noble gases, released during any calendar quarter to 1.25 curies or less per unit.

3.9.5 As far as practicable, the releases of liquid waste shall be coordinated with the operation of the Keowee Hydro unit.

3.9.6 Liquid waste discharged from the liquid waste disposal system shall be continuously monitored during release. The liquid effluent monitor reading shall be compared with the expected reading of each discharge batch. The monitor shall be tested daily or prior to releases and calibrated at refueling intervals. The calibration procedure shall consist of exposing the detector to a referenced calibration source in a controlled, reproducible geometry. The sources and geometry shall be referenced to the original monitor calibration which provides the applicable calibration curves.

3.9.7 The effluent control monitor shall be set to alarm and automatically close the waste discharge valve such that the appropriate requirements of the specification are met.

3.9.8 In addition to the continuous monitoring requirements, liquid radioactive waste sampling and activity analysis shall be performed in accordance with Table 4.1.3. Records shall be maintained and reports of the sampling and analysis shall be submitted in accordance with Section 6.6 of these Technical Specifications.

Bases

It is expected that the releases of radioactive materials and liquid wastes will be kept within the design objective levels and will not exceed the concentration limits specified in 10CFR20. These levels provide the reasonable assurance that the resulting annual exposure to the whole body or any individual body organ will not exceed 5 millirem per year. At the same time, the licensee is permitted the flexibility of operation compatible with considerations of health and safety to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than design objective levels but still within the concentration limits specified in 10CFR20. It is expected that when using this operational flexibility under unusual operating conditions, the licensee shall exert every effort to keep the levels of radioactive materials and liquid wastes as low as practicable and that annual releases will not exceed a small fraction of the annual average concentration limits specified in 10CFR20.

The anticipated annual releases from the three Oconee units have been developed taking into account a combination of variables including fuel failures, primary system leakage, primary-to-secondary leakage, and the performance of the various waste treatment systems. The actual magnitude of these parameters are as follows:

a. Maximum expected reactor coolant corrosion product concentrations.
b. Reactor coolant fission product concentration corresponding to 0.25 percent fuel cladding defects.
c. Steam generator primary-to-secondary leakage rate of 20 gpd.
d. 255,160 gallons per year processed by the waste disposal system in a 30-day hold-up.
e. 1,060,800 gallons per year processed by the reactor coolant bleed treatment system.
f. A decontamination factor of 10^4 for all radionuclides except tritium for the coolant bleed and waste evaporators and a decontamination factor of 10 for the demineralizers except for tritium which had an assumed decontamination factor of 1 for evaporation-demineralization.
g. No removal by demineralization for Cs, Mo, and Y. A decontamination factor of 10^3 was used for the evaporation of iodine.
h. The decay time of the reactor coolant bleed system was 30 days.

The application of the above estimates results in the radionuclide discharge concentrations and rates shown in Table III-12 of the "Final Environmental Statement Related to Operation of Oconee Nuclear Station Units 1, 2, and 3". These concentrations are based on an annual average flow in the Keowee River of 1,100 cfs.

Operating procedures will identify all equipment installed in the liquid waste handling and treatment systems and will specify detailed procedures for operating and maintaining this equipment.

The lowest practicable liquid release objectives expressed in this specification are based on the guidelines contained in the proposed Appendix I of 10CFR50. Since these guidelines have not been adopted as yet, the release objectives of this specification will be reviewed at the time Appendix I becomes a regulation to assure that this specification is based upon the guidelines contained therein.

3.10 RELEASE OF GASEOUS RADIOACTIVE WASTE

Applicability

Applies to the controlled release of all gaseous waste discharged from the station which may contain radioactive materials.

Objective

To establish conditions in which gaseous waste containing radioactive materials may be released and to assure that all such releases are within the concentration and dose limits specified in 10CFR20. In addition, to assure that the releases of gaseous radioactive wastes (above background) to unrestricted areas meet the as low as practicable concept, the following objectives shall apply:

1. Averaged over a yearly interval, the release rate of noble gases and other radioactive isotopes, except I-131 and particulate radio-isotopes with half lives greater than eight days, discharged at the unit vent, shall be limited as follows:

$$\Sigma \frac{Qi}{(MPC)i} \leq 5560 \text{ m}^3/\text{sec}$$

   where Qi is the annual controlled release rate (Ci/sec) of radio-isotope i and (MPC) is the permissible concentration for unrestricted areas in units of Ci/m^3 ($\mu Ci/ml = Ci/m^3$) for any radionuclide given in Column 1, Table II of Appendix B to 10CFR20.

2. Averaged over a yearly interval, the release rate of I-131 and other particulate radio-isotopes with half lives longer than eight days, discharged at the unit vent, shall be limited as follows:

$$\Sigma \frac{Qi}{(MPC)i} \leq 117 \text{ m}^3/\text{sec}$$

   where Qi and (MPC)i are as defined above.

Specifications

3.10.1 If the experienced rate of release of radioactive materials in gaseous wastes when averaged over a calendar quarter is such that these quantities if continued at the same release rate for a year would exceed twice the annual objective, the licensee shall:

a. Make an investigation to identify the causes for such releases;
b. Define and initiate a program of action to reduce such release rates to the design levels;
c. Describe these actions in a report to the Commission within 30 days after incurring the reporting obligation.

3.10.2 If the experienced rate of release of radioactive materials in gaseous wastes, when averaged over a calendar quarter, is such that these quantities if continued at the same release rate for a year would exceed

eight times the annual objectives, the licensee shall define and initiate a program of action to assure that such release rates are reduced and shall submit a report to the AEC within seven days after incurring the reporting requirement, describing the causes for such release rates and the course of action taken to reduce them.

3.10.3 The rate of release of radioactive materials and gaseous wastes from the station (except I-131 and particulate radio-isotopes with half lives greater than eight days) shall be controlled such that the maximum release rate averaged over any one-hour period shall not exceed:

$$\Sigma \frac{Qi}{(MPC)i} = 3.0 \times 10^5 \ m^3/sec$$

3.10.4 During release of radioactive gaseous waste from the gaseous waste tanks to the unit vent, the following conditions shall be met:

a. The gaseous radioactivity monitor, the iodine monitor, and the particulate monitor in the unit vent shall be operable.

b. The waste gases and particulates shall be passed through the high efficiency particulate filters and charcoal filters except, when under unusual conditions the filter system is inoperable gaseous wastes shall be held up for the maximum period practicable prior to release. Every reasonable effort shall be made to return inoperable filters to the operable condition before releases to the environment are made.

3.10.5 a. The gaseous waste tanks shall be maintained and operated for the purposes of keeping releases within the objectives of these specifications and shall process all radioactive gases from the vent gas header prior to their release in order to limit the activity released during any calendar quarter to one fourth the annual release quantities or less as determined by the objectives of this specification.

b. The maximum activity to be contained in one gaseous waste tank shall not exceed $17,200/\bar{E}$ curies. $\bar{E}$ will be assumed to be the same as the $\bar{E}$ of the noble gases in the reactor coolant system as determined in accordance with Table 4.1-3 of Specification 4.1.2.

c. As far as practicable, release of radioactive gas will be co-ordinated with favorable meteorological conditions.

3.10.6 During power operation, whenever the air ejector off-gas monitor is inoperable, grab samples shall be taken from the air ejector discharge and analyzed for gross radioactivity daily.

3.10.7 Gases discharged through the unit vent shall be continuously monitored for gross noble gas and particulate activity. Whenever either of these monitors is inoperable, appropriate grab samples shall be taken and analyzed daily.

3.10.8 The reactor building shall not be purged unless the following conditions are met;

a. Reactor building purge shall be through the high efficiency particulate filters and charcoal filters until the activity concentration is below the occupational limit inside the reactor building, at which time bypass may be initiated.

b. If reactor building is purged, the purge shall be through the high efficiency particulate filters whenever irradiated fuel is being handled or any objects are being handled over irradiated fuel in the reactor building.

3.10.9 In addition to the above continuous sampling and monitoring requirements, gaseous radioactive waste sampling and activity analysis shall be performed in accordance with Table 4.1-3. Records shall be maintained and reports of the sampling and analysis results shall be submitted in accordance with Section 6.6 of these specifications.

Bases

It is expected that the releases of radioactive materials and gaseous wastes will be kept within the design objective levels and will not exceed on an instantaneous basis the dose rate limits specified in 10CFR20.

These levels provide reasonable assurance that the resulting annual exposure from noble gases to the whole body or any organ of an individual will not exceed 10 mRem per year. At the same time, the licensee is permitted the flexibility of operation compatible with considerations of health and safety to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10CFR20. It is expected that using this operational flexibility under unusual operating conditions, the licensee shall exert every effort to keep levels of radioactive materials and gaseous wastes as low as practicable and that annual releases will not exceed a small fraction of the annual average concentration limits specified in 10CFR20. These efforts shall include consideration of meteorological conditions during releases.

The anticipated annual releases from the three Oconee reactor units have been developed taking into account a combination of system variables including fuel failure, primary system leakage, and the performance of radio-isotope removal mechanisms. The values assumed for these variables include the following:

a. Reactor coolant fission product concentration corresponding to 0.25 percent fuel cladding defects;

b. Steam generator primary-to-secondary leakage rate of 20 gpd;

c. Reactor coolant leakage to the containment building of 120 gpd and 12 containment vents per year;

d. Primary coolant stripped 12 times per year;

e. Decay time of the waste gas processing system - 30 days;

f. Decontamination factor of 1,000 for iodine in the evaporator;

g. Charcoal filter decontamination factor of 10 for iodine removal in the purge exhaust system.

The application of the above estimates results in the radio-gas discharge rates shown in Table III-13 of the "Final Environmental Statement Related to Operation of Oconee Nuclear Station Units 1, 2, and 3".

The noble gas release rates stated in the objectives are based on a X/Q value from the annual meteorological data. The dispersion factor used, 3.6×10^{-6} sec/m^3, is conservative and the release rate is controlled to a small fraction of 10CFR20 requirements at the exclusion area boundary (.02 of 10CFR20 = less than 10 mRem per year).

The I-131 and particulate release rates stated in the objectives limits the concentration at the exclusion area boundary to much less than 1 percent of the MPC listed in 10CFR20. The release rate also controls the expected iodine dose due to the milk pathway (using concentration factor in the milk pathway of 700) at the nearest cow and the nearest dairy (taken as five miles west, $X/Q = 1.22 \times 10^{-7}$ sec/m^3) to less than five millirem per year. This meets the intent of proposed Appendix I to 10CFR50. A survey will be conducted once per year to assure that no milk producing cows are within a five mile radius of the plant.

The maximum one-hour release rate limits the dose rate at the exclusion area boundary to less than 2 mRem per hour even during periods of unfavorable meteorology (using conservative meteorological conditions, i.e., two hour X/Q of 1.16×10^{-4} sec/m^3 accident meteorology).

The maximum activity in a gaseous waste tank is specified as 17,200 curies/$\overline{E}$ based on a postulated tank rupture that allows all of the contents to escape to the atmosphere. This specification limits the maximum off-site dose to well below the limits of 10CFR100.

The lowest practicable gaseous release objectives expressed in this specification are based on the guidelines contained in the proposed Appendix I of 10CFR50. Since these guidelines have not yet been adopted, the release objectives of this specification will be reviewed at the time Appendix I becomes a regulation to assure that this specification is based upon the guidelines contained therein.

3.11 MAXIMUM POWER RESTRICTION

Applicability

Applies to the nuclear steam supply system of Units 1 and 2 reactors.

Objective

To maintain a power and core life margin in reserve until the system has performed under operating conditions and design objectives for a significant period of time.

Specification

3.11.1 Unit 1 power level may not be increased above 2452 MWt until operated in the range of 2352 MWt to 2452 MWt for 30 days, except that 50 percent of the time the power can be as low as 2,000 MWt, and a subsequent approval is granted by the Directorate of Licensing.

3.11.2 The first reactor core in Unit 1 may not be operated beyond 7500 effective full power hours until supporting analyses and data pertinent to fuel clad collapse under fuel densification conditions have been approved by Directorate of Licensing staff.

3.11.3 Unit 2 power level may not be increased above 2452 MWt until the approval specified in Specification 3.11.1 has been granted by the Directorate of Licensing.

3.11.4 The first reactor core in Unit 2 may not be operated beyond 11,040 effective full power hours until supporting analysis and data pertinent to fuel clad collapse under fuel densification conditions have been approved by the Directorate of Licensing.

Bases

The Preliminary Safety Analysis Report section of the application for a construction permit was based on a maximum power level of 2452 MWt. Subsequent safety evaluations done as part of the Final Safety Analysis Report were done for power levels of 2568 MWt. However, since this is the first nuclear steam supply system of this design to go into service, a power margin of 116 MWt is temporarily being held in reserve until the system has performed at significant power levels for a reasonable period of time. Following evaluation of the summary report of plant startup and power escalation test programs and evaluations, (required by these Technical Specifications) and in the absence of any significant deviation in plant performance from that predicted by design and required for safety, it is expected that this temporary restriction will be lifted.

