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REGULATORY DOCKET FILE COPY

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

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

The Commission has issued the enclosed Amendments Nos. 81, 81, and 78 for 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 and are in response to your request dated August 6, 1979, as supplemented August 22, 1979, December 31, 1979, and January 28, 1980.

These amendments revise the Technical Specifications to support the operation of Oconee Unit No. 1 at full rated power during Cycle 6. The amendments also revise the Technical Specifications for Units Nos. 1, 2, and 3 in regard to engineered safety features.

Oconee Unit No. 1, during Cycle 5, was operating under an October 23, 1978 Exemption to 10 CFR 50.46, the Emergency Core Cooling System (ECCS) rule. The enclosed Safety Evaluation and our letter of December 13, 1978, provide the bases for terminating the Exemption, as your ECCS modifications and operating procedures have met the provisions of the Exemption as confirmed by your letter of January 28, 1980.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Original signed by
Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 81 to DPR-38
2. Amendment No. 81 to DPR-47
3. Amendment No. 78 to DPR-55
4. Safety Evaluation
5. Notice of Issuance

cc w/enclosures: See next page

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

February 22, 1980

Dockets Nos. 50-269
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cc w/enclosures: See next page

Duke Power Company

cc w/enclosure(s):

Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

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

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Mr. Francis Jape
U. S. Nuclear Regulatory Commission
P. O. Box 7
Seneca, South Carolina 29678

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

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

cc w/enclosure(s) and incoming dtd:
8/6/79, 8/22/79, 12/31/79, & 1/28/80

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

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. 81
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 August 6, 1979, as supplemented August 22, 1979, December 31, 1979, and January 28, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 81 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8003060692

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: February 22, 1980

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. 81
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 August 6, 1979, as supplemented August 22, 1979, December 31, 1979, and January 28, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 81 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8003060699

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: February 22, 1980

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. 78
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 August 6, 1979, as supplemented August 22, 1979, December 31, 1979, and January 28, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 78, 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

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: February 22, 1980

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 81 TO DPR-38

AMENDMENT NO. 81 TO DPR-47

AMENDMENT NO. 78 TO DPR-55

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

Revise Appendix A as follows:

Remove the following pages and insert the revised identically numbered pages.

ii

2.1-2 & 2.1-3

2.1-7

2.3-2 & 2.3-3

2.3-8

2.3-11

3.1-4

3.2-2

3.3-1 - 3.3-7*

3.5-15 & 3.5-15a

3.5-18 & 3.5-18a

3.5-21 & 3.5-21a

3.5-24 & 3.5-24a

4.1-1 & 4.1-2

The new pages and changes on the revised pages are identified by marginal lines.

* Pages 3.3-5 - 3.3-7 are new pages

can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 correlation has 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 operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 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 2.1-1A represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 106.5 percent of 131.3 x 10^6 lbs/hr.). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, which result in a more conservative DNBR than any other shape that exists during normal operation.

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

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.05 kw/ft for Unit 1. (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 2.1-2A correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1A is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.

The maximum thermal power for three-pump operation is 87.2 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 3). The maximum thermal power for other coolant pump conditions are produced in a similar manner.

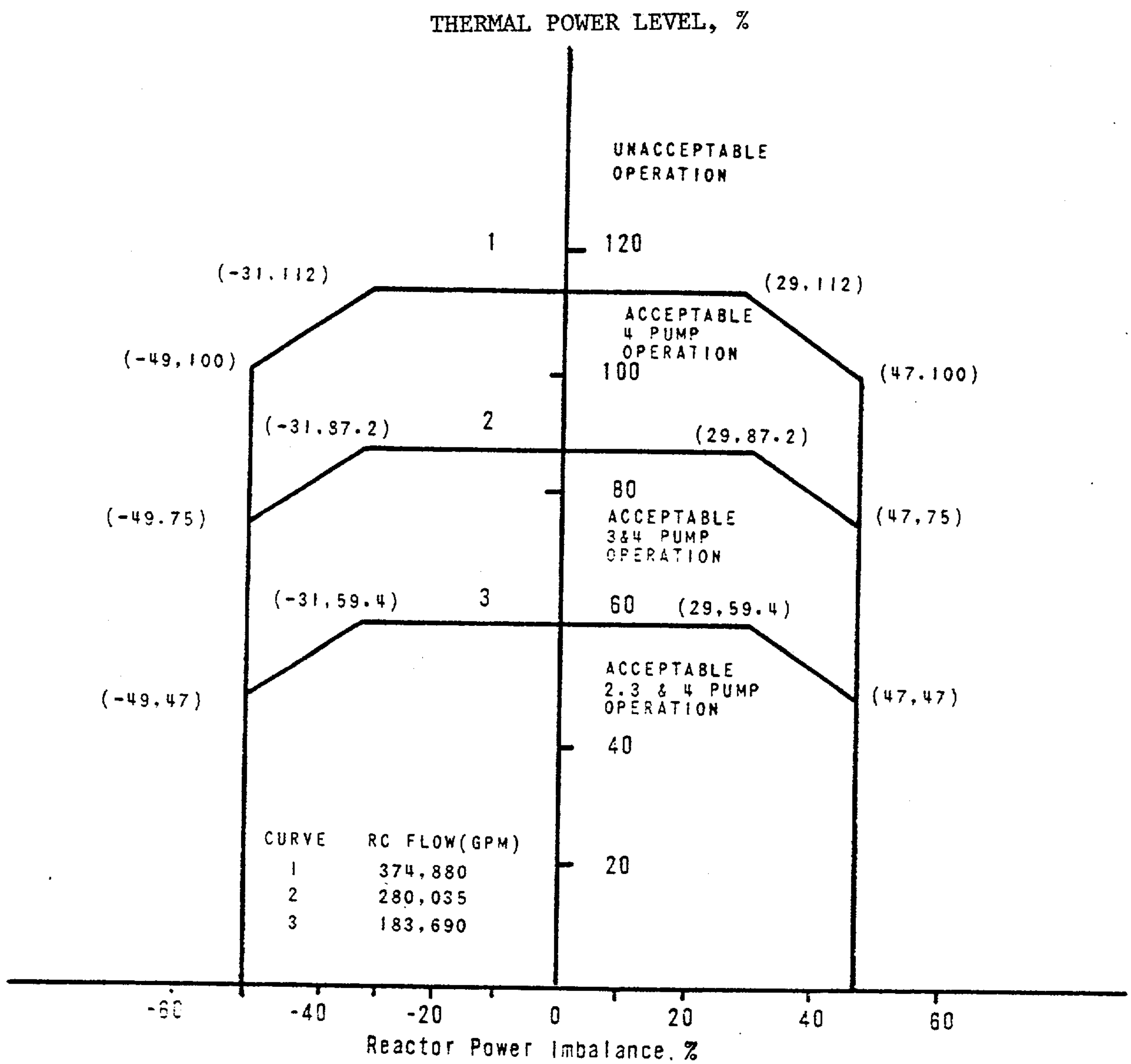
For Figure 2.1-3A, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30.

References

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

(2) Oconee 1, Cycle 4 - Reload Report - BAW-1447, March, 1977.

(3) Oconee 1, Cycle 6 - Reload Report - BAW-1552, July, 1979.



