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September 19, 1985

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Dockets Nos. 50-269, 50-270
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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. 142 , 142 , and 139 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 May 31, 1985.

These amendments revise the TSs to support the operation of Oconee Unit 3 at full rated power during the upcoming Cycle 9. 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).

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

Sincerely,

ORIGINAL SIGNED BY:

Helen Nicolaras, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 142 to DPR-38
2. Amendment No. 142 to DPR-47
3. Amendment No. 139 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram	HNicolaras;cf	JStolz	S E Johnson	GLainas
9/3/85	9/3/85	9/5/85	9/10/85	9/13/85

Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

cc:
Mr. William L. Porter
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Liberman, Cook, Purcell & Reynolds
1200 Seventeenth Street, N.W.
Washington, D.C. 20036

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 33515

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Regional Administrator
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 3100
Atlanta, Georgia 30303

Mr. Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. Paul F. Guill
Duke Power Company
Post Office Box 33189
422 South Church Street
Charlotte, North Carolina 28242

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. 142
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 May 31, 1985, 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:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 142 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: September 19, 1985

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. 142
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 May 31, 1985, 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:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 142 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: September 19, 1985

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. 139
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 May 31, 1985, 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:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 139 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: September 19, 1985

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 142 TO DPR-38

AMENDMENT NO. 142 TO DPR-47

AMENDMENT NO. 139 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
2.1-3c	2.1-3c
2.1-3d	2.1-3d
2.1-9	2.1-9
2.3-3	2.3-3
2.3-10	2.3-10
2.3-13	2.3-13
3.2-1	3.2-1
3.2-2	3.2-2
3.5-17 (3 pages)	3.5-17 (1 page)
3.5-20 (3 pages)	3.5-20 (1 page)
3.5-23 (3 pages)	3.5-23 (1 page)
3.5-26 (2 pages)	3.5-26 (1 page)
3.5-29 (2 pages)	3.5-29 (1 page)

Bases - Unit 3

The safety limits presented for Oconee Unit 3 have been generated using the BAW-2 and BWC critical heat flux correlations[1,3] and the Reactor Coolant System flow rate at 106.5 percent of the design flow (design flow is 131.32 x 10^6 lbs/hr for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate[2].

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the CHF correlations[1,3]. The BAW-2 and BWC correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operation transients, and anticipated transients is limited to 1.30 (BAW-2) or 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; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1C represents the conditions at which a minimum allowable DNBR is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 139.86 X 10^6 lbs/hr). This curve is based on the following nuclear power peaking factors with potential fuel densification and fuel rod bowing effects:

$$F_q^N = 2.565; \; F_{\Delta H}^N = 1.71^{(3)} F_z^N = 1.50$$

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

The curves of Figure 2.1-2C are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing:

1. The combination of the radial peak, axial peak and position of the axial peak that yields no less than the CHF correlation limit.

2. The combination of radial and axial peak that causes central fuel melting of the hot spot. The limit is 20.15 kw/ft for fuel rod burnup less than or equal to 1,000 MWD/MTU and 21.2 kw/ft after 1,000 MWD/MTU.

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

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

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.

A B&W topical report discussing the mechanisms and resulting effects of fuel rod bow has been approved by the NRC[4]. The report concludes that the DNBR penalty due to rod bow is insignificant and unnecessary, because the power production capability of the fuel decreases with irradiation. Therefore, no rod bow DNBR penalty needs to be considered for thermal-hydraulic analyses.

The maximum thermal power for three-pump operation is 88.07 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.92 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions is 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 the CHF correlation limit or a local quality at the point of minimum DNBR less than the CHF correlation quality limit 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 combination shown in Figure 2.1-3C.

