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

Posted
Amdt. 147
to DPR-47

April 25, 1986

Dockets Nos. 50-269, 50-270
and 50-287

Mr. Hal B. Tucker
Vice President - Nuclear Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 147, 147 and 144 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated November 19, 1985, as revised on January 14, 1986, and February 14, 1986.

These amendments revise the TSs to support the operation of Oconee Unit 1 at full rated power during the upcoming Cycle 10. The amendments change the following areas: 1) Core Protection Safety Limits (TS 2.1); 2) Protective System Maximum Allowable Setpoints (TS 2.3); 3) Rod Position Limits (TS 3.5.2); and 4) Power Imbalance Limits (TS 3.5.2). These amendments also make administrative changes to clarify the TSs.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance of the enclosed amendments will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Helen Pastis

Helen N. Pastis, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 147 to DPR-38
2. Amendment No. 147 to DPR-47
3. Amendment No. 144 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

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Oconee Nuclear Station
Units Nos. 1, 2 and 3

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. DPR-38

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

   A. The application for amendment by Duke Power Company (the licensee) dated November 19, 1985, as revised on January 14 and February 14, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 147 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 25, 1986

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 147
License No. DPR-47

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

   A. The application for amendment by Duke Power Company (the licensee) dated November 19, 1985, as revised on January 14 and February 14, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 147, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 25, 1986

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 144
License No. DPR-55

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

   A. The application for amendment by Duke Power Company (the licensee) dated November 19, 1985, as revised on January 14 and February 14, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 144, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 25, 1986

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO.147 TO DPR-38

AMENDMENT NO.147 TO DPR-47

AMENDMENT NO.144 TO DPR-55

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

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

Remove Pages	Insert Pages
vi	vi
vii	vii
viii	viii
ix	ix
2.1-1 through 2.1-12	2.1-1 through 2.1-5
2.3-1 through 2.3-13	2.3-1 through 2.3-7
3.1-19a	3.1-19a
3.1-20	3.1-20
3.5-2	3.5-2
3.5-4	3.5-4
3.5-5	3.5-5
3.5-5c	3.5-5c
3.5-12	3.5-12
3.5-15 (3 pages)	3.5-15
3.5-18 (3 pages)	3.5-18
3.5-21 (3 pages)	3.5-21
3.5-24 (3 pages)	3.5-24

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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

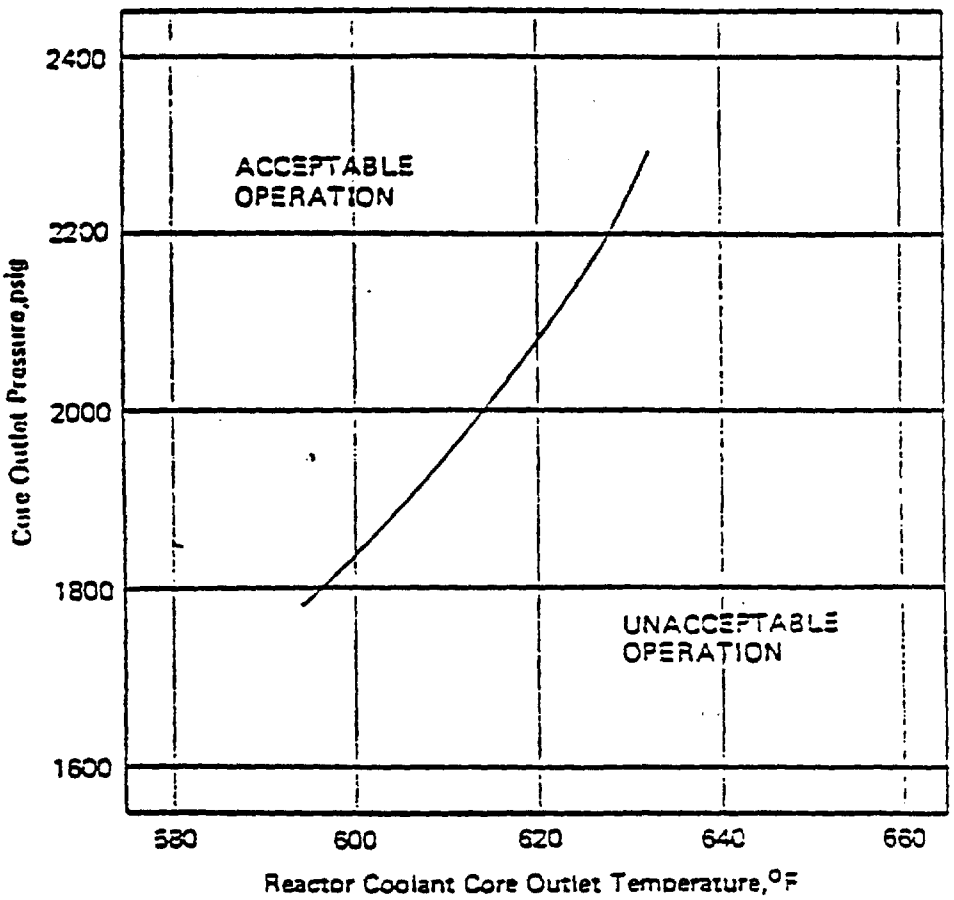
(1) Correlation of Critical Heat Flux in a Bundle cooled by Pressurized Water, BAW-10000, March 1970.

