Duke Power Company

Oconee Nuclear Station

Attachment 1

Proposed Techncial Specification Revision

Oconee 1 Cycle 10

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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 2.1-1. If the actual pressure/temperature point is below 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 2.1-2. If the actual reactor-thermal-power/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 and anticipated transients. This is accomplished by operating within the nucleate boiling heat transfer regime where the heat transfer coefficient is large and the cladding temperature is only slightly greater than the coolant temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation, but neutron power and reactor coolant pressure and temperature can be related to DNB using a critical heat flux (CHF) correlation. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB.

The BAW-2 and BWC CHF correlations[1,2] have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The curve presented in Figure 2.1-1[3] represents the conditions at which the minimum allowable DNBR is predicted to occur for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based upon the design nuclear peaking factors which include the potential effects of fuel densification[4]:

$$F_{\Delta H}^{N} = 1.71$$

$$F_{Z}^{N} = 1.50$$

Since power peaking is not a directly measurable quantity, DNBR limited power peaks and fuel melt limited power peaks are separately correlated to measurable reactor power and power imbalance. The reactor power imbalance limits, Figure 2.1-2[5], define the values of reactor power as a function of axial imbalance that correspond to the more restrictive of two thermal limits - MDNBR equal to the DNBR limit or the linear heat rate equal to the centerline fuel melt limit.

The core protection safety limits are based on an RCS flow of 374,880 gpm (4 pump operation). Three and two pump operation are analyzed assuming 74.7 percent and 49.0 percent of four pump flow, respectively. The maximum thermal power for three pump operation is 88.07 percent (Figure 2.1-2) due to a power level trip produced by the flux/flow ratio (74.7 percent flow x 1.07 = 79.92 percent power = 88.07 percent power adding the maximum calibration and instrument error). The maximum thermal power for 2 pump operation, 60.63 percent, is produced in a similar manner.

REFERENCES

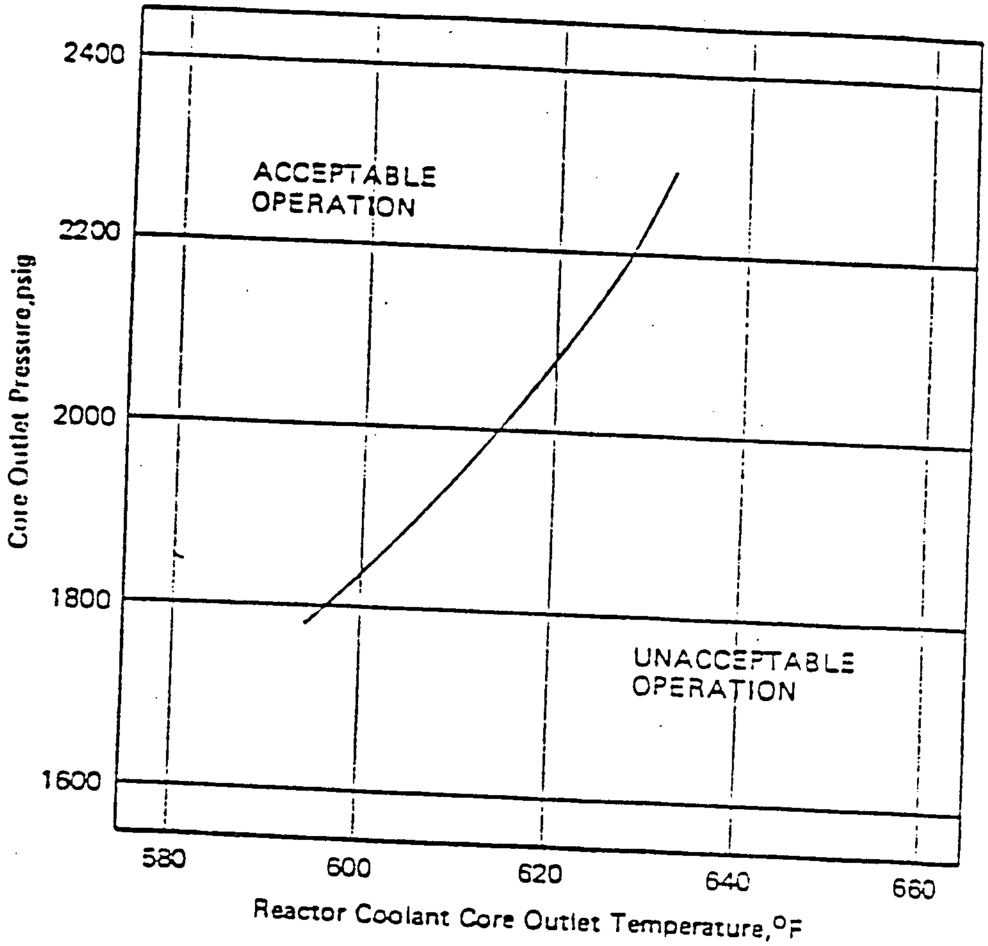
(1) Correlation of Critical Heat Flux in a Bundle cooled by Pressurized Water, BAW-10000, March 1970.

(2) Correlation of 15 x 15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143P, Part 2, August 1981.

(3) Oconee Unit 3, Cycle 7 - Reload Report, DPC-RD-2001, Rev. 1, Duke Power Company, July 1982.

(4) Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002, Duke Power Company, March 1985.

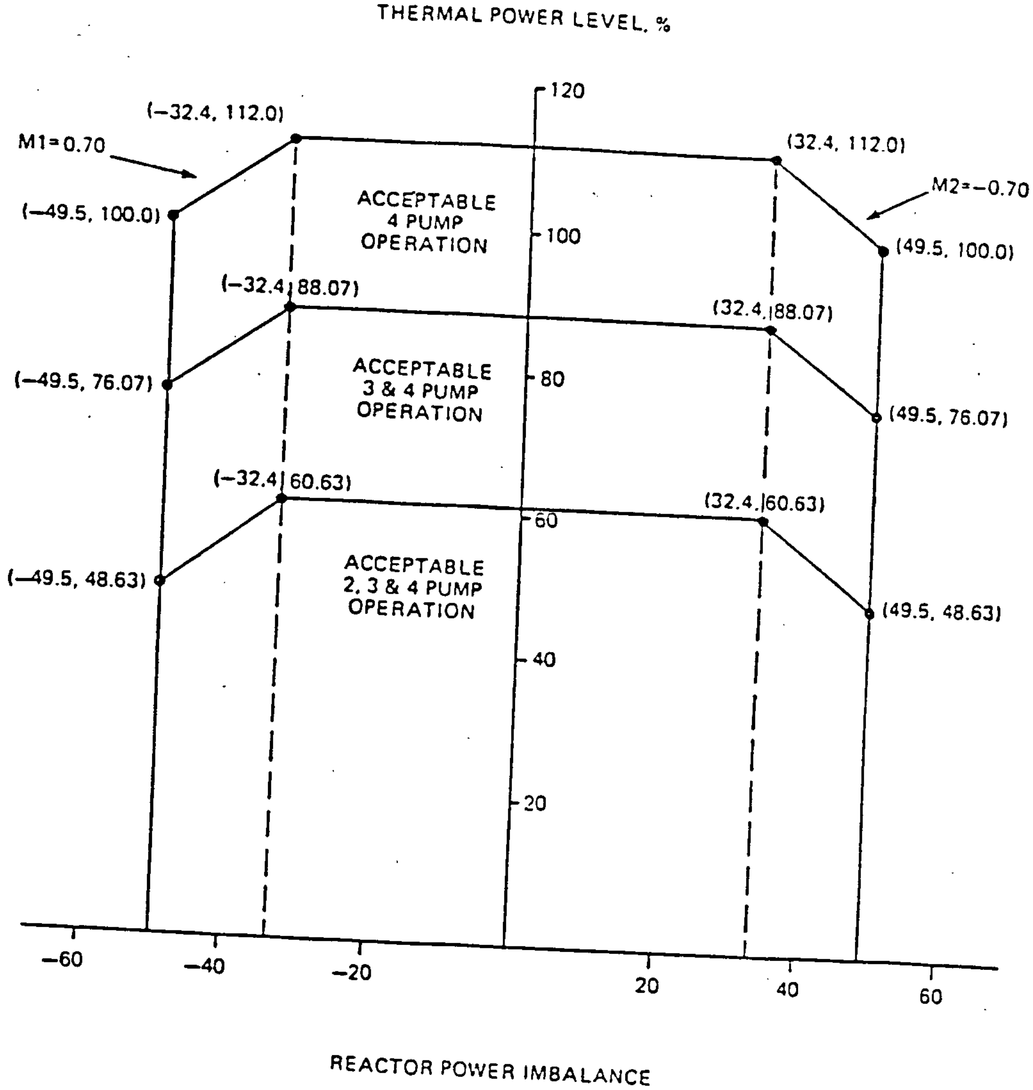
(5) Oconee Unit 2, Cycle 7 - Reload Report, DPC-RD-2002, Duke Power Company, September 1983.



