Attachment 1

Duke Power Company
Oconee Nuclear Station

Proposed Technical Specification Revision
Oconee 3 Cycle 9

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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.

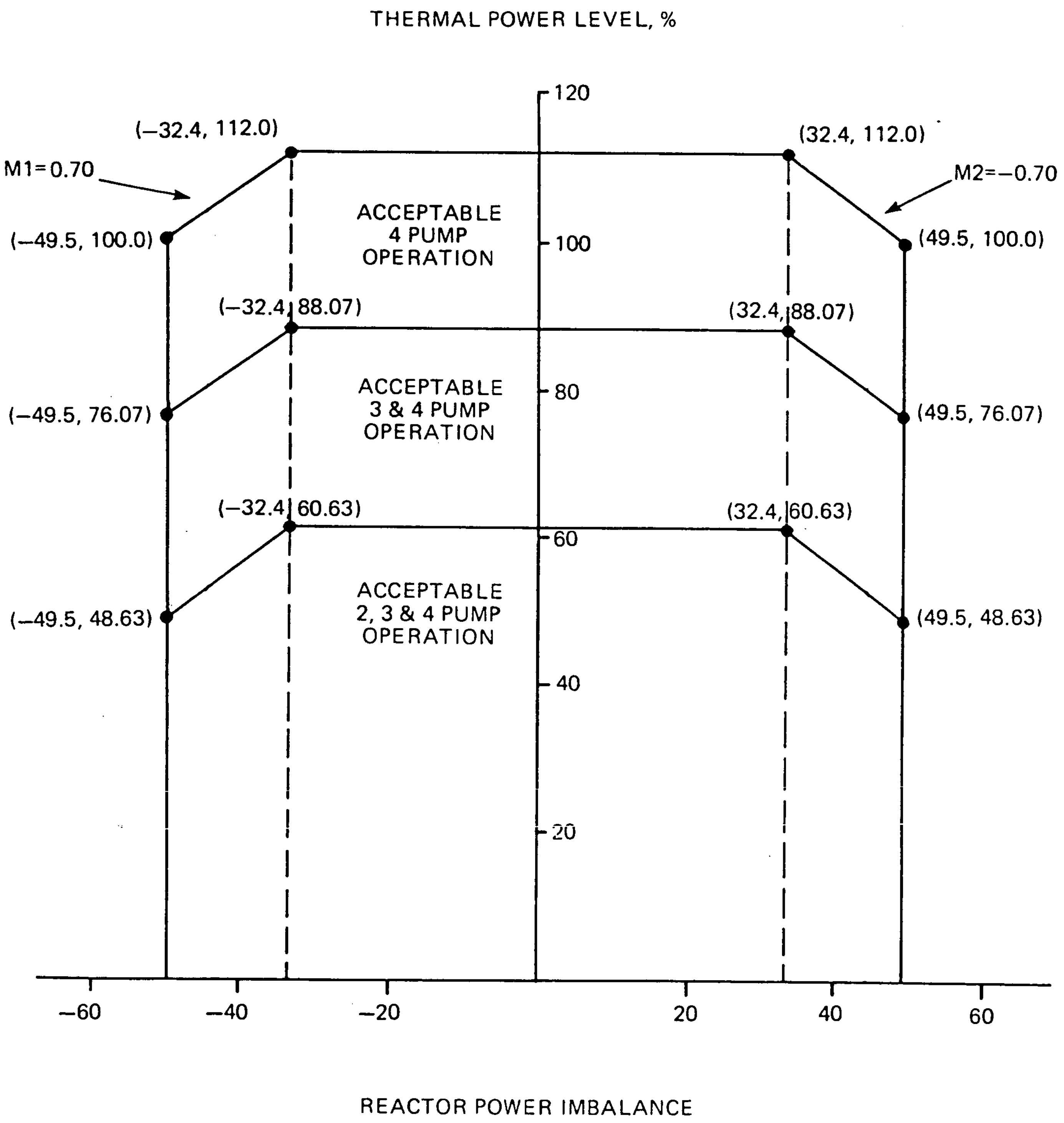
References

(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)