(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.

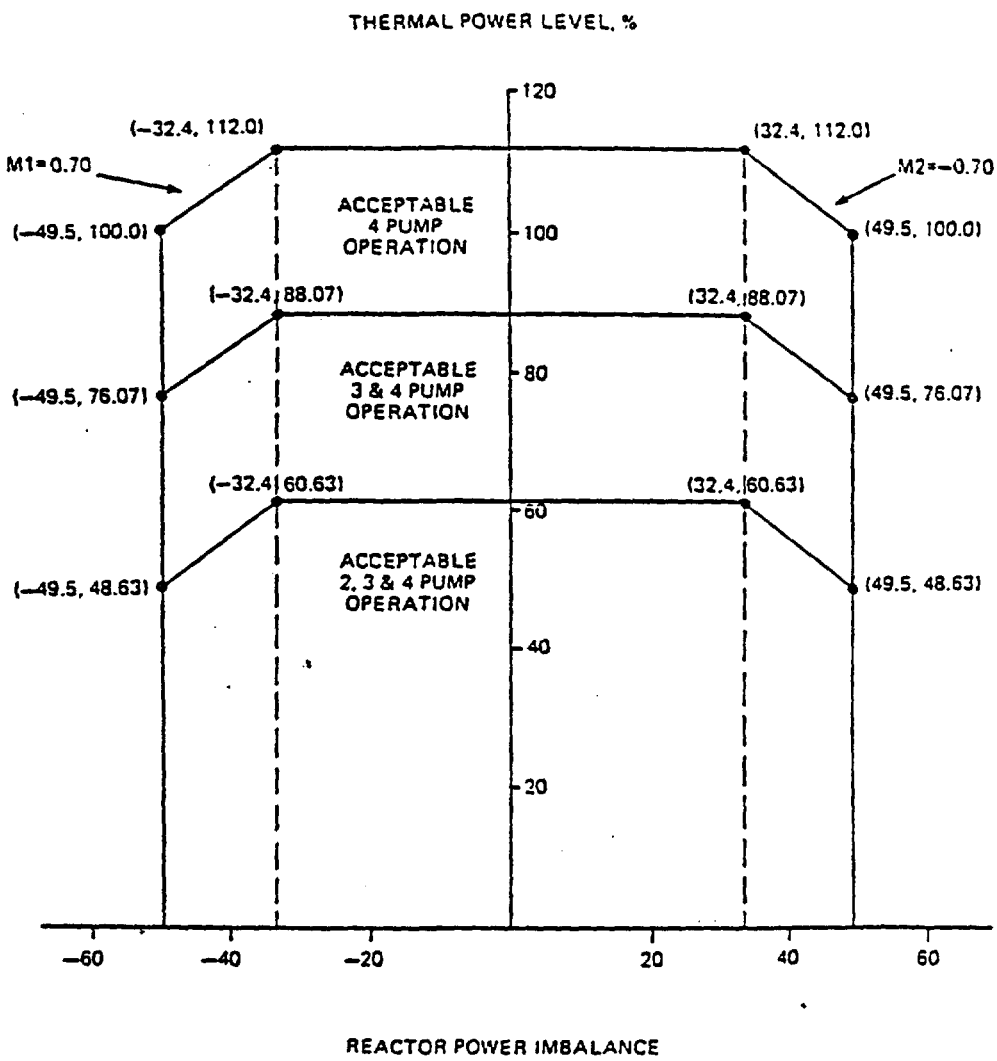
(5) Oconee Unit 2, Cycle 7 - Reload Report, DPC-RD-2002, Duke Power Company, September 1983.



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

Amendments Nos. 147 , 147 , & 144

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 of $\leq$ 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