The licensing staff has reviewed the effects of fuel densification for the first core in Oconee Units 1 and 2 and concluded that clad collapse will not take place within the first fuel cycle (7500 effective full power hours for Unit 1 and 11040 effective full power hours for Unit 2). However, the clad collapse model used is questionable for extrapolation of clad collapse time out beyond the first fuel cycle because of limited experimental verification.

3.12 REACTOR BUILDING POLAR CRANE AND AUXILIARY HOIST

Applicability

Applies to the use of the reactor building polar crane over the steam generator compartments and the fuel transfer canal and the Auxiliary Hoist over the fuel transfer canal.

Objective

To identify those conditions for which the operation of the reactor building polar crane and auxiliary hoist are restricted.

Specification

3.12.1 The reactor building polar crane shall not be operated over the fuel transfer canal when any fuel assembly is being moved.

3.12.2 The auxiliary hoist shall not be operated over the fuel transfer canal when any fuel assembly is being moved unless the hoist is being used to move that assembly.

3.12.3 During the period when the reactor vessel head is removed and irradiated fuel is in the reactor building and fuel is not being moved, the reactor building polar crane and auxiliary hoist shall be operated over the fuel transfer canal only where necessary and in accordance with approved operating procedures stating the purpose of such use.

3.12.4 When the reactor vessel head is removed and the polar crane is being operated in areas away from the fuel transfer canal, the flagman shall be located on top of the secondary shield wall when the polar crane hook is above the elevation of the fuel transfer canal.

3.12.5 During the period when the reactor coolant system is pressurized above 300 psig, and is above 200°F, and fuel is in the core, the reactor building polar crane shall not be operated over the steam generator compartments.

Bases

Restriction of use of the reactor building polar crane and auxiliary hoist over the fuel transfer canal when the reactor vessel head is removed to those operations necessary for the fuel handling and core internals operations is to preclude the dropping of materials or equipment into the reactor vessel and possibly damaging the fuel to the extent that an escape of fission products would result. The fuel transfer canal will be delineated by readily visible markers at an elevation above which the reactor building polar crane would not normally handle loads.

Restriction of use of the reactor building polar crane over the steam generator compartments during the time when steam could be formed from dropping a load on the steam generator or reactor coolant piping resulting in rupture of the system is required to protect against a loss of coolant accident.

3.13 SECONDARY SYSTEM ACTIVITY

Applicability

Applies to the limiting conditions of secondary system activity for operation of the reactor.

Objective

To limit the maximum secondary system activity.

Specification

The iodine-131 activity in the secondary side of a steam generator shall not exceed 1.4 μCi/cc.

Bases

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following a loss of load accident is considered. As stated in FSAR Section 14.1.2.8.2, 148,000 pounds of water is released to the atmosphere via the relief valves. A site boundary dose limit of 1.5 rem is used.

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser air ejector, thus, in the event of a loss of load incident there are only small quantities of these gases which would be released.

I-131 is the significant isotope because of its low MPC in air and because the other iodine isotopes have shorter half-lives, and therefore, cannot build up to significant concentrations in the secondary coolant, given the limitations on primary system leak rate and technical specification limiting activity. One-tenth of the contained iodine is assumed to reach the site boundary, making allowance for plateout and retention in water droplets. I-131 is assumed to contribute 70% of the total thyroid dose based on the ratio of I-131 to the total iodine isotopes given in Table 11-3 of the FSAR.

The maximum inhalation dose at the site boundary is then as follows:

Dose (rem) = $Ci \cdot V \cdot B \cdot DCF \cdot (0.1) \cdot X/Q$
C = Secondary coolant activity (2.0 uCi/cc I-131 equivalent)
V = Secondary water volume released to atmosphere (90 m^3)
B = Breathing rate (3.47×10^{-4} m^3/sec)
X/Q = Ground level release dispersion factor (1.16×10^{-4} sec/m^3)
DCF = 1.48×10^6 rem/Ci
0.1 = Fraction of activity released

The resultant dose is 1.15 rem compared to the Radiation Protection Guide of 1.5 rem for an annual individual exposure in an unrestricted area.

4 SURVEILLANCE STANDARDS

Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

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

Objective

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

Specification

4.1.1 The minimum frequency and type of surveillance required for reactor protective system and engineered safety feature protective system instrumentation when the reactor is critical 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 detector system, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten (10) effective full power days.

Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures 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

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be calibrated (during steady state operating conditions) when indicated neutron power and core thermal power differ by more than 2 percent. During non-steady state operation, the nuclear flux channels amplifiers shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

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 of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

On-line testing of reactor protective channels is required once every four weeks on a rotational or perfectly staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protective channels is as follows:

Channels A, B, C & D	Before startup if the reactor has been shutdown for greater than seven days
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protective system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every four weeks. The trip test checks all logic combinations and is to be performed on a rotational basis. Discovery of an unsafe failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the equipment and systems in a safe operational status.

Power Distribution Mapping

The incore instrumentation detector system will provide a means of assuring that axial and radial power peaks and the peak locations are being controlled by the provisions of the Technical Specifications within the limits employed in the safety analysis.

REFERENCE

FSAR, Section 7.1.2.3.4

TABLE 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
1. Protective Channel Coincidence Logic	NA	M	NA	
2. Control Rod Drive Trip Breaker	NA	M	NA	
3. Power Range Amplifier	D(1)	NA	(1)	(1) Heat Balance Check daily. Heat balance calibration whenever indicated neutron power & core thermal power differ by more than 2%.
4. Power Range Channel	S	M	M(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper & lower chambers after each startup if not done previous week.
5. Intermediate Range Channel	S(1)	P	NA	(1) When in service
6. Source Range Channel	S(1)	P	NA	(1) When in service
7. Reactor Coolant Temperature Channel	S	M	R	
8. High Reactor Coolant Pressure Channel	S	M	R	
9. Low Reactor Coolant Pressure Channel	S	M	R	
10. Flux-Reactor Coolant Flow Comparator	S	M	R	
11. Reactor Coolant Pressure Temperature Comparator	S	M	R	
12. Pump Flux Comparator	S	M	R	

TABLE 4.1-1 Cont.

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
13. High Reactor Building Pressure Channel	D	M	R	
14. High Pressure Injection Logic Channel	NA	M	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S	M	R	
b. Reactor Building 4 psig Channel	S	M	R	
16. Low Pressure Injection Logic Channel	NA	M	NA	
17. Low Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S	M	R	
b. Reactor Building 4 psig Channel	S	M	R	
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	M	NA	
19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
a. Reactor Building 4 psig Channels	S	M	R	

TABLE 4.1-1 Cont.

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
20. Reactor Building Spray System Logic Channel	NA	M	NA	
21. Reactor Building Spray System Analog Channels				
a. Reactor Building High Pressure Channels	NA	M	R	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	R(2)	(1) Check with Relative Position Indicator (2) Calibrate rod misalignment channel
24. Control Rod Relative Position	S(1)	NA	R(2)	(1) Check with Absolute Position Indicator (2) Calibrate rod misalignment channel
25. Core Flooding Tanks				
a. Pressure Channels	S	NA	R	
b. Level Channels	S	NA	R	
26. Pressurizer Level Channels	S	NA	R	
27. Letdown Storage Tank Levels Channels	D	NA	R	
28. Radiation Monitoring Systems	W(1)	M	Q	(1) Check functioning of self-checking feature on each detector.
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

TABLE 4.1-1 Cont.

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
30. Borated Water Storage Tank Level Indicator	W	NA	R	
31. Boric Acid Mix Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
32. Concentrated Boric Acid Storage Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
33. Containment Temperature	NA	NA	R	
34. Incore Neutron Detectors	M(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout
35. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check
36. Environmental Monitors	M(1)	NA	R	(1) Check Functioning
37. Reactor Manual Trip	NA	P	NA	
38. Reactor Building Emerg. Sump Level	NA	NA	R	
39. Steam Generator Water Level	W	NA	R	
40. Turbine Overspeed Trip	NA	NA	R	
41. Engineered Safeguards Channel 1 HP Injection Manual Trip	NA	R	NA	

TABLE 4.1-1 Cont.

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
42. Engineered Safeguards Channel 2 HP Injection Manual Trip	NA	R	NA	
43. Engineered Safeguards Channel 3 LP Injection Manual Trip	NA	R	NA	
44. Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	R	NA	
45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	R	NA	
46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	R	NA	
47. Engineered Safeguards Channel 7 Spray Manual Trip	NA	R	NA	
48. Engineered Safeguards Channel 8 Spray Manual Trip	NA	R	NA	

S - Each Shift

D - Daily

W - Weekly

M - Monthly

R - Each Refueling Period

NA- Not Applicable

Q - Quarterly

P - Prior to each startup if not done previous week

Table 4.1-2

Minimum Equipment Test Frequency

	Item	Test	Frequency
1.	Control Rods	Rod Drop Times of all full length rods	Each Refueling shutdown
2.	Control Rod Movement (1)	Movement of each rod	Every two weeks
3.	Pressurizer Safety Valves	Setpoint	50% each refueling period
4.	Main Steam Safety Valves	Setpoint	25% each refueling period
5.	Refueling System Interlocks	Functional	Each refueling period
6.	Turbine Steam Stop Valves(1)	Movement of each stop valve	Monthly
7.	Reactor Coolant System(2) Leakage	Evaluate	Daily
8.	Charcoal and High Efficiency Filters for Penetration Room, Control Room, and RB Purge Filters	DOP Test on HEPA filters. Freon Test on Charcoal Filter Units	Each refueling period and at any time work on filters could alter their integrity.
9.	Condenser Cooling Water System Gravity Flow Test	Functional	Each refueling period
10.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
11.	Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
12.	Hydraulic snubbers on safety related systems	Visual Inspection	Each refueling period

(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.

TABLE 4.1-3

MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency
1. Reactor Coolant	a. Gamma Isotopic Analysis	a. Monthly*
	b. Radiochemical Analysis for Sr 89, 90	b. Monthly*
	c. Tritium	c. Monthly*
	d. Gross Beta & Gamma Activity (1)	d. 5 times/week*
	e. Chemistry (Cl, F and O2)	e. 5 times/week*
	f. Boron Concentration	f. 2 times/week**
	g. Gross Alpha Activity	g. Monthly*
	h. $\bar{E}$ Determination (2)	h. Semi-annually
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly* and after each makeup
3. Core Flooding Tank	Boron Concentration	Monthly* and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly*** and after each makeup
5. Secondary Coolant	a. Gross Beta & Gamma Activity	a. Weekly*
	b. Iodine Analysis (3)	
6. Concentrated Boric Acid Tank	Boron Concentration	Twice weekly*

*Not applicable if reactor is in a cold shutdown condition for a period exceeding the sampling frequency.
**Applicable only when fuel is in the reactor.
***Applicable only when fuel is in the spent fuel pool.

TABLE 4.1-3 Cont.

MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency	Sensitivity of Waste Analysis in Lab
7. Low Activity Waste Tank & Condensate Test Tank	a. Gross Beta & Gamma Activity	a. Prior to release of each batch	a. $<10^{-7}$ μCi/ml
	b. Radiochemical Analysis Sr 89, 90	b. Monthly	b. $<10^{-8}$ μCi/ml
	c. Gamma Analysis including Dissolved Noble Gases	c. Monthly	c. Gamma Nuclides $<5x10^{-7}$ μCi/ml Dissolved Gases $<10^{-5}$ μCi/ml
	d. Tritium	d. Monthly	d. $<10^{-5}$ μCi/ml
	e. Gross Alpha Activity	e. Monthly	e. $<10^{-7}$ μCi/ml
	f. Ba-La-140, I-131	f. Weekly Proportional	f. $<5x10^{-7}$ μCi/ml
8. Waste Gas Decay Tank	a. Gamma Isotopic Analysis	a. Prior to release of each batch	a. $<10^{-4}$ μCi/cc
	b. Gross Gamma Activity	b. Prior to release of each batch	b. $<10^{-11}$ μCi/cc
	c. Tritium	c. Prior to release of each batch	c. $<10^{-6}$ μCi/cc
9. Unit Vent Sampling	a. Iodine Spectrum(4)	a. Weekly	a. $<10^{-10}$ μCi/cc
	b. Particulates (4)		
	1) Gross Beta & Gamma Activity	1) Weekly	1) $<10^{-11}$ μCi/cc
	2) Gross Alpha Activity	2) Quarterly on a sample of one week duration	2) $<10^{-11}$ μCi/cc

TABLE 4.1-3 Cont.

MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency	Sensitivity of Waste Analysis in Lab
	3) Gamma Isotopic Analysis	3) Monthly Composite	3) $<10^{-10}$ μCi/cc
	4) Radiochemical Analysis Sr 89, 90	4) Quarterly Composite	4) $<10^{-11}$ μCi/cc
	5) Ba-La-140, I-131	5) Weekly	5) $<10^{-10}$ μCi/cc
10. Keowee Hydro Dam Dilution Flow	Measure Leakage Flow Rate	Annually	
11. Condenser Air Ejector Partition Factor	Measure Iodine Partition Factor in Condenser	One time if and when primary to secondary leaks develop	
12. Reactor Building Purge	a. Gamma Isotopic Analysis	a. Each Purge	a. $<10^{-4}$ μCi/cc
	b. Gross Gamma Activity	b. Each Purge	b. $<10^{-11}$ μCi/cc
	c. Tritium	c. Each Purge	c. $<10^{-6}$ μCi/cc

(1) When radioactivity level is greater than 10 percent of the limits of Specification 3.1.4, the sampling frequency shall be increased to a minimum of once each day.

(2) $\bar{E}$ determination will be started when gross beta-gamma activity analysis indicates greater than 10 μCi/ml and will be redetermined each 10 μCi/ml increase in gross beta-gamma activity analysis. A radiochemical analysis for this purpose shall consist of a quantitative measurement of 95% of radionuclides in reactor coolant with half lives of >30 minutes. This is expected to consist of gamma isotopic analysis of primary coolant including dissolved gaseous activities, radiochemical analysis for Sr 89, 90, and tritium analysis.

TABLE 4.1-3 Cont.

MINIMUM SAMPLING FREQUENCY

(3) When gross activity increases by a factor of two above background, an iodine analysis will be made and performed thereafter when the gross beta-gamma activity increases by 10 percent.

(4) When activity level exceeds 10 percent of the limits of Specification 3.9, the sampling frequency shall be increased to a minimum of once each day. This can be done by RIA-44 (Unit Vent Iodine) monitor. When the gross activity release rate exceeds one percent of maximum release rate and the average gross activity release rate increases by 50 percent over the previous day, an analysis shall be performed for iodines and particulates. This can be done by RIA-44 (Unit Vent Iodine Monitor) and RIA-43 (Unit Vent Particulate Monitor).

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the reactor coolant system pressure boundary.

Objective

To assure the continued integrity of the reactor coolant system pressure boundary.

Specification

4.2.1 Prior to initial unit operation, an ultrasonic test survey shall be made of reactor coolant system pressure boundary welds as required to establish preoperational integrity and base line data for future inspections.

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

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

4.2.3 The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a

result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated, including evaluation of comparable areas of the reactor coolant system.

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

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

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

4.2.7 The inspection of each reactor coolant pump flywheel shall include: a volumetric examination, in place, of the areas of higher stress concentration at the bore and key way at approximately three year intervals. A surface examination of exposed surfaces, and a complete volumetric examination at approximately 10 year intervals when disassembly and/or flywheel removal is required for maintenance or repair. Disassembly or flywheel removal is not required to perform these examinations.

4.2.8 "A 'B' Type vessel specimen capsule shall be withdrawn after one year of operation. An 'A' Type capsule shall be withdrawn after 11, 17, and 22 years of operation. The withdrawal schedules may be modified to coincide with those refueling outages or unit shutdowns most closely approaching the withdrawal schedule. Specimens thus withdrawn shall be tested in accordance with ASTM-E-185-70. A report of the test results shall be forwarded to the AEC within 90 days of withdrawal."

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

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, 1970, including 1970 Winter addenda, edition. The program

places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

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

A Type	B Type
Weld Material	Haz Material
Haz Material	Base Line Material
Base Line Material	

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

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2285 psig prior to the reactor being made critical.

4.3.3 The limitations of Specification 3.1.2 shall apply.

Bases

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as USAS B31.7, code for pressure piping, Nuclear Power Piping dated February, 1968, and as corrected for errata under date of June, 1968, and ASME Boiler and Pressure Vessel Code, Section XI, IS-400, dated January, 1970.

REFERENCES

FSAR, Section 4

4.4 REACTOR BUILDING

4.4.1 Containment Leakage Tests

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the reactor building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leakage Rate Tests

4.4.1.1.1 Design Pressure Leakage Rate

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

4.4.1.1.2 Testing at Reduced Pressure

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

a. Prior to reactor operation the initial value of the integrated leakage rate of the reactor building shall be measured at design pressure and at the reduced pressure to be used in the periodic integrated leakage rate tests. The leakage rates thus measured shall be identified as L_{pm} and L_{tm} respectively.

b. L_t shall not exceed $L_a \left[\frac{L_{tm}}{L_{pm}} \right]$ for values of $\frac{L_{tm}}{L_{pm}}$ below 0.7.

c. L_t shall not exceed $L_a \sqrt{\frac{P_t}{P_p}}$ for values of $\frac{L_{tm}}{L_{pm}}$ above 0.7.

d. If L_{tm}/L_{pm} is less than 0.3, the initial integrated test results shall be subject to review by the AEC to establish an acceptable value of L_t.

4.4.1.1.3 Conduct of Tests

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

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

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

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

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

4.4.1.1.4 Frequency of Test

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

4.4.1.1.5 Conditions for Return to Criticality

a. If L_t is less than 50% of the value permitted in 4.4.1.1.2, local leakage rate testing need not be completed prior to return to criticality following a periodic integrated leakage rate test.

b. If L_t is between 50 and 100% of the value permitted in 4.4.1.1.2 the return to criticality will be permitted conditioned upon demonstrating that local leakage rate into the penetration room measured at full design pressure, accounts for all leakage above 50% of that permitted by 4.4.1.1.2. If this cannot be demonstrated within 30 days of returning to criticality, the reactor shall be shutdown.

4.4.1.1.6 Corrective Action and Retest

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

4.4.1.1.7 Report of Test Results

Each integrated leak rate test will be the subject of a summary technical report which will include a description of test methods used and a summary of local leak detection tests. Sufficient data and analysis shall be included to show that a stabilized leak rate was attained and to identify all significant required correction factors such as those associated with humidity and barometric pressure, and all significant errors such as those associated with instrumentation sensitivities and data scatter.

4.4.1.2. Local Leakage Rate Tests

4.4.1.2.1 Scope of Testing

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

(a) Personnel hatch

(b) Emergency hatch

(c) Equipment hatch seals

(d) Fuel transfer tube seals

(e) Reactor building normal sump drain line

(f) Reactor coolant pump seal outlet line

(g) Reactor coolant pump seal inlet line

(h) Quench Tank drain line

(i) Quench Tank return line

(j) Quench Tank vent line

(k) Normal makeup to reactor coolant system

(l) High pressure injection line

(m) Electrical penetrations

(n) Reactor building purge inlet line

(o) Reactor building purge outlet line

(p) Reactor building sample lines

(q) Reactor coolant letdown line

4.4.1.2.2 Conduct of Tests

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

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

4.4.1.2.3 Acceptance Criteria

The total leakage from all penetrations and isolation valves shall not exceed 0.125% of the reactor building atmosphere per 24 hours.

4.4.1.2.4 Corrective Action and Retest

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

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

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed at a frequency of at least each refueling period, except that:

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

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

4.4.1.3 Isolation Valve Functional Tests

Every three months, remotely operated reactor building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The latter valves shall be tested during each refueling period.

4.4.1.4 Annual Inspection

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The dis-

covery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests, and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

4.4.1.5 Reactor Building Modifications

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

Bases(1)

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

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner. The test pressure of 29.5 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leakage rate test at 29.5 psig. The specification provides a relationship for relating the measured leakage of air at 29.5 psig to the potential leakage at 59 psig. The minimum of 24 hours was specified for the integrated leakage rate test to help stabilize conditions and thus improve accuracy and to better evaluate data scatter. The frequency of the periodic integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.25% leakage rate at 59 psig during pre-operational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at design pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.125%) of leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the containment is maintained.

Leakage to the penetration room, which is permitted to be up to 50% of the total allowable containment leakage, is discharged through high efficiency particulate air (HEPA) and charcoal filters to the unit vent. The filters are conservatively said to be 90% efficient for iodine removal.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage than the reactor building liner due to the mechanical closure involved. Particular attention is given to testing those penetrations with resilient sealing materials, penetrations that vent directly to the reactor building atmosphere, and penetrations that connect to the reactor coolant system pressure boundary. The basis for specifying a maximum leakage rate of 0.125% from penetrations and isolation valves is that one-half of the actual integrated leakage rate is expected from those sources. Valve operability tests are specified to assure proper closure or opening of the reactor building isolation valves to provide for isolation of functioning of Engineered Safety Features systems. Valves will be stroked to the position required to fulfill their safety function unless it is established that such testing is not practical during operation. Valves that cannot be full-stroke tested will be part-stroke tested during operation and full-stroke tested during each normal refueling shutdown.

REFERENCES

(1) FSAR, Sections 5 and 13.

4.4.2 Structural Integrity

Applicability

Applies to the structural integrity of the reactor buildings.

Objective

To define the inservice surveillance program for the reactor buildings.

Specification

4.4.2.1 Tendon Surveillance

For the initial surveillance program, covering the first five years of operation, nine tendons shall be selected for periodic inspection for symptoms of material deterioration or force reduction. The surveillance tendons shall consist of three horizontal tendons. One in each of three 120° sectors of the containment; three vertical tendons located at approximately 120° apart; and three dome tendons located approximately 120° apart. The following nine tendons have been selected as the surveillance tendons:

Dome	1D28
	2D28
	3D28
Horizontal	13H9
	51H9
	53H10
Vertical	23V14
	45V16
	61V16

4.4.2.1.1 Lift-Off

Lift-off readings shall be taken for all 9 surveillance tendons.

4.4.2.1.2 Wire Inspection and Testing

One surveillance tendon of each directional group shall be relaxed and one wire from each relaxed tendon shall be removed as a sample and visually inspected for corrosion or pitting. Tensile tests shall also be performed on a minimum of three specimens taken from the ends and middle of each of the three wires. The specimens shall be the maximum length acceptable for the test apparatus to be used and shall include areas representative of signaficant corrosion or pitting.

After the wire removal, the tendons shall be retensioned to the stress level measured at the lift-off reading and then checked by a final lift-off reading.

Should the inspection of one of the wires reveal any significant corrosion (pitting or loss of area), further inspection of the other two sets in that directional group will be made to determine the extent of the corrosion and its significance to the load-carrying capability of the structure. The sheathing filler will be sampled and inspected for changes in physical appearance.

Wire samples shall be selected in such a manner that with the third inspection, wires from all 9 surveillance tendons shall have been inspected and tested.

4.4.2.2 Inspection Intervals and Reports

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

A quantitative analytical report covering results of each inspection shall be submitted (required by Technical Specification 6.6.5.7.e) and shall especially address the following conditions, should they develop:

(1) Broken wires.

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

(3) Unexpected changes in corrosion conditions or sheathing filler properties.

4.4.2.3 End Anchorage Concrete Surveillance

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

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

c. The inspections made shall include:

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

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

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

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

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

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

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

e. The acceptance criteria shall be as follows:

If the inspections determine that the conditions are favorable in comparison with experience and predictions, the close inspections will be terminated by the last of the inspections stated in the schedule and a report will be prepared which documents the findings and recommends the schedule for future inspections, if any. If the inspections detect symptoms of greater than normal cracking or movements, an immediate investigation will be made to determine the cause.

4.4.2.4 Liner Plate Surveillance

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

a. Location of areas which have inward deformations. The magnitude of the inward deformations shall be measured and recorded. These areas shall be permanently marked for future reference and the inward deformations shall be measured between the angle stiffeners which are on 15-inch centers. The measurements shall be accurate to $\pm$ .01 inch. Temperature readings shall be obtained on both the liner plate and outside containment wall at the locations where inward deformations occur.

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

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

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

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

Bases

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

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

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

Vertical - Three tendons spaced approximately 120° apart.

Dome - Three tendons spaced approximately 120° apart.

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

REFERENCE

FSAR, Section 5.6.2.2

4.4.3 Hydrogen Purge System

Applicability

Applies to testing Reactor Building Hydrogen Purge System.

Objective

To verify that this system and components are operable.

Specification

4.4.3.1 Operating Tests

An in-place system test shall be performed annually using the written emergency procedures. These tests shall consist of visual inspection, hook-up of system to either the Unit 1 or Unit 2 reactor building, a flow measurement using flow instruments in the portable purging station and pressure drop measurements across the filter bank. Flow shall be design flow or higher, and pressure drops across the filter bank shall not exceed two times the pressure drop when new. Fan motors shall be operated continuously for at least one hour, and valves shall be proven operable. This test shall demonstrate that under simulated emergency conditions the system can be taken from storage and placed into operation within 48 hours.