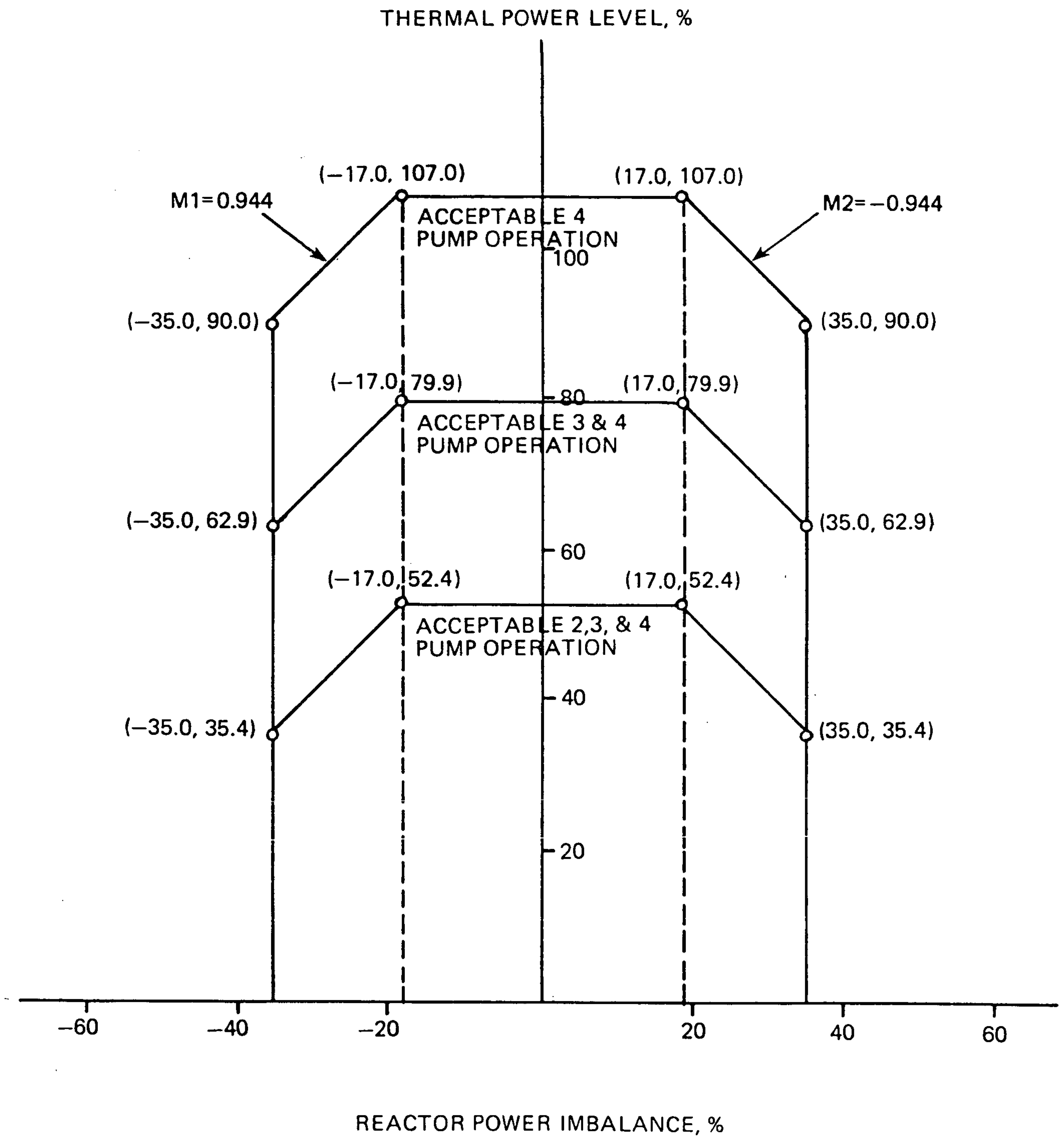
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)
($11.14\ T_{out}$ - 4746)

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

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 Imblance, (% 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. Bases 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 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 an be borated to a shutdown 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