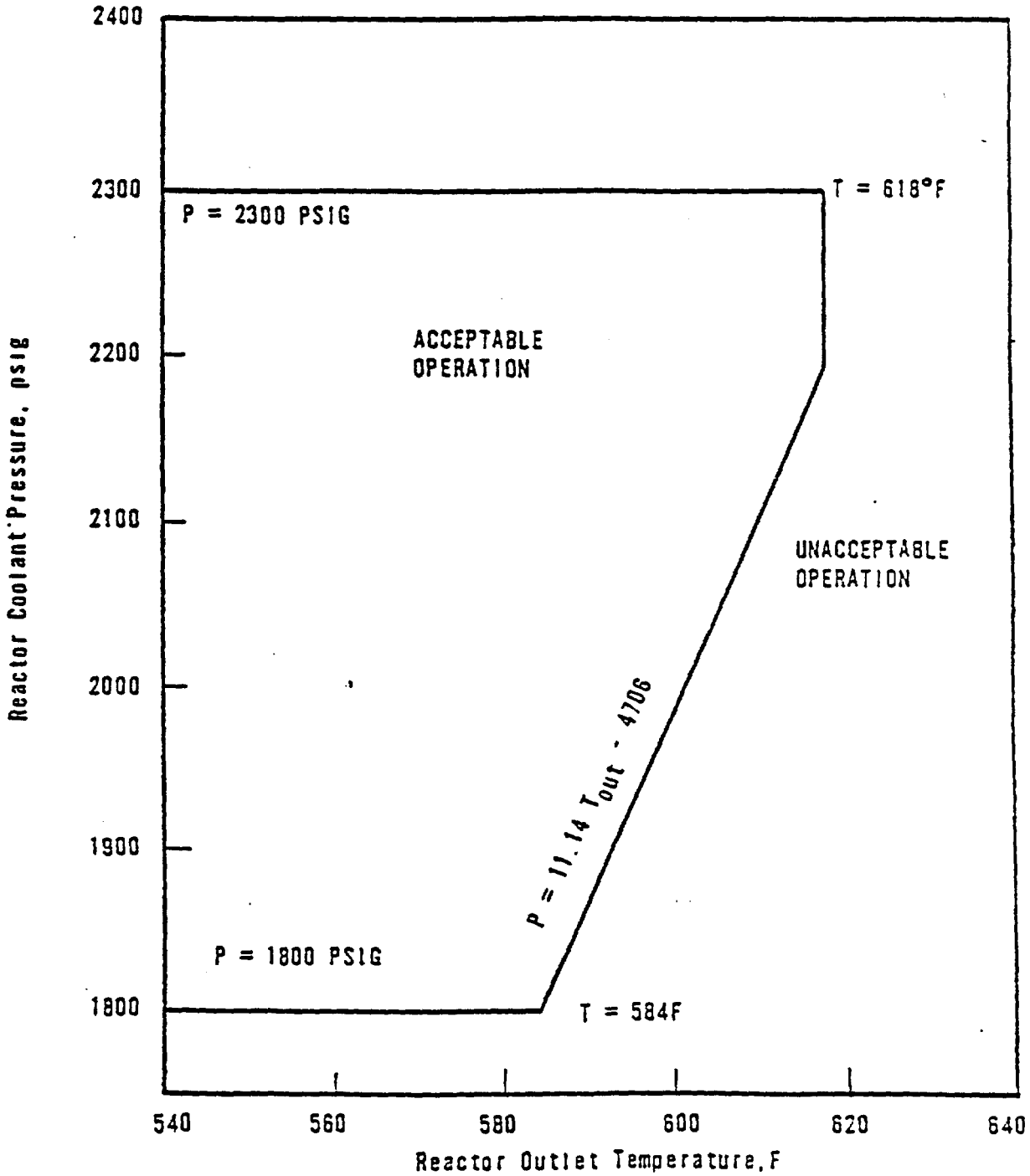
(1) FSAR, Section 15.3

(2) FSAR, Section 15.2

(3) FSAR, Section 15.7

(4) FSAR, Section 15.8

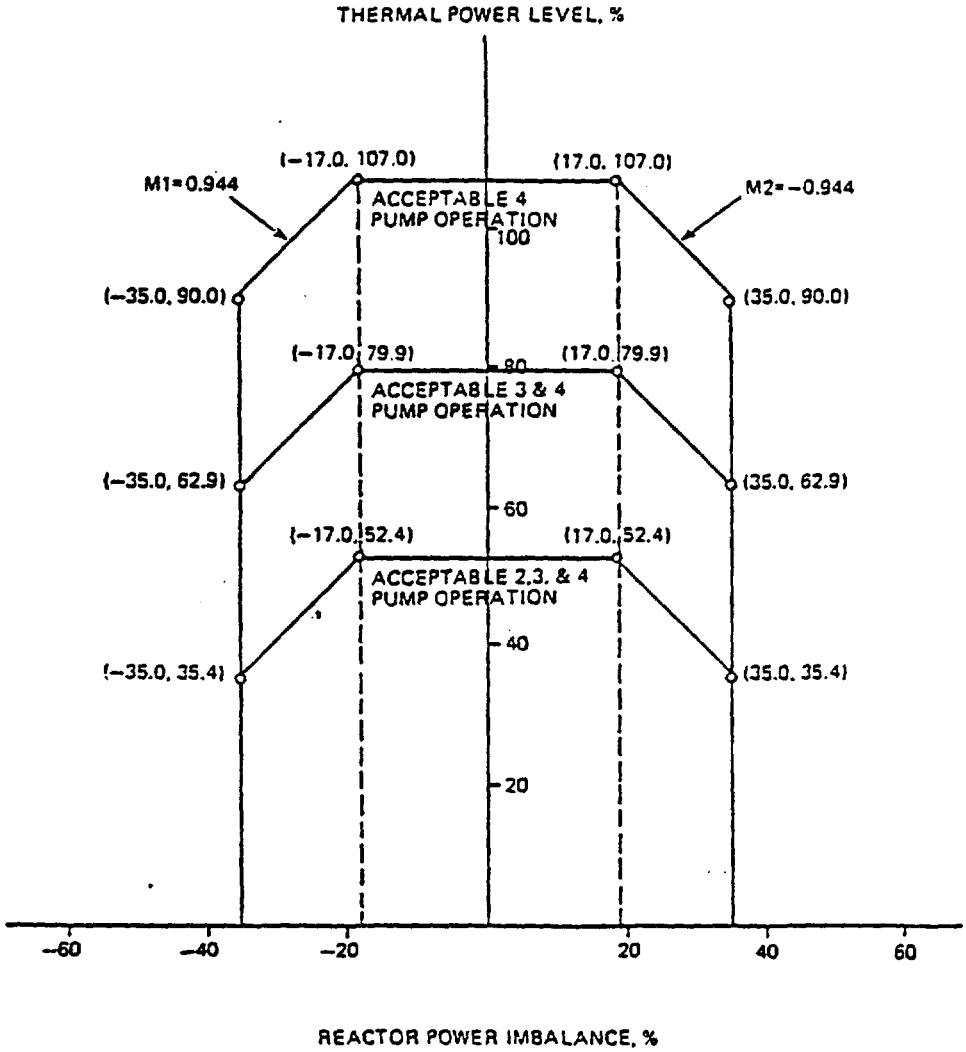
(5) FSAR, Section 15.6



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

Figure 2.3-1
OCONEE NUCLEAR STATION

Amendments Nos. 147 , 147 , & 144



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) = (11.14 T_{out} - 4706)(3)	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).

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%$\Delta k/k$ 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.

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 is 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 Instrument 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.

The rod position limits 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.

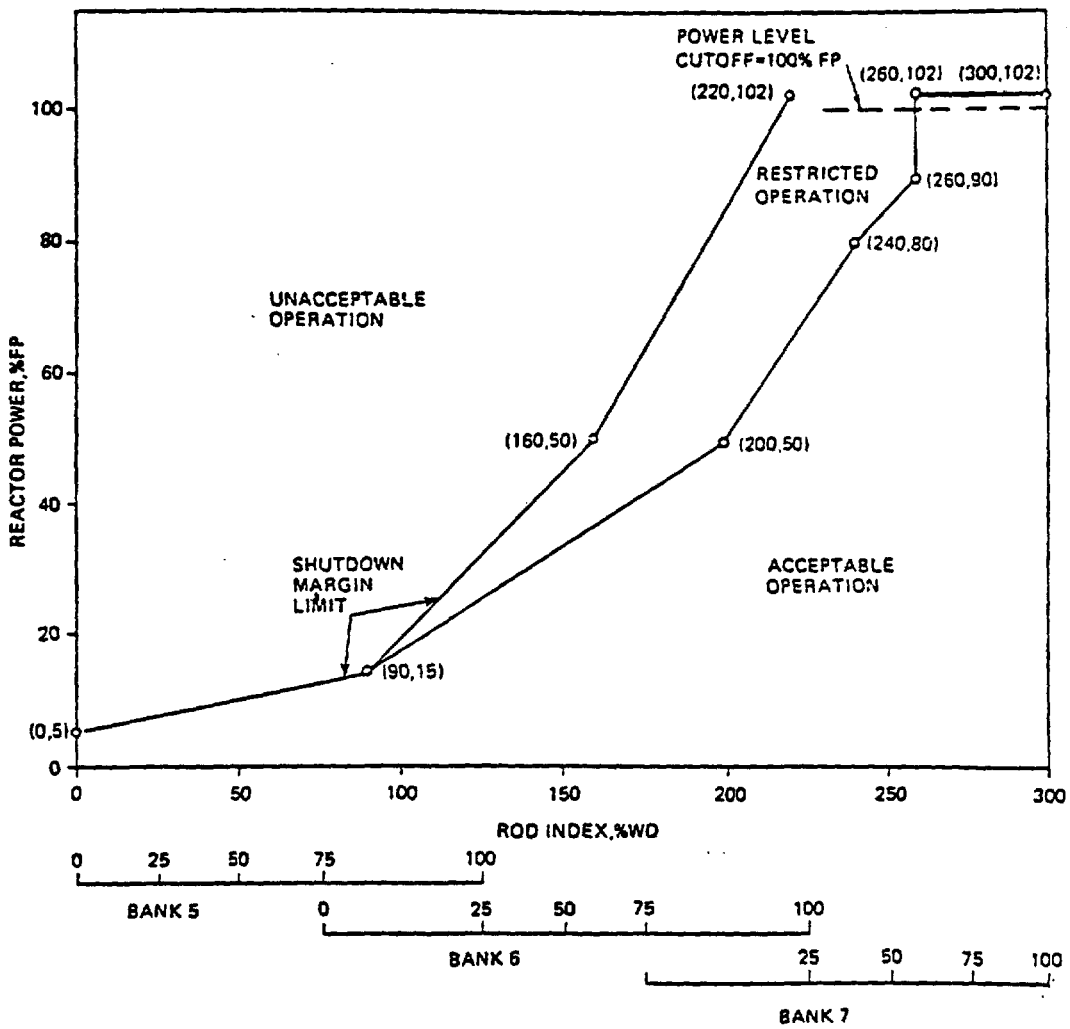
The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding
7.50% for Unit 1. The limits shown in Specification 3.5.2.4
7.50% for Unit 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 transient xenon power peaking, including xenon-free startup, are inherently included in the limits of Sections 3.5.2.5 (Control Rod Positions) and 3.5.2.7 (Reactor power imbalance) for transient peaking behavior bounded by the following factors. For feed and bleed (unrodded) operation, a 5% peaking increase is applied to calculated peaks at equilibrium conditions for powers at and above 90% FP. A 13% increase is applied below 90% FP. For rodded operation an 8% peaking increase is applied at and above 90% FP and an 18% increase is applied below 90% FP. If these values, checked every cycle, conservatively bound the peaking effects of all transient xenon, then the need for any hold at a power level cutoff below 100% FP is precluded. If not, either the power level at which the requirements of 3.5.2.6 must be satisfied or the above listed factors will be suitably adjusted to preserve the ECCS power peaking criteria. (Reference 6)