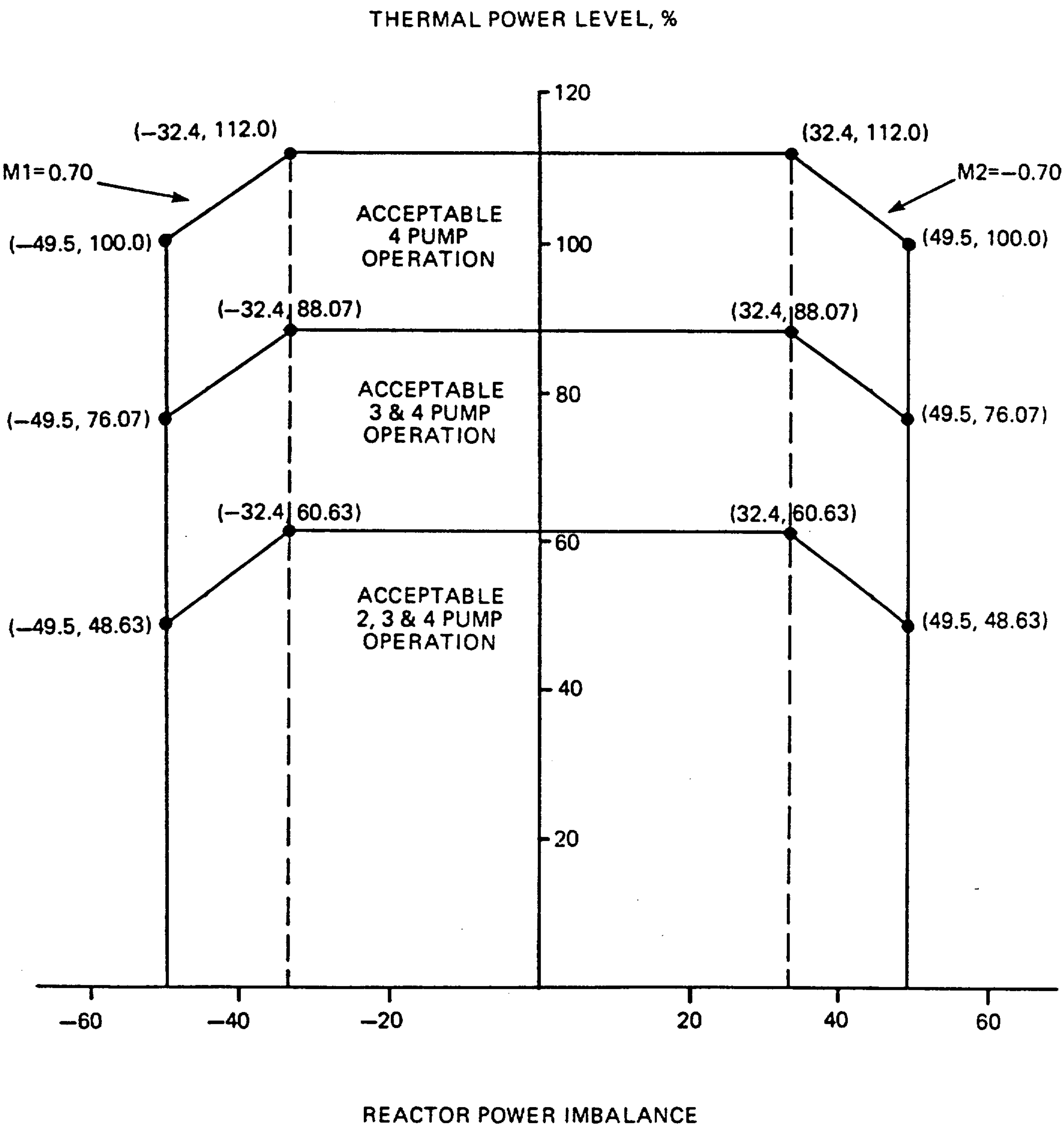
References

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

(4) Fuel Rod Bowing in Babcock & Wilcox Designs, BAW-10147P-A, Rev. 1, Babcock & Wilcox, May 1983.



CORE PROTECTION SAFETY LIMITS
Unit 3

OCONEE NUCLEAR STATION

FIGURE 2.1-2C

level trip and associated reactor power/reactor power-imbalance boundaries by 1.08% - Unit 1 for 1% flow reduction.
1.07% - Unit 2
1.07% - Unit 3

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below the minimum allowable value 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 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}$-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 to DNB
2.3-1B
2.3-1C
ratio greater than or equal to the minimum allowable value for those design accidents that result in a pressure reduction. (2,3)