4.4.3.2 Filter Tests

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

4.4.3.3 H_2 Detector Test

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

Bases

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

The purge rate is controlled through the use of a portable purging station (Insert, Figure 14A-5.1 of the FSAR). The station consists of a purge blower, dehumidifier, filter train, purge flowmeter, sample connection and flowmeter and associated piping and valves.

The blower is a rotitive type rated 60 scfm. The dehumidifier consists of two redundting elements inserted in a section of ventilation duct. The function dehumidifier is to sufficiently increase the temperature of the enterinp assure 70 percent relative humidity entering the filter train with lent saturated air entering the dehumidifier. The purpose of the dehum is to assure optimum charcoal filter efficiency. Heating element cont provided by a thermoswitch. Humidity indication is provided downstrthe heating elements by a humidity readout gage. The filter train prprefiltration, high efficiency particulate filtration and charcoal fon. The filter train assembly is identical in design to the waste lter train assembly which is rated at 200 scfm, thus conservatively capaperforming the assigned function. Face velocity to the charcoal filter y low. The charcoal filter is composed of a module consisting of two in double tray carbon cells. The purge flow to the unit vent is metered a 0-60 scfm rotometer. The purge sample flow is metered using a 0-1 rotometer. Both of these rotometers have an accuracy of ± two percent of cale, and each has remote readout capability. The purge discharge ratentrolled by a blower discharge throttling valve. The purge sample acts can be collected, counted and analyzed in the radio-chemistry labo Makeup air to the Reactor Building is supplied by a compressed air connection to one of the aforementioned existing vent and pressurizatnnections.

That portion of the ation room ventilation system piping and valves which is used as a part of rge system is permanently installed and is designed for seismic loading the existing vent and pressurization connections. The remainder of the system is the portable purging station which is stored in an area wh earthquake will not damage it. Following a LOCA, there is adequate tbre purging is required to permit checkout of the portable purging stand to optimize the system operation to minimize the total dose to the pu

References
FSAR Section 14A

4.5 EY CORE COOLING SYSTEM AND REACTOR BUILDING CGSYSTEM PERIODIC TESTING

4.5.1 ncy Core Cooling Systems

Applicabilit

Applies to p testing requirement for emergency core cooling systems.

Objective

To verify th emergency core cooling systems are operable.

Specificatic

4.5.1.1 Systs

4.5.1.1.1 h Pressure Injection System

During each refueling period, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the high pressure injection system for emergency core cooling operation.

The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.

4.5.1.1.2 Pressure Injection System

During each refueling period, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to demonstrate actuation of the low pressure injection system for emergency core cooling operation.

(2) Verification of the engineered safety features function of the low pressure service water system which supplies cooling water to the low pressure coolers shall be made to demonstrate operability of the coolers.

The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel.

4.5.1.1.3 looding System

uring each refueling period, a system test shall be onducted to demonstrate proper operation of the system. uring pressurization of the Reactor Coolant System, erification shall be made that the check and isolation alves in the core flooding tank discharge lines operate roperly.

ne test will be considered satisfactory if control oard indication of core flood tank level verifies that ll valves have opened.

4.5.1.2 Compones

4.5.1.2.1

ervals not to exceed 3 months, the high pressure and ssure injection pumps shall be started and operated to proper operation. Acceptable performance will be ed if the pump starts, operates for fifteen minutes, discharge pressure and flow are within ±10% of a n the pump head curve.

4.5.1.2.2 - Power Operated

intervals not to exceed three months each engineered fety features valve in the emergency core cooling stems and each engineered safety features valve sociated with emergency core cooling in the low pres- re service water system shall be tested to verify erability.

e acceptable performance of each power operated valve ll be that motion is indicated upon actuation by propriate signals.

ting each refueling period, low pressure injection p discharge (Engineered Safety Features) valves, pressure injection discharge throttling valves, and pressure injection discharge header crossover valves ll be cycled manually to verify the manual operability these power operated valves.

Bases

The emergency coing systems are the principle reactor safety features in the event of af coolant accident. The removal of heat from the core provided by ystems is designed to limit core damage.

The high pressureion system under normal operating conditions has one pump operating. t once per month, operation will be rotated to another high pressjection pump. This will help verify that the high pressure injectio are operable.

The requirements of the low pressure service water system for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis will verify that two pumps are operable.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

Testing the manual operability of power operated valves in the low pressure injection system gives assurance that flow can be established in a timely manner even if the capability to operate a valve from the control room is lost.

With the reactor shut down, the valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

REFERENCE

FSAR, Section 6

4.5.2 Reactor Building Cooling Systems

Applicability

Applies to testing of the reactor building cooling systems.

Objective

To verify that the reactor building cooling systems are operable.

Specification

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

(a) During each refueling period a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the reactor building spray system (except for reactor building inlet valves to prevent water entering nozzles). Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

(b) Station compressed air will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every five years.

(c) The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly; the appropriate pump breakers shall have closed, all valves shall have completed their travel.

4.5.2.1.2 Reactor Building Cooling System

(a) During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the reactor building cooling system for reactor building cooling operation.

(2) Verification of the engineered safety features function of the low pressure service water system which supplies coolant to the reactor building coolers shall be made to demonstrate operability of the coolers.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate pump breakers have completed their travel, fans are running at half speed, LPSW flow through each cooler exceeds 1400 GPM and air flow through each fan exceeds 40,000 CFM.

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

At intervals not to exceed 3 months the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for 15 minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

4.5.2.2.2 Valves

At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building cooling system and each engineered safety features valve associated with reactor building cooling in the low pressure service water system shall be tested to verify that it is operable.

Bases

The reactor building cooling systems and reactor building spray system are designed to remove the heat in the containment atmosphere to prevent the building pressure from exceeding the design pressure.

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or fog can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the reactor building cooling system are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

One low pressure service water pump will normally be operating. At least once per month, the operation will be rotated to another low pressure service water pump. Testing will be, therefore, unnecessary.

The reactor building fans are normally operated periodically, constituting the test that these fans are operable.

REFERENCE

FSAR, Section 6

4.5.3 Penetration Room Ventilation System

Applicability

Applies to testing of the reactor building penetration room ventilation system.

Objective

To verify that the penetration room ventilation system is operable.

Specification

4.5.3.1 System Tests

4.5.3.1.1 At intervals not to exceed 3 months, a system test shall be conducted to demonstrate proper operation of the system. This test shall consist of visual inspection, a flow measurement using the flow instrument installed at the outlet of each unit and pressure drop measurements across each filter unit. In addition, a test signal will be applied to demonstrate proper actuation of the penetration room ventilation system. Fan motors shall be operated continuously for at least one hour, and the louvers and other mechanical system shall be proven operable and adjustable from their remote location.

4.5.3.1.2 The test will be considered satisfactory if control board indication verifies that all components have responded properly to the actuation signal, if flow rate through the system is design flow or higher, and if pressure drops across any filter bank do not exceed two times the pressure drop which existed when the filters were new.

4.5.3.2 Filter Tests

No less frequently than each normal refueling period, "in-place" leakage tests using DOP on HEPA units and Freon-112 (or equivalent) on charcoal units shall be performed at design flow on each filter train. Removal of 99.5% DOP by each entire HEPA filter unit and removal of 99.0% Freon-112 (or equivalent) by each entire charcoal absorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the structural integrity of the filtration system units.

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration room leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of a sealed penetration room, two redundant filter trains, and two redundant fans discharging to the unit vent. The entire system is activated by a 4 psig reactor building pressure engineered safety features signal and initially required no operator action.

Each filter train is constructed with a prefilter, an absolute filter, and a charcoal filter in series. The design flow rate through each of these filters is 1000 scfm, which is significantly higher than the approximately 15 scfm maximum leakage rate from the reactor building at a leak rate of 0.25% per day. Except for periodic ventilation of the penetration room, the penetration room ventilation system is not normally used. Quarterly testing of this system will show that the system is available for its engineered safety features function. During this test, the system will be inspected for such things as water, oil, or other foreign material, gasket deterioration, and unusual or excessive noise or vibration when the fan motor is running.

Less frequent testing will verify the efficiency of the absolute and charcoal filters.

4.5.4 Low Pressure Injection System Leakage

Applicability

Applies to Low Pressure Injection System leakage.

Objective

To maintain a preventative leakage rate for the Low Pressure Injection System which will prevent significant offsite exposures.

Specification

4.5.4.1 Acceptance Limit

The maximum allowable leakage from the Low Pressure Injection System components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

4.5.4.2 Test

During each refueling period the following tests of the Low Pressure Injection System shall be conducted to determine leakage:

a. The portion of the Low Pressure Injection System, except as specified in (b), that is outside the containment shall be tested either by use in normal operation or by hydrostatically testing at 350 psig.

b. Piping from the containment emergency sump to the low pressure injection pump suction isolation valve shall be pressure tested at no less than 59 psig.

c. Visual inspection shall be made for excessive leakage from components of the system. Any excessive leakage shall be measured by collection and weighing or by another equivalent method.

Bases

The leakage rate limit for the Low Pressure Injection System is a judgment value based on assuring that the components can be expected to operate without mechanical failure for a period on the order of 200 days after a Loss of Coolant Accident. The test pressure (350 psig) achieved either by normal system operation or by hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the pressure test for the return lines from the containment to the Low Pressure Injection System (59 psig) is equivalent to the design pressure of the containment. The dose to the thyroid calculated as a result of this leakage is .76 rem for a 2 hr. exposure at the site boundary.

REFERENCES

FSAR, Section 14.2.2.4.4.

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTING

Applicability

Applies to the periodic testing and surveillance of the emergency power system.

Objective

To verify that the emergency power sources and equipment will respond promptly and properly when required.

Specification

4.6.1 At intervals not to exceed one month, a test of the Keowee Hydro units shall be performed to verify proper operation of these emergency power sources and associated equipment. This test shall assure that:

a. Each hydro unit can be automatically started from the Oconee Control Room.

b. Each hydro unit can be synchronized through the 230 kV overhead circuit to the startup transformers.

c. Each hydro unit can energize the 13.8 kV underground feeder.

4.6.2 During each calendar year at a refueling outage, the Keowee Hydro Units will be started using the emergency start circuits in each control room to verify that each hydro unit and associated equipment is available to carry load within 25 seconds of a simulated requirement for engineered safety features.

4.6.3 Annually a simulated emergency transfer to the 4160 volt main feeder buses shall be made to transformers CT1, CT2, and CT3, and to the 4160 volt standby buses to verify proper operation.

4.6.4 Quarterly the External Grid Trouble Protection System Logic shall be tested to demonstrate its ability to provide an isolated power path between Keowee and Oconee.

4.6.5 Annually it shall be demonstrated that a Lee Station combustion turbine can be started and connected to the 100 kV line. It shall be demonstrated that the 100 kV line can be separated from the rest of the system and supply power to the 4160 volt main feeder buses.

4.6.6 Batteries in the Instrument and Control, Keowee Station, and Switching Station 125 volt DC systems shall be tested as follows:

a. The voltage and temperature of a pilot cell in each bank shall be measured and recorded daily, five days/week.

b. The specific gravity and voltage of each cell shall be measured and recorded every month.

c. Before initial operation and at each refueling outage, a one-hour discharge test at the required maximum safeguards load will be made.

4.6.7 The operability of the individual diode monitors in Instrument and Control and Keowee Station 125V DC system shall be verified on a monthly basis by imposing a simulated diode failure signal on the monitor.

4.6.8 The peak inverse voltage capability of each auctioneering diode in the Instrument and Control, Switchyard, and Keowee Station 125V DC system shall be measured and recorded every six months.

4.6.9 The tests specified in 4.6.6, 4.6.7, and 4.6.8 will be considered satisfactory if control room indication and/or visual examination demonstrate that all components have operated properly.

Bases

The Keowee Hydro units, in addition to serving as the emergency power sources for the Oconee Nuclear Station, are power generating sources for the Duke system requirements. As power generating units, they are operated frequently, normally on a daily basis at loads equal to or greater than required by Table 8.5 of the FSAR for ESF bus loads. Normal as well as emergency startup and operation of these units will be from the Oconee Unit 1 Control Room. The frequent starting and loading of these units to meet Duke system power requirements assures the continuous availability for emergency power for the Oconee auxiliaries and engineered safety features equipment. It will be verified that these units are available to carry load within 25 seconds, including instrumentation lag, after a simulated requirement for engineered safety features. To further assure the reliability of these units as emergency power sources, they will be, as specified, tested for automatic start on a monthly basis from the Oconee control room. These tests will include verification that each unit can be synchronized to the 230 kV bus and that each unit can energize the 13.8 kV underground feeder.

The interval specified for testing of transfer to emergency power sources is based on maintaining maximum availability of redundant power sources.

Starting a Lee Station gas turbine, separation of the 100 kV line from the remainder of the system, and charging of the 4160 volt main feeder buses are specified to assure the continuity and operability of this equipment.