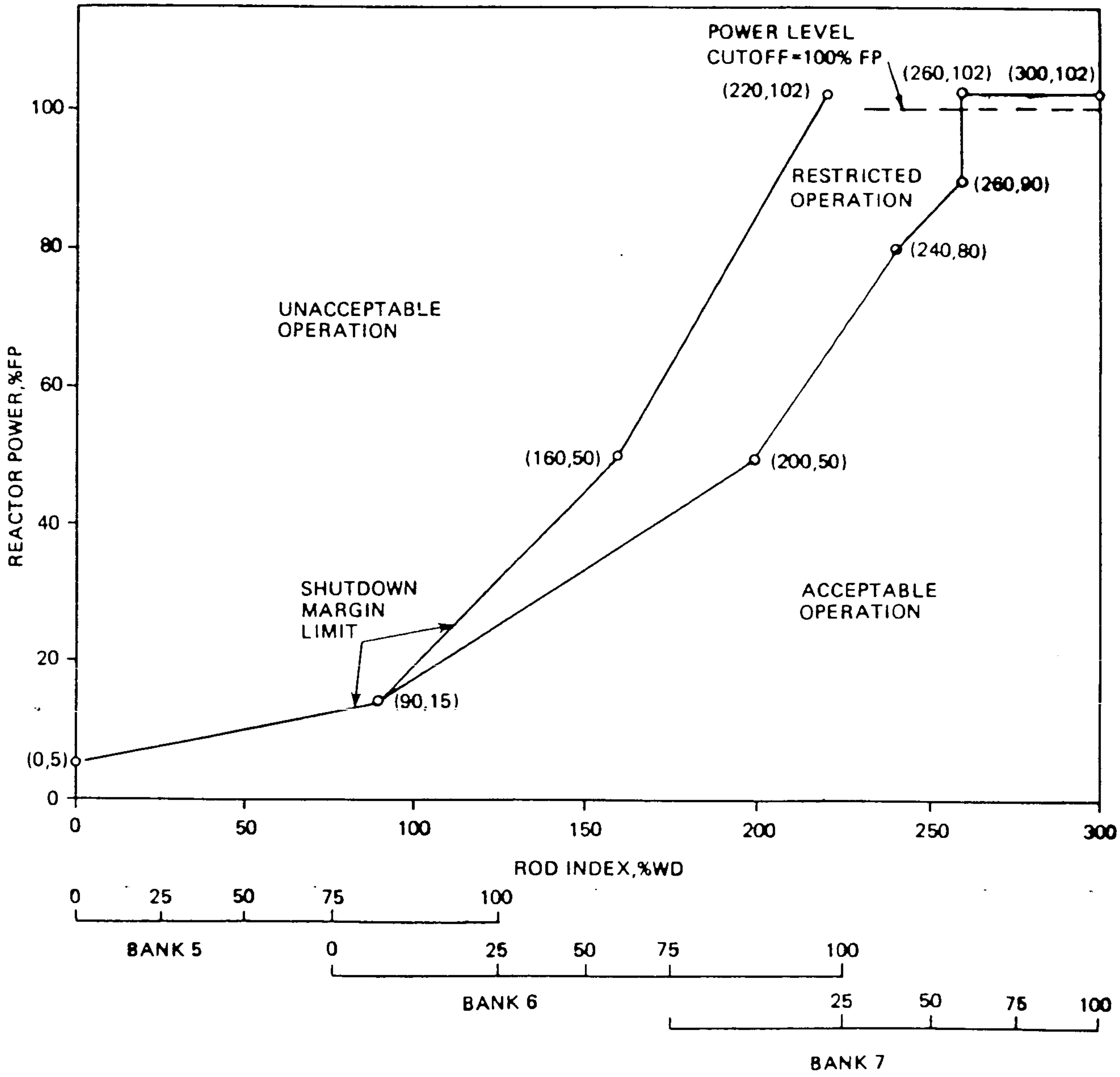
The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or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

