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RECEIVED NOV 12 1981 U.S. NUCLEAR REGULATORY COMMISSION

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

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 102 , 102, and 99 to 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 applications dated September 8 and September 10, 1981.

These amendments revise the TSs to reflect current calculated string errors used in the determination of Reactor Protective System setpoints and upgrade the format of the Operational Safety Instrumentation Table.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Philip C. Wagner, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 102 to DPR-38
2. Amendment No. 102 to DPR-47
3. Amendment No. 99 to DPR-55
4. Safety Evaluation
5. Notice

cc w/enclosures:
See next page

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E.K & F.R Amendment Notice only as required.

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD		
SURNAME	RIngram	PWagner;cf	JStolz	TNovak	KETCHEN		
DATE	10/24/81	10/23/81	10/27/81	10/28/81	10/30/81		

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. 102
License No. DPR-38

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

   A. The applications for amendment by Duke Power Company (the licensee) dated September 8 and September 10, 1981, comply 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 applications, 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:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.102 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 2, 1981

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.102
License No. DPR-47

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

   A. The applications for amendment by Duke Power Company (the licensee) dated September 8 and September 10, 1981, comply 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 applications, 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:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.102 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 2, 1981

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. 99
License No. DPR-55

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

   A. The applications for amendment by Duke Power Company (the licensee) dated September 8 and September 10, 1981, comply 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 applications, 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:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 99 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 2, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 102 TO DPR-38

AMENDMENT NO. 102 TO DPR-47

AMENDMENT NO. 99 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
2.1-3d	2.1-3d
2.1-9	2.1-9
2.1-12	2.1-12
2.3-2	2.3-2
2.3-3	2.3-3
2.3-5	2.3-5
2.3-6	2.3-6
2.3-7	2.3-7
2.3-10	2.3-10
2.3-11	2.3-11
2.3-12	2.3-12
2.3-13	2.3-13
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3
3.5-4	3.5-4
3.5-5	3.5-5
3.5-5a	3.5-5a
-	3.5-5b
4.5-3	4.5-3

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 3.

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

The specified flow rates for Curves 1, 2 and 3 of Figure 2.1-2C correspond to the expected minimum flow rates with four pumps, three pumps and one pump in each loop, respectively.

The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup independent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis (4).

All plant operating limits are presently based on an original method of calculating rod bowing penalties that are more conservative than those that would be obtained with new approved procedures (4). For Cycle 6 operation, this subrogation results in a 10% DNBR margin, which is partially used to offset the reduction in DNBR due to fuel rod bowing.

The maximum thermal power for three-pump operation is 90.65 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.08 = 80.7 percent power plus the maximum calibration and instrument error (Reference 4). The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For each curve of Figure 2.1-3C a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The curve of Figure 2.1-1C is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3C.