CORE PROTECTION
SAFETY LIMITS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.1-2A

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% 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 92.6% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.7% and reactor flow rate is 74.7% or flow rate is 69.4% 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 52.9% and reactor flow rate is 49.0% or flow rate is 45.4% 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.
1.055% - 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 (2355 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 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)

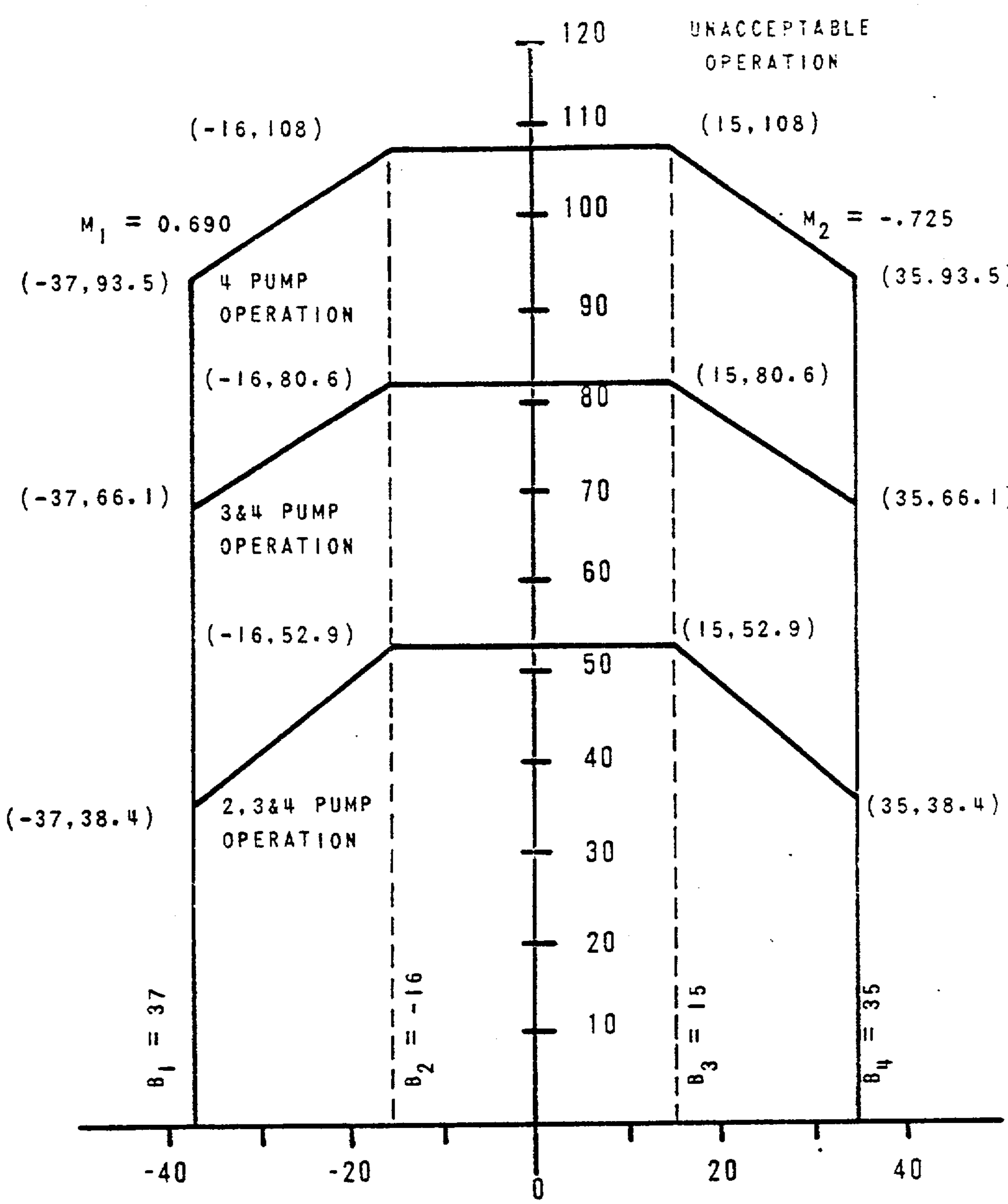
Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (619°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.

THERMAL POWER LEVEL, %



Reactor Power Imbalance, %

PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.3-2A

Amendments Nos. 81, 81, & 78

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)	105.5	105.5	105.5	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.	2355	2355	2355	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14T_{out}-4706)$(1)	$(11.14T_{out}-4706)$(1)	$(11.14T_{out}-4706)$(1)	Bypassed
7.	Reactor Coolant Temp. F., Max.	619	619	619	619
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.

Bases - Units 1, 2 and 3

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, startup and shutdown operations, and inservice leak and hydrostatic tests. The various categories of load cycles used for design purposes are provided in Table 4.8 of the FSAR.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1436[1], BAW-1437[2] and BAW-1438[3].

The Figures specified in 3.1.2.1, 3.1.2.2 and 3.1.2.3 present the pressure-temperature limit curves for normal heatup, normal cooldown and hydrostatic tests respectively. The limit curves are applicable up to the indicated effective full power years of operation. These curves are adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations.

The cooldown limit curves are not applicable to conditions of off-normal operation (e.g., small LOCA and extended loss of feedwater) where cooling is achieved for extended periods of time by circulating water from the HPI through the core. If core cooling is restricted to meet the cooldown limits under other than normal operation, core integrity could be jeopardized.

The pressure-temperature limit lines shown on the figure specified in 3.1.2.-1 for reactor criticality and on the figure referred to in 3.1.2.3 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region, or in test reactors.

The limitation on steam generator pressure and temperature provide protection against nonductile failure of the secondary side of the steam generator. At metal temperatures lower than the RT_{NDT} of +60°F, the protection against nonductile failure is achieved by limiting the secondary coolant pressure to 20 percent of the preoperational system hydrostatic test pressure. The limitations of 110°F and 237 psig are based on the highest estimated RT_{NDT} of +40°F and the preoperational system hydrostatic test pressure of 1312 psig. The average metal temperature is assumed to be equal to or greater than the coolant temperature. The limitations include margins of 25 psi and 10°F for possible instrument error.

The spray temperature difference is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

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, Oconee 1 Cycle 6, Oconee 2 Cycle 4, and Oconee 3 Cycle 5 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 reevaluated with each reload. A minimum of 995 ft^3 of 8,700 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallons of 1800 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 12.25 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 8,700 ppm in the concentrated boric acid storage tank corresponds to a crystallization temperature of 77°F and therefore a temperature requirement of 87°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

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING COOLING, REACTOR BUILDING SPRAY, AND LOW PRESSURE SERVICE WATER SYSTEMS

Applicability

Applies to the emergency core cooling, reactor building cooling, reactor building spray, and low pressure service water systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building cooling, reactor building spray and low pressure service water systems.

Specification

3.3.1 High Pressure Injection (HPI) System

a. Prior to initiating maintenance on any component of the HPI system, the redundant component shall be tested to assure operability.

b. When the reactor coolant system (RCS), with fuel in the core, is in a condition with temperature above 350°F and reactor power less than 60% FP:

(1) Two independent trains, each comprised of an HPI pump and a flowpath capable of taking suction from the borated water storage tank and discharging into the reactor coolant system automatically upon Engineered Safeguards Protective System (ESPS) actuation (HPI segment) shall be operable.

(2) Test or maintenance shall be allowed on any component of the HPI system provided one train of the HPI system is operable. If the HPI system is not restored to meet the requirements of Specification 3.3.1.b(1) above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.1.b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS temperature below 350°F within an additional 24 hours.

c. For Unit 2, when reactor power is greater than 60% FP:

(1) In addition to the requirements of Specification 3.3.1.b(1) above, the remaining HPI pump shall be operable and valves HP-99 and HP-100 shall be open.

(2) HPI Pump Operability

(a) Tests or maintenance shall be allowed on any one HPI pump, provided two trains of HPI system are operable.

(b) If the inoperable HPI pump is not restored to operable status within 72 hours, reactor power shall be reduced below 60% FP within an additional 12 hours.

(3) HPI Flowpath Operability

(a) If one automatic HPI flowpath becomes inoperable, then either restore the inoperable flowpath to operable status within one hour, or reactor power shall be reduced to below 60% FP within an additional 2 hours.

d. For Units 1 and 3, when reactor power is greater than 60% FP:

(1) In addition to the requirements of Specification 3.3.1.b(1) above, the remaining HPI pump and valves 3HP-409 and 3HP-410 shall be operable and valves HP-99 and HP-100 shall be open.

(2) Tests or maintenance shall be allowed on any component of the HPI system, provided two trains of HPI system are operable. If the inoperable component is not restored to operable status within 72 hours, reactor power shall be reduced below 60% FP within an additional 12 hours.

3.3.2 Low Pressure Injection (LPI) System

a. Prior to initiating maintenance on any component of the LPI system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

(1) Two independent LPI trains, each comprised of an LPI pump and a flowpath capable of taking suction from the borated water storage tank and discharging into the RCS automatically upon ESPS actuation (LPI segment), together with two LPI coolers and two reactor building emergency sump isolation valves (manual or remote-manual) shall be operable.