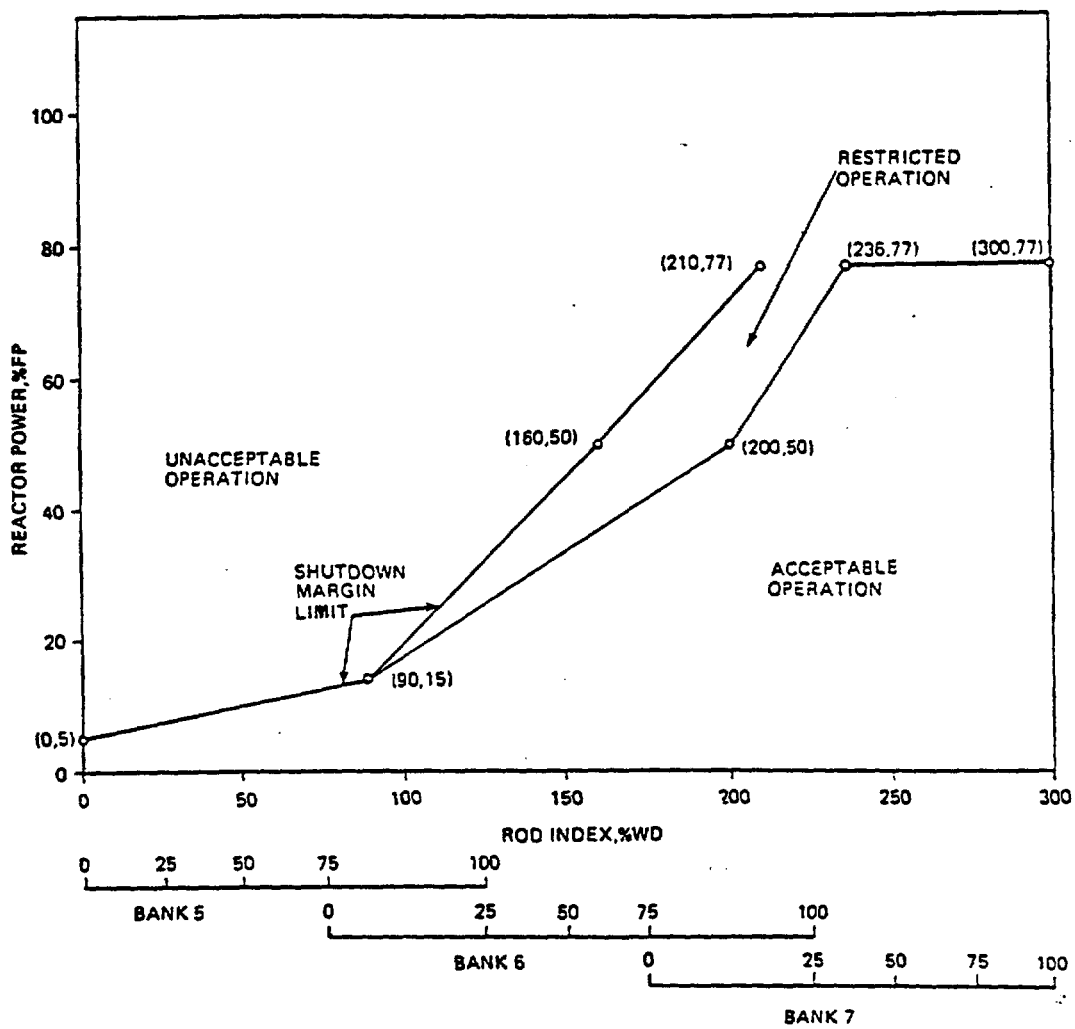


ROD POSITION LIMITS FOR FOUR PUMP OPERATION
FROM 0 EFPD TO EOC UNIT 1

Figure 3.5.2-1

OCONEE NUCLEAR STATION

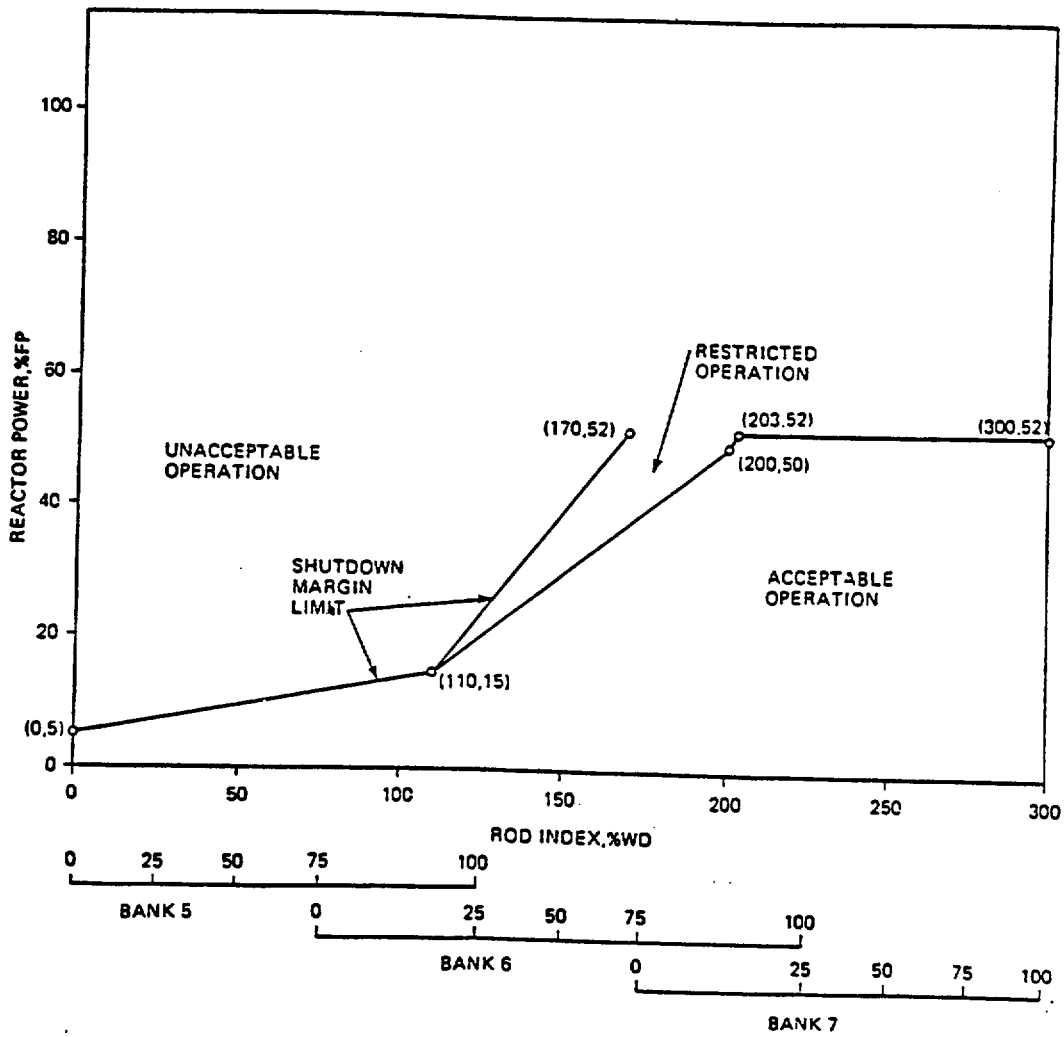
Amendments Nos. 147 , 147, & 144



ROD POSITION LIMITS FOR THREE PUMP OPERATION
FROM 0 EFPD TO EOC UNIT 1

Figure 3.5.2-4
OCONEE NUCLEAR STATION

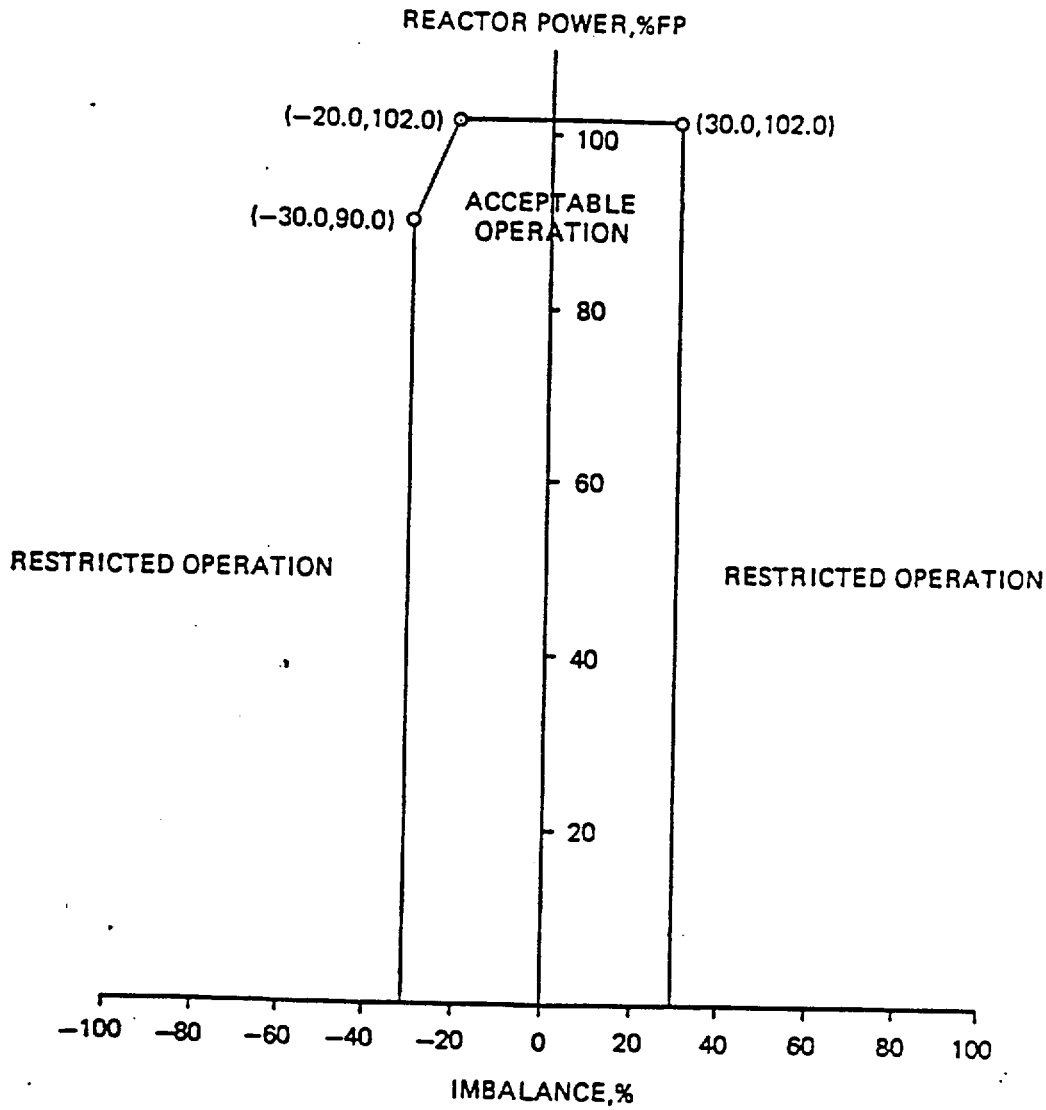
Amendments Nos. 147 , 147 , & 144



ROD POSITION LIMITS FOR TWO PUMP OPERATION
FROM 0 EFPD TO EOC UNIT 1

Figure 3.5.2-7
OCONEE NUCLEAR STATION

Amendments Nos. 147 , 147 , & 144



OPERATIONAL POWER IMBALANCE ENVELOPE
FROM 0 EFPD TO EOC UNIT 1

Figure 3.5.2-10
OCONEE NUCLEAR STATION

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 144 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

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

INTRODUCTION

By letter dated November 19, 1985, as revised on January 14, 1986, and February 14, 1986 (Refs. 1, 6 and 5), Duke Power Company (the licensee) proposed changes to the Technical Specifications (TSs) of Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2 and 3. These amendments would consist of changes to the Station's common TSs.

These amendments would authorize proposed changes to the Oconee Nuclear Station TSs which are required to support the operation of Oconee Unit 1 at full rated power during the upcoming Cycle 10. The proposed amendments would change the following areas: 1) Core Protection Safety Limits (TS 2.1); 2) Protective System Maximum Allowable Setpoints (TS 2.3); 3) Rod Position Limits (TS 3.5.2); and 4) Power Imbalance Limits (TS 3.5.2). These amendments would also make administrative changes to clarify the TSs.