CORE PROTECTION SAFETY LIMITS UNITS 1, 2, AND 3

Figure 2.1-1

OCONEE NUCLEAR STATION



CORE PROTECTION SAFETY LIMITS UNITS 1, 2, AND 3

Figure 2.1-2

OCONEE NUCLEAR STATION

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 setpoints and the permissible bypasses for the instrument channels shall be as stated in Table 2.3.1 and Figure 2.3-2.

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.

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

Bases

The reactor trip setpoints for reactor protective system (RPS) instrumentation are given in Table 2.3-1. The trip setpoints have been selected to ensure that the core and reactor coolant system are prevented from exceeding their safety limits. The various reactor trip circuits automatically open the reactor trip breakers whenever a parameter monitored by the RPS deviates from an allowed range. The RPS consists of four instrument channels for redundancy. The plant safety analyses are based on the trip setpoints given in Table 2.3-1 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, a reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in the trip setpoint due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis. (1)

Overpower Trip Based on Flow and Imbalance

Following the loss of one or more reactor coolant pumps, the core is prevented from violating the minimum DNBR criterion by a reactor trip initiated by exceeding the allowable reactor power to reactor coolant flow (flux/flow) ratio setpoint. Loss of one or more reactor coolant pumps is also detected by the pump monitors. The power level trip produced by the flux/flow ratio provides DNB protection for all modes of pump operation.

The power level trip setpoint produced by the flux/flow ratio provides both high power level and low flow protection. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible flow rate. Typical power level and flow rate combinations for different pump situations are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.46% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.93% and reactor flow rate is 74.7% or flow rate is 70.09% and power level is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.43% and reactor flow rate is 49.0% or flow rate is 45.79% and the power level is 49%.

   The analysis to determine the flux/flow setpoint accounts for calibration and instrument errors and the variation in RC flow in such a manner as to ensure a conservative setpoint. A Monte-Carlo simulation technique is used to determine the combined effects of calibration and instrument uncertainties with the final string uncertainties used in the analysis corresponding to the 95/95 tolerance limits.

The reactor power imbalance (power in the top half of the core minus the power in the bottom half) reduces the power level trip produced by the flux/flow ratio as shown in Figure 2.3-2. The flux/flow ratio reduces the power level trip and associated power-imbalance boundaries by 1.07% for a 1% flow reduction. The power-imbalance boundaries shown in Figure 2-3.2 are established to prevent fuel thermal limits; DNBR and centerline fuel melt limits, from being exceeded.

Pump Monitors

The pump monitors trip the reactor due to the loss of reactor coolant pump(s) to ensure the DNBR remains above the minimum allowable DNBR. The pump monitors provide redundant trip protection for DNB; tripping the reactor on a signal diverse from that of the flux/flow trip. The pump monitors also restrict the power level depending on the number of operating reactor coolant pumps.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the reactor coolant system (RCS) high pressure setpoint is reached before the nuclear overpower trip setpoint. The high RCS pressure trip setpoint (2300 psig) ensures that the pressure remains below the safety limit (2750 psig) for any design transient.(2) The low pressure (1800 psig) and variable low pressure ($11.14\ T_{out} - 4706$) trip setpoints shown in Figure 2.3-1 ensure that the minimum DNBR is greater than or equal to the minimum allowable DNBR for those accidents that result in a reduction in pressure.(3,4) The limits shown in Figure 2.3-1 bound the pressure-temperature curves calculated for 4, 3, and 2 pump operation.

Accounting for calibration and instrumentation errors, the safety analyses used a variable low RCS pressure trip setpoint of ($11.14\ T_{out} - 4746$).

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setpoint (618°F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures. Accounting for calibration and instrumentation errors, the safety analyses used a trip setpoint of 620°F.

Reactor Building Pressure

The high reactor building pressure trip setpoint (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 startup the reactor and to be able to perform control rod drive tests and zero power physics tests (see Technical Specification 3.1.9), there is provision for bypassing certain segments of the reactor protective system (RPS). The RPS segments which can be bypassed are given in Table 2.3-1. Two conditions are imposed when the RPS is bypassed:

1. By administrative control the Nuclear overpower trip setpoint is reduced to a value of $\leq$ 5.0% of rated power.

2. The high reactor coolant system pressure trip setpoint is automatically lowered to 1720 psig.

The high RCS pressure trip setpoint is reduced to prevent normal operation with part of the RPS bypassed. The reactor must be tripped before the bypass is initiated since the high pressure trip setpoint is lower than the normal low pressure trip setpoint (1800 psig).

The overpower trip setpoint is $\leq$ 5.0% prevents any significant reactor power from being produced when performing physics tests. If no reactor coolant pumps are operating, sufficient natural circulation would be available to remove 5.0% of rated power.(5)

Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped and is subject to the limitations set forth in Specification 3.1.8. The RPS trip setpoints and permissible instrument channel bypasses will be confirmed prior to single loop operation.