REFERENCE

FSAR, Section 8

4.7 REACTOR CONTROL ROD SYSTEM TESTS

4.7.1 Control Rod Drive System Functional Tests

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at reactor coolant full flow conditions or 1.40 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine (9) inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a 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.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located with absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14.

Each control rod drive mechanism shall be exercised by a movement of approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod

deviates from its group average position by more than nine (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.[2]

REFERENCES

(1) FSAR, Section 14

(2) Technical Specification 3.5.2

4.7.2 Control Rod Program Verification (Group vs. Core Positions)

Applicability

Applies to surveillance of the control rod systems.

Objective

To verify that the designated control rod (by core position 1 through 69) is operating in its programmed functional position and group. (Rod 1 through 12, Group 1-8)

Specification

4.7.2.1 Whenever the control rod drive patch panel is locked (after inspection, test, reprogramming, or maintenance) each control rod drive mechanism shall be selected from the control room and exercised by a movement of approximately two inches to verify that the proper rod has responded as shown on the unit computer printout of that rod.

4.7.2.2 Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection shall be performed.

4.7.2.3 Any rod found to be improperly programmed shall be declared inoperable until properly programmed.

Bases

Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer identified by a unique number (1 through 69) associated with only one core position. The other set of outputs goes to a programmable bank of 69 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or to the control room meter bank are improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check by (1) selecting a specific rod from one group (e.g. Rod 1 in Regulating Group 6) (2) noting that the program-approved core position for this rod of the group (assume the approved core position is No. 53) (3) exercise the selected rod and (4) note that (a) the computer prints out both absolute and relative position response for the approved core position (assumed to be position No. 53) (b) the proper meter in the control room display bank (assumed to be Rod 1 in Group 6) in both absolute and relative meter positions. This type of comparative check will not assure detection of improperly connected cables inside the reactor building. For these, (Spec. 4.7.2.2) it will be necessary for a responsible person, other than the one doing the work, to verify by appropriate means that each cable has been matched to the proper control rod drive assembly.

4.8 MAIN STEAM STOP VALVES

Applicability

Applies to the main steam stop valves.

Objective

To verify the ability of the main steam stop valves to close upon signal and to verify the leak tightness of the main steam stop valves.

Specification

4.8.1 Using Channels A and B the operation of each of the main steam stop valves shall be tested no less frequently than the normal refueling period interval to demonstrate a closure time of one second or less in Channel A and a closure time of 15 seconds or less for Channel B.

4.8.2 The leak rate through the main steam stop valves shall not exceed 25 cubic feet per hour at a pressure of 59 psig and shall be tested no less frequently than the normal refueling period.

Bases

The main steam stop valves limit the reactor coolant system cooldown rate and resultant reactivity insertion following a main steam line break accident. Their ability to promptly close upon redundant signals will be verified at each scheduled refueling shutdown. Channel A solenoid valves are designed to close all four turbine stop valves in 240 milliseconds. The backup Channel B solenoid valves are designed to close the turbine stop valves in approximately 12 seconds.(1)

Using the maximum 15 second stop val ~ closing time, the fouled steam generator inventories, and the minimum ' ipped rod worth with the maximum stuck rod worth, an analysis similar to that presented in FSAR Section 14.1.2.9, (but considering a blowdown of both steam generators) shows that the reactor will remain subcritical after reactor trip following a double-ended steam line break.

The main stop valves would become isolation valves in the unlikely event that there should be a rupture of a reactor coolant line concurrent with rupture of the steam generator feedwater header. The allowable leak rate of 25 cubic feet per hour is approximately 25 percent of total allowable containment leakage from all penetrations and isolation valves.(2)

REFERENCES

(1) FSAR, Supplement 2, Page 2-7

(2) Technical Specifications 4.4.1

4.9 EMERGENCY FEEDWATER PUMP PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine driven emergency feedwater pump.

Objective

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

Specification

4.9.1 Test

On a three-month basis, the turbine driven emergency feedwater pump shall be operated on recirculation to the upper surge tank for a minimum of one hour.

4.9.2 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 three (3) month testing frequency will be sufficient to verify that the turbine driven emergency feedwater pump is operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pump.

REFERENCE

FSAR, Section 10.2.2

FSAR, Section 14.1.2.8.3

4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require the evaluation of reactivity anomalies of a specified magnitude occurring during the operation of the unit.

Specification

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of discrepancy shall be made and reported to the Atomic Energy Commission.

Bases

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn position is always maintained.

4.11 ENVIRONMENTAL SURVEILLANCE

Applicability

Applies to the routine testing of the station environs for radiation and radioactive materials attributable to station operation and waste releases.

Objective

To establish a sampling schedule for the purpose of detecting, measuring and evaluating any significant effects of station operation and waste releases on the environment.

Specification

4.11.1 Environmental samples taken in accordance with Table 2-1a of the FSAR shall be collected and processed in accordance with Table 4.11-1 of these specifications. In addition to these samples, soil samples will be collected near the location of the terrestrial vegetation samples. Further, data will be included from two air particulate and air iodine samplers (near Warpath Recreation Area, and near boat dock at R. W. Gudger's Office) in the South Carolina State Board of Health Environmental Surveillance Program.

4.11.2 Thermoluminescent dosimeters will be installed at various locations within the exclusion area including shoreline areas, and read quarterly.

4.11.3 The location of cows and/or goats used as a direct source of milk for individuals, and the location of any dairies within five miles of the plant will be determined semi-annually by survey and reported with the results of the radiological environmental monitoring program. These locations will be included in the milk sampling program as soon as the necessary arrangements can be made. For these locations the sampling frequency will be weekly with analysis for I-131. Monthly composites will receive gamma analysis and analysis for radiostrontium.

Bases

The program will be conducted in accordance with Section 2.7 of the FSAR.

The sensitivity of counting used in the analyses (90% confidence level when counted for 20 minutes) is based on nominal background levels of 0.5 cpm alpha and 1.0 cpm beta.

Environmental monitoring results will be correlated with information on station radioactive waste releases, site meteorological data and radiological controls, and with information obtained from the installed process radiation monitoring system.

The Environmental Surveillance Program will provide means of detecting significant changes in levels of radioactivity. The results will demonstrate the effectiveness of station control over radioactive waste disposal operations and of compliance with Federal and State regulations for disposal of these materials.

The thermoluminescent dosimeter data will be used in conjunction with routine surveillance of recreational activities on the lake within the exclusion area to assure that the exposure of the public will be maintained to levels within the limits of 10 CFR 20 and as low as practicable for unrestricted areas.

REFERENCES

FSAR, Section 2.7

TABLE 4.11-1
ENVIRONMENTAL SURVEILLANCE PROGRAM

Type Samples	Schedule	Analysis	
Water (1) (3)	Monthly	Gross Alpha & Beta Activity	Gamma Analysis*(4)
Rain, Settled Dust	Monthly	"	Gamma Analysis (4)
Air Particulate	Monthly,	"	Gamma Analysis (4)
	Charcoal Filter Weekly	"	Gamma Analysis for I^{131}
Vegetation, Terrestrial(2)	Quarterly	"	Gamma Analysis Quarterly
Aquatic Organiams	Quarterly	"	Gamma Analysis Quarterly
	Semi-annually	Composite Sr^{89}, Sr^{90}(5)	
Bottom Sediment	Quarterly	Gross Alpha & Beta Activity	Gamma Analysis Quarterly
	Semi-annually	Sr^{89}, Sr^{90}	
Radiation Dose & Dose Rate	Quarterly	mr & mr/hr	
Animals	Quarterly	Gamma Analysis	Sr^{89}, Sr^{90} Cs^{137}
Fish	Quarterly	Gamma Analysis	Sr^{89}, Sr^{90} Cs^{137}, K^{40}, I^{131}
Milk	Bi Weekly	Gamma Analysis	I^{131}(6)
	Monthly (composite)	Gamma Analysis	Sr^{89}, Sr^{90}, Cs^{137}, K^{40}
Soil	Annually	Gamma Analysis	

*Dependent on gross activity for individual samples.

(1) Lakes Keowee and Hartwell will be sampled annually for tritium analysis.

(2) Commercial Crops will be substituted when available.

(3) A composite sample from the Keowee River, below the liquid waste effluent release point, will be analyzed monthly for gross Alpha and Beta and undergo an isotopic analysis.

(4) A gamma analysis on a quarterly composite sample.

(5) If sufficient sample quantity is available for analysis.

(6) Sensitivity will be sufficient to detect 0.5 $pc_i/l \pm 25\%$ at time of sampling.

4.12 CONTROL ROOM FILTERING SYSTEM

Applicability

Applies to control room filtering system components.

Objective

To verify that these systems and components will be able to perform their design functions.

Specification

4.12.1 Operating Tests

System tests shall be performed at approximately quarterly intervals. These tests shall consist of visual inspection, a flow measurement at the outlet of each unit and pressure drop measurements across each filter bank. Pressure drop across pre-filter shall not exceed 1" H_2O and pressure drop across HEPA shall not exceed 2" H_2O. Fan motors shall be operated continuously for at least one hour, and all louvers and other mechanical systems shall be proven operable.

4.12.2 Filter Tests

Annually for the Unit 1 and 2 control room, an "in-place" leakage test using DOP on HEPA units and Freon-112 (or equivalent) on charcoal units shall be performed at design flow on each filter train. Removal of 99.5% DOP by each entire HEPA filter unit and removal of 99.0% Freon-112 (or equivalent) by each entire charcoal adsorber unit shall constitute acceptable performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing.

Bases

The purpose of the control room filtering system is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with two 100 percent capacity filter trains each of which consists of a prefilter, high efficiency particulate filters, charcoal filters and a booster fan to pressurize the control room with outside air.

Since these systems are not normally operated, a periodic test is required to insure their operability when needed. Quarterly testing of this system will show that the system is available for its safety action. During this test the system will be inspected for such things as water, oil, or other foreign material; gasket deterioration, adhesive deterioration in the HEPA units; and unusual or excessive noise or vibration when the fan motor is running.

Annual testing will verify the efficiency of the charcoal and absolute filters.

4.13 FUEL SURVEILLANCE

Applicability

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

Objective

To specify the fuel surveillance program for fuel rods.

Specification

4.13.1 Visual Inspection

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

4.13.2 Dimensional Examination

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

a. One assembly after the first cycle.

b. Four assemblies after the second cycle.

c. Two assemblies after the third cycle.

Bases

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

4.14 REACTOR BUILDING PURGE SYSTEM

Applicability

Applies to testing Reactor Building Purge Filters.

Objective

To verify that the Reactor Building Purge Filters will perform their design function.

Specification

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

Bases

The reactor building purge filter is constructed with a prefilter, an absolute filter and a charcoal filter in series. This test will verify the efficiency of the absolute and charcoal filters.

4.15 IODINE RADIATION MONITORING FILTERS

Applicability

Applies to the Iodine Radiation Monitoring Filters.

Objective

To assure that the Iodine Radiation Monitoring Filters perform their intended function.

Specification

4.15.1 The Iodine Radiation Monitoring Charcoal Filters will be removed and replaced after every 30 days of operation in RIA 44.

4.15.12 All spare Iodine Radiation Monitoring Charcoal Filters will be discarded after 2 years of shelf-life.

Bases

The purpose of this specification is to assure the reliability of the Iodine Radiation Monitoring Charcoal Filters. This specification was submitted in compliance to a request by the AEC.

5 DESTURES

5.1 S

5.1.1 Tee Nuclear Station is approximately eight miles northeast oa, South Carolina. Figure 2-3 of the Oconee FSAR shows t of the site. The minimum distance from the reactor cine to the boundary of the exclusion area and to the outer b of the low population zone as defined in 10 CFR 100.3, s one mile and six miles respectively.

5.1.2 Fpurpose of satisfying 10 CFR Part 20, the "Restricted Aor gaseous release purposes only, is the same as the en area as defined above except that the temporary con-sn quarters (2) in the east southeast section of the en area shall not, when occupied, be deemed to be within tricted area.

REFERENCE

(1) FSAR, 2.2

(2) Technicification 3.9

5.2 CONTAINMENT

Specification

The containment for this unit consists of three systems which are the reactor building, reactor building isolation system, and penetration room ventilation system.

5.2.1 Reactor Building

The reactor building completely encloses the reactor and its associated reactor coolant system. It is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and flat foundation slab. The cylindrical portion is prestressed by a post tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post tensioning system. The structure can withstand the loss of 3 horizontal and 3 vertical tendons in the cylinder wall or adjacent tendons in the dome without loss of function. The foundation slab is conventionally reinforced with high strength reinforcing steel. The entire structure is lined with 1/4" welded steel plate to provide vapor tightness.