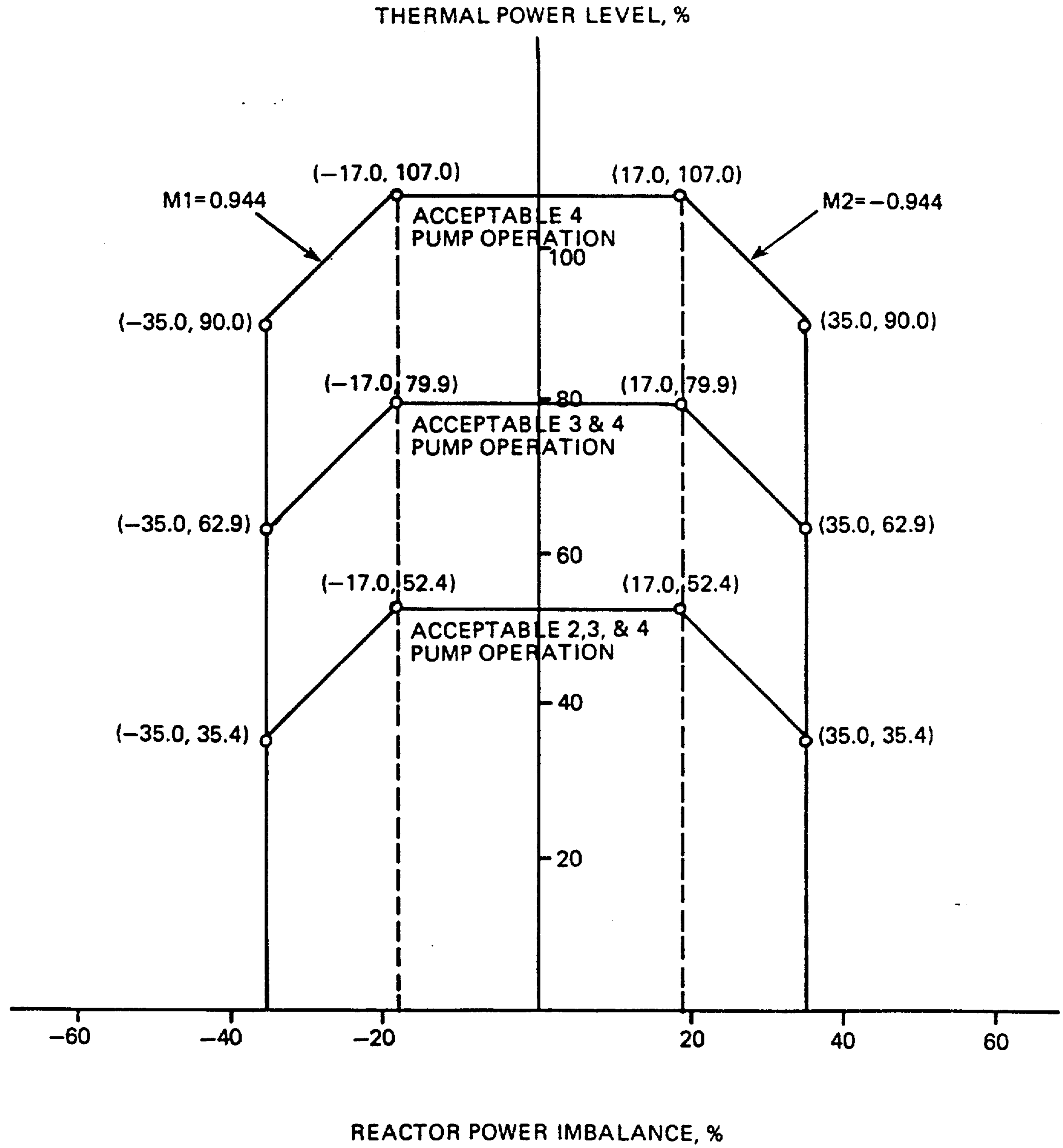
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)

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 3

OCONEE NUCLEAR STATION

FIGURE 2.3-2C

Amendments Nos. 142, 142, & 139

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)	105.5	105.5	105.5	5.0[(3)]
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	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.

3.2. HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS

Applicability

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

Objective

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

Specification

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

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

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

This source will be the concentrated boric acid storage tank containing at least the equivalent of 1020 ft^3 of 11,000 ppm boron as boric acid solution with a temperature at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall be operable and shall have the same temperature requirement as the concentrated boric acid storage tank. At least one channel of heat tracing capable of meeting the above temperature requirement shall be in operation. One associated boric acid pump shall be operable.

If the concentrated boric acid storage tank with its associated flowpath is unavailable, but the borated water storage tank is available and operable, the concentrated boric acid storage tank shall be restored to operability within 72 hours or the reactor shall be placed in a hot shutdown condition
margin equivalent to 1% $\Delta k/k$ at 200°F within the next twelve hours; if the concentrated boric acid storage tank has not been restored to operability within the next 7 days the reactor shall be placed in a cold shutdown condition within an additional 30 hours.

If the concentrated boric acid storage tank is available but the borated water storage tank is neither available nor operable, the borated water storage tank shall be restored to operability within one hour or the reactor shall be placed in a hot shutdown condition within 6 hours and in a cold shutdown condition within an additional 30 hours.