REFERENCES

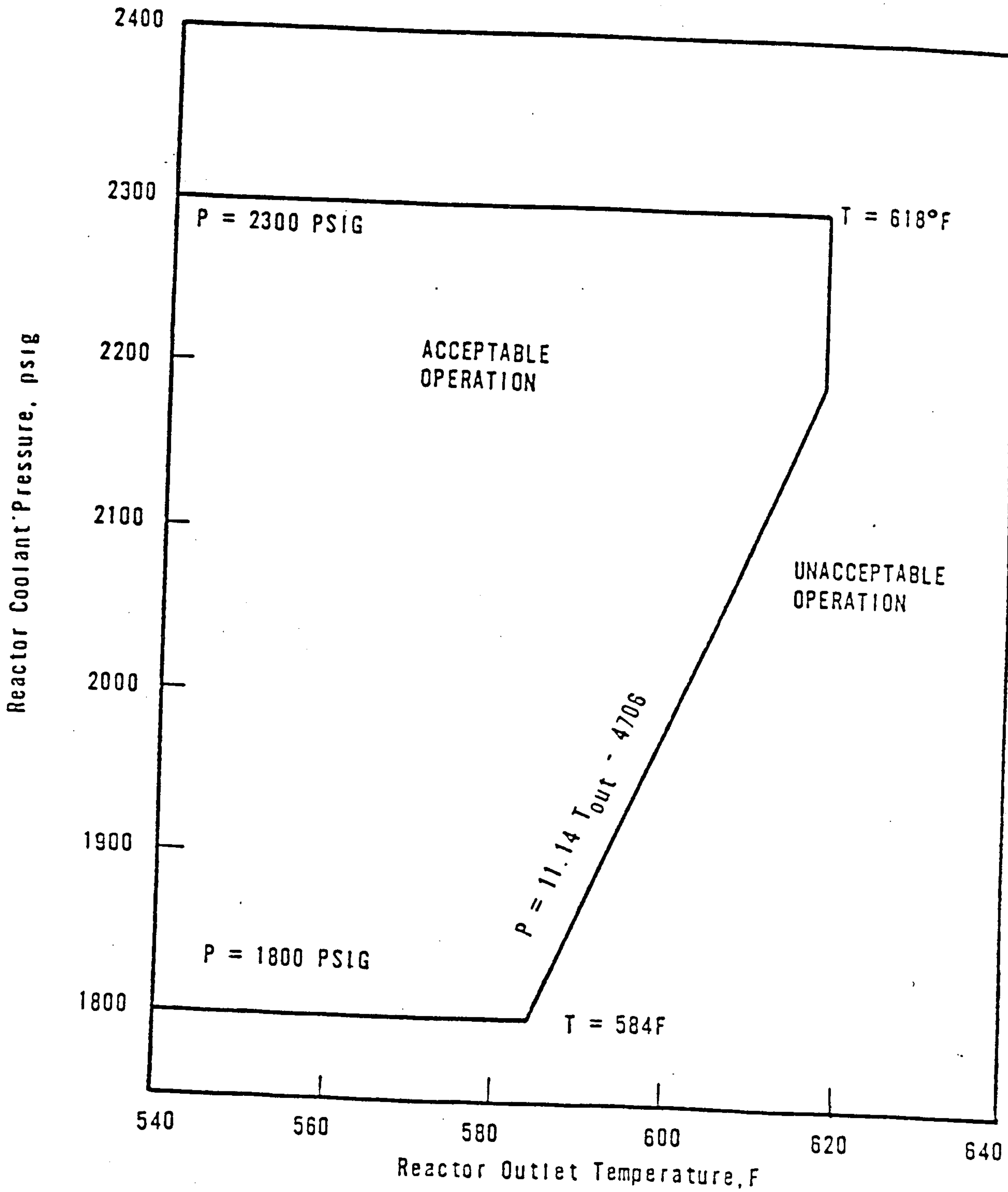
(1) FSAR, Section 15.3

(2) FSAR, Section 15.2

(3) FSAR, Section 15.7

(4) FSAR, Section 15.8

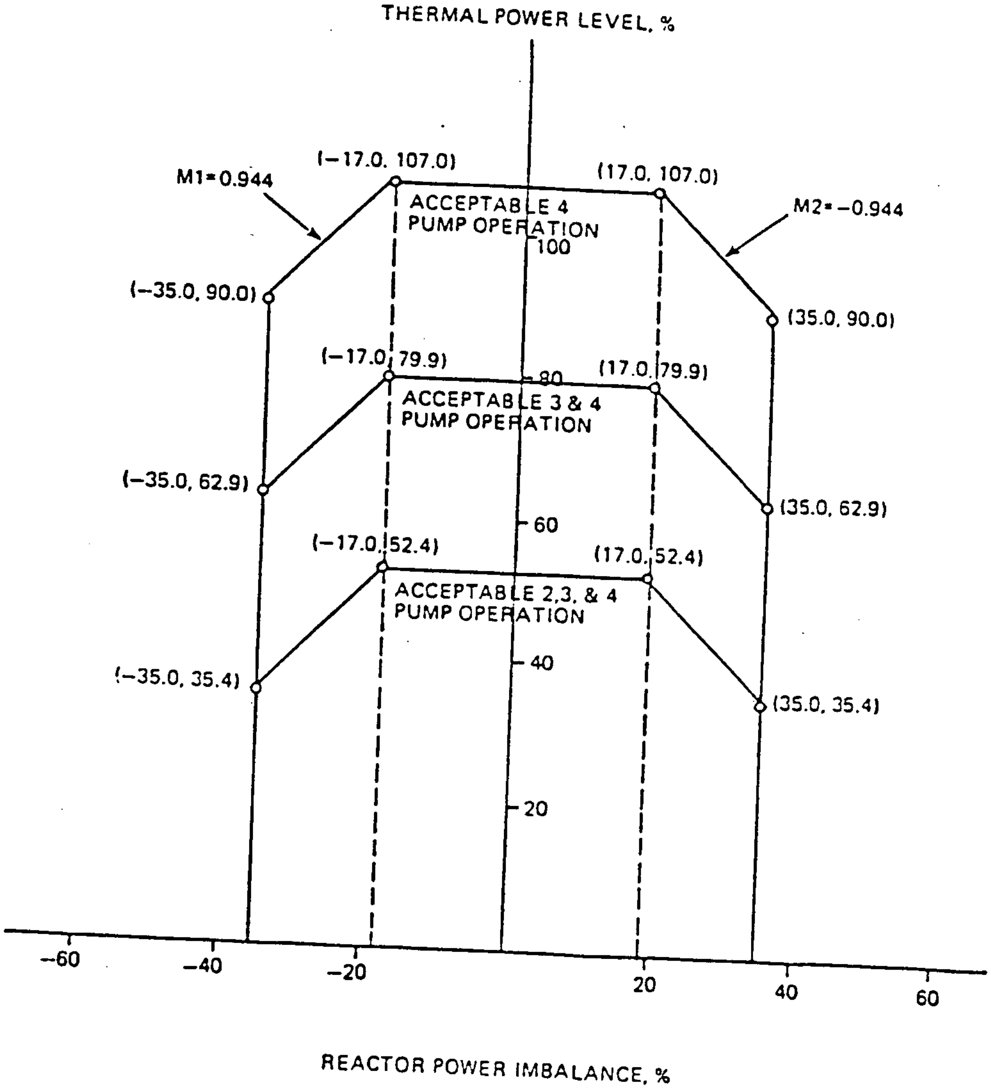
(5) FSAR, Section 15.6



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS UNITS 1, 2, AND 3

Figure 2.3-1

OCONEE NUCLEAR STATION



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS UNITS 1, 2, AND 3

Figure 2.3-2

OCONEE NUCLEAR STATION

TABLE 2.3-1

Reactor Protective System Trip Setting Limits

	RPS Trip		RPS Trip Setpoint	Shutdown Bypass
1.	Nuclear Overpower		105.5% Rated Power	5.0% Rated Power(1)
2.	Flux/Flow/Imbalance		1.07	Bypassed
3.	Pump Monitors	a.	> 0% Rated Power loss of two pumps in one reactor coolant loop	Bypassed
		b.	> 55% Rated Power loss of two pumps	
		c.	> 0% Rated Power loss of one or two pumps during two pump operation	
4.	High Reactor Coolant System Pressure		2300 psig	1720(2)
5.	Low Reactor Coolant System Pressure		1800 psig	Bypassed
6.	Variable Low Reactor Coolant System Pressure		P (psig)(3) = (11.14 T_{out} - 4706)	Bypassed
7.	High Reactor Coolant Temperature		618°F	618°F
8.	High Reactor Building Pressure		4 psig	4 psig

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

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

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

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 the criticality limit of
3.1.2-1A (Unit 1)
3.1.2-1B (Unit 2)
3.1.2-1C (Unit 3)

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. The regulating rods shall then be positioned within their position limits defined by the Core Operational Limits Report of Specification 6.6.1.1 prior to deboration.