The internal volume of the reactor building is approximately 1.91 x 10^6 cu. ft. The approximate inside dimensions are: diameter--116'; height--208 1/2'. The approximate thickness of the concrete forming the buildings are: cylindrical wall--3 3/4'; dome--3 1/4'; and the foundation slab--8 1/2'.

The concrete containment structure provides adequate biological shielding for both normal operation and accident situations. Design pressure and temperature are 59 psig and 286°F, respectively.

The reactor building is designed for an external atmospheric pressure of 3.0 psi greater than the internal pressure. This is greater than the differential pressure of 2.5 psig that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. Since the building is designed for this pressure differential, vacuum breakers are not required.

Penetration assemblies are seal welded to the reactor building liner. Access openings, electrical penetrations, and fuel transfer tube covers are equipped with double seals. Reactor building purge penetrations and reactor building atmosphere sampling penetrations are equipped with double valves having resilient seating surfaces. (1)

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in FSAR Section 14 with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks.

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. (2)

5.2.3 Penetration Room Ventilation System

This system is designed to collect, control, and minimize the release of radioactive materials from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slight negative pressure will be maintained in the penetration room to assure inleakage. (3)

REFERENCES

(1) FSAR Section 5.1

(2) FSAR Section 5.2

(3) FSAR Section 5.3

5.3 REACTOR

Specification

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 93 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies, all of which are prepressurized with Helium.

5.3.1.2 The fuel assemblies shall form an essentially cylindrical lattice with an active height of 144 in. and an equivalent diameter of 128.9 in. (2)

5.3.1.3 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSR) distributed in the reactor core as shown in FSAR Figure 3-46. The full-length CRA contain a 134 inch length of silver-indium-cadmium alloy clad with stainless steel. The APSR contain a 36 inch length of silver-indium-cadnium alloy. (3)

5.3.1.4 Initial core and reload fuel assemblies and rods shall conform to design and evaluation described in FSAR and shall not exceed an enrichment of 3.5 percent of U-235.

5.3.2 Reactor Coolant System

5.3.2.1 The design of the pressure components in the reactor coolant system shall be in accordance with the code requirements.(4)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670°F.(5)

5.3.2.3 The maximum reactor coolant system volume shall be 12,200 ft^3.

REFERENCES

(1) FSAR Section 3.2.1

(2) FSAR Section 3.2.2

(3) FSAR Section 3.2.4

(4) FSAR Section 4.1.3

(5) FSAR Section 4.1.2

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Specification

5.4.1 New Fuel Storage

5.4.1.1 New fuel will normally be stored in the spent fuel pool serving the respective unit. The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21 inches in both directions. This spacing is sufficient to maintain a K effective of less than .9 when flooded with unborated water, based on fuel with an enrichment of 3.5 weight percent U^{235}.

5.4.1.2 New Fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in five racks in a row having a nominal center to center distance of 2' 1 3/4". One rack is oversized to receive a failed fuel assembly container. The other four racks are normal size and are capable of receiving new fuel assemblies.

5.4.1.3 New fuel may also be stored in shipping containers.

5.4.2 Spent Fuel Storage

5.4.2.1 Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pool, which is located in its respective auxiliary building. Each pool is sized to accommodate a full core of irradiated fuel assemblies in addition to the concurrent storage of the largest quantity of new and spent fuel assemblies predicted by the fuel management program.

5.4.2.2 Whenever there is fuel in the pool (except the initial core loading), the spent fuel pool is filled with water borated to the concentration that is used in the reactor cavity and fuel transfer canal during refueling operations.

5.4.2.3 Spent fuel may also be stored in storage racks in the fuel transfer canal when the canal is at refueling level.

5.4.2.4 The spent fuel pool and fuel transfer canal racks are designed for an earthquake force of 0.1g ground motion.

REFERENCES

FSAR, Section 9.7

6 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION, REVIEW AND AUDIT

Introduction

Administrative controls relate to the organization and management procedures, record keeping, review and audit systems, and reporting that are considered necessary to provide the assurance and evidence that the station will be managed in a dependable manner.

The administrative controls specify the administrative tools and functions necessary for safe operation. They also define the administrative action to be taken in the event operating limits or safety limits are exceeded.

Specification

These administrative controls in regard to operations, review and audit become effective at the time of issuance of the facility license by the AEC, and after normal design and construction activities have been essentially completed.

6.1.1 Organization

6.1.1.1 The Superintendent is directly responsible for the safe operation of the facility.

6.1.1.2 In all matters pertaining to actual operation and maintenance and to these Technical Specifications, the Superintendent shall report to and be directly responsible to the Assistant Vice President, Steam Production. The organization is shown in Figure 6.1-2.

6.1.1.3 The station organization for operation, Technical Support, and Maintenance shall be functionally as shown in Figure 6.1-1.

6.1.1.4 Incorporated in the staff of the station shall be supervisory and professional personnel meeting the minimum requirements specified in FSAR Section 12.2 encompassing the training and experience described in Section 4 of the ANSI 18.1, "Selection and Training of Nuclear Power Plant Personnel."

6.1.1.5 Retraining and replacement of station personnel shall be in accordance with Section 5.5 of the ANSI 18.1, "Selection and Training of Nuclear Power Plant Personnel."

6.1.1.6 Shift staffing shall be provided as follows:

a. Units 1 and 2 minimum shift staffing for other than cold shutdown will be according to Table 6.1-1. This minimum operating staff shall be permitted to assist in the pre-licensing activity of Unit 3 only to the extent that it does not affect their full availability for Units 1 and 2 operation.

b. At least one licensed Reactor Operator per unit and two Senior Reactor Operators shall be at the station at all times when there is fuel in two reactor vessels. One licensed operator per unit shall be in the control room for that unit. At no time will the shift crew be less than three persons per unit.

c. Two licensed reactor operators shall be in the control room during startup and scheduled shutdown of the reactor, and during recovery from reactor trips.

d. At least one licensed reactor operator shall be in the reactor building when fuel handling operations are in progress in the reactor building. An operator holding a Senior Reactor Operators license and assigned no other concurrent operational duties shall be in direct charge of refueling operations.

e. At least one operator per shift for each unit will have sufficient training to perform routine health physics requirements.

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

6.1.2 Review and Audit

In matters of nuclear safety and radiation exposure, review and audit of station operation, maintenance and technical matters shall be provided by two committees as follows (Reference Figure 6.1.2):

6.1.2.1 Station Review Committee

a. Membership

Assistant Superintendent--Chairman
Operating Engineer
Technical Support Engineer
At least two other members of the station supervisory staff appointed by the Superintendent

The Superintendent shall appoint an acting chairman in the absence of the Assistant Superintendent.

b. Meeting Frequency

This committee shall meet at least once each month and as required on call by the chairman.

c. Quorum

The chairman plus two members shall constitute a quorum.

d. Responsibilities

The committee shall have the following responsibilities:

(1) Review all new procedures or proposed significant, safety-related changes to existing procedures as determined by the Station Superintendent.

(2) Review station operation and safety considerations.

(3) Review abnormal occurrences, unusual events, and violations of Technical Specifications and make recommendations to prevent recurrence.

(4) Review proposed tests determined by the Station Superintendent to significantly affect the nuclear safety of the facility.

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

e. Authority

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

f. Records

Minutes of all meetings of the Committee shall be kept at the station, and copies shall be sent to the Superintendent, Assistant Vice President Steam Production, and the chairman of the Nuclear Safety Review Committee.

6.1.2.2 Nuclear Safety Review Committee

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

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

Subjects within the purview of the committee.

Responsibility and authority.

Mechanisms for convening meetings.

Provisions for use of specialists or sub-groups.

Authority for access to station records.

Reporting requirements.

Identification of management position to which the group reports.

Provisions for assuring that the committee is kept informed of matters within its purview.

c. The committee shall be composed of:

Chairman

At least two members from the Steam Production Department. (May include Superintendent or Assistant Superintendent but not other Oconee Nuclear Station personnel)

At least two members from the Engineering Department.

Others deemed advisable. (May include consultant from outside the company)

The committee shall elect a vice chairman.

Qualified alternates shall be appointed or other provisions shall be made for covering the absence of fulltime members of the group. The use of alternates shall be restricted to legitimate and unavoidable absences of principals.

d. Qualifications:

At least one-half of the members of the committee (and/or alternates attending a specific meeting) shall have extensive nuclear experience and all members and alternates shall be engineering or science graduates. No more than a minority of the members or alternates shall have direct line responsibility for station operation. All members shall have a minimum of three years' professional level experience in nuclear services, nuclear plant operation, or nuclear engineering and the necessary overall nuclear background to detect when to call consultants and contractors for dealing with complex problems beyond the scope of their own organization.

e. Members of the committee shall collectively have the capability required to review the areas of:

(1) Nuclear power plant operations
(2) Nuclear Engineering
(3) Chemistry and Radiochemistry
(4) Metallurgy
(5) Instrumentation and Control
(6) Radiological Safety
(7) Mechanical and Electrical Engineering
(8) Other appropriate fields associated with the unique characteristics of the Oconee Nuclear Station.

When the nature of a particular situation dictates, special consultants shall be utilized to provide expert advice to the committee.

f. Meeting Frequency:

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

g. Quorum

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

h. Meeting Minutes:

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

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

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

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

(3) All requests to the AEC/DOL for changes in Technical Specifications or license that involve unreviewed safety questions as defined in 10CFR50.59.

(4) Violations of Statutes, Regulations, Orders, Technical Specifications, License Requirements, or Internal Procedures, or Instructions having Safety Significance.

(5) Significant operating abnormalities or deviations from normal performance of unit equipment.

(6) Abnormal occurrences as defined in 1.8 of these Specifications.

(7) Station operating records, logs, reports, and tests on a periodic basis.

(8) Station Review Committee Minutes

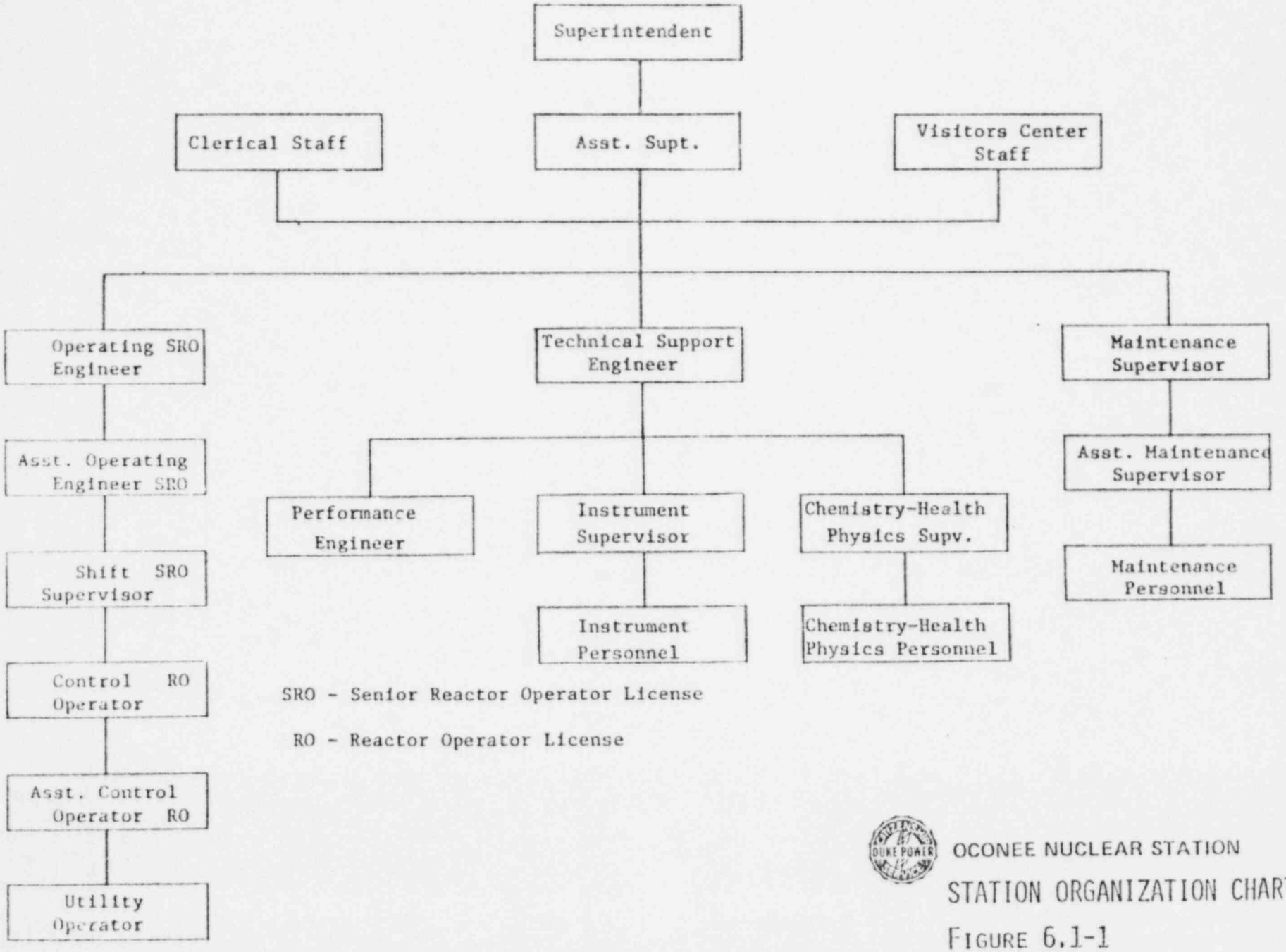
(9) Conduct special reviews or investigations as required by the Assistant Vice President, Steam Production or the Station Superintendent.