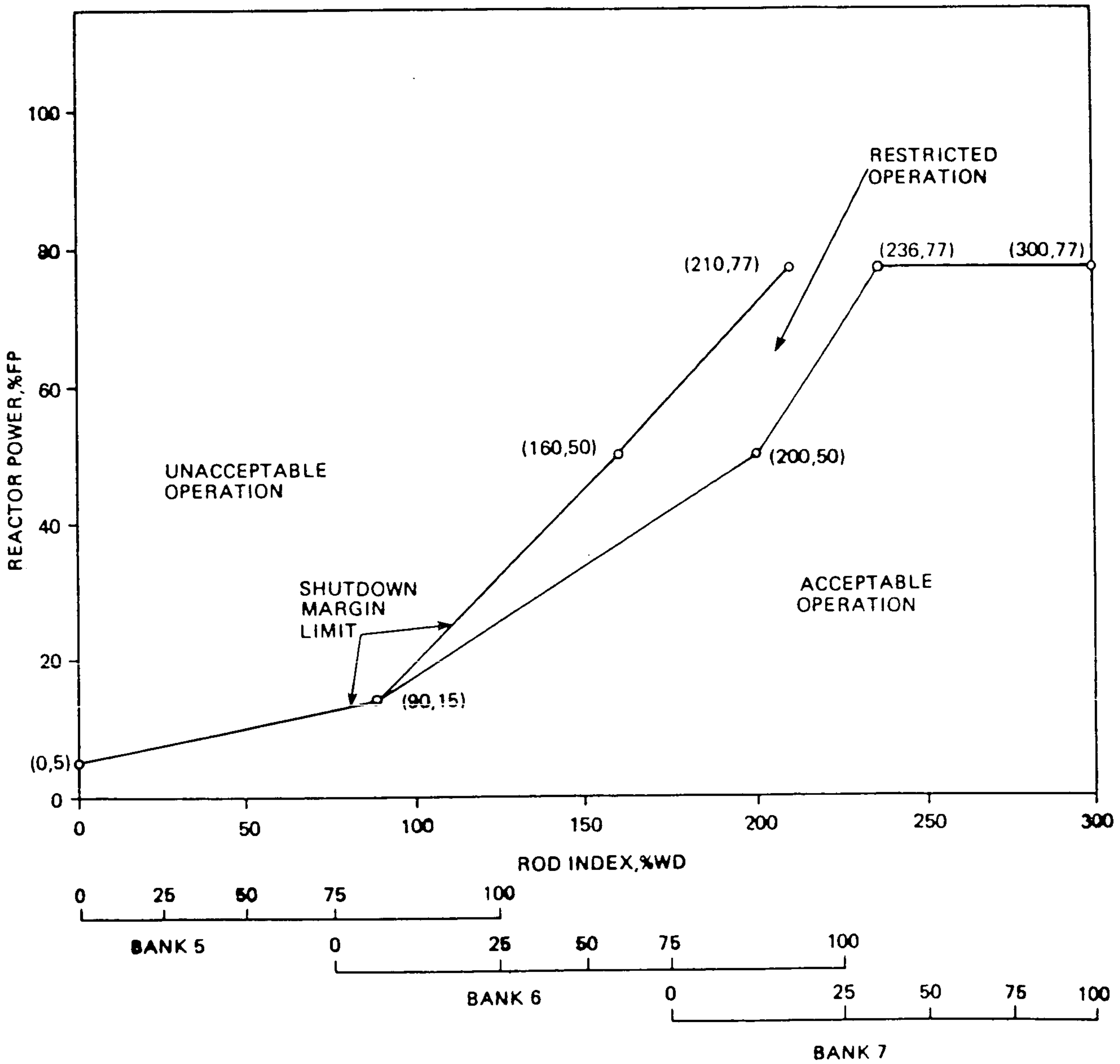
The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above 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, Section 9.1; 9.2
(2) FSAR, Figure 6.2
(3) Technical Specification 3.3