Bases

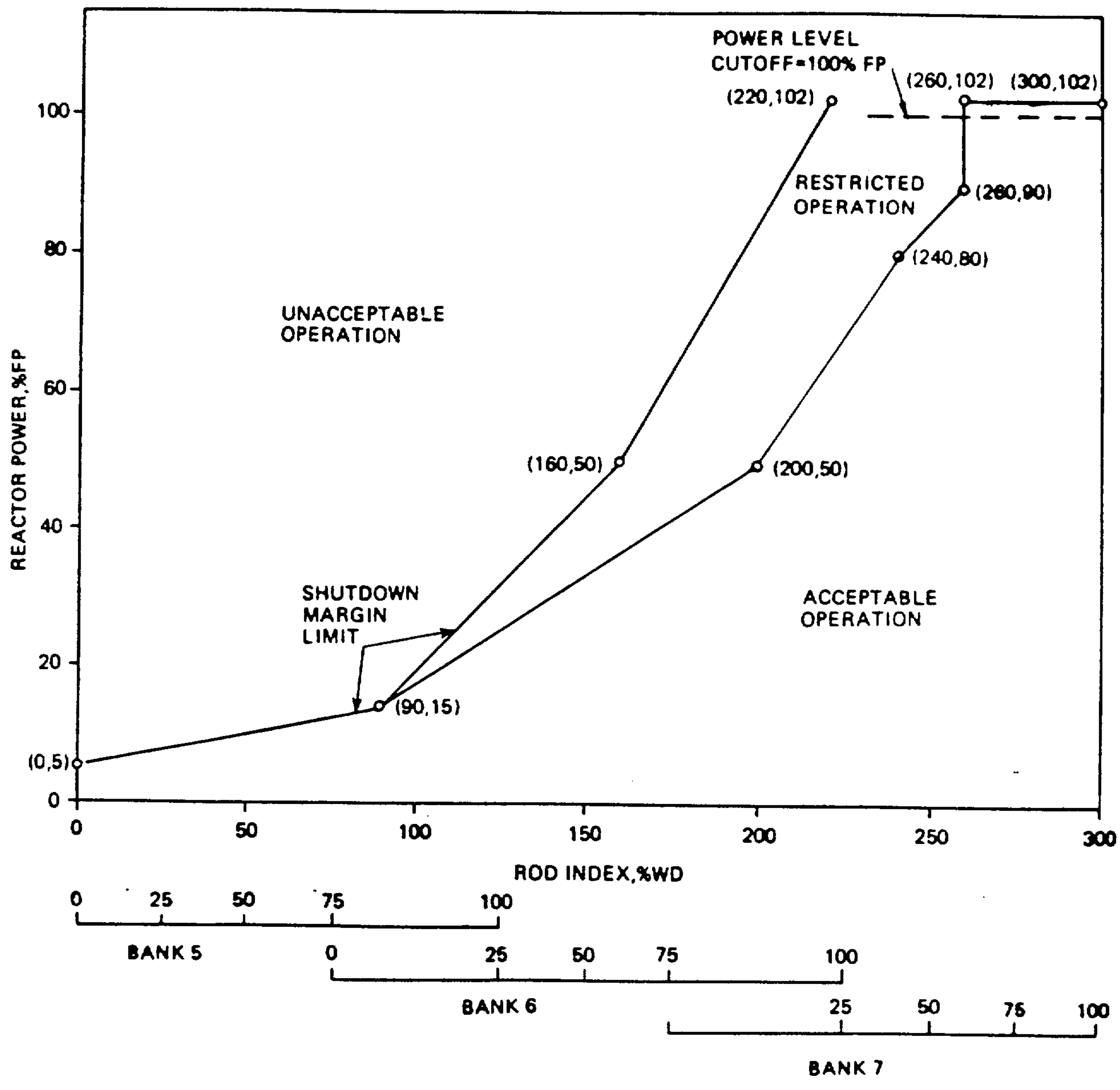
The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the concentrated boric acid storage tank. An alternate method of boration will be the use of the high pressure injection pumps taking suction directly from the borated water storage tank.(2)

The quantity of boric acid in storage in the concentrated boric acid storage tank or the borated water storage tank is sufficient to borate the reactor coolant system to a 1% $\Delta k/k$ subcritical margin at cold conditions (70°F) with the maximum worth stuck rod and no credit for xenon at the worst time in core life. The current cycles for each unit were analyzed with the most limiting case selected as the basis for all three units. Since only the present cycles were analyzed, the specifications will be re-evaluated with each reload. A minimum of 1020 ft^3 of 11,000 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1835 ppm boric acid in the borated water storage tank (3) will satisfy the requirements. The volume requirements include a 10% margin and, in addition, allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. Using only one 10 gpm boric acid pump taking suction from the concentrated boric acid storage tank would require approximately 12.7 hours to inject the required boron. An alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. The required boric acid can be injected in less than six hours using only one of the makeup pumps.