(10) Non-Routine Reports to the AEC and AEC Responses

The committee shall make recommendations relating to the review of the above items to the appropriate members of management to prevent or reduce the probability of recurrence. Copies of these recommendations shall be included in the meeting minutes.

(11) Activities of the Steam Production Quality Assurance Organization.

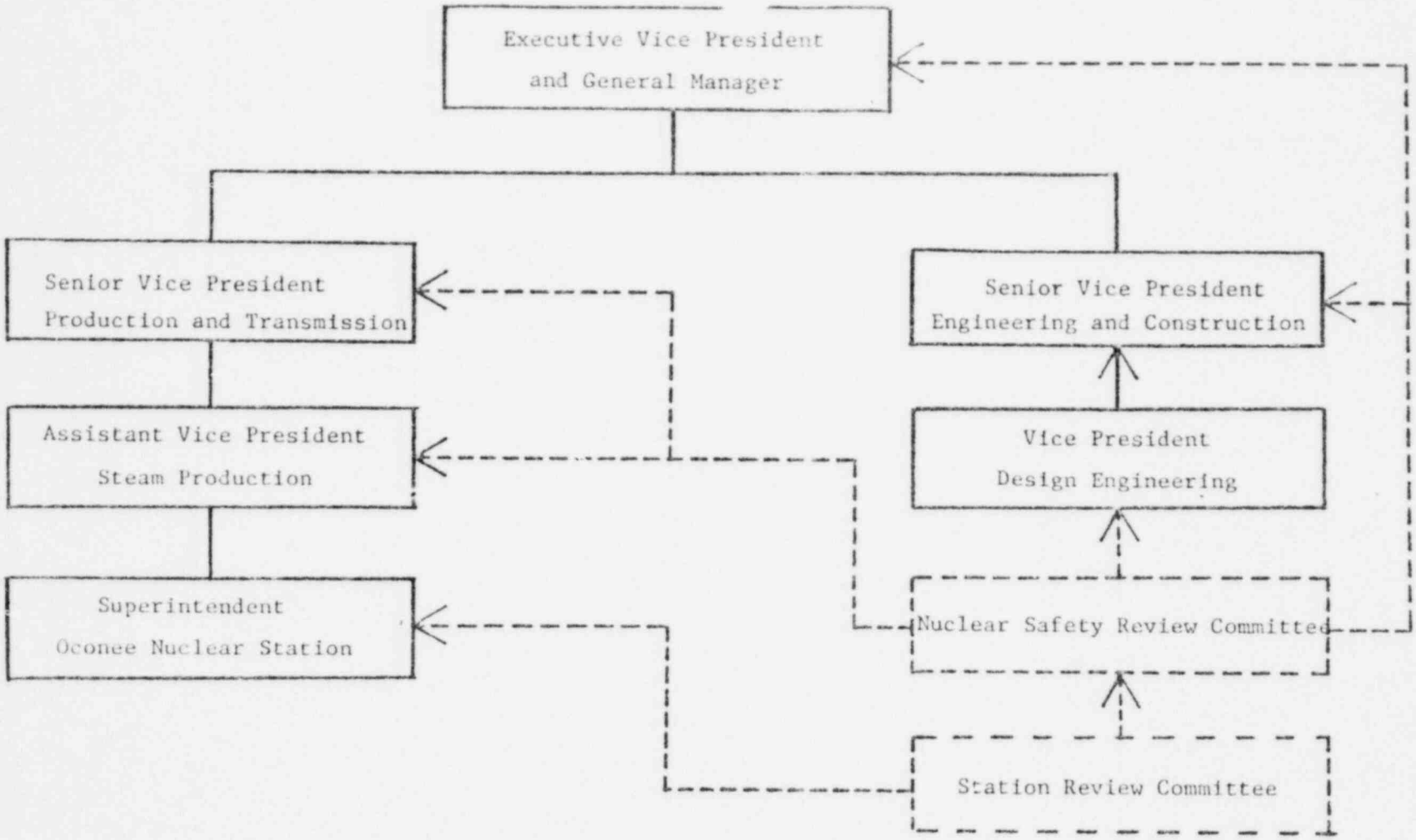
j. It is the intent of this specification to fulfill the requirements and recommendations of Proposed Standard ANS-3.2, Section 4, entitled, "Standard for Administrative Controls for Nuclear Power Plants," - Draft 8, November 2, 1972.



OCONEE NUCLEAR STATION

STATION ORGANIZATION CHART

FIGURE 6.1-1



OCONEE NUCLEAR STATION

MANAGEMENT ORGANIZATION CHART

FIGURE 6.1-2

TABLE 6.1-1

OCONEE NUCLEAR STATION

MINIMUM OPERATING SHIFT REQUIREMENTS

UNITS 1 & 2

Responsibility	Minimum Qualifications	Number/ Shift
Shift Supervisor	SRO	1
Assistant Shift Supervisor	SRO	1
Control Operator	RO	1
Asst. Control Operator	RO	2
Utility Operator		2
Total/Shift		7

SRO - AEC Senior Reactor Operator License
RO - AEC Reactor Operator License

6.2 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE OR UNUSUAL EVENT

Specification

6.2.1 Any abnormal occurrence or unusual event shall be investigated promptly by the Superintendent.

6.2.2 The Superintendent shall promptly notify the Assistant Vice President, Steam Production, of any abnormal occurrence or unusual event and shall cause the Station Review Committee to perform a review and prepare a written report which shall describe the circumstances leading up to and resulting from the occurrence and shall recommend appropriate action to prevent or minimize the probability of a recurrence.

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

6.2.4 The Senior Vice President, Production-Transmission, shall report the circumstances of any abnormal occurrence or unusual event to the AEC as specified in Section 6.6, Station Reporting Requirements.

6.3 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

Specification

If a safety limit is exceeded:

6.3.1 The reactor shall be shut down immediately and maintained in a safe shutdown condition until otherwise authorized by the AEC.

6.3.2 The Superintendent shall make an immediate report to the Assistant Vice President, Steam Production; the Senior Vice President, Production and Transmission; and the Chairman of the Nuclear Safety Review Committee.

6.3.3 The circumstances shall be promptly reported to the AEC by the Senior Vice President, Production and Transmission as indicated in Section 6.6.2.1, Station Reporting Requirements.

6.3.4 The Superintendent shall direct the Station Review Committee to perform an analysis of the circumstances leading up to and resulting from the situation together with recommendations to prevent a recurrence. The report covering this analysis shall be sent to the Nuclear Safety Review Committee for review and approval. Copies of this report shall also be submitted to the Superintendent; Assistant Vice President, Steam Production; the Senior Vice President, Production and Transmission; the Chairman of the Nuclear Safety Review Committee; the Senior Vice President, Engineering and Construction; Vice President, Design Engineering; and the Executive Vice President and General Manager. Appropriate analyses or reports shall be submitted to the AEC by the Senior Vice President, Production and Transmission as indicated in Section 6.6.2.1, Station Reporting Requirements.

6.4 STATION OPERATING PROCEDURES

Specification

6.4.1 The station shall be operated and maintained in accordance with approved procedures. Detailed written procedures with appropriate check-off lists and instructions shall be provided for the following conditions:

a. Normal startup, operation and shutdown of the complete facility and of all systems and components involving nuclear safety of the facility.

b. Refueling operations.

c. Actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety and radiation levels, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

d. Emergency procedures involving potential or actual release of radioactivity.

e. Preventive or corrective maintenance which could affect nuclear safety or radiation exposure to personnel.

f. Station survey following an earthquake.

g. Radiation control procedures.

h. Operation of radioactive waste management systems.

i. Control of pH in recirculated coolant after loss-of-coolant accident. Procedure shall state that pH will be measured and the addition of appropriate caustic to coolant will commence within 30 minutes after switchover to recirculation mode of core cooling to adjust the pH to a range of 7.0 to 8.0 within 24 hours.

j. Nuclear safety-related periodic test procedures.

k. Long-term emergency core cooling systems. Procedures shall include provision for remote or local operation of system components necessary to establish low pressure injection within 15 minutes after a line break.

6.4.2 Quarterly selected drills shall be conducted on site emergency procedures including assembly preparatory to evacuation off site and a check of the adequacy of communications with off-site support groups.

6.4.3 Respiratory protective program approved by AEC shall be in force.

6.5 STATION OPERATING RECORDS

Specification

6.5.1 The following records shall be prepared and permanently retained in a manner convenient for review:

a. Records of modifications to the station as described in the FSAR.

b. Special nuclear material physical inventory records.

c. Special nuclear material isotopic inventory records.

d. Radiation monitoring records, including records of radiation and contamination surveys.

e. Records of off-site environmental surveys.

f. Personnel radiation exposure records as required by 10CFR20.

g. Records of radioactive releases and waste disposal.

h. Records of reactor coolant system in-service inspections.

i. Preoperational testing records.

j. Records of special reactor tests or experiments.

k. Records of changes to safety-related operating procedures.

6.5.2 The following records shall be prepared and retained for a minimum of six (6) years in a manner convenient for review:

a. Switchboard Record.

b. Reactor Operations Logbook.

c. Shift Supervisor Logbook.

d. Maintenance histories for station safety-related structures, systems and components.

e. Records of safety-related inspections, other than reactor coolant system in-service inspections.

f. Records of abnormal occurrences.

g. Records of unusual events.

h. Periodic testing records and records of other periodic checks, calibrations, etc. performed in accordance with surveillance requirements for safety-related parameters, structures, systems and components.

i. By-product material inventory records.

j. Records of the activities of the Station Review Committee.

k. Minutes of Nuclear Safety Review Committee meetings.

l. Training records.

6.6 STATION REPORTING REQUIREMENTS

6.6.1 Routine Reports

6.6.1.1 Operating Reports

The following reports shall be submitted to the Directorate of Licensing, USAEC, Washington, D. C., 20545

A. Startup Report

Upon receipt of a new operating license or amendment to a facility license involving the planned increase in reactor power level or the installation of a new core, a summary report of unit startup and power escalation test programs and evaluations of results thereof shall be submitted within 60 days following commencement of commercial power, (i.e. following synchronization of the turbo-generator to produce commercial power).

B. First Year Operation Report

A report submitted within 60 days of completion of one year of commercial operation covering:

(1) An evaluation of unit performance to date in comparison with design specifications.

(2) A reassessment of the validity of prior accident analyses in light of measured operating characteristics, which may affect consequences; and system, component, and personnel performance which may affect accident probabilities.

(3) A progress and status report on all items identified in the operating license review as requiring further effort.

(4) An assessment of the performance of structures, systems, and components important to safety.

C. Semi-Annual Operating Report

A Semi-Annual Station Operations Report shall be prepared and submitted within 60 days after the end of each reporting period. The report shall provide the following information (summarized on a monthly basis) and shall cover the six month period or fraction thereof, ending June 30 and December 31. The due date for the first report shall be calculated from the date of initial criticality.

(1) Operations Summary

(a) A narrative summary of operating experience and of changes in facility design that relate to safe operation, performance characteristics (including fuel performance) and operating procedures related to safety occurring during the reporting period.

(b) A summary of results of surveillance tests and inspections.

(c) The results of any periodic containment leak rate tests performed during the reporting period.

(d) A brief summary of those changes, tests, and experiments requiring authorization from the Commission pursuant to 10CFR50.59a.

(e) Any changes in plant operating organization which involve positions for which minimum qualifications are specified in the Technical Specifications.

(2) Power Generation

A summary of the nuclear and electrical output of the unit during the reporting period, and the cumulative total outputs since initial criticality, including:

(a) Gross thermal power generated (in MWH).

(b) Gross electrical power generated (in MWH).

(c) Net electrical power generated (in MWH).

(d) Number of hours the reactor was critical.

(e) Number of hours the generator was on line.

(f) Histogram of thermal power versus time.

(3) Shutdowns

Descriptive material covering all outages occurring during the reporting period. The following information shall be provided for each outage:

(a) The cause of the outage.

(b) The method of shutting down the reactor; e.g., trip, automatic rundown, or manually controlled deliberate shutdown.