(2) Tests or maintenance shall be allowed on any component of the LPI system provided the redundant train of the LPI system is operable. If the LPI system is not restored to meet the requirements of Specification 3.3.2.b(1) above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.2.b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

3.3.3 Core Flood Tank (CFT) System

When the RCS is in a condition with pressure above 800 psig both CFT's shall be operable with the electrically operated discharge valves open and breakers locked open and tagged; a minimum level of 13 ± .44 feet (1040 ± 30 ft.^{3}) and one level instrument channel per CFT; a minimum concentration of borated water in each CFT of 1,800 ppm boron; and pressure at 600 ± 25 psig with one pressure instrument channel per CFT.

3.3.4 Borated Water Storage Tank (BWST)

When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

a. The BWST shall have operable two level instrument channels.

(1) Tests or maintenance shall be allowed on one channel of BWST level instrumentation provided the other channel is operable.

(2) If the BWST level instrumentation is not restored to meet the requirements of Specification 3.3.4.a above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.4.a are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

b. The BWST shall contain a minimum level of 46 feet of water having a minimum concentration of 1,800 ppm boron at a minimum temperature of 40°F. The manual valve, LP-28, on the discharge line shall be locked open. If these requirements are not met, the BWST shall be considered unavailable and action initiated in accordance with Specification 3.2.

3.3.5 Reactor Building Cooling (RBC) System

a. Prior to initiating maintenance on any component of the RBC system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical:

(1) Two independent RBC trains, each comprised of an RBC fan, associated cooling unit, and associated ESF valves shall be operable.

(2) Tests or maintenance shall be allowed on any component of the RBC system provided one train of the RBC and one train of the RBS are operable. If the RBC system is not restored to meet the requirements of Specification 3.3.5.b(1) above within 24 hours, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

c. When the reactor is critical:

(1) In addition to the requirements of Specifications 3.3.5.b(1) above, the remaining RBC fan, associated cooling unit, and associated ESF valves shall be operable.

(2) Tests or maintenance shall be allowed on one RBC train under either of the following conditions:

(a) One RBC train may be out of service for 24 hours.

(b) One RBC train may be out of service for 7 days provided both RBS trains are operable.

(c) If the inoperable RBC train is not restored to meet the requirements of Specification 3.3.5.c(1) within the time permitted by Specification 3.3.5.c(2)(a) or (b), the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.5.c(1) are not met within an additional 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

3.3.6 Reactor Building Spray (RBS) System

a. Prior to initiating maintenance on any component of the RBS system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical:

(1) One RBS train, comprised of an RBS pump and a flowpath capable of taking suction from the LPI system and discharging through the spray nozzle header automatically upon ESPS actuation (RBS segment) shall be operable.

(2) Tests or maintenance shall be allowed on any component of the RBS system under the following conditions:

(a) One RBS train may be out of service for 24 hours provided two RBC train are operable.

(b) If the inoperable RBS train is not restored to meet the requirements of Specification 3.3.6.b(1) within 24 hours, the reactor shall be placed in a condition with the RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

c. When the reactor is critical:

(1) In addition to the requirements of Specification 3.3.6.b(1) above, the other RBS train comprised of an RBS pump and a

flowpath capable of taking suction of the LPI system and discharging through the spray nozzle header automatically upon ESPS actuation (RBS segment) shall be operable.

(2) Tests or maintenance shall be allowed on one RBS train under either of the following conditions:

(a) One RBS train may be out of service for 24 hours.

(b) One RBS train may be out of service for 7 days provided all three RBC trains are operable.

(c) If the inoperable RBS train is not restored to meet the requirements of Specification 3.3.6.c(1) above within the time permitted by Specification 3.3.5.c.(2)(a) or (b), the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.6.c(1) are not met within an additional 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

3.3.7 Low Pressure Service Water (LPSW)

a. Prior to initiating maintenance on any component of the LPSW system, the redundant component shall be tested to assure operability.

b. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

(1) Two LPSW pumps for the shared Unit 1, 2 LPSW system and two LPSW pumps for the Unit 3 LPSW system shall be operable with valves LPSW-108, 2LPSW-108, and 3LPSW-108 locked open.

(2) Tests or maintenance shall be allowed on any component of the LPSW system provided the redundant train of the LPSW system is operable. If the LPSW system is not restored to meet the requirements of Specification 3.3.7.b(1) above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.7.b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250° within an additional 24 hours.

Bases

Specification 3.3 assures that, for whatever condition the reactor coolant system is in, adequate engineered safety feature equipment is operable.

For operation up to 60% FP, two high pressure injection pumps are specified. Also, two low pressure injection pumps and both core flood tanks are required. In the event of the need for emergency core cooling, adequate protection will be provided by one high pressure injection pump, one low pressure injection pump, and both core flood tanks. In the event of a main coolant loop severance, the above equipment will limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad volume. (1) Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.

The requirement to have three HPI pumps and two HPI flowpaths operable during power operation above 60% FP is based on considerations of potential small breaks at the reactor coolant pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling.(2) The analysis of these breaks indicates that for operation at or below 60% FP only a single train of the HPI system is needed to provide the necessary core cooling.

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent freezing. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70°F without any control rods in the core. This concentration is 1,338 ppm boron while the minimum value specified in the tanks is 1,800 ppm boron.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the Reactor Building design pressure will not be exceeded with one spray and two coolers operable.(4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor Building spray systems are operable for seven days or one Reactor Building spray system provided all three Reactor Building cooling units are operable.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Units 1, 2, and 3. One low pressure service water pump per unit is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

Prior to initiating maintenance on any of the components, the redundant component(s) shall be tested to assure operability. Operability shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated immediately prior to removal. The basis of acceptability is a likelihood of failure within 24 hours following such demonstration.