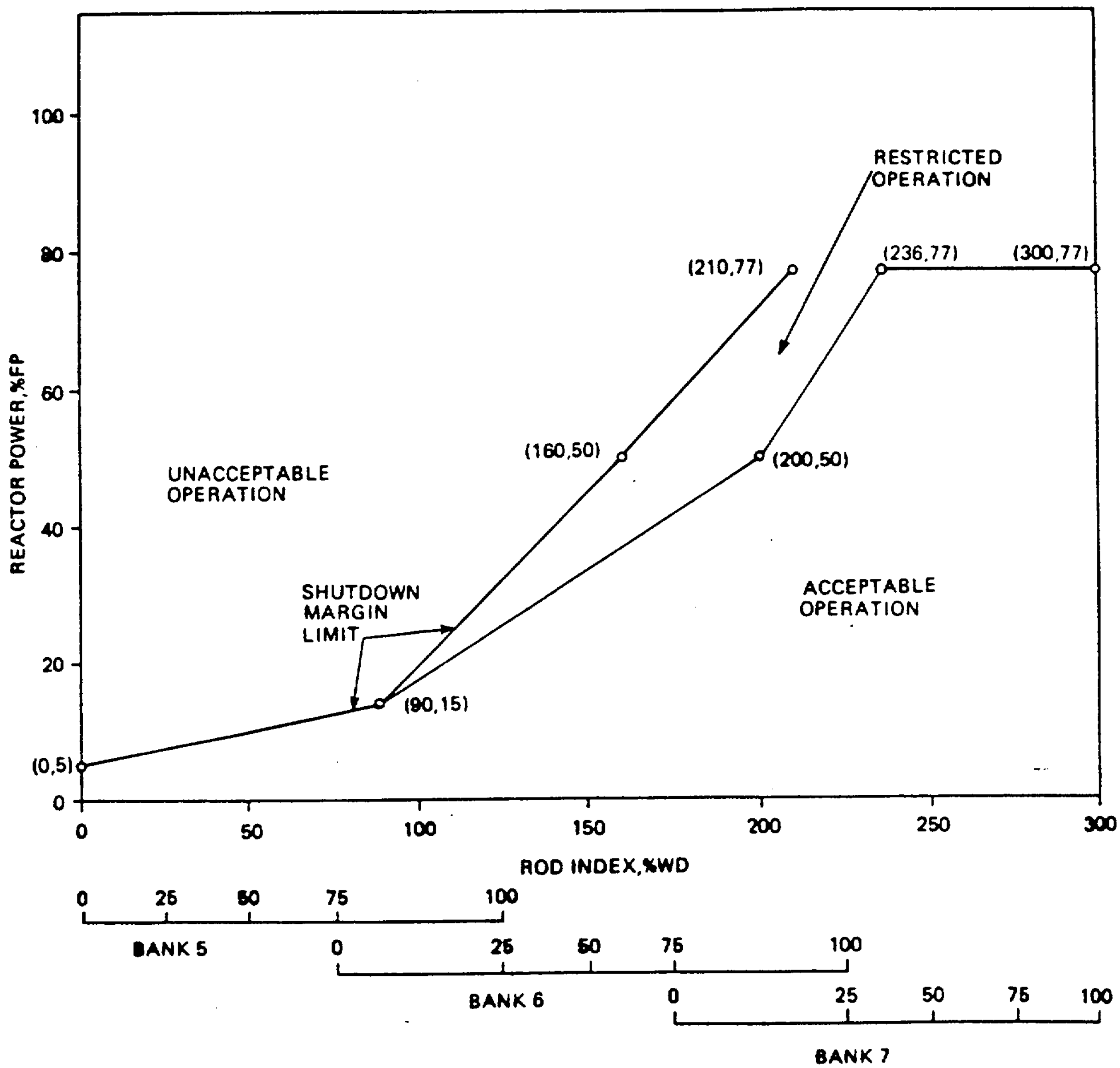
The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above the crystallization temperature for the concentration present. The boric acid concentration of 11,000 ppm in the concentrated boric acid storage tank corresponds to a crystallization temperature of 88°F and therefore a temperature requirement of 98°F. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

(1) FSAR, Sections 9.3.1, and 9.3.2
(2) FSAR, Figure 6.0.2
(3) Technical Specification 3.3



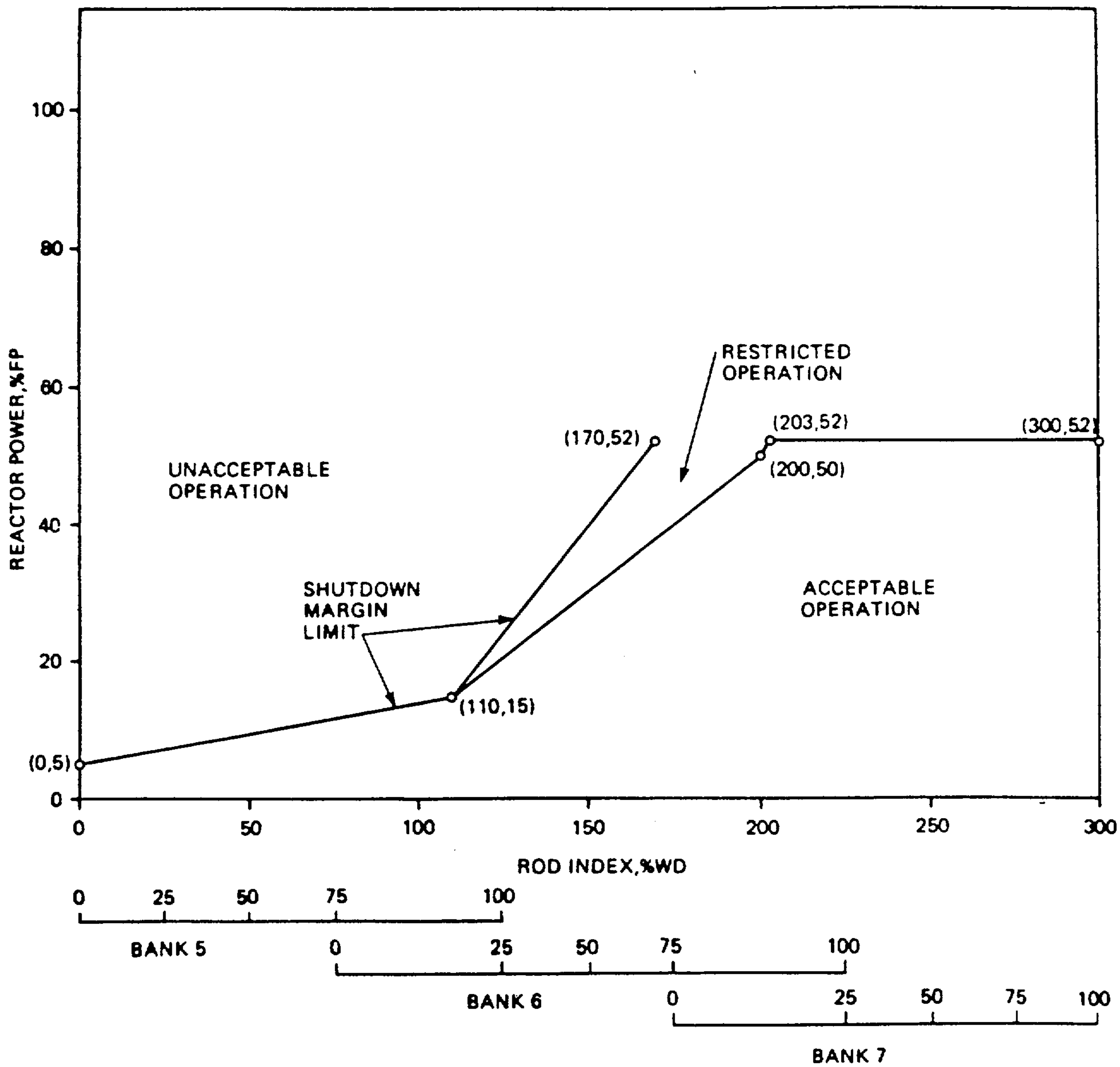
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 EFPD TO EOC
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3
(1 of 1)