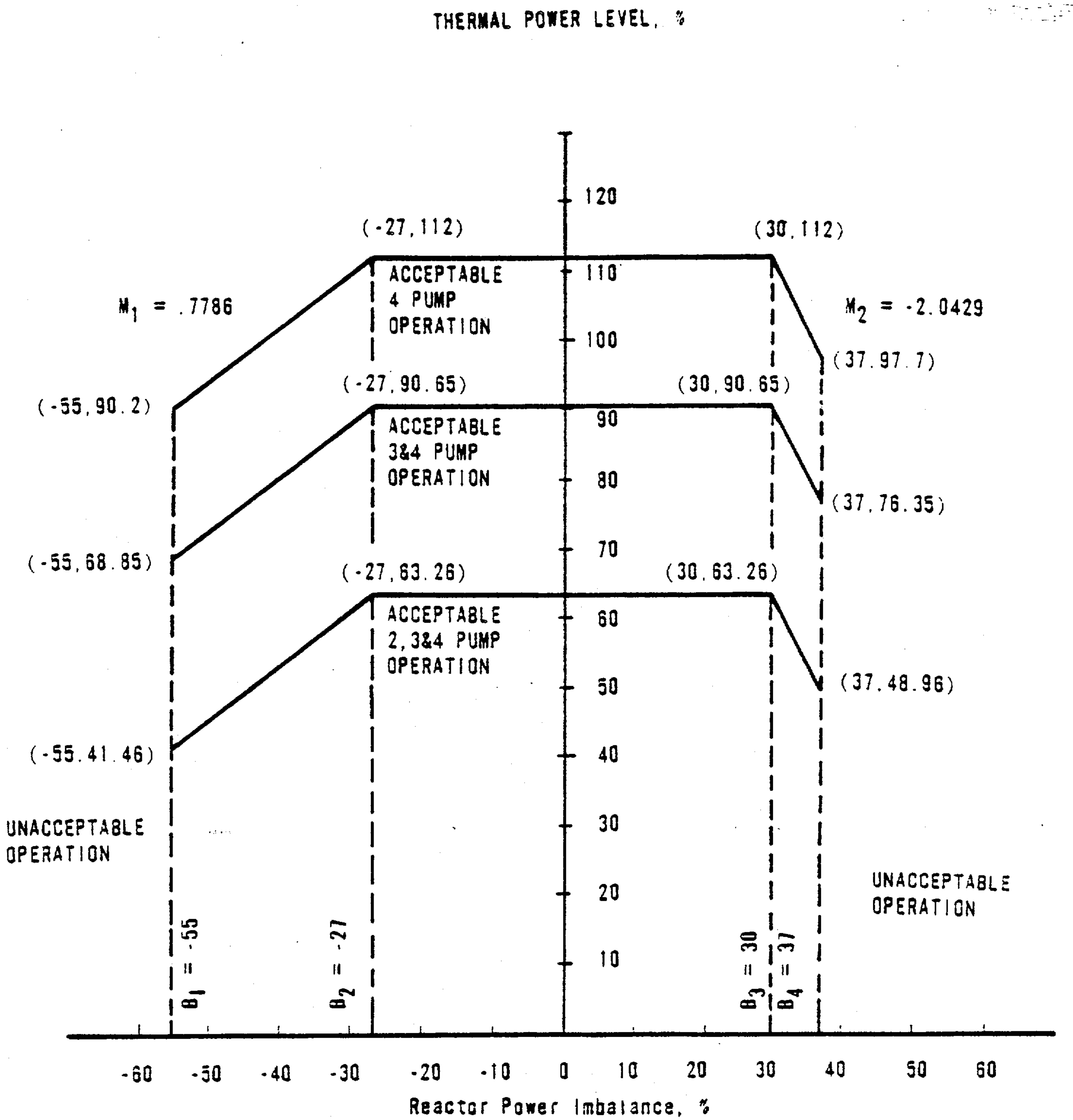
References

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

(2) Oconee 3, Cycle 3 - Reload Report - BAW-1453, August, 1977.

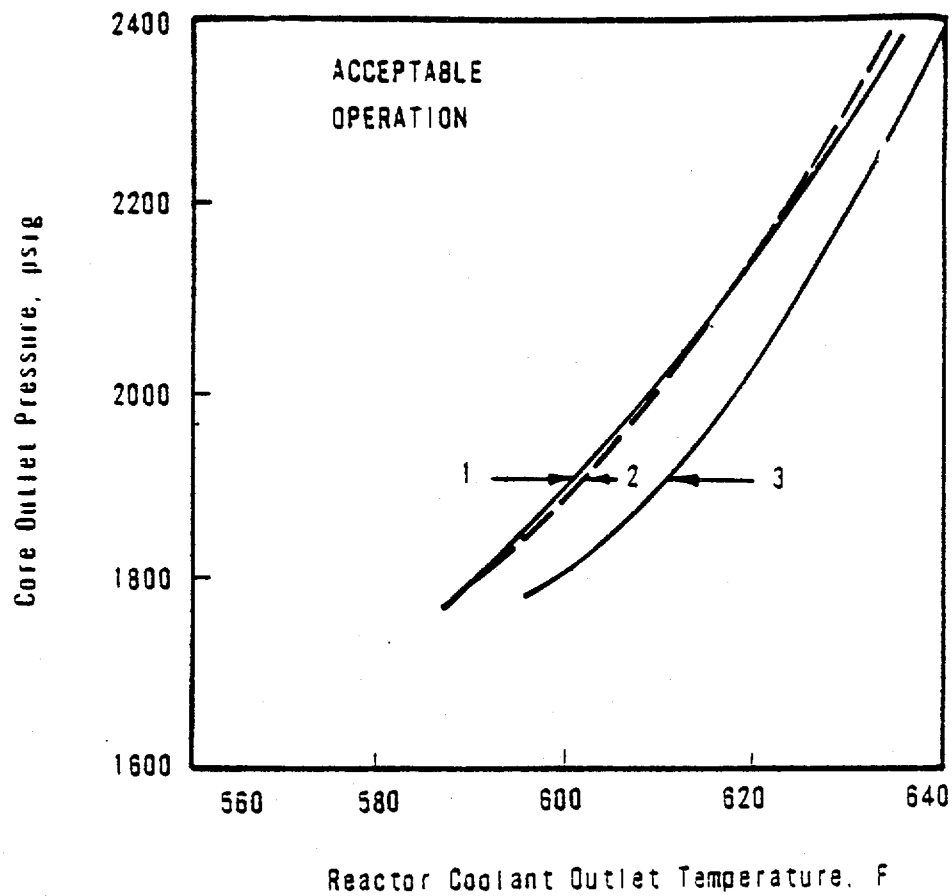
(3) Amendment 1 - Oconee 3, Cycle 4 - Reload Report - BAW-1486, June 12, 1978.