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 more 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 the limits of Specification 3.1.2.1 provides increased assurance that the proper rela-

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 setpoint and reactor coolant outlet temperature trip setpoint) can be conservatively adjusted taking into account instrument errors.

Limiting the pump monitor trip setpoint to 50 percent of rated power and the reactor coolant outlet temperature trip setpoint to 610°F to perform this confirmatory testing assures operation well within the core protective safety limits shown in Figure 2.1-1.

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

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 2.3-1.

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

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.

3.1.9.3 Shutdown margin may not be reduced below 1.0%Δk/k as required by Specification 3.5.2.1 with the exception that the stuck rod worth criterion does not apply during rod worth measurements.

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.

3.1.11 Shutdown Margin

Specification

The available shutdown margin during all system conditions except refueling shall be greater than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

Bases

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

During power operation and startup the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits provided in the Core Operational Limits Report of Specification 6.6.1.1.

During refueling conditions equivalent protection is provided in the requirements of Specification 3.8.4.

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 three channels each of the following are operable: reactor coolant temperature, reactor coolant pressure, pressure-temperature, flux-imbalance flow, power-number of pumps, and high reactor building pressure. The engineered safety features actuation system must have three analog channels and two digital channels functioning correctly prior to a startup. Additional operability requirements are provided by Technical Specifications 3.1.12 and 3.4 for equipment which are not part of the RPS or ESFAS.

Operation at rated power is permitted as long as the systems have at least the minimum number of operable channels given in Column C (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. The minimum number or operable channels required is three. While a bypassed channel is considered inoperable, a channel placed in the tripped condition is considered operable. Thus, only one channel may be placed in bypass at any one time in order to maintain the minimum number of required channels. This results in a trip logic of two out of three. It should be noted that, for a limited period of time, an effective trip logic of one out of two can be achieved by placing one channel in bypass and one channel in the tripped condition.

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

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

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

Power is normally supplied to the control rod drive mechanisms from two separate parallel 600 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
1. Nuclear Instrumentation Intermediate Range Channels	2	NA	1	Bring to hot shutdown within 12 hours (b)
2. Nuclear Instrumentation Source Range Channels	2	NA	1	Bring to hot shutdown within 12 hours (b) (c)
3. RPS Manual Pushbutton	1	1	1	Bring to hot shutdown within 12 hours
4. RPS Power Range Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
5. RPS Reactor Coolant Temperature Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
6. RPS Pressure-Temperature Instruments Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
7. RPS Flux Imbalance Flow Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
8. RPS Reactor Coolant Pressure				
a. High Reactor Coolant Pressure Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
b. Low Reactor Coolant Pressure Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
9. RPS Power-Number of Pumps Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours (h)

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

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
10. RPS High Reactor Building Pressure Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
11. RPS Anticipatory Reactor Trip System				
a. Loss of Turbine	4	2	3(a)	Bring to hot shutdown within 12 hours
b. Loss of Main Feedwater	4	2	3(a)	Bring to hot shutdown within 12 hours
12. ESF High Pressure Injection System and Reactor Building Isolation (Non-essential Systems)				
a. Analog Reactor Coolant Pressure Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
c. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
d. Digital Logic Channels (1 and 2)	2	1	2	Bring to hot shutdown within 24 hours (e)

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic 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) (Deleted)

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

(f) (Deleted)

(g) (Deleted)

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state.

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 Shutdown Margin

a. The available shutdown margin shall be greater than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

b. If the shutdown margin is less than 1% $\Delta k.k$, then within 1 hour initiate and continue boration until the required shutdown margin is restored. The requirements of the Core Operational Limits Report of Specification 6.6.1.1 shall be met.

3.5.2.2 Movable Control Assemblies

a. All control (safety and regulating) rods shall be operable and positioned within nine (9) inches of their group average height.

b. A control rod shall be declared inoperable if any of the following conditions exist for that rod:

   1. The control rod cannot be moved due to excessive friction or mechanical interference, or cannot perform its intended trip function.

   2. The control rod cannot be located by either absolute or relative position indication or by in or out limit lights.

   3. The control rod is misaligned with its group average by more than nine (9) inches.

   4. The control rod does not meet the exercise requirements of Specification 4.1.

   5. The control rod does not meet the rod trip insertion times of Specification 4.7.1.

   6. The control rod does not meet the rod program verification of Specification 4.7.2.

c. If a control rod is declared inoperable by being immovable due to excessive friction or mechanical interference or known to be untrippable then:

1. Within 1 hour verify that the shutdown margin requirement of Specification 3.5.2.1 is satisfied and,

2. Within 12 hours place the reactor in the hot standby condition.

d. If a control rod is declared inoperable due to causes other than addressed in 3.5.2.2.c above then:

1. Within 1 hour either restore the rod to operable status or,

2. Continue power operation with the control rod declared inoperable and

a. Within 1 hour verify the shutdown margin requirement of Specification 3.5.2.1 with an additional allowance for the withdrawn worth of the inoperable rod and,

b. Either reactor thermal power shall be reduced to less than 60% of the allowable power for the reactor coolant pump combination within 1 hour and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow/imbalance, shall be reduced within the next 4 hours to 65.5% of thermal power value allowable for the reactor coolant pump combination or,

c. Position the remaining rods in the affected group such that the inoperable rod is maintained within allowable group average limits of Specification 3.5.2.2.a and the withdrawal limits provided in the Core Operational Limits Report of Specification 6.6.1.1.

e. If more than one control rod is inoperable or misaligned, the reactor shall be shut down to the hot standby condition within 12 hours.

3.5.2.3 The worths of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the control rod position limits defined in the Core Operational Limits Report of Specification 6.6.1.1.

3.5.2.4 Quadrant Power Tilt

a. Except for physics tests, the maximum positive quadrant power tilt shall not exceed the Steady State Limit of Table 3.5-1 during power operation above 15% full power.

b. If the maximum positive quadrant power tilt exceeds the Steady State Limit but is less than or equal to the Transient Limit of Table 3.5-1, then:

1. Either the quadrant power tilt shall be reduced within 2 hours to within its Steady State Limit or

2. The reactor thermal power shall be reduced below the power level cutoff (as specified in the Core Operational Limits Report of Specification 6.6.1.1) and further reduced 2% thermal power for each 1% of quadrant power tilt in excess of the Steady State Limit, and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within 4 hours by 2% thermal power for each 1% tilt in excess of the Steady State Limit. If less than four reactor coolant pumps are in operation, the allowable thermal power for the reactor coolant pump combination shall be reduced by 2% for each 1% excess tilt.

c. Quadrant power tilt shall be reduced within 24 hours to within its Steady State Limit or,

1. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

d. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit of Table 3.5-1 and if there is a simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 30 minutes at least 2% for each 1% of the quadrant power tilt in excess of the Steady State Limit.

2. Either quadrant power tilt shall be reduced within 2 hours to within its Transient Limit or,

3. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

e. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit of Table 3.5-1, due to causes other than simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be

reduced within the next 2 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

f. If the maximum positive quadrant power tilt exceeds the Maximum Limit of Table 3.5-1, the reactor shall be shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.

g. Quadrant power tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% full power.

3.5.2.5 Control Rod Positions

a. Technical Specification 3.1.3.5 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. Except for physics tests, operating rod group overlap shall be 25% ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours or the reactor shall be placed in a hot shutdown condition within an additional 12 hours.

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are provided in the Core Operational Limits Report of Specification 6.6.1.1 pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are provided in the Core Operational Limits Report of Specification 6.6.1.1.

If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times.