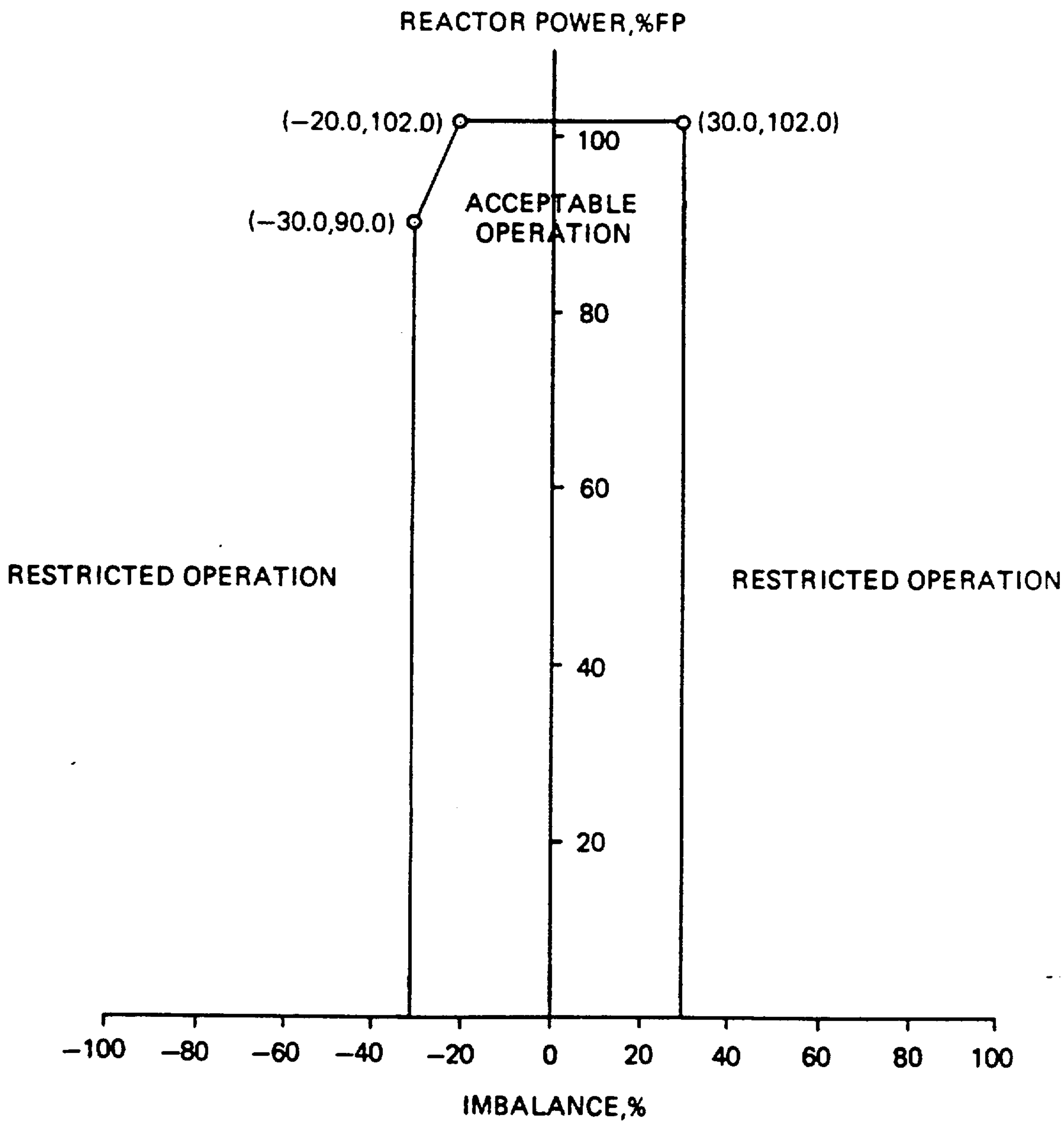
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 EFPD TO EOC
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-6
(1 of 1)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 EFPD TO EOC
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-9
(1 of 1)



OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 EFPD TO EOC
UNIT 3

OCONEE NUCLEAR STATION

FIGURE 3.5.2-12
(1 of 1)

Figure 3.5.2-15
(Deleted)

[Note that no rod position limits exist for Unit 3 axial power shaping rods.]