(c) Duration of the outage in hours.

(d) Unit status during the outage; e.g., cold shutdown, hot shutdown, or hot standby.

(e) Corrective and preventive action taken to preclude recurrence of each unplanned outage.

(4) Maintenance

A discussion of safety-related maintenance (excluding preventive maintenance) performed during the reporting period on systems and components that are designed to prevent or mitigate the consequences of postulated accidents or to prevent the release of significant amounts of radioactive material. For any malfunction for which corrective maintenance was required, information should be provided on:

(a) The system or component involved.

(b) The cause of the malfunction.

(c) The results and effect on safe operation.

(d) Corrective action taken to prevent repetition.

(e) Precautions taken to provide for reactor safety during repair.

(5) Changes, Tests, and Experiments

A summary of all changes in the facility design and procedures that relate to the safe operation of the facility should be included in the Operations Summary section of this semi-annual report. This section should include a brief description and the summary of the safety evaluation for those changes, tests, and experiments carried out without prior Commission approval, pursuant to the requirements of 10 CFR 50.59(b).

(6) Reporting of Radioactive Effluent Releases

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

(a) Gaseous Releases

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

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

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

(4) Percentage applicable limits released.

(b) Iodine Releases

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

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

(3) Percentage of limit.

(c) Particulate Releases

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

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

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

(4) Percentage of limit.

(d) Liquid Releases

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

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

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

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

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

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

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

(8) Percentage of limit for total activity released.

(e) Solid Waste

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

(2) Estimated total radioactivity (in curies):

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

a. Packaged

b. Shipped

(f) Environmental Monitoring

(1) For each medium sampled during the six-month period, the following information shall be provided.

a. Number of sampling locations.

b. Total number of samples

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

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

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

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

6.6.1.2 Personnel Exposure and Monitoring Reports

A. This report shall be submitted to the Directorate of Licensing, USAEC, Washington, D. C., 20545 within the first quarter of each calendar year.

(1) A report of the total number of individuals for whom personnel monitoring was provided during the calendar year.

(2) A report of individuals, 18 years of age or older whose annual radiation dose exceeded the applicable quarterly numerical values, and for each individual under 18 years of age whose annual dose exceeded 10 percent of the applicable quarterly numerical values will be submitted.

B. A report shall be submitted to the employee and the Directorate of Licensing, USAEC, Washington, D. C., within 30 days after the exposure determination or 90 days from the termination of employment, whichever comes first, on the total exposure to radiation and radioactive material received during the period of employment.

6.6.1.3 Material Status

A. The licensee will file Form AEC-742 within 30 days of December 31 and June 30 to report the status of all special nuclear materials.

B. The licensee will file Form AEC-741 within 10 days of shipping or receiving special nuclear material.

6.6.2 Non-Routine Reports

6.6.2.1 Reporting of Abnormal Occurrences & Unusual Events

A. Events requiring notification within 24 hours (by telephone or telegraph to the Director of Region II Regulatory Operations Office followed by a written report within 10 days to the Directorate of Licensing, USAEC, Washington, D. C. 20545 (copy to the Directorate of Regulatory Operations, Region II, Atlanta, Georgia).

(1) Abnormal occurrences specified in Section 1.8 of the Technical Specifications.

(2) Any significant variation of measured values in a non-conservative direction from corresponding predicted values of safety connected parameters during initial criticality.

The written report, and to the extent possible the preliminary telephone or telegraph report, shall describe, analyze, and evaluate safety implications, and outline the corrective actions and measures taken or planned to prevent recurrence of (1) and (2) above.

B. Unusual Events as defined in Section 1.9 of the Technical Specifications shall be reported within 30 days to the Directorate of Licensing the Directorate of Regulatory Operations, Region II, Atlanta, Georgia.

6.6.2.2 Radiation Exposure and Monitoring

The licensee will report any over exposure, excessive radiation level or concentration to the Directorate of Regulatory Operations, USAEC, Washington, D. C. 20545, and Regulatory Operations, Region II, Atlanta, Georgia, per 10CFR20.

6.6.2.3 Loss of Licensed Material

A. The licensee will report immediately by telephone or telegraph the theft or loss of any licensed material in such quantities and under such circumstances that a substantial hazard may result to persons in an unrestricted area.

B. Within 30 days after the loss of a quantity licensed material, the licensee will file a written report with the Directorate of Licensing, USAEC, Washington, D. C., and the Regulatory Operations, Region II Office, Atlanta, Georgia, containing the following information:

(1) A description of the licensed material involved, including kind, quantity, chemical and physical form;

(2) A description of the circumstances under which the loss or theft occurred;

(3) A statement of disposition or probable disposition of the licensed material involved;

(4) Radiation exposures to individuals, circumstances under which the exposures occurred, and the extent of possible hazard to persons in unrestricted areas;

(5) Actions which have been taken, or will be taken, to recover the material; and

(6) Procedures or measures which have been or will be adopted to prevent a recurrence of the loss or theft of licensed material.

6.6.2.4 Accidental Criticality

The licensee will report promptly any accidental criticality to Regulatory Operations, Region II, Atlanta, Georgia.

6.6.2.5 Incidents Involving Licensed Material

In the event of an incident involving licensed material, the licensee will immediately notify Regulatory Operations, Region II, Atlanta, Georgia.

6.6.3 Special Reports

The following reports shall be prepared and submitted to the Directorate of Licensing, USAEC, Washington, D. C., 20545.

6.6.3.1 Authorization of Changes, Test, and Experiments

The licensee will file a request with the Directorate of Licensing, USAEC, Washington, D. C., for authorization of a change in technical specifications or any change, test or experiment which requires authorization by the Commission.

6.6.3.2 Reactor Building Integrated Leak Rate Test

The initial reactor building integrated leak rate test shall be the subject of a summary technical report and shall be submitted within 90 days and shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications. Other containment leak rate tests that fail to meet the acceptance criteria shall be the subject of a special summary report.

6.6.3.3 Single Loop Operation Report

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

(1) Coolant flows in the idle loop and operating loop are as predicted in FSAR Supplement 7, Tables 1-1 and 1-2.

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

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

6.6.3.4 Reactor Building Structural Tests

The results of the initial reactor building structural tests (as specified in FSAR Section 5.6.1.2 including the Structural Instrumentation Report contained in Amendment No. 25, dated December 30, 1970) shall be reported within 90 days following completion of the test.

6.6.3.5 Reactor Building Structural Integrity Report

A reactor building structural integrity shall be submitted within 90 days of completion of each of the following tests covered by Technical Specification 4.4* (the integrated leak rate test is covered in 6.6.3.2 above): *May be included in Semi-Annual Operations Report.

(1) Annual Inspection
(2) Tendon Stress Surveillance
(3) End Anchorage Concrete Surveillance
(4) Liner Plate Surveillance

6.6.3.6 Inservice Inspection Program Report

The Inservice Inspection program shall be performed as specified in Technical Specification 4.2.

6.6.3.7 Fuel Surveillance Program Report

Report to be submitted upon completion of program on fuel surveillance per Specification 4.13.

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

TABLE 6.6-1

REPORT OF RADIOACTIVE EFFLUENTS

Year _____

I. Liquid Releases

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

TABLE 6.6-1 (CONTINUED)

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

REPORT OF RADIOACTIVE EFFLUENTS

Year _____

II. Airborne Releases	Units	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	TOTAL
1. Total noble gases	Curies													
2. Total halogens	Curies													
3. Total particulate gross radioactivity (β, γ)	Curies													
4. Total tritium	Curies													
5. Total particulate gross alpha radioactivity	Curies													
6. Maximum noble gas release rate	μCi/sec													
7. Percent of applicable limit for:														
a. noble gases	%													
b. halogens	%													
c. particulates	%													
8. Fission and activation products released:	Curies													
Particulates														
Cs-137														
Sr-89 + Sr-90														
Na-24														
Mn-54														
Sb-124														
Halogens														
I-131														
I-132														
I-133														
Gases														
Kr-85														
Xe-133														
Kr-88														
Kr-87														
Kr-85m														
Xe-138														
Xe-135m														
Xe-135														
Ar-41														

6.7 RADIOLOGICAL CONTROLS

Specification

6.7.1 The radiation protection program shall be organized with the following exceptions, to meet the requirements of 10 CFR 20.

a. Pursuant to 10 CFR 20.103(c)(1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this station in determining whether individuals in the Restricted Area are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

1. Notwithstanding any exposure limit provided herein, the licensee shall, as a precautionary procedure, use process or other engineering controls, to the extent practicable, to limit concentrations of radioactive materials in air to levels below those which delimit an airborne radioactivity area as defined in 20.203(d)(1).

2. When it is impracticable to apply process or other engineering controls to limit concentrations of radioactive materials to levels below those which delimit an airborne radioactivity area as defined in 20.203(d)(1), and respiratory protective equipment is used to limit the inhalation of airborne radioactive material, the licensee may make allowance for such use in estimating exposures of individuals to such materials provided:

(a) Intake of radioactive material by any individual within any period of seven consecutive days will not exceed that which would result from inhalation 1/2/3/ of such material 40 hours per week, at uniform concentrations specified in Appendix B, Table I, Column 1 of 10 CFR Part 20.

1/ Since the concentration specified for tritium oxide vapor assumes equal intakes by skin absorption and inhalation, the total intake permitted is twice that which would result from inhalation alone at the concentration specified for H^3, S in Appendix B, Table I, Column 1 for 40 hours

2/ For radioactive materials designated "Sub" in the "Isotope" column of the table, the concentration value specified is based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in ¶20.101. These materials shall be subject to applicable precautionary procedures of Paragraph 6.7.1.a.1 above.

3/ For modes of intake other than inhalation, such intakes must be controlled, evaluated, and accounted for by techniques and procedures as may be appropriate to the circumstances of the occurrence with proper consideration of critical organs and limiting doses.

(b) Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent values specified in Appendix B, Table I, Column 1 of 10 CFR Part 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be initially estimated by dividing the ambient airborne concentration by the protection factor specified in Table 6.7-1 attached hereto for the respiratory protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been greater than that initially estimated, the greater quantity shall be used in evaluating exposures; if it is less than that initially estimated, the lesser quantity may be used in evaluating exposures.

(c) The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

(d) The licensee maintains a respiratory protective program adequate to assure that the requirements of paragraphs 1 and 2 above are met.

Such a program shall include:

(1) Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

(2) Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

(3) Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.

(4) Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

(5) Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

(6) Bioassays and/or whole body counts of individuals, and other surveys, as appropriate, to evaluate individual exposures and to assess protection actually provided.

(7) Records sufficient to permit periodic evaluation of the adequacy of the respiratory protective program.

(e) The licensee uses equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.7-1 below. Equipment not approved under U. S. Bureau of Mines Approval Schedules may be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type, as specified in Table 6.7-1 below.

(f) Unless otherwise authorized by the Commission, the licensee does not assign protection factors in excess of those specified in Table 6.7-1 below in selecting and using respiratory protective equipment.

(g) These specifications with respect to the provisions of 20.103 shall be superseded by adoption of proposed changes to 10 CFR 20, Section 20.103, which would make this specification unnecessary.

6.7.2 Exposure of individuals to concentrations of radioactive noble gases may be controlled in accordance with the dose limits and requirements of Section 20.101, instead of 20.103 consistent with requirements of Specification 6.7.1a.2.a and footnote 2 referenced therein.

TABLE 6.7-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES 1/	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE 3/	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 7/	NP	5	21B 30 CFR 14.4(b)(4)
Facepiece, full 7/	NP	100	21B 30 CFR 14.4(b)(5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR 12.2(c)(2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR 12.2(c)(2) Type C(i)
Facepiece, full 7/	D	100	19B 30 CFR 12.2(c)(2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR 12.2(c)(2) Type C(iii)
Hood	CF	5/ See note	6/
Suit	CF	5/ See note	6/
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full 7/	D	100	13E 30 CFR 11.4(b)(2)(i)
Facepiece, full	PD	1,000	13E 30 CFR 11.4(b)(2)(ii)
Facepiece, full	R	1,000	13E 30 CFR 11.4(b)(1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	19 B CFR 12.2(e) or applicable schedules as listed above

1/, 2/, 3/, 4/, 5/, 6/, 7/, [These notes are on the following pages]

1/ See the following symbols:

CF: continuous flow
D : demand
NP: negative pressure (i.e., negative phase during inhalation)
PD: pressure demand (i.e., always positive pressure)
PP: positive pressure
R : recirculating (closed circuit)

2/ (a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency [above 99.9% removal efficiency by U. S. Bureau of Mines type dioctyl phthalate (DOP) test] particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

3/ Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote 5/ below, concerning supplied-air suits and hoods.

4/ Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR, Part 20.

5/ Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

6/ No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

7/ Only for clean-shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U. S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U. S. Bureau of Mines in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10 CFR Part 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.