3.5.2.6 Xenon Reactivity

Except for physics tests, reactor power shall not be increased above the power level-cutoff provided in the Core Operational Limits Report of Specification 6.6.1.1 unless one of the following conditions is satisfied:

1. Xenon reactivity did not deviate more than 10 percent from the equilibrium value for operation at steady state power.

2. Xenon reactivity deviated more than 10 percent but is now within 10 percent of the equilibrium value for operation at steady state rated power and has passed its final maximum or minimum peak during its approach to its equilibrium value for operation at the power level cutoff.

3. Except for xenon free startup (when 2. applies), the reactor has operated within a range of 87 to 92 percent of rated thermal power for a period exceeding 2 hours.

3.5.2.7 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined in the Core Operational Limits Report of Specification 6.6.1.1. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.8 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.

3.5.2.9 The Core Operational Limits Report of Specification 6.6.1.1 is valid for a nominal design cycle length, as defined for the appropriate unit and cycle. Operation beyond the nominal design cycle length is permitted provided that an evaluation is performed to verify that the operational limit curves are valid for extended operation. If the operational limit curves are not valid for the extended period of the operation, appropriate limits will be established and the Core Operational Limits Report will be revised per Specification 6.6.1.1.

Bases

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

The power-imbalance envelope defined in the Core Operational Limits Report of Specification 6.6.1.1 is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbably because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification power spike factors
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

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

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

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

The rod position limits provided in the Core Operational Limits Report of Specification 6.6.1.1 are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position(1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2,3,4, 5) of hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ 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 Group 7 to be partially inserted.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding
7.50% for Unit 1. The limits shown in Specification 3.5.2.4
7.50% for Unit 2
7.50% for Unit 3
are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.7, 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.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Technical Specification 3.5.2.6 provides the ability to prevent excessive power peaking by transient xenon at rated power.

Operating restrictions resulting from xenon transients and power maneuvers are inherently included in the limits provided in the Core Operational Limits Report of Specification 6.6.1.1.

Figure 3.5.2-16
(Deleted)

[Note that the information provided by this figure is provided in the Core Operational Limits Report.]

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	≦30 psig
	High-Pressure Injection	≦4 psig
	Low-Pressure Injection	≦4 psig
	Start Reactor Building Cooling & Reactor Building Isolation (Essential and Non-essential Systems)	≦4 psig
	Penetration Room Ventilation	≦4 psig
Lower Reactor Coolant System Pressure	High Pressure Injection(1) & Reactor Building Isolation (Non-essential systems)	≧1500 psig
	Low Pressure Injection(2)	≧500 psig

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

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

Bases

High Reactor Building Pressure

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

Low Reactor Coolant System Pressure

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

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)

REFERENCE

(1) FSAR, Section 15.14.

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

3.5.4.1 At or above 80 percent of the power allowable for the existing reactor coolant pump operating combination, incore detectors shall be operable as necessary to meet the following:

a. For axial imbalance measurements:

At least three detectors in each of at least three strings shall lie in the same axial plane, with one plane in each axial core half. The axial planes in each core half shall be symmetrical about the core mid-plane. The detector strings shall not have radial symmetry.

b. For quadrant power tilt measurements:

At least two sets of at least four detectors shall lie in each axial core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane. Detectors in the same plane shall have quarter core radial symmetry.

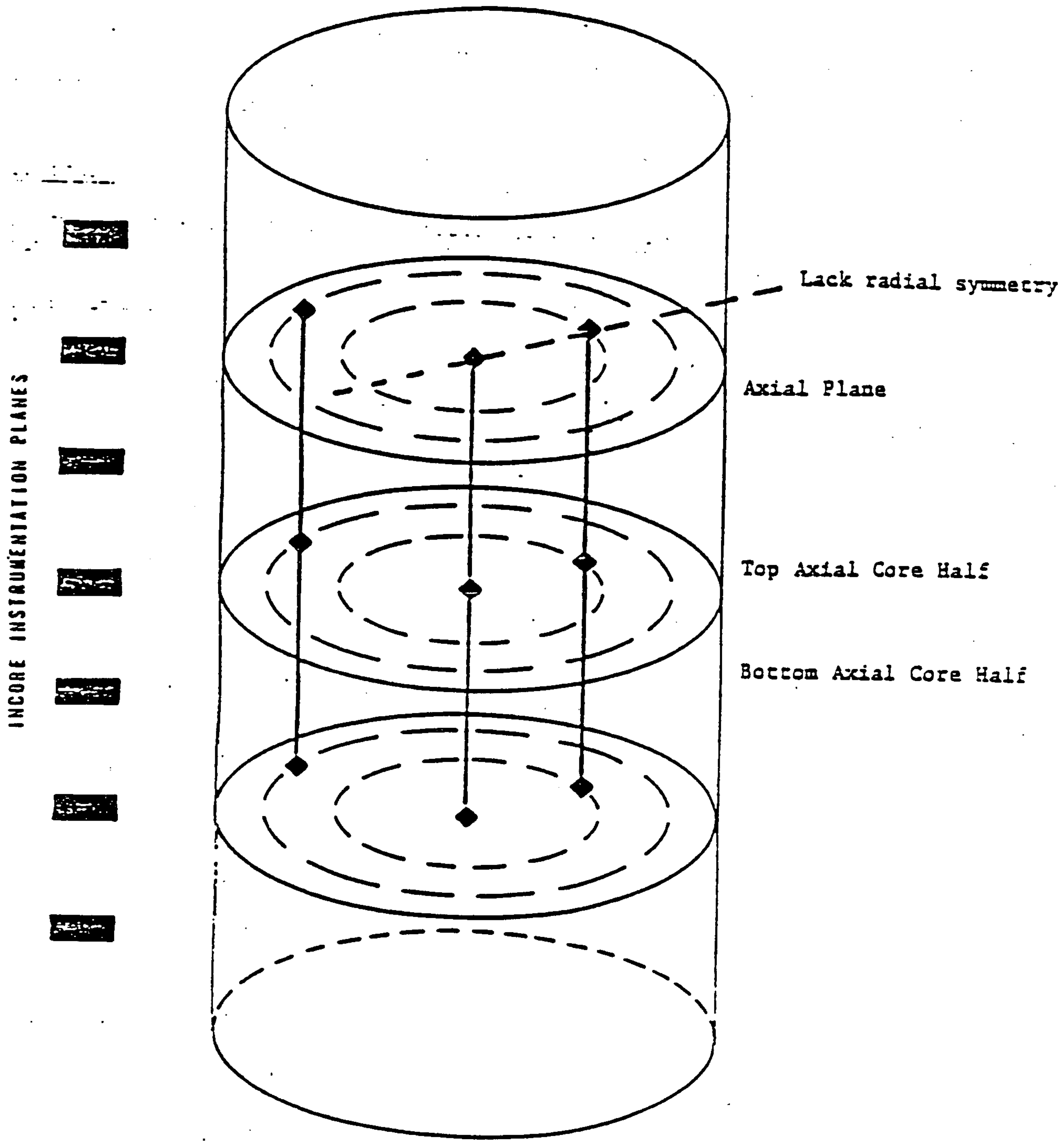
3.5.4.2 If requirements of 3.5.4.1 are not met, power shall be reduced below 80 percent of the power allowable for the existing reactor coolant pump combination within eight hours and incore detector measurements shall not be used to determine axial imbalance or quadrant power tilt.

Bases

The operability of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Figures 3.5.4-1, 3.5.4-2, and 3.5.4-3 for satisfactory incore detector arrangements.

The safety of reactor operation at or below 80 percent of the power allowable for the reactor coolant pump combination[1] without the axial imbalance trip system has been determined by extensive 3-D calculations, and was verified during the physics startup testing program.