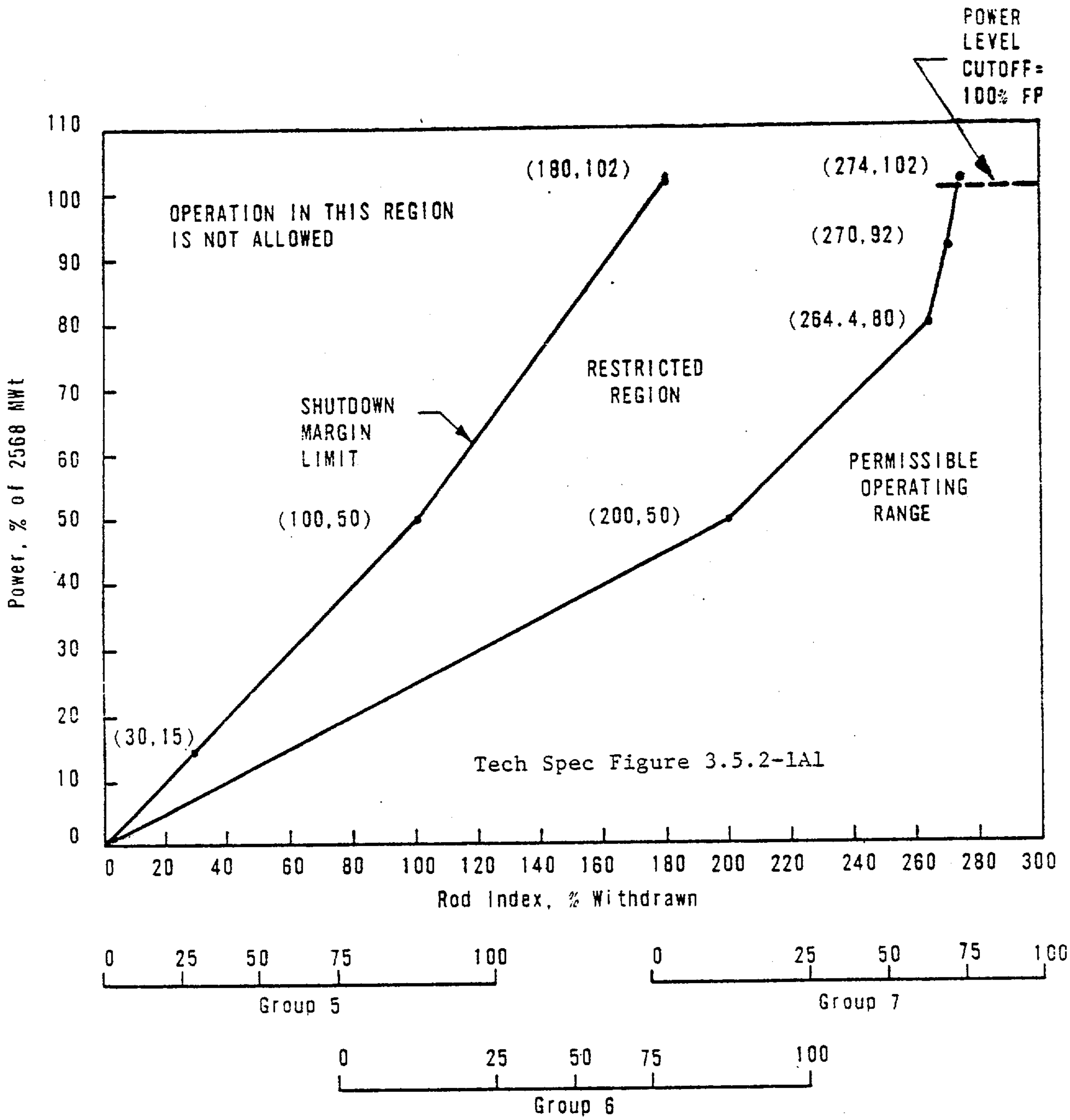
REFERENCES

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

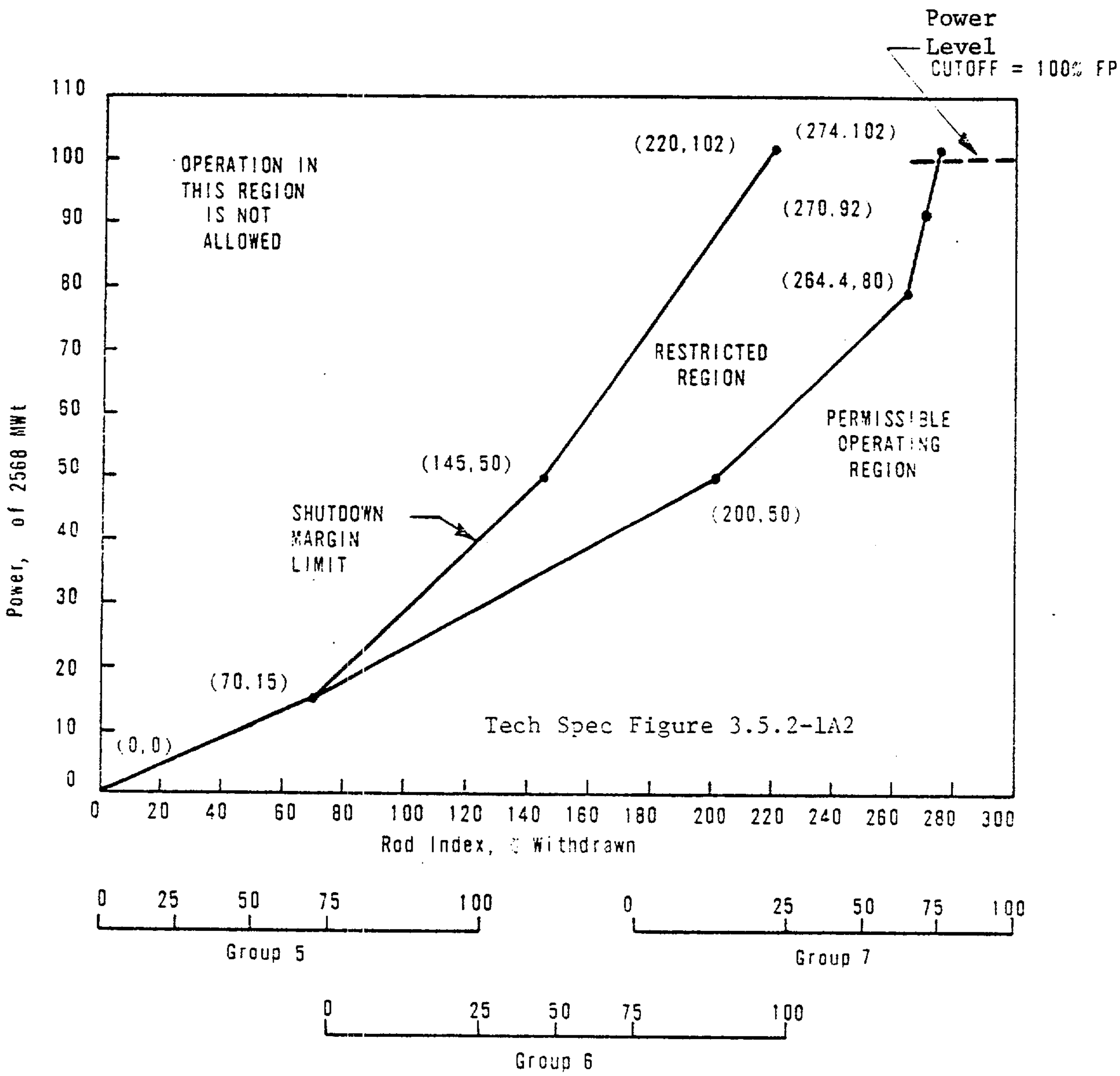
(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) FSAR, Section 9.5.2