(4) Oconee 3, Cycle 6 - Reload Report - BAW-1634, August, 1980.



CORE PROTECTION SAFETY LIMITS
UNIT 3
OCONEE NUCLEAR STATION

Figure 2.1-2C



CURVE	COOLANT FLOW, GPM	POWER, %	PUMPS OPERATING	TYPE OF LIMIT
1	374,880 (100%)*	112	4	DNBR
2	280,035 (74.7%)	90.65	3	DNBR
3	183,690 (49.0%)	63.26	2	QUALITY

*106.5% OF FIRST-CORE DESIGN FLOW.

CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.1-3C

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis. (4)

Overpower Trip Based on Flow and Imbalance

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

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

1. Trip would occur when four reactor coolant pumps are operating if power is 108% and reactor flow rate is 100%, or flow rate is 92.59% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.68% and reactor flow rate is 74.7% or flow rate is 69.44% 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 52.92 and reactor flow rate is 49.0% or flow rate is 45.37% and the power level is 49%.

   The flux-to-flow ratios account for the maximum calibration and instrument errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

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

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A - Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2B - Unit 2
2.3-2C - Unit 3

level trip and associated reactor power/reactor power-imbalance boundaries by 1.08% - Unit 1 for 1% flow reduction.
1.08% - Unit 2
1.08% - Unit 3

Pump Monitors

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

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2300 AF) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}$-4706) trip
(1800) psig ($11.14\ T_{out}$-4706)
(1800) psig ($11.14\ T_{out}$-4706)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