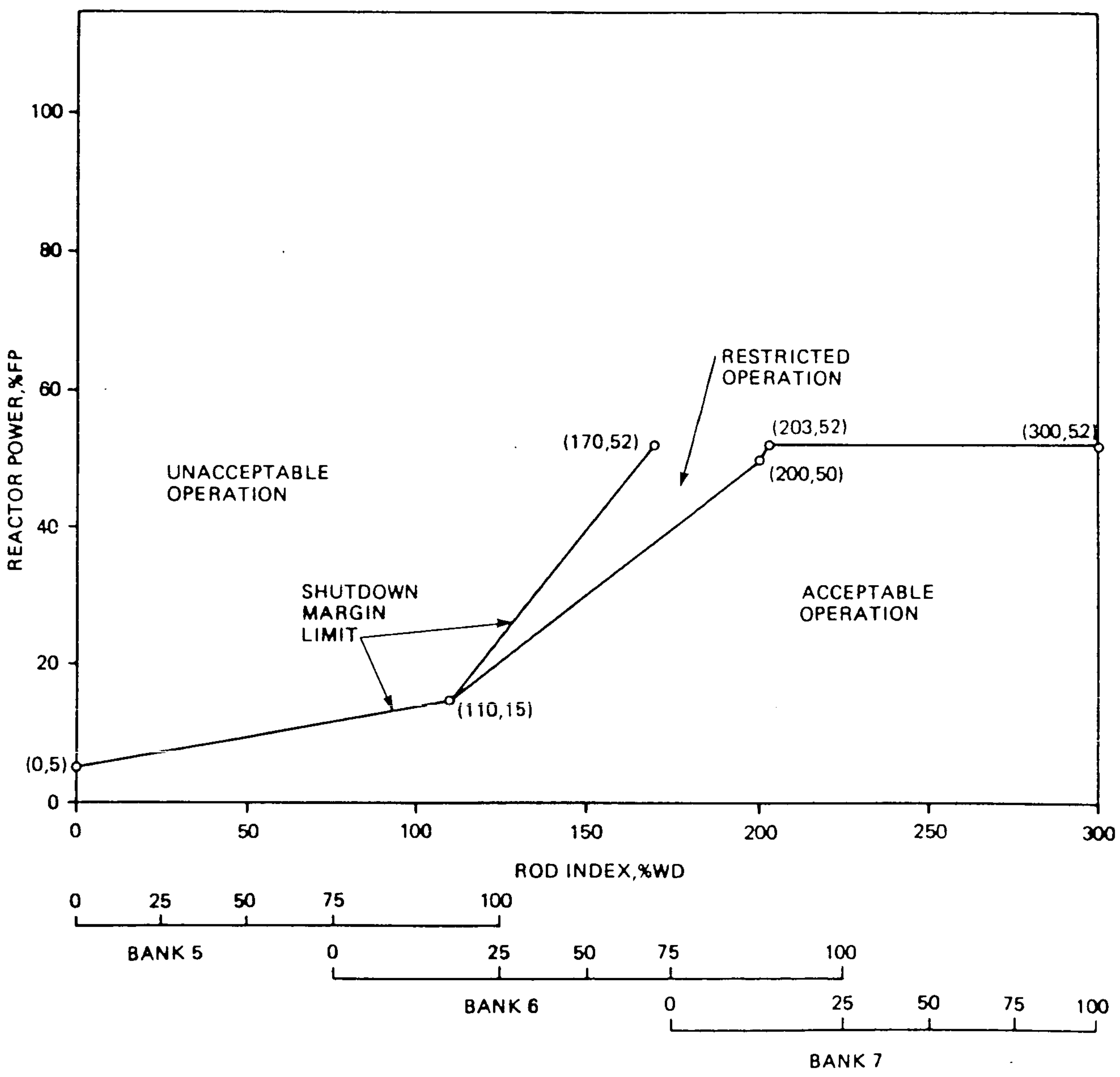
ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 EFPD TO EOC
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3
(1 of 1)