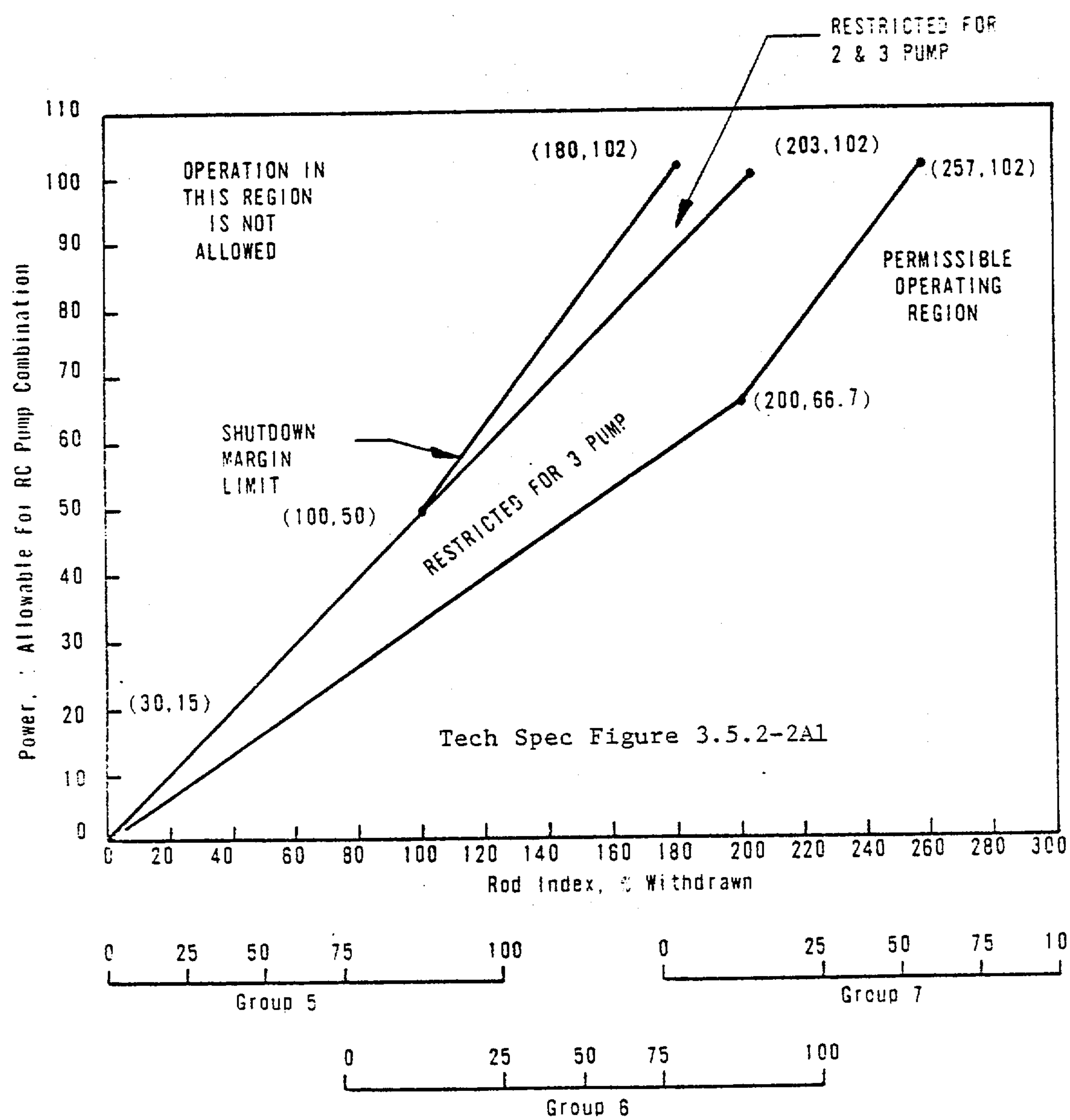
(4) FSAR, Supplement 13



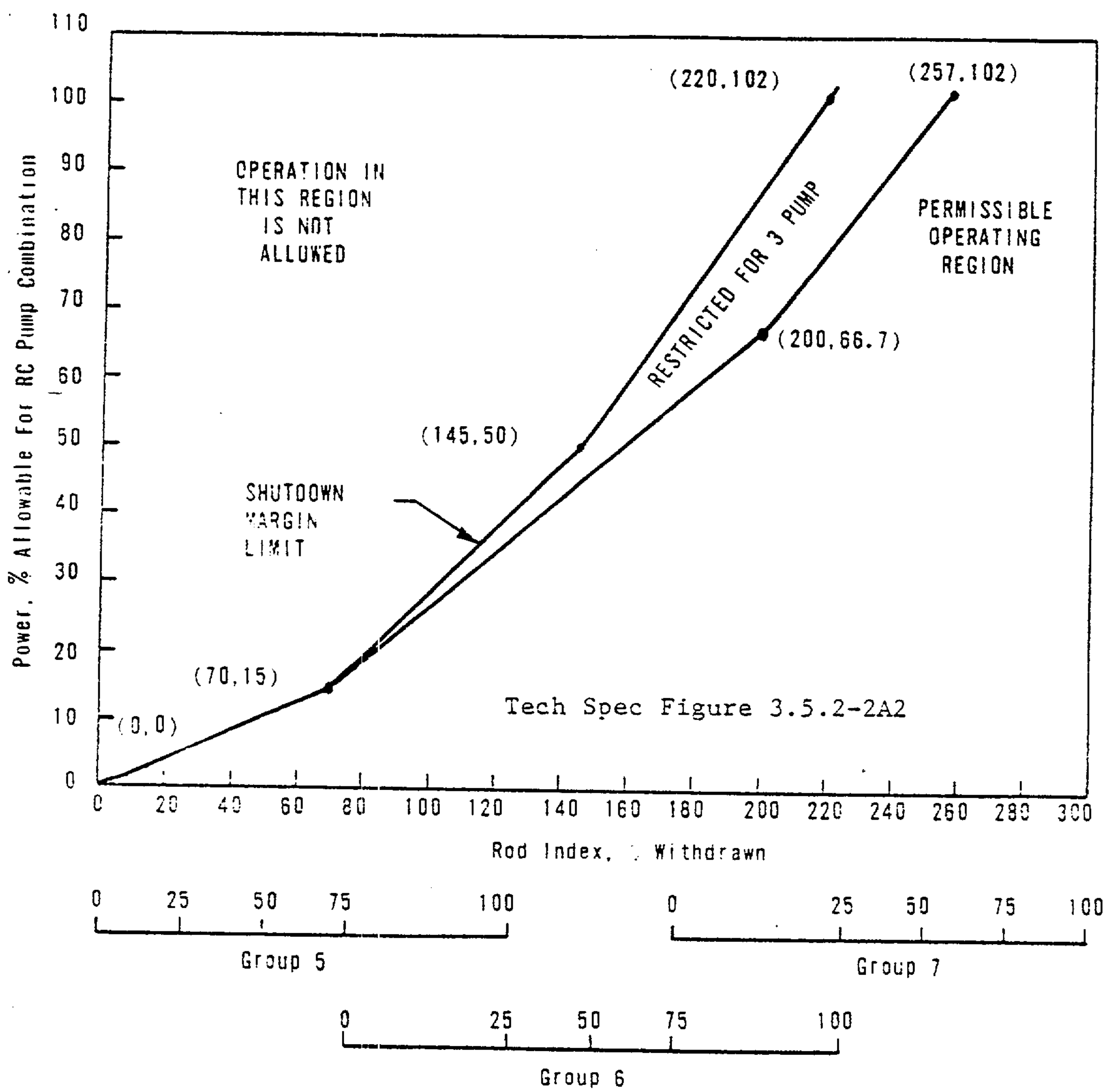
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 200 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-1A1



ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 200 ± 10 EFPD to 372 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-1A2

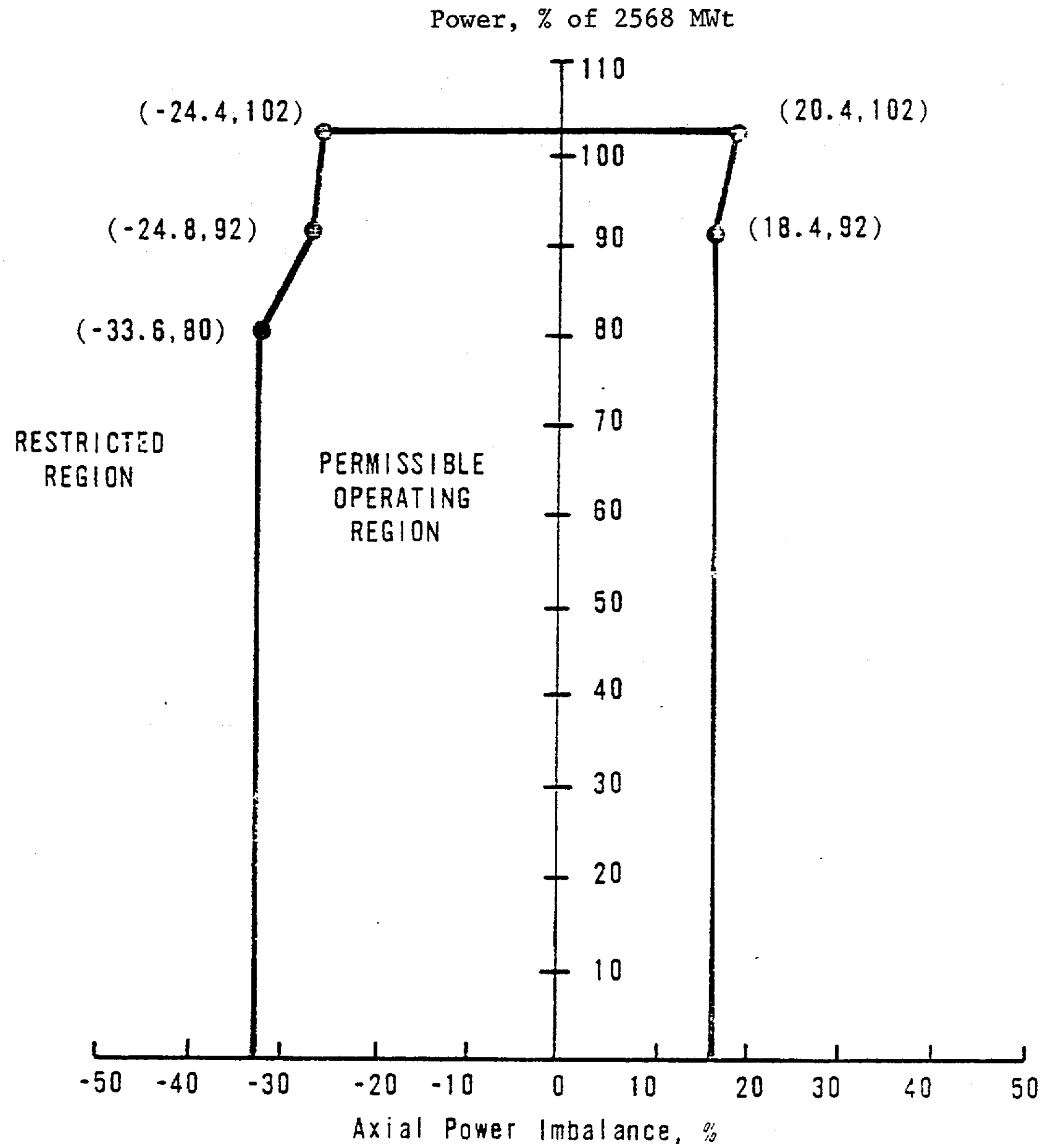


ROD POSITION LIMITS
FOR TWO- & THREE-PUMP OPERATION
FROM 0 TO 200 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-2A1