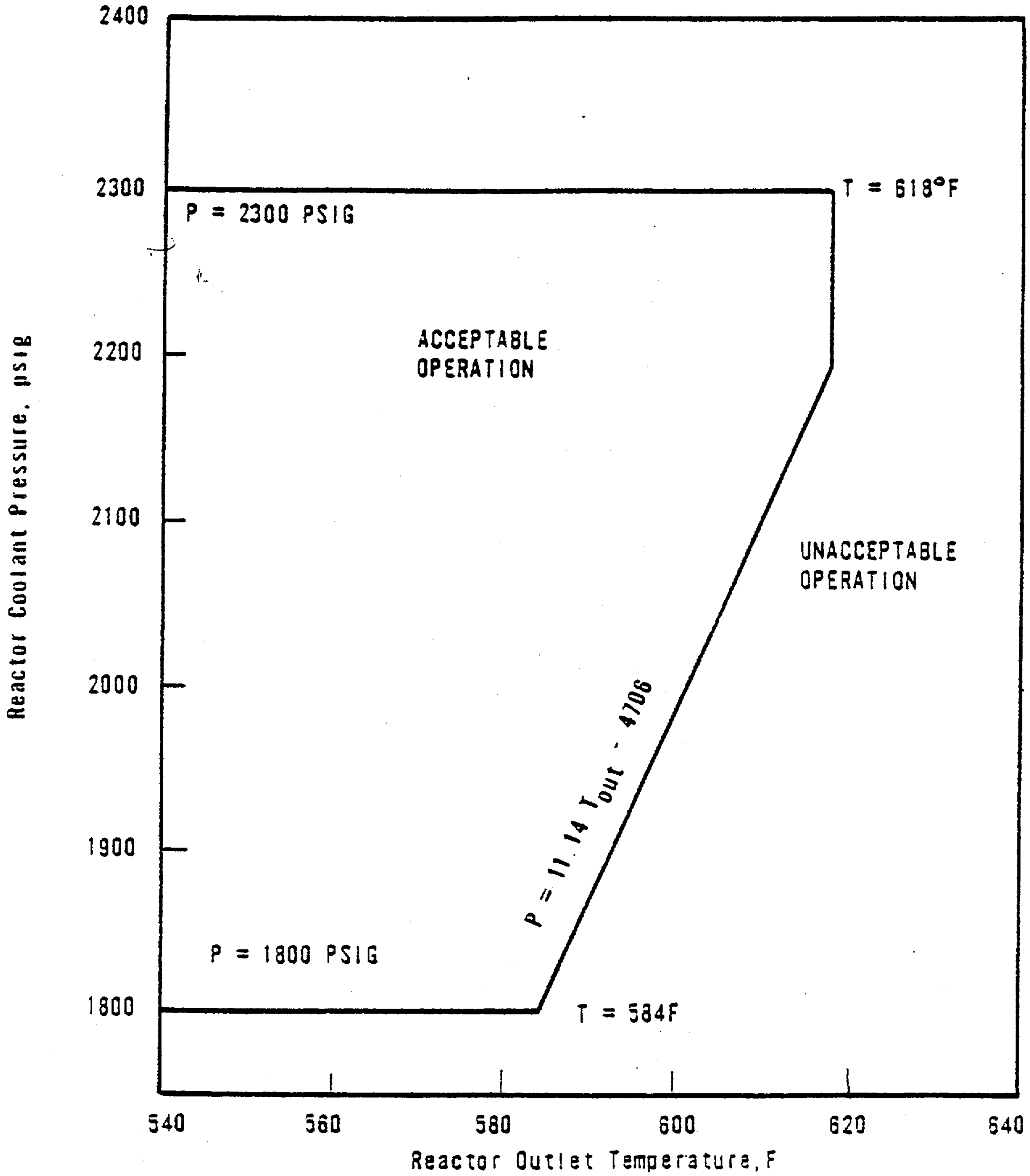
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618°F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620°F.