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

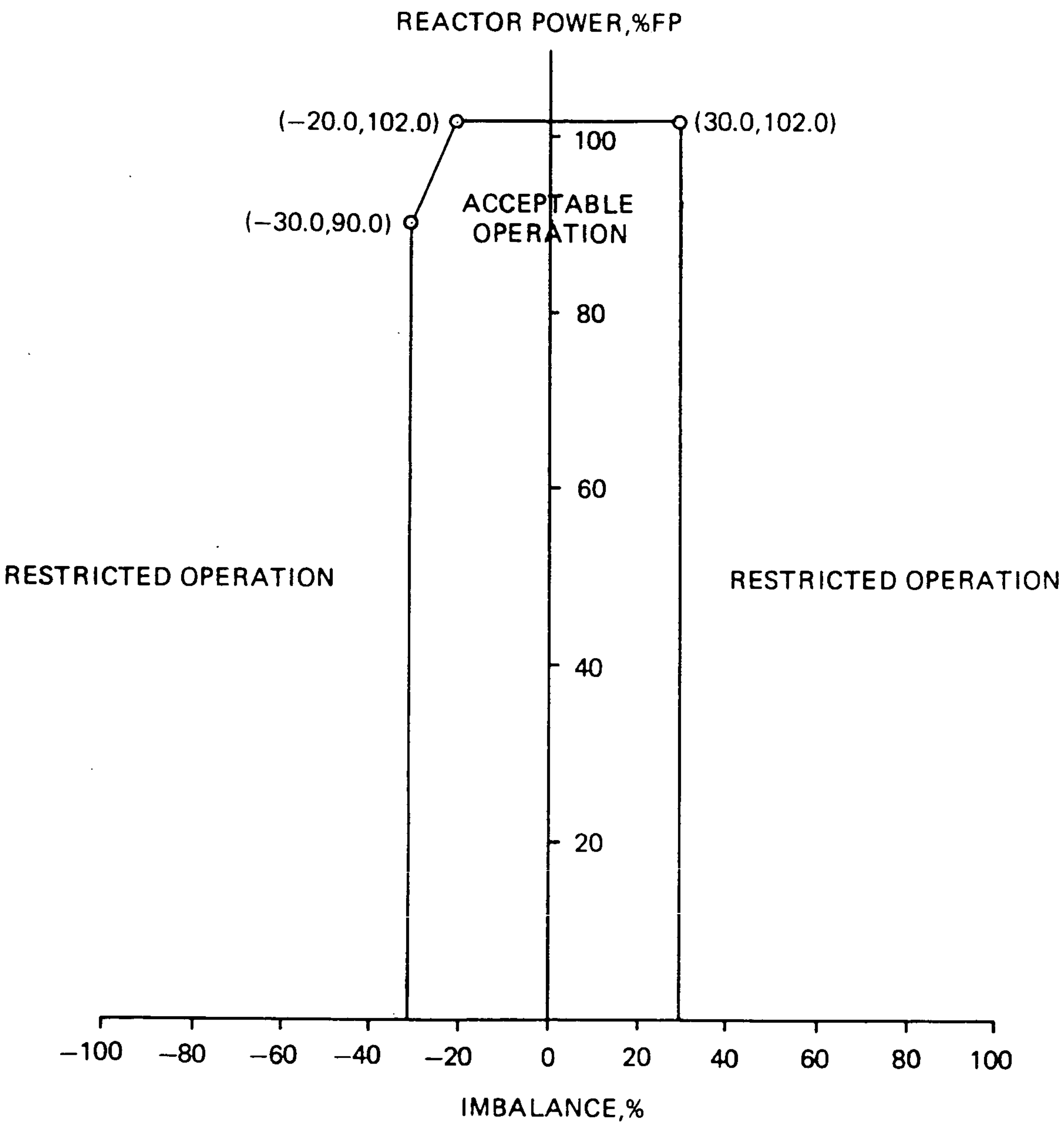
OCONEE NUCLEAR STATION

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.]

Attachment 2

Duke Power Company
Oconee Nuclear Station

No Significant Hazards Consideration Evaluation

NO SIGNIFICANT HAZARDS CONSIDERATION
EVALUATION FOR OCONEE 3 CYCLE 9 RELOAD

Duke Power has made the determination that this amendment request involves no significant hazards under the Commission's regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

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

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

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

Guidance has been supplied by the Commission concerning the application of these standards by providing certain examples (48 FR 14870). Example (iii) of the types of amendments not likely to involve significant hazards considerations applies in this case as the reload is for a nuclear power reactor. This assumes:

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

2) no significant changes are made to the acceptance criteria for the Technical Specifications; and

3) that the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed; and

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

This reload does not involve fuel assemblies significantly different from those found previously acceptable to the NRC. In this reload all of the 177 fuel assemblies to be inserted into the core are similar to fuel assemblies previously used and found acceptable by the NRC.

There are no significant changes to the acceptance criteria for the Technical Specifications. The Technical Specification revisions required for Cycle 9 operation were made in accordance with methods and procedures found acceptable through previously submitted reloads. The analytical methods used are consistent with the approved methodologies of the Oconee Nuclear Station Reload Design Methodology Technical Report (NFS-1001A, Duke Power Company, April 1984).