To support the license amendment application, the licensee submitted a Duke Power Company report, DPC-RD-2006, "Oconee Unit 1, Cycle 10 Reload Report" (Ref. 2), as an attachment to Reference 1. A summary of the Cycle 10 operating parameters is included in the report, along with safety analyses.

The Cycle 10 core consists of 177 fuel assemblies (FA), each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches except the batch containing the gadolinia lead test assemblies (LTAs) have an average nominal fuel loading of 426.9 kilograms of uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 4-1 (Ref. 2). The Cycle 10 core loading diagram, enrichments, control rods and burnable poison rod assembly (BPRA) locations and enrichments are also given in Reference 2.

Cycle 10 will operate in a rods-out, feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 52 BPRAS. In addition to the full-length control rods, eight Inconel gray axial power shaping rods (APSRs) are provided for additional control of axial power distribution. Since gray APSRs are being utilized, there are eight control rods in group seven and twelve in group five to reduce the negative offset response to the group seven rod movement.

The present reload involves no significant changes to the acceptance criteria for TSs. Proposed revisions to the TSs required for Cycle 10 operation were made in accordance with methods and procedures found acceptable in connection with previous reloads and are the result of minor cycle-to-cycle fuel changes.

EVALUATION

Evaluation of Fuel System Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 1 Cycle 10 are listed in Table 4-1 (Ref. 2). All fuel assemblies are mechanically interchangeable. Four regenerative neutron sources will be used in the Mark BZ fuel assemblies. The Cycle 10 core contains only fuel designs which have been previously loaded in the Oconee Unit 1 reactor and have been previously approved by the NRC staff. The fuel rod design, cladding collapse, cladding stress and strain, and the thermal design fuel analyses for Cycle 10 fuel designs, including the gadolinia LTAs and the gray APSRs, are either bounded by conditions previously analyzed for Oconee 1 or were analyzed specifically for Cycle 10 using methods and limits previously reviewed and approved by the NRC staff. We conclude that the overall fuel system design for Oconee 1 Cycle 10 is acceptable.

Nuclear Design

Table 5-1 (Ref. 2) compares the core physics parameters of Cycle 10 with those of the reference Cycle 9. The values for Cycle 9 were calculated by Babcock and Wilcox (B&W) (Ref. 3), and Cycle 10 values were generated by Duke Power Company using the reload design methods described in Reference 4. The primary reasons for the differences in the physics parameters between Cycle 9 and 10 are the decreased number of fresh fuel assemblies, different BPRA loadings, and different shuffle patterns. Core design changes for Cycle 10 include a reduction in cycle length to 400 effective full power days and a reduction in the number of fresh Mark BZ assemblies from 64 to 60. Another important difference is that the Duke Power calculational methods are used to obtain the important nuclear design parameters for this cycle.