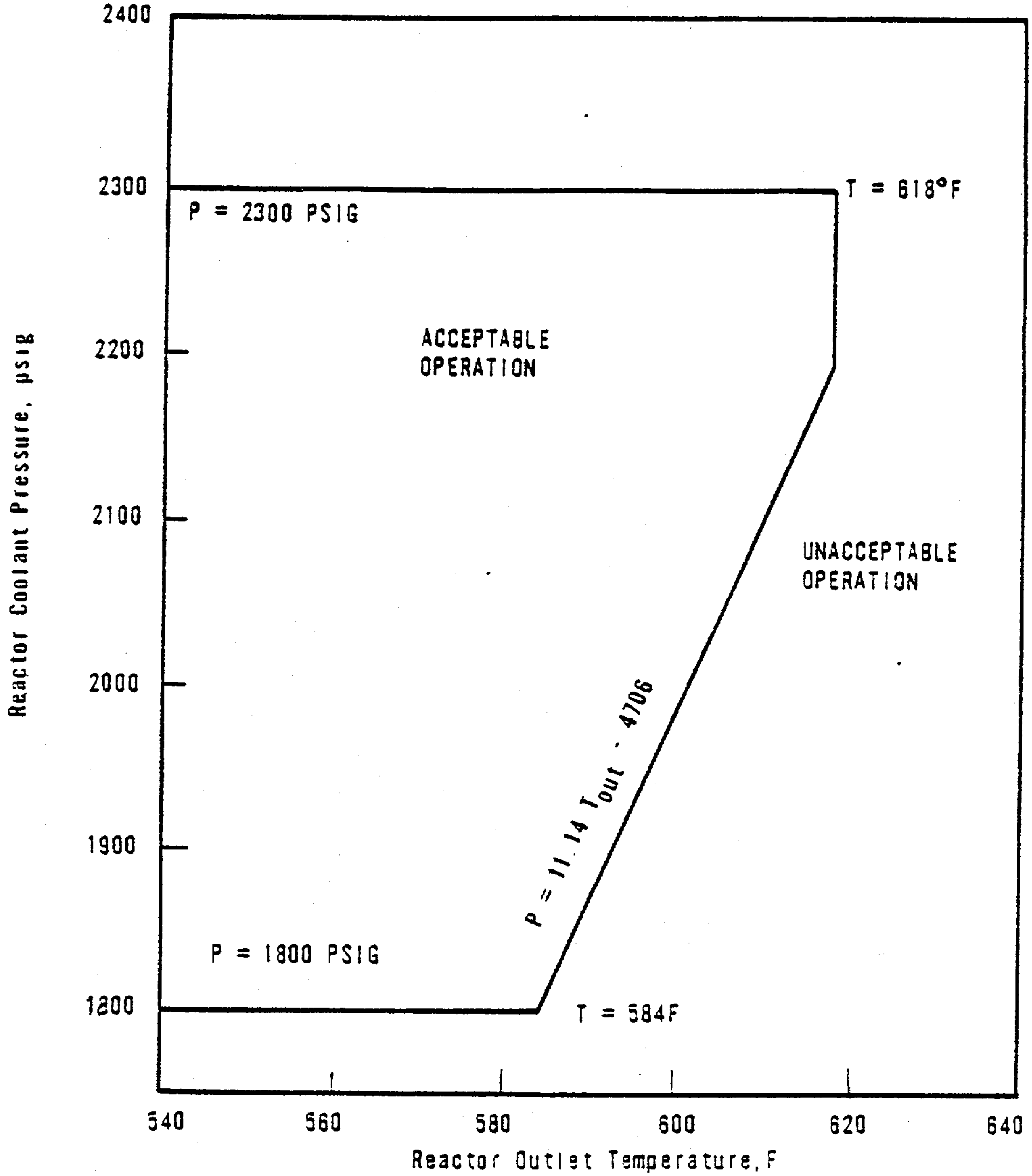
Reactor Building Pressure

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



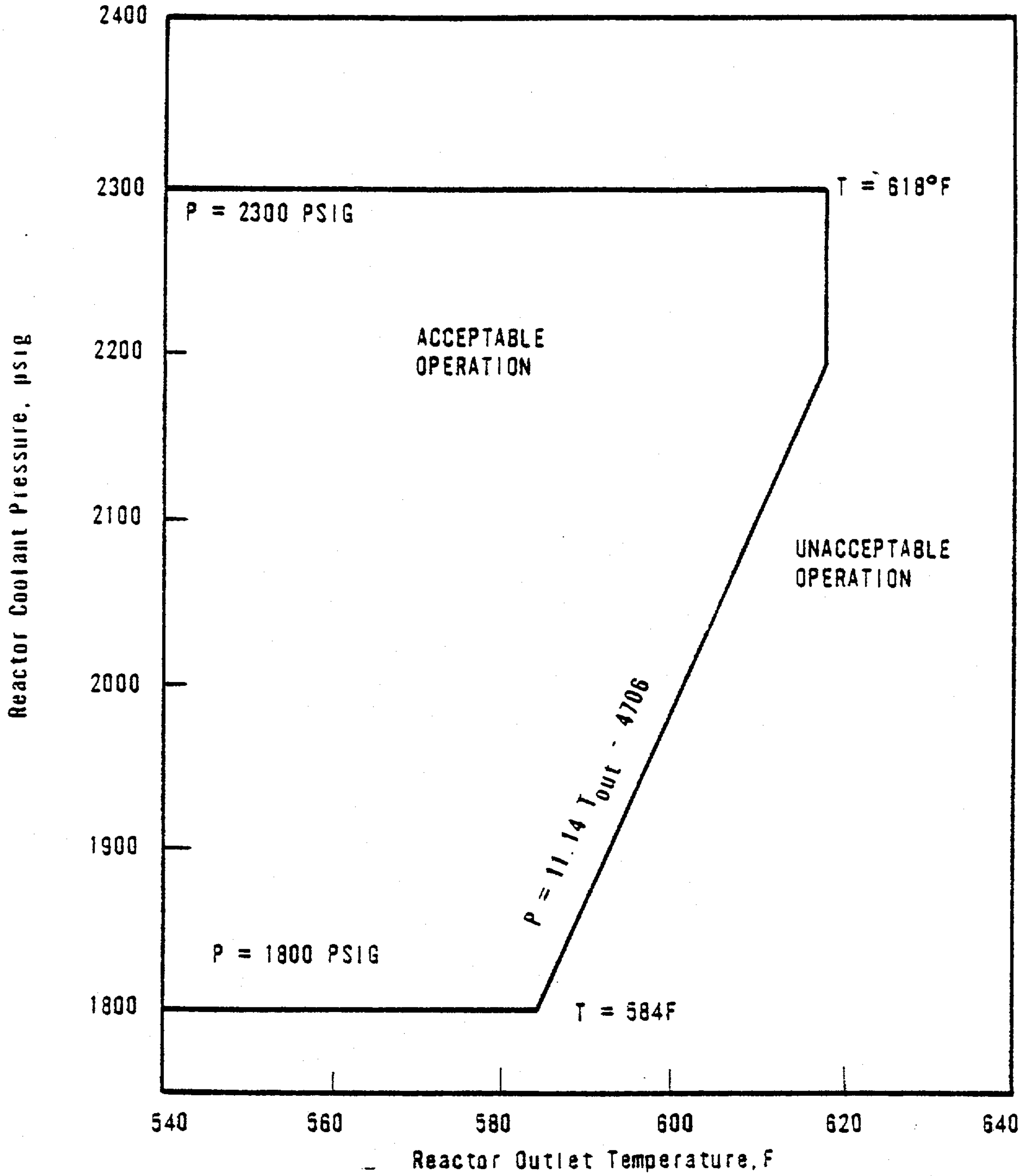
PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 1
OCONEE NUCLEAR STATION