(1) FSAR, Section 5.1.2.3

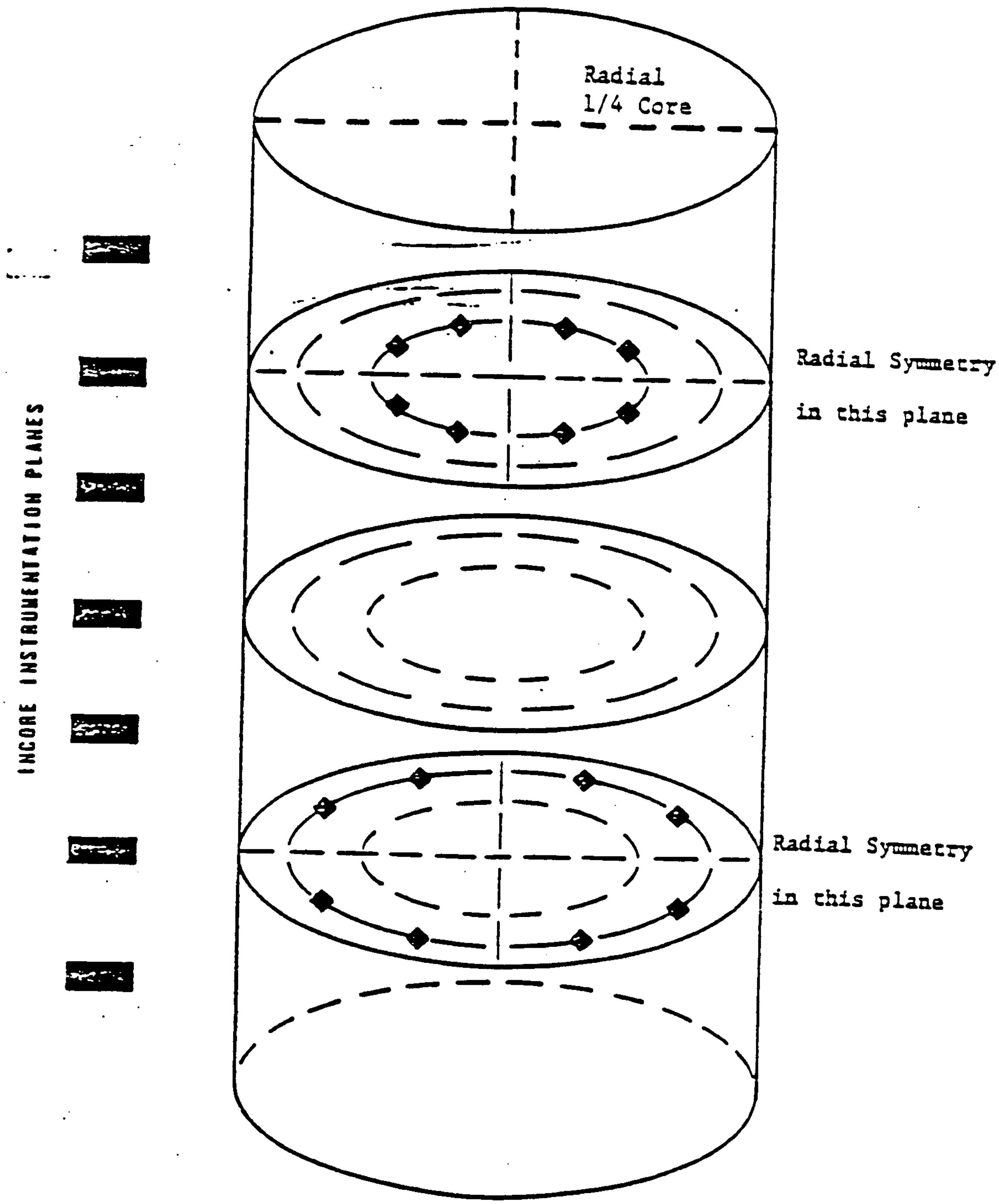


INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION

INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION

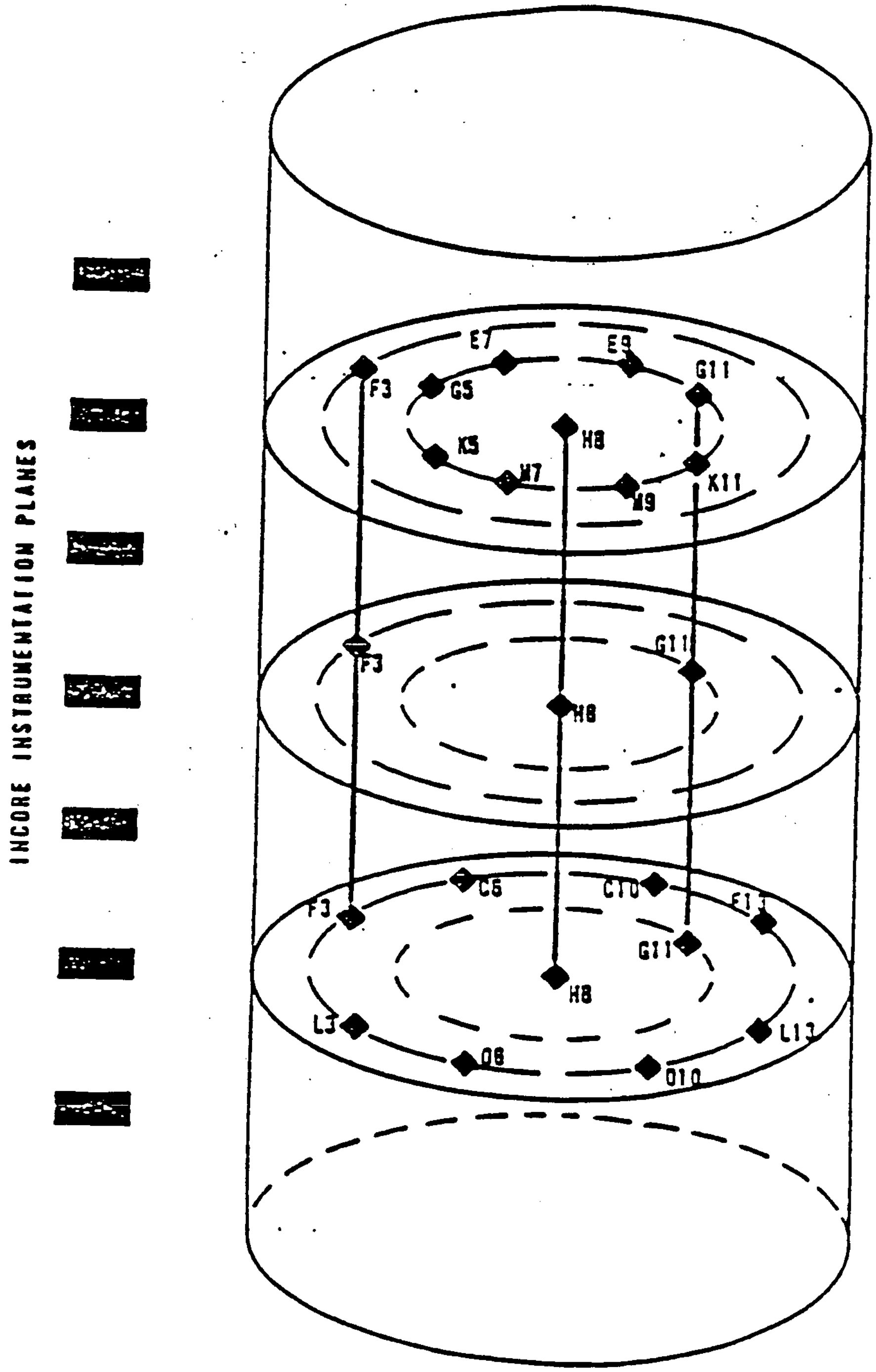
OCONEE NUCLEAR STATION

Figure 3.5.4-1



INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX TILT INDICATION

INCORE INSTRUMENTATION
SPECIFICATION
RADIAL FLUX TILT INDICATION
OCONEE NUCLEAR STATION
Figure 3.5.4-2



INCORE INSTRUMENTATION SPECIFICATION

INCORE INSTRUMENTATION SPECIFICATION

OCONEE NUCLEAR STATION

Figure 3.5.4-3

3.5.5 Radioactive Effluent Monitoring Instrumentation

Applicability

Applies to radioactive liquid effluent, gaseous effluent, and gaseous process monitoring instrumentation.