The Oconee 3 Cycle 9 reload report (Attachment 3) justifies the operation of the ninth cycle at the rated core power of 2568 MWt. Included in the report are the required analyses as outlined in the USNRC document "Guidance

Fuel System Design Consideration

The Fuel System Design consideration is described in Section 4 of Attachment 3. The most limited fuel assemblies, from the collapse of the clad standpoint are the batch 9B assemblies because of their longer incore exposure time. The batch 9B assembly power histories were analyzed, and the most limiting fuel assembly was used to perform creep collapse analysis. The analysis performed is based on the methods and procedures described in the B&W report, "Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse" (BAW-10084A, Rev. 2). The analysis conservatively predicts a collapse time of more than 31,400 EFPH which exceeds the maximum projected residence time of 30,460 EFPH.

The stress parameters for the Oconee 3 fuel rods were conservatively analyzed in accordance with the guidelines set forth in Section III, Division 1 - Subsection NB of the ASME Boiler pressure vessel code. Compliance with ASME code criteria verify the structural integrity of the cladding throughout the most limiting design conditions.

Analysis of the cladding strain in accordance with approved methodology (NFS-1001A) demonstrated that the uniform, circumferential strain of the cladding was within 1.0%.

All fuel in the Cycle 9 core is thermally similar. The incoming fresh batch 11 fuel inserted for Cycle 9 operation introduces no significant differences in fuel thermal performance relative to the fuel remaining in the core. The burnup dependent linear heat rate (LHR) capability and the average fuel temperature for each batch are shown in Table 4-2 of Attachment 3. The maximum assembly average burnup is predicted to be 39,758 MWD/MTU and the maximum fuel rod burnup is predicted to be 40,912 MWD/MTU. The maximum fuel rod internal pressure has been conservatively evaluated to be less than the nominal reactor coolant system pressure of 2200 psia.

Nuclear Design Consideration

The nuclear design analysis was performed using the methodology which has been accepted by the NRC (NFS-1001A). Section 5 of Attachment 3 describes the core physics parameters of Cycle 9 and a comparison to the physics parameters of Cycle 8. The primary reasons for the differences in the physics parameters between Cycles 8 and 9 are the different control rod pattern, BPRA loadings, shuffle patterns and the reduced number of Group 7 control rods and the introduction of gray APSRs.

The use of gray APSRs, the new control rod group pattern, and the introduction of 68 Mark BZ assemblies are new core design changes for Cycle 9. Gray APSRs, which are longer and use a weaker Inconel absorber, replace the silver-indium-cadmium APSRs used in all previous cycles. Calculations with the standard three-dimensional model verified that these APSRs provide adequate axial power distribution control.

The required shutdown margin is 1.00% ΔK/K. The shutdown margin at the beginning of the cycle and at the end of the cycle is 3.68%ΔK/K and 2.74 %ΔK/K, respectively. The calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod position limits. All safety criteria associated with these rod worths are met. Thus, the analysis shows that the Cycle 9 design meets all criteria including those applicable to radial power peaking, ejected rod worths and shutdown margin.

Thermal-Hydraulic Design Consideration

The generic Mark-B and Mark-BZ thermal-hydraulic design supporting Cycle 9 operation were performed using the methods and models described in Section 6 of Attachment 3, which have been found to be acceptable by the NRC through previously submitted reloads.

The Cycle 9 transition core includes 68 fresh Mark-BZ Batch 11 fuel assemblies, 60 of which will contain BPRAs. Two assemblies will contain regenerative neutron sources, leaving 46 fuel assemblies with open guide tubes. This results in a conservative bypass flow of 7.9% of the total system flow. A description of thermal-hydraulic design of the Mark-BZ fuel is given in Section 6 of Attachment 3.

A flux to flow setpoint of 1.07 is used for operation of Cycle 9. The minimum DNBR determined in the Mark-BZ transition core flux to flow analysis is greater than the BWC CHF correlation limit of 1.18 (BWC Correlation, BAW-10143, Part 2, Babcock & Wilcox, March 1980). The minimum DNBR for Mark-B fuel has been determined to be greater than the BAW-2 CHF correlation limit of 1.30. All other plant operating limits based on DNBR criteria include a minimum of 10.2% margin from the 1.30 design limit.

One can conclude from the examination of these sections, and the Cycle 9 core thermal and kinetics properties that this core reload will not significantly reduce the ability of the Oconee 3 unit to operate safely during Cycle 9.

In summary, Duke has determined and submits that the proposed reload described herein does not involve a significant safety hazard.

Attachment 3

Duke Power Company
Oconee Nuclear Station

Oconee Unit 3, Cycle 9 Reload Report