Figure 2.3-1 A



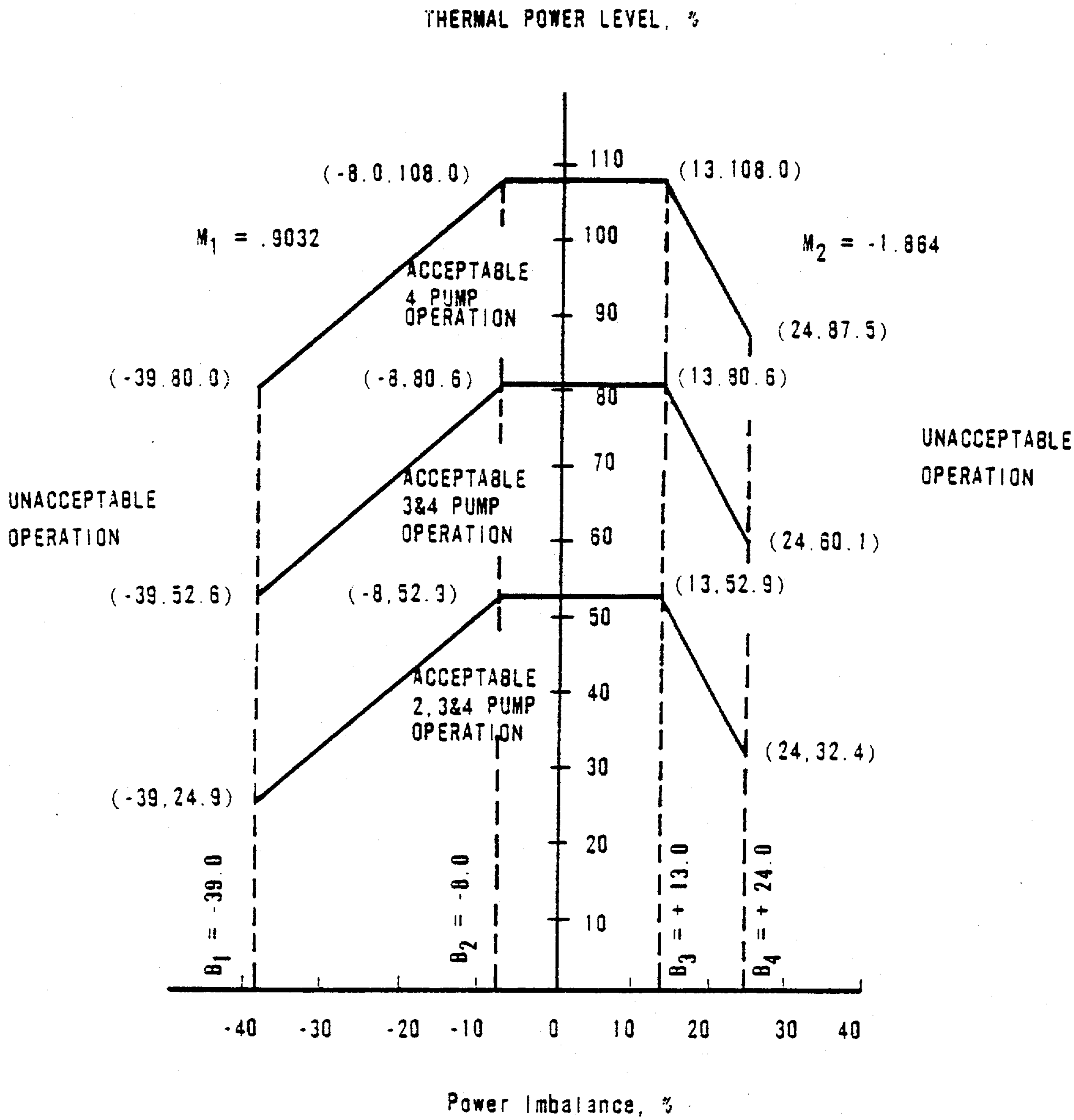
PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 2
OCONEE NUCLEAR STATION