Specifications

3.5.5.1 Liquid Effluents

a. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.5.5-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.1 are not exceeded.

b. If a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

c. In the event that the number of operable radioactive liquid effluent monitoring instrumentation channels falls below the limit given under Table 3.5.5-1, Column A, action shall be as shown in Column B. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.5.2 Gaseous Process and Effluents

a. The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.5.5-2 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.10.1 are not exceeded.

b. If a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

c. In the event that the number of radioactive gaseous process or effluent monitoring instrumentation channels falls below the limit given under Table 3.5.5-2, Column A, action shall be taken as shown in Column B. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.5.3 Setpoints

The setpoints shall be determined in accordance with the methodology described in the ODCM and shall be recorded. Setpoint correction may be permitted without declaring the channel inoperable.

3.5.5.4 The provisions of Technical Specification 3.0 do not apply.

Bases

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentration of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

Table 3.5.5-1
LIQUID EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
1. Monitors Providing Automatic Termination of Release			
Liquid Radwaste Effluent Line Monitors			
1 RIA-33	1	*	(a)
Turbine Building Sump			
1 RIA-54 (Units 1 & 2)	1	*	(b)
3 RIA-54 (Unit 3)	1	*	(b)
2. Monitors not Providing Automatic Termination of Release			
Low Pressure Service Water			
1 RIA-35	1	*	(d)
2 RIA-35	1	*	(d)
3 RIA-35	1	*	(d)
3. Flow Rate Measuring Devices			
Liquid Radwaste Effluent Line	1	*	(c)
Keowee Hydroelectric Station Tailrace Discharge **	NA	NA	NA
4. Continuous Composite Sampler			
#3 Chemical Treatment Pond Composite Sampler and Sampler Flow Monitor (Turbine Building Sumps Effluent)	1	*	(d)

*At all times.
**Flow determined from number of hydro units operating; if hydro is not operating, leakage flow, which is measured periodically, is used.

Table 3.5.5-1 NOTES

(a) Effluent releases may continue provided that prior to initiating a release:

1. Two independent samples are analyzed in accordance with Specification 3.9 and;

2. Two independent data entry checks for release rate calculations and valve lineups of the effluent pathway are conducted.

Otherwise, suspend release of radioactive effluents by this pathway.

(b) Effluent releases may continue provided that prior to each discrete release of the sump, grab samples are collected and analyzed for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml.

(c) Effluent releases may continue provided flow rate is estimated at least once per four hours during actual releases.

(d) Effluent releases may continue provided that grab samples are collected and analyzed for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml every 12 hours.

Table 3.5.5-2
GASEOUS PROCESS AND EFFLUENT
MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
1. Waste Gas Holdup Tanks			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination Of Release (RIA-37, - 38)	1	**	(a)
b. Effluent Flow Rate Monitor (Waste Gas Discharge Flow)	1	**	(b)
2. Unit Vent Monitoring System			
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Containment Purge Release (RIA - 45)	1	*	(a)
b. Iodine Sampler	1	*	(d)
c. Particulate Sampler	1	*	(d)
d. Effluent Flow Rate Monitor (Unit Vent Flow)	1	*	(b)
e. Sampler Flow Rate Monitor	1	*	(e)
f. Effluent Flow Rate Monitor (Containment Purge)	1	**	(b)
3. Interim Radwaste Building Ventilation Monitoring System			
a. Noble Gas Activity Monitor (RIA - 53)	1	*	(c)
b. Iodine Sampler#	1	*	(d)
c. Particulate Sampler#	1	*	(d)

Table 3.5.5-2 (Cont'd)
GASEOUS PROCESS AND EFFLUENT
MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
d. Effluent Flow Rate Monitor (Interim Radwaste Exhaust)#	1	*	(b)
e. Sampler Flow Rate Monitor#	1	*	(e)
4. Hot Machine Shop Ventilation Monitoring System			
a. Iodine Sampler#	1	*	(d)
b. Particulate Sampler#	1	*	(d)
c. Effluent Flow Rate Monitor (Hot Machine Shop Exhaust)#	1	*	(b)
d. Sampler Flow Rate Monitor#	1	*	(e)

* At all times.
** During waste gas holdup tank releases and/or containment purge operation.
\# Effective upon installation of equipment.

Table 3.5.5-2 NOTES

(a) Effluent releases from waste gas tanks or containment purges may continue provided that prior to initiating a release:

1. Two independent samples are analyzed and;

2. Two independent data entry checks for release rate calculations and valve lineups of the effluent pathway are conducted and;

Effluent release from ventilation system or condenser air ejectors may continue provided that grab samples are taken once per 8 hours and these samples are analyzed for gross activity (beta and/or gamma) within 24 hours, or continuously monitor through the unit vent. Otherwise, suspend release of radioactive effluents via this pathway.

(b) Effluent releases may continue provided the flow rate is estimated at least once per 4 hours.

(c) Effluent releases may continue provided grab samples are taken once per 8 hours and these samples are analyzed for gross activity (beta and/or gamma) within 24 hours.

(d) Effluent releases may continue provided samples are continuously collected with auxiliary sampling equipment for periods not to exceed 7 days and analyzed within 48 hours of the end of sample collection.

(e) Alarms indicating low flow may be substituted for flow measuring devices.

6.6 STATION REPORTING REQUIRMENTS

6.6.1 Routine Reports

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator Region II unless otherwise noted.

6.6.1.1 Core Operational Limits Report

A report providing the following core operational limits shall be provided to the Regional Administrator of the Regional Office of the NRC with a copy to the Director, NRR, Attention: Chief, Core Performance Branch, USNRC at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter:

1. Regulating Control Rod Insertion/Withdrawal Limits
2. Axial Power Shaping Control Rod Insertion/Withdrawal Limits
3. Power Level Cutoff
4. Power/Power Imbalance Operational Limits

In addition, in the event that the limits should change requiring a new submittal or amended submittal of the Core Operational Limits Report, it shall be submitted at least 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support the Core Operational Limits Report will be requested from the NRC and need not be included in the report.

6.6.1.2 Startup Report

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

6.6.1.3 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C., 20555, with a copy to the appropriate Regional Office, to be submitted by the fifteenth of each month following the calendar month covered by the report.

6.6.1.4 Personnel Exposure and Monitoring Report

Prior to March 1 of each year, a tabulation shall be submitted to the NRC of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total body dose received from external sources shall be assigned to specific major work functions.

6.6.1.5 Radioactive Effluent Release Report

Routine Radioactive Effluent Release Reports covering the operating of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

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

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

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

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

a. A description of the event and equipment involved.

b. Cause(s) for the unplanned release.

c. Actions taken to prevent recurrence.

d. Consequences of the unplanned release.

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

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

a. container volume,

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

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

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

e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and

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

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

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed Member Of The Public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Methods for calculating the dose contribution from liquid and gaseous effluents are given in the ODCM.