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 139 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 May 31, 1985 (Ref. 1), 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 3 at full rated power during the upcoming Cycle 9. 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).

To support the license amendment application, the licensee submitted a Duke Power Company report, DPC-RD-2005 (Ref. 2), "Oconee Unit 3, Cycle 9 Reload", as an attachment to Reference 1. A summary of the Cycle 9 operating parameters is included in the report, along with safety analyses. The fuel system design, the nuclear design, the thermal-hydraulic design, and the accident and transient analysis of this reload are presented in the Reference 2 report. An evaluation of this analysis and the proposed TS changes follows.

The Oconee Unit 3, Cycle 9 reload consists of 177 fuel assemblies, 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 average nominal fuel loading is 463.6 kg of uranium. The undensified nominal active fuel length is 141.8 inches; the initial mean density is 95% of theoretical. The fuel pellet outside diameter is .3686 inches and the initial enrichment is 3.22 w/o U-235. The new loading will contain 68 new fuel assemblies designated as Mark B5-Z, Batch 11. Cycle 9 will operate in a rods-out, feed-and-bleed mode. Reactivity control is supplied by soluble boron, full-length Ag-In-Cd control rods, burnable poison rod assemblies and Inconel axial power shaping rods. The design of Cycle 9 was compared to the design of Cycle 8.

EVALUATION

1.0 Evaluation of the Fuel System Design

1.1 Fuel Assembly Mechanical Design

Batch 11 uses intermediate spacer grids made of Zircaloy-4. All of the 68 new fuel assemblies are mechanically interchangeable into any core position. The Cycle 9 reload will include two regenerative neutron sources (built into the poison rod assemblies). The analysis methodology is that of Reference 3 which has been approved by the NRC staff.

1.2 Cladding Stress, Strain and Collapse

Fuel Batch 9B has been shown to be the most limiting for cladding creep collapse for Cycle 9 due to its longer previous incore exposure. The Batch 9B assembly power histories were analyzed and the most limiting assembly was used to perform the creep collapse analysis using the CROV code and the procedures described in Reference 4. The TACO2 (Ref. 5) code was used to calculate internal pin pressure and clad temperature used as input to CROV. The collapse time for the most limiting assembly was estimated to be 31,400 effective full power hours which is greater than the estimated residence time of 30,460 effective full power hours for Cycle 9.

The cladding stress was estimated in a conservative and generic manner as described in Reference 3 and in compliance with the provisions of Section III of the ASME Boiler and Pressure Vessel Code. Exception in the methodology of Reference 3 is the static stress analysis which complies with the requirements of ASME Code Article III-2000 for the static stress analysis. For the stress calculation, conservative cladding dimensions were assumed, combined with high external pressure (110% of design), low internal pressure and the maximum possible radial temperature gradient through the clad.

The strain was estimated using the TACO2 code (Ref. 5), and it demonstrated that the uniform circumferential strain of the cladding is within the limit of 1.0%.

Based on the above results for the cladding stress, strain and collapse, it was found that the cladding design is acceptable.

1.3 Fuel Thermal Design

The Cycle 9 fuel analysis was performed using the approved TACO2 code (Ref. 5). The design of the Batch 11 fuel, which is the new fuel in Cycle 9, is such as to be equivalent to the other batches present in Cycle 9 in the remainder of the core. Conservative parameters were used to determine for each fuel batch in the core the fuel melt limits. The maximum average assembly burnup was estimated to be 39,758 MWD/MTU and the maximum fuel rod burnup to be 40,912 MWD/MTU. The fuel rod internal pressure was evaluated using TACO2 and was found to be less than the nominal reactor coolant system pressure of 2,200 psi. The results of the fuel thermal design are acceptable.

2.0 Evaluation of the Nuclear Design

Cycle 9 differs from Cycle 8 in that there are now 68 new Mark B5-Z assemblies, the use of gray axial power shaping rods and the use of new control rod group patterns. The nuclear design calculations were carried out using the approved methods of Reference 3. The burnable poison rod assemblies are now being loaded in a different pattern as a result of the fuel assembly shuffle pattern; Cycle 9 is a transition cycle toward low leakage loadings. This affects also the power distribution and control rod worths. Analysis of the shutdown margin indicates that the minimum value is 2.74% Δ k/k compared to the required shutdown margin of 1.0% Δ k/k.

The results of the Cycle 9 physics analysis were found to be acceptable.