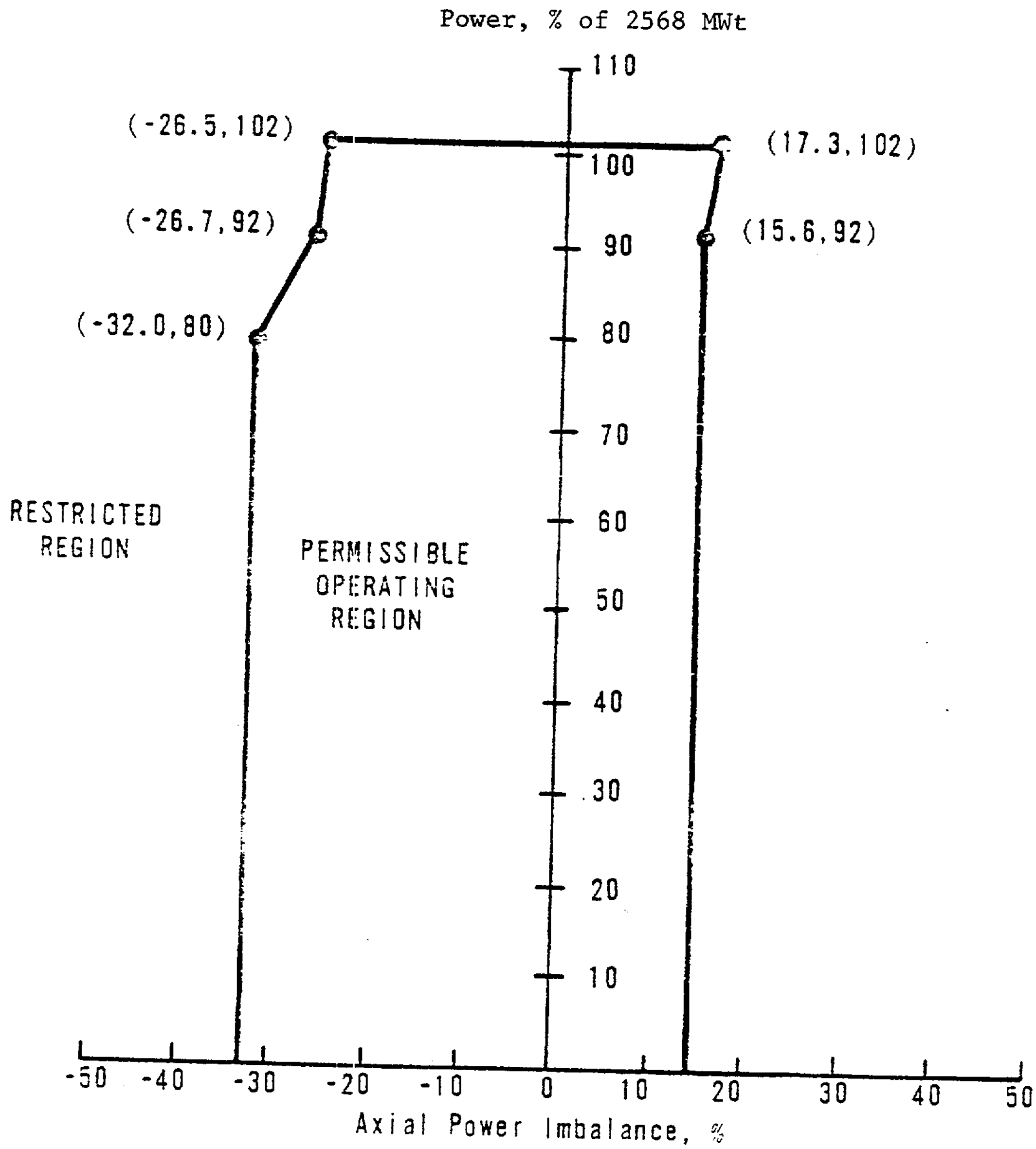


ROD POSITION LIMITS
FOR TWO- & THREE- PUMP OPERATION
AFTER 200 ± 10 EFPD to 372 ± 10 EFPD
UNIT 1

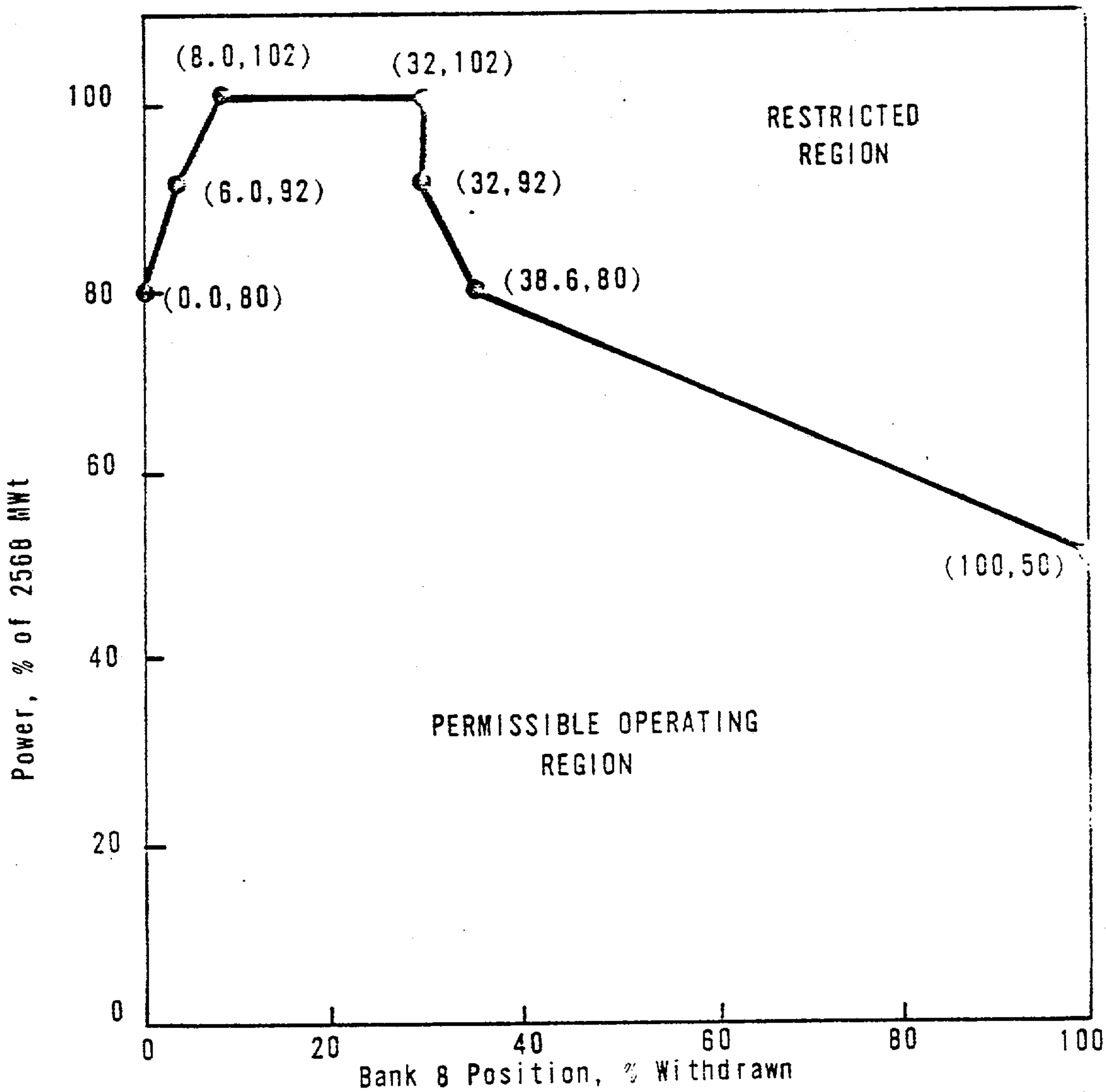
OCONEE NUCLEAR STATION
Figure 3.5.2-2A2



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 0 TO 200 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-3A1