6.6.1.6 Radiological Environmental Monitoring

Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Report shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses required by Specification 4.11. If harmful effects are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The Annual Radiological Environment Operating Report shall include a summary of the results obtained as part of the required Interlaboratory Comparison Program and in accordance with the ODCM. Alternatively, participants in

the EPA cross-check program shall provide the EPA program code designation for the unit.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the radiological environmental samples required by Specification 4.11 taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as practical in a supplementary report.

The initial report shall also include the following: a summary description of the radiological environmental monitoring program including sampling methods for each sample type, size and physical characteristics of each sample type, sample preparation methods, analytical methods, and measuring equipment used; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the result of land use censuses required by Specification 4.11. Subsequent reports shall describe all substantial changes in these aspects.

6.6.2 Non-Routine Reports

6.6.2.1 Reportable Events

Reporting requirements for Licensee Event Reports are contained in 10 CFR 50, ¶50.73.

6.6.2.2 Environmental Monitoring

a. If individual milk samples show I-131 concentrations of 10 picocuries per liter or greater, a plan shall be submitted within one week advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 45 mrem/yr to the thyroid of any individual.

b. If milk samples collected over a calendar quarter show average concentrations of 4.8 picocuries per liter or greater, a plan shall be submitted within 30 days advising the NRC of the proposed action to ensure the plant related annual doses will be within the design objective of 45 mrem/yr to the thyroid of any individual.

Duke Power Company
Oconee Nuclear Station

Attachment 2

No Significant Hazards Consideration Evaluation

No Significant Hazards Consideration Evaluation
for Oconee Unit 1, Cycle 10 Reload

Duke Power has determined that the present amendment request poses no significant hazards as defined by NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

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

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

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

The commission has provided guidelines pertaining to the application of the three standards by listing specific examples in 48 FR 14870. Example (iii) of the types of amendments not likely to involve significant hazards considerations expressly applies inasmuch as the proposed amendment involves a nuclear power reactor core reload.

Example (iii) of amendments <u>not</u> likely to involve a significant hazards consideration concerns a core reload, assuming that:

(1) no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved,

(2) no significant changes are made to the acceptance criteria for the technical specifications,

(3) the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and

(4) the NRC has previously found such methods acceptable.

The mark BZ fuel assembly, 60 of which comprise Batch 12, is an improved version of the Mark B fuel assembly, in that the six Inconel Intermediate Spacer grids are replaced with Zircaloy grids. The Mark BZ spacer grids have the same functional design as the Mark B grids, but are slightly different dimensionally to accommodate Zircaloy's material properties. The other fuel assembly components such as the fuel rods, end grid, end fittings, and guide tubes are the same for both designs. The interfaces with the control rods and fuel handling equipment are unchanged, ensuring compatibility with the present reactor site operational procedures. The Mark BZ fuel assembly has been analyzed to ensure conformance with the standard review plan acceptance criteria. Therefore, on the whole, the Mark BZ fuel assembly is a minor design change from the Mark B -- with the Zircaloy spacer grids and related changes being the principal differences. The use of the Mark BZ fuel assembly in the Oconee Unit 1, Cycle 9 and Oconee 2 Cycle 8 core reload was accepted by

the NRC via the Staff's approval of the Unit 1, Cycle 9 amendment request dated November 23, 1984 and the Unit 2, Cycle 8 amendment request dated April 18, 1985.

As was used in Cycle 9, gray (less-absorbing) axial power shaping rods (APSR's) are to be utilized. The staff approved the use of gray APSRs in the Oconee Unit 1, Cycle 9 and Oconee Unit 2, Cycle 8 reload amendment request.

The present reload involves no significant changes to the acceptance criteria for the Technical Specifications. Revisions of the Technical Specifications required for Cycle 10 operation were made in accordance with methods and procedures found acceptable in connection with previous reloads. The final acceptance criteria of the ECCS limits will not be exceeded, and thermal design criteria will be satisfied.

The Oconee Unit 1, Cycle 10 Reload Report (Attachment 3) justifies the operation of the tenth cycle at the rated core power of 2568 MWt. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1985. The Reload Report employs analytical techniques and design bases established in reports submitted for previous reloads which were accepted by USNRC and its predecessor. These techniques are described in the Reload Report references.

Example (i) of 48 FR 14870 is also applicable to several of the attached Tech Spec revisions. This example involves amendment requests that are considered to be purely administrative in nature. The revisions to Section 2 will allow for the consolidation of certain Figures and a table for simplicity and clarity and should be considered administrative. With the information found in this section now generic for all three units, the elimination of repetitive information is needed.

The revisions to the bases of Section 3.5.1 are also administrative and are being made to achieve consistency throughout the tech spec. The bases were not rewritten when an administrative revision to Table 3.5.1-1 occurred. In addition, Table 3.5.1-1 Footnote A is being revised to clarify its intent.

Example (vi) of 48 FR 14870 is applicable to the deletion of the rod position limits and operation imbalance envelope curves. This example involves a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria.

The removal of Specific Cycle dependent figures from the Technical Specification and the use of the Core Operational Limits Report has no impact upon plant operation or safety. The Technical Specification shall continue to require operation within rod position and operational imbalance limits provided by the Core Operational Limits Report. Appropriate actions to be taken if limits are violated shall also remain in the Technical Specifications.

The development of the rod position and operational imbalance limits shall continue to be performed by methods reviewed and approved by the NRC. The NRC shall be provided a copy of the Core Operational Limits Report at least 60 days prior to initial criticality of the cycle. This should be sufficient time to allow the NRC to review the document and to make inquiries if necessary.

Based on the above considerations, Duke contends that the removal of these particular curves (rod position limits and operational imbalance limits) will not involve a significant increase in the probability or consequences of accidents previously considered, nor create the possibility of a new or different kind of accident and will not involve a significant reduction in a safety margin. Therefore, Duke concludes that there is no significant hazards considerations involved with the deletion of these curves.

With supporting reference to previously performed analyses, the following evaluation measures aspects of the Unit 1, Cycle 10 reload against the §50.92 (c) requirements to demonstrate that all three standards are satisfied.

First Standard

> (Amendment would not) involve a significant increase in the probability or consequences of an accident previously evaluated.

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in Cycle 9 parameters to determine the effect of the Cycle 10 reload and to ensure that thermal performance during hypothetical transients is not degraded. The transient evaluation of Cycle 10 is considered to be bound by previously accepted analyses. Section 7 of the Reload Report addresses "Accident and Transient Analysis" for this core reload. This analysis ensures that the proposed reload will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

> (Amendment would not) create the possiblity of a new or different kind of accident from any accident previously evaluated.

The analyses performed in support of this reload are in accordance with the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1975. The conclusion of the overall analysis is that the proposed reload does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

> (Amendment would not) involve a significant reduction in a margin of safety.

The issue of margin of safety for a reload modification involves the following areas:

1. Fuel System Design considerations,

2. Nuclear Design considerations, and

3. Thermal-Hydraulic Design considerations.

Sections 4, 5 and 6 of the Oconee Unit 1, Cycle 10 Reload Report addresses the above areas, respectively. The value limits and margins discussed in these areas are well within the allowable limits and requirements, and reflect no significant reductions to any margins of safety. One can conclude from the examination of these sections, and the Cycle 10 core thermal and kinetic properties (with respect to previous cycle values), that this core reload will not significantly reduce the ability of Oconee Unit 1 to operatoe safely during Cycle 10.

The above evaluation, with its accompanying references, shows that the three §50.92(c) standards are satisfied. In summary, Duke has determined and submits that the proposed reload described herein does not represent any significant hazards.

Duke Power Company
Oconee Nuclear Station

Attachment 3

Core Operational Limits Report
for
O1C10, O2C8, O3C9