3.0 Evaluation of the Thermal-Hydraulic Design

The methods described in the Oconee Station Final Safety Analysis Report, the Oconee reload methodology (Ref. 3), the Unit 3 Cycle 8 reload report (Ref. 6) and the Oconee Fuel Densification Report (Ref. 7) were utilized in the thermal-hydraulic analysis of Unit 3 Cycle 9. Of the 68 new fuel assemblies in the Cycle 9 core, six have open guide tubes. Counting their contribution, the total core bypass flow is estimated to be 7.9% which, however, is less than the 8.2% assumed in the generic analysis.

The Mark BZ fuel assembly has a slightly higher pressure drop than the Mark B which constitutes the remainder of the core (109 assemblies). Therefore, the limiting hot channel of the Mark B assemblies will receive more coolant than a full Mark B core. The generic analyses based on the B&W - 2 critical heat flux correlation are bounding and applicable to the Cycle 9 core. The Mark BZ assemblies minimum departure from nucleate boiling ratio (DNBR) for the transition core is greater than 1.18 which is the BWC critical heat flux correlation limit (Ref. 8).

No fuel rod bow penalty was included in the DNBR limit used in the generic analysis. This was justified and approved in Reference 9.

The methods used in the analysis and evaluation of the Oconee 3 Cycle 9 loading have been previously approved. Based on the results, we find the thermal-hydraulic design of Cycle 9 acceptable.

4.0 Evaluation of the Accident and Transient Analysis

A generic loss of coolant accident (LOCA) analysis for the B&W 177 fuel assembly reactors has been performed using the final acceptance criteria in the emergency core cooling system (ECCS) evaluation model (Ref. 10). In this analysis, the limiting parameter values for all plants were used. The values of the fuel temperature (as a function of the linear heat rate) and the pin pressure calculated for the Oconee 3 Cycle 9 are conservative compared to the corresponding values of the generic analyses (Ref. 10). Therefore, the analysis and the LOCA limits reported in Reference 10 provide conservative results for Cycle 9. The lower pre-pressurization of the Batch 11 assemblies has a negligible effect on the LOCA analysis (Ref. 11). The theoretical density of the fuel in Batch 11 is higher than that considered in the densification report (Ref. 7). Finally, there was no need to recalculate doses because the estimates of Oconee 1 Cycle 9 are applicable to Oconee 3 Cycle 9 (Ref. 12).

From the review of the accident analyses for Oconee 3 Cycle 9 and on the bases of the parameters used with methods which have been previously approved, we conclude that the transient and accident analyses have been treated properly and are acceptable.

5.0 Evaluation of Technical Specification Changes

The changes discussed above necessitated Technical Specification changes to account for the differences in power peaking and control rod worths. The changes are such that neither the thermal design criteria nor the ECCS acceptance criteria are violated.

The following Specifications have been affected:

2.1 Safety Limits, Reactor Core

The modifications pertain to the value of the design flow (the Mark BZ assemblies have slightly higher hydraulic resistance), and the new radial and axial peaking values and the linear heat generation rates due to the use of the gray axial power shape rods. The revised Figure 2.1-2C specifies the acceptable limits of the thermal power level versus the reactor power imbalance. The analyses of the Mark BZ assemblies, the power distribution and the thermal-hydraulics have been performed with methods which have been approved and with a range of parameters which are acceptable.

2.3 Limit Safety System Settings, Protective Instrumentation

The power imbalance boundaries are established to prevent reactor thermal limits from being exceeded. The power imbalance affects the power level trip established by the power to flow ratio. The revised power level versus power imbalance limits are specified in Figure 2.3-2C. The power level and flow rates have been estimated using approved methods and acceptable parameter ranges.

Therefore, we find the specified power level versus power imbalance limits acceptable.

3.2 High Pressure Injection and Chemical Addition System

Due to the change in the control rod configuration, the boron solution concentration in the boric acid storage tank was changed to 11,000 parts per million (ppm). The nuclear analysis and estimation of this value was performed with approved codes and used an acceptable range of parameters. Therefore, we find that the proposed change will be adequate to bring the reactor to a cold shut-down condition and is acceptable.

3.5.2 Control Rod Group and Power Distribution Limits

The introduction of the gray axial power shaping control rods affected the control rod position limits for two, three, or four pump operation versus burnup. The rod position limits for the axial power shaping rods are no longer needed. The proposed control rod position limits are shown in Figures 3.5.2-3, -6, -9. The operational power imbalance envelope versus burnup is shown in Figure 3.5.2-12. The nuclear characteristics of Cycle 9 have been estimated with approved codes using acceptable ranges of parameters. Therefore, we find the control rod position limits and the operating power imbalance envelope acceptable.

6.0 Evaluation Findings

We have reviewed the fuels, physics, thermal-hydraulic and accident analysis information presented in the Oconee Unit 3, Cycle 9 reload report as stated above. We find the proposed reload and the associated modified Technical Specifications acceptable.

ENVIRONMENTAL CONSIDERATION

These amendments involve a change 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: September 19, 1985

Principal Contributor: L. Lois

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