Based on our review, we conclude that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of Oconee 1 Cycle 10 is acceptable.

Evaluation of Thermal-Hydraulic Design

The generic Mark B and Mark BZ thermal-hydraulic design analyses supporting Cycle 10 operation were performed by Duke Power Company using the methods described in Reference 4. The Cycle 9 and Cycle 10 maximum design conditions are summarized in Table 6-1 (Ref. 4).

The Cycle 10 transition core will include 60 fresh Mark BZ, Batch 12 fuel assemblies, 52 of which will contain BPRAs, leaving 56 fuel assemblies with open guide tubes. This results in a core bypass flow of 8.2% of the total system flow, which is the bypass flow assumed in the generic thermal-hydraulic analyses. The core will also contain four gadolinia LTAs which are geometrically and hydraulically identical to the Mark BZ assemblies.

The Mark BZ fuel assembly has a slightly higher pressure drop than the Mark B assembly as a result of the increased flow resistance of the Zircaloy spacer grids. The presence of Mark BZ and Mark B assemblies in a core results in less coolant flow in the Mark BZ fuel than would occur in an all Mark BZ core. The generic Mark BZ analyses conservatively account for this transition core effect.

In a Mark BZ transition core, the limiting Mark B hot channel will receive more coolant and yield better departure from nucleate boiling (DNB) performance than would be predicted for a full Mark B core. Thus, the generic Mark B analyses, based on the B&W-2 critical heat flux (CHF) correlation, are bounding and are applicable to the Cycle 10 transition core.

Based on our review, we conclude that approved methods have been used, that the thermal-hydraulic design parameters meet the DNB ratio safety limit using approved CHF correlations and that the thermal-hydraulic design of Oconee 1 Cycle 10 is acceptable.

Safety Analyses

The important kinetics parameters for Cycle 10 have been compared to the values used in the Final Safety Analysis Report (FSAR) and/or the densification report. The licensee has shown that the Cycle 10 values are bounded by those previously used. The licensee has also determined that the initial conditions of the transients in Cycle 10 are bounded by either the FSAR, the fuel densification report, previous reload analyses, or analyses using approved methods.

B&W has performed a generic loss of coolant accident (LOCA) analysis for the B&W 177-FA, lowered-loop nuclear steam supply system using the final acceptance criteria Emergency Core Cooling System evaluation model. The combination of average fuel temperature as a function of linear heat rate (LHR) and the

lifetime pin pressure data is conservative relative to those calculated for this cycle. These results are based upon a bounding analytical assessment of NUREG-0630 on LOCA and operating LHR limits performed by B&W. The B&W analyses have been approved by the NRC staff, and the LHR limits are satisfactorily incorporated into the TSs for Cycle 10 through the operating limits on rod index and axial power imbalance.

Technical Specification Modifications

Oconee Unit 1 Cycle 10 TSs have been modified to account for normal cycle-to-cycle fuel changes in power peaking and control rod worths. We have reviewed the proposed TS revisions for Cycle 10. These changes concern the (1) Core Protection Safety Limits of Specification 2.1, (2) Protective System Maximum Allowable Setpoints of Specification 2.3, (3) Rod Position Limits and Operational Power Imbalance Limits of Specification 3.5.2. On the basis that approved methods were used to obtain these limits, we find these TS modifications acceptable.

The licensee also proposed to clarify some of the TSs. Some of the figures and a Table in Section 2, such as the rod position limits and operational power imbalance which have been individually given for each Oconee unit, are being combined into one TS. The Reactor Protective System setpoints have been assigned the same values and thus Section 2 would be written such that it is generic to all Oconee units. Also, the Bases for Section 2 have been revised to simplify and clarify this section. A discrepancy was found between TS 3.5.1 and its Bases. It appears that the Bases for this TS were not reworded when the licensee previously requested a revision to Table 3.5.1-1. The footnote allowing a one-out-of-two logic for up to four hours in the power range instrumentation is being clarified.

We have reviewed these proposed changes and have determined that they are administrative in nature and that they clarify the TSs. We therefore find them acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments involve changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding.

Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

CONCLUSION

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

Dated: April 25, 1986

Principal Contributor: G. Schwenck

References

1. Letter, H. B. Tucker (Duke) to H. R. Denton (NRC), "Oconee Nuclear Station Unit 1," November 19, 1985.

2. Report, "Oconee Unit 1, Cycle 10 Reload Report," DPC-RD-2006, Duke Power Company, November 1985.

3. Report, "Oconee Unit 1, Cycle 9 - Reload Report, BAW-1841, Babcock & Wilcox, August 1984.

4. Report, "Oconee Nuclear Station Reload Design Methodology II," DPC-NE-1002, Duke Power Company, Charlotte, North Carolina, March 1985.

5. Letter, H. B. Tucker (Duke) to H. R. Denton (NRC), "Oconee Nuclear Station Unit 1," February 14, 1986.

6. Letter, H. B. Tucker (Duke) to H. R. Denton (NRC), "Oconee Nuclear Station Unit 1," January 14, 1986.