Figure 2.3-1 B



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 3
OCONEE NUCLEAR STATION

Figure 2.3-1C



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.3-2C

Table 2.3-1A
Unit 1

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	104.9	104.9	104.9	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	Bypassed
7.	Reactor Coolant Temp. F., Max.	618	618	618	618
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

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

(2) Reactor Coolant System Flow, %.

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

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

Table 2.3-1B
Unit 2

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	104.9	104.9	104.9	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	Bypassed
7.	Reactor Coolant Temp. F., Max.	618	618	618	618
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

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

(2) Reactor Coolant System Flow, %.

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

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

Table 2.3-1C
Unit 3

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	104.9	104.9	104.9	5.0[3]
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720[4]
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure,psig, Min.	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	Bypassed
7.	Reactor Coolant Temp. F., Max.	618	618	618	618
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

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

(2) Reactor Coolant System Flow, %.

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

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

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objective

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

Specifications

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

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

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

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light.

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

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

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and 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 redundancy requirements of Column C (Table 3.5.1-1). A tripped channel is considered to be operable. This is in agreement with redundancy and single failure criteria of IEEE-279 as described in FSAR Section 7.

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

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

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

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

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

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

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

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

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

REFERENCE

FSAR, Section 7.1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS

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	Bring to hot shutdown within 12 hours
6.	RPS Pressure-Temperature Instruments Channels	4	2	3	Bring to hot shutdown within 12 hours
7.	RPS Flux Imbalance Flow Instrument Channels	4	2	3	Bring to hot shutdown within 12 hours
8.	RPS Reactor Coolant Pressure				
	a. High Reactor Coolant Pressure Instrument Channels	4	2	3	Bring to hot shutdown within 12 hours
	b. Low Reactor Coolant Pressure Channels	4	2	3	Bring to hot shutdown within 12 hours
9.	RPS Power-Number of Pumps Instrument Channels	4	2	3	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	Bring to hot shutdown within 12 hours
11. RPS Anticipatory Reactor Trip System (g)				
a. Loss of Turbine	4	2	3	Bring to hot shutdown within 12 hours
b. Loss of Main Feedwater	4	2	3	Bring to hot shutdown within 12 hours
12. ESF High Pressure Injection System and Reactor Building Isolation (Non-essential Systems)				
a. Reactor Coolant Pressure Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
c. Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
13. ESF Low Pressure Injection System				
a. Reactor Coolant Pressure Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)

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
b. Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
c. Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
14. ESF Reactor Building Isolation (Essential Systems) & Reactor Building Cooling System				
a. Reactor Building 4 PSIG Instrument Channel	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
15. ESF Reactor Building Spray System				
a. Reactor Building High Pressure Instrument Channel	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
16. Turbine Stop Valves Closure	2	1	2	Bring to hot shutdown within 12 hours (f)

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

(a) For channel testing, calibration, or maintenance, one of the three minimum operable channels may be put into manual bypass leaving a 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 condition within 24 hours.