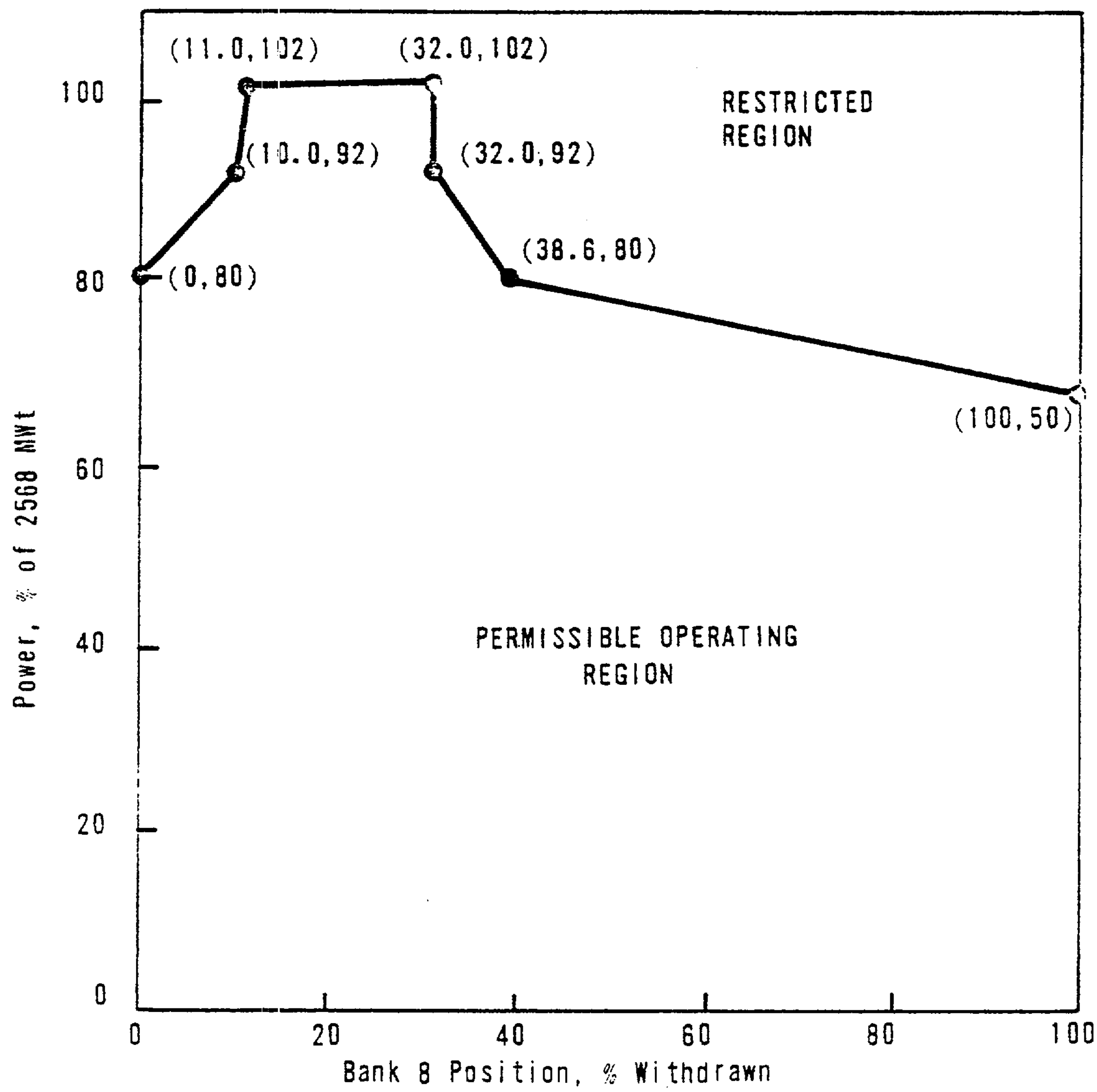
OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 200 ± 10 EFPD
UNIT 1 to 372 ± 10 EFPD
OCONEE NUCLEAR STATION
Figure 3.5.2-3A2



APSR POSITION LIMITS
FOR OPERATION FROM
0 TO 200 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-4A1

Amendments Nos. 81 , 81, & 78



APSR POSITION LIMITS
FOR OPERATION AFTER 200 $\pm$ 10 EFPD
UNIT 1 to 372 $\mp$ 10 EFPD
OCONEE NUCLEAR STATION
Figure 3.5.2-4A2

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days.

Bases

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Substantial calibration shifts within a channel (essentially a channel failure) are revealed during routine checking and testing procedures. Thus, the minimum calibration frequencies set forth are considered acceptable.

Periodic use of the Incore Instrumentation System for power mapping is sufficient to assure that axial and radial power peaks and the peak locations are controlled in accordance with the provisions of the Technical Specifications.

REFERENCE

(1) FSAR, Section 7.1.2.3.4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 78 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 August 6, 1979, as supplemented August 22, 1979, December 31, 1979, and January 28, 1980 (References 1, 2, 16 and 18) Duke Power Company (DPC) requested amendment of the Oconee Nuclear Station Technical Specifications (TSs). The DPC submittal of August 6, 1979, included Babcock & Wilcox (B&W) Report BAW-1552, dated July 1979, to support the Oconee Unit 1 operation at full power during Cycle 6. The B&W report describes the fuel system design, the accident analyses, and the startup test program. The design length of the proposed Cycle 6 operation is 372 effective full power days (EFPDs). At the end of Cycle 5, a total of 68 burned fuel assemblies (FAs) will be discharged and 68 fresh FAs will be loaded. The five batch 4D FAs, the sixty batch 5 FAs and three of batch 6 FAs will be discharged. The remaining 53 batch 6 FAs, designated 6B, and the fresh batch 8A and 8B (20 and 48 FAs respectively) with initial enrichments of 2.79, 2.97 and 3.07 wt% U-235 respectively will be loaded into the central portion of the core. The batch 7 (56 FAs) fuel, with an initial enrichment of 3.02 wt%U-235, will occupy primarily the core periphery as in the reference Cycle 5. Reactivity is controlled by 61 full-length Ag-In-Cd control rods, soluble boron shim, and 60 burnable poison rod assemblies (BPRAs). The latter being required to offset the increased excess reactivity built in for the longer Cycle 6 (372 EFPDs vs. 303.6 EFPDs for Cycle 5). In addition to the full-length control rods, eight axial power shaping rods are provided for axial power distribution control. The control rod-Cycle 6 locations and group designations are identical to those of the reference cycle. The following sections present the evaluations of any changes to the fuel system design, the nuclear and thermal-hydraulic design, the accident and transient analysis, the proposed TS changes to accommodate Cycle 6 operation, and the startup physics test program. A change is also proposed for TS 3.3 dealing with four engineered safety features.

2.0 Evaluation of Core Design Modifications and Changes

2.1 Fuel System Design

2.1.1 General and Mechanical Design

Modified BPRA retainers (Reference 19) are to be used in Cycle 6 to ensure positive retention of the BPRAs. These retainers have been previously approved in another B&W reactor core, for retention of Orifice Rod Assemblies (ORAs). Mechanical and thermal-hydraulic compatibility of the BPRA retainers has been previously reviewed and accepted. Use of the modified BPRA retainers will ensure positive retention of the BPRAs. DPC has committed to the removal of the retainers at the end of Cycle 6 in the absence of NRC approval authorizing the use of this component for multiple cycles of operation.

To achieve the longer Cycle 6, an in-out-in fuel management scheme and an average core fuel enrichment increase are applied. To ensure that achieved core power distributions conform with the values assumed in the safety and setpoint analyses, monthly incore power maps are to be compared with predicted distributions and deviations are to be reported to the NRC. The mechanical design of the Cycle 6 fresh FAs is the same as the Cycle 5 fuel and is thus acceptable.

2.1.2 Rod Design

The fuel pellet end configuration has changed from a spherical dish for batches 1 through 8A to a truncated cone dish for batch 8B. The new design reduces pellet end laminations during manufacturing. We conclude the fuel performance will not be adversely affected by this change and is thus acceptable.

2.1.3 Enhanced Fission Gas Release

During the last several years, data have begun to indicate that the fission gas release rate from LWR fuel pellets is increased (enhanced) with burnup.

The effect of enhanced fission gas release on ECCS performance was significant for B&W fuel. Enhanced release at high burnup affects the fuel rod internal pressure and the pellet volumetric average temperature which are important inputs to the B&W LOCA analyses. B&W calculates these inputs using the TAFY-3 fuel performance code which was approved prior to identification of enhanced fission gas release at high burnup. Another B&W fuel performance code, TACO, includes the effects of enhanced release and was approved by the NRC staff. B&W states that both the rod pressure and volumetric average fuel temperature calculated by TAFY-3 conservatively envelope those calculated by TACO between 2,000 and 42,000 MWD/MTU peak fuel rod burnup. The limiting LOCA calculations for all operating B&W reactors occur at burnups within this range. Thus, the use of TAFY-3 to calculate the fuel rod pressure and volumetric average temperature input for the LOCA analyses conservatively bounds the effects of enhanced fission gas release. Therefore, no immediate licensing action is required on fueled operating B&W reactors.

Inasmuch as the current reload has been and all future Oconee reloads will be evaluated against fuel vendors' revised fuel performance codes which provide for increase in fission gas release at higher burnups, we consider this a satisfactory resolution of this concern.

2.1.4 Cladding Creep Collapse

Due to its longer accumulated incore exposure, the fuel of batch 6B is more limiting than the fuel in other batches. The batch 6B assembly power histories were analyzed and the most limiting assembly was used to perform the creep collapse analysis using the CROV computer code and procedures described in Reference 3. The collapse time for the limiting FA was determined to be more than 35,000 effective full power hours (EFPHs), which is greater than the maximum projected residence time for Cycle 6 operation. We conclude that cladding creep collapse will not occur during Cycle 6.

2.1.5 Cladding Stress and Strain

For design evaluation, the primary stress is less than two-thirds of the minimum specified unirradiated yield strength, and all stresses (primary and secondary) are less than the minimum specified unirradiated yield strength. DPC states that (a) the stress analysis has a margin in excess of 30%, and (b) the Oconee 1 stress parameters are enveloped by that stress analysis.

The fuel design criteria specify a limit of 1.0% on cladding circumferential plastic strain. The pellet design is established for plastic cladding strain of less than 1.0% at maximum design local pellet burnup (55,000 MWD/MTU) and heat generation rate (20.15 KW/FT). Oconee 1 fuel is not expected to experience those maximum values. We conclude that the clad stresses and strains are within acceptable limits and thus acceptable.

2.1.6 Thermal Design

Fresh batch 8 FAs added to Cycle 6 are thermally similar to fuel remaining from previous cycles except for an increase of batch 8B initial nominal pellet density to 95% of theoretical density (TD) compared to 94% TD for other fuel in the core. Linear heat rate (LHR) capabilities are based on centerline fuel melt and were established using the TAFY-3 code (Reference 4) considering fuel densification. We conclude that the indicated thermal LHR limits are acceptable for preventing center melt and that the limits will not be exceeded.

2.2 Nuclear Design

The core design physics parameters for Cycle 6 were generated using the B&W version of PDQ07 (References 5, 6 and 7) and compared to the Cycle 5 parameters (Reference 1, Table 5-1).

The initial BPRA loading, the longer Cycle 6 design life and the different shuffle pattern (Reference 1, Figure 5-1) make it difficult to directly compare the physics parameters between Cycles 5 and 6. The critical boron concentrations for Cycle 6 are higher to compensate for the additional reactivity necessary for the longer cycle, not completely offset by the BPRA. The control rod worths differ between cycles due to changes in radial flux and burnup distributions. Cycle 6 shutdown margin is calculated to be 3.38% $\Delta K/K$ and 2.29% $\Delta K/K$ for beginning of cycle (BOC) and end of cycle (EOC) conditions, respectively, with the maximum worth rod stuck. The required shutdown margin is 1.0% $\Delta K/K$. We conclude that the nuclear design does not differ in a significant way from earlier cycles, that the nuclear parameters have been calculated by acceptable methods and are within the range of values expected for a cycle approaching an equilibrium cycle, and that the nuclear design has resulted in an adequate shutdown margin. The nuclear design for Oconee 1 Cycle 6 is, therefore, acceptable.

2.3 Thermal-Hydraulic Design

The incoming batch 8 fuel is hydraulically identical to the fuel used in previous cycles. The thermal-hydraulic methodologies and models used to support Cycle 6 operation are described in References 8, 9 and 10; these models have been previously found acceptable by the NRC staff. The main differences between Cycle 6 and the reference Cycle 5 are discussed below.

2.3.1 Core Bypass Flow

The effect of removing all ORAs in Cycle 5 was a maximum core bypass flow (CBF) of 10.4%. For Cycle 6 operation, 60 BPRAs will be inserted, leaving 46 vacant FAs, thus reducing the CBF to 8.1% resulting in a net increase in core flow. This provides for increased heat removal in both normal operation or in accident conditions and is thus acceptable.

2.3.2 BPRA Retainers

The retainers added to provide positive holddown of BPRAs introduce a small departure from nucleate boiling ratio (DNBR) penalty as discussed in Reference 11. However, the increase in core flow due to the BPRA insertion (Section 2.3.1) more than compensates for the decrease in DNBR due to the BPRA retainers and is thus acceptable.

2.3.3 Rod Bow DNBR Penalty

To determine the DNBR penalty due to fuel rod bow, DPC referenced the B&W interim procedure (Reference 13) which indicates there is no DNBR penalty for fuel burnup to approximately 21,300 MWD/MTU. That procedure was not accepted by the NRC staff and a modified procedure (Reference 14) that imposes a DNBR penalty at 21,300 MWD/MTU, was agreed to by the NRC and B&W. Even though the modified procedure was not used to calculate the rod bow DNBR penalty for Cycle 6, the latter has operational limits (based on minimum DNBR) that contain a 10.2% margin above the 1.30 DNBR criterion.

Therefore, we find the rod bow effect covered by the excess DNBR margin. Table 6-1 of Reference 1 lists the thermal-hydraulic parameters for Cycle 6 and the reference Cycle 5. That table shows the similarity between the two cycles. It also shows the increase in DNBR margin due to the decrease in densification penalty. The decrease in penalty is a result of the generally denser fuel used in Cycle 6 over that used in Cycle 5. Therefore, we conclude DPC's fuel rod bowing calculations to be acceptable.

3.0 Evaluation of Accidents and Transients

DPC has reviewed each Final Safety Analysis Report (FSAR) accident analysis with respect to changes in Cycle 6 parameters to determine their effect on the plant thermal performance during the analyzed accidents and transients. The key parameters having the greatest effect on the outcome of a transient or accident are the core thermal parameters, thermal-hydraulic parameters, and physics and kinetics parameters. Fuel thermal analysis values are listed in Table 4-2 of Reference 1 for all fuel batches in Cycle 6. Table 6-1 of the same reference compares the thermal-hydraulic parameters for Cycles 5 and 6. These parameters are exactly the same except for the higher value of design minimum DNBR (MDNBR) for Cycle 6 (2.05 as compared to 1.98 for Cycle 5). A comparison of the key kinetic parameters from the FSAR and Cycle 6 is provided in Table 7-1 of Reference 1. These comparisons indicate no significant changes (Table 4-1 above compared to Table 4-2 of Reference 9) or changes in the conservative direction (Tables 6-1, 7-1 of Reference 1). The effects of fuel densification on the FSAR accident analyses have been evaluated in Reference 10. Since batch 8B fuel of Cycle 6 contains fuel pellets with theoretical density that