(f) One channel may be inoperable for no more than 24 hours before going to the hot shutdown condition.

(g) This requirement is applicable as follows:

Unit 1 - following Summer 1981 refueling outage

Unit 2 - following Fall 1981 refueling outage

Unit 3 - April 1, 1981

(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 High Pressure Injection System under normal operating conditions has one pump operating. At least once per month, operation is rotated to another high pressure injection pump. This verifies that the high pressure injection pumps are operable.

The requirements of the Low Pressure Service Water System for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis verifies that two pumps are operable.

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

Testing the manual operability of power-operated valves in the Low Pressure Injection System gives assurance that flow can be established in a timely manner even if the capability to operate a valve from the control room is lost.

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

Power Operated Valves LP-17 and LP-18, are boundary valves between high pressure and low pressure design piping. As such, functional testing of these valves is performed during cold shutdown conditions when the Reactor Coolant System pressure is below the design pressure of the Low Pressure Injection System piping and the potential for over-pressurization of the low pressure system is eliminated. Check Valves CF-12, CF-14, LP-47, and LP-48 are located on the high pressure piping and therefore can be leak tested with the Reactor Coolant System at hot shutdown conditions.

<u>REFERENCE</u>

(1) FSAR, Section 6

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 99 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

1.0 Introduction

By letter dated September 8, 1981, Duke Power Company (Duke) requested a change to the format of Table 3.5.1-1 "Instruments Operating Conditions" to the format of the Standard Technical Specifications (STS). This administrative change would resolve interpretation problems which have existed. Also included in this request was a revision to the bases for TS 4.5 to remove incorrect wording.

Duke submitted an additional application by letter dated September 10, 1981. This application requested changes to Safety Limits and Reactor Protective System (RPS) setpoints to reflect the current calculated string errors. The newly calculated RPS errors were based on the Oconee Nuclear Station (ONS) Unit 3. Based on instrumentation comparisons, these errors were determined to bound those for ONS Units 1 and 2 for the maximum power and temperature setpoints. The remaining changes apply only to Unit 3; similar changes for Units 1 and 2 will be handled separately.

2.0 Discussion and Evaluation

2.1 Instrumentation Table Format Change

By letter dated September 8, 1981, Duke requested that the format of Table 3.5.1-1 "Instruments Operating Conditions" be changed to a format similar to the STS. This change is administrative in nature and was requested to eliminate confusion of the requirements.

We have reviewed the proposed change and have determined that the requirements remain unchanged and that only the format is revised. Since the requirements remain the same, we find this change acceptable.

Included in this request was a revised wording to the bases for specification 4.5 "Emergency Core Cooling Systems." The change corrects the wording contained in the last paragraph and is acceptable.

2.2 RPS String Errors

By letter dated September 10, 1981, Duke requested changes to TSs 2.1 "Safety Limits, Reactor Core" and 2.3 "Limiting Safety System Settings, Protective Instrumentation." These changes are necessary to reflect the current calculated string errors used in the determination of RPS setpoints. The effect of these errors requires more restrictive setpoints for some reactor trip functions.

The errors used in the determination of maximum nuclear power and reactor coolant temperature were calculated for Oconee 3 and comparisons of the instruments installed in Oconee 1 and 2 determined that the errors were bounded for these Units. Therefore, these new setpoints apply to all three units and since they are conservative, they have been implemented on Oconee Units 2 and 3 and will be implemented on Oconee Unit 1 prior to restart from the current refueling outage.

The changes to Section 2.1 and the Protective System Maximum Allowable Setpoints in Section 2.3 were calculated for Oconee Unit 3 and apply only to that Unit. Similar changes will be incorporated in the reload applications for Oconee Units 1 and 2.

We have reviewed these proposed changes and find: 1) the analysis procedures used are the same as those previously found acceptable for similar plants, 2) the changes are in the conservative direction, and 3) the changes are the same reductions found acceptable for other similar plants. Therefore, we conclude that these changes are acceptable.

3.0 Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: November 2, 1981

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 102 , 102 , & 99 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised the Technical Specifications (TSs) for operation of the Oconee Nuclear Station, Units Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of the date of issuance.

These amendments revise the TSs to reflect current calculated string errors used in determining the Reactor Protective System setpoints and upgrade the format of the Operational Safety Instrumentation Table.

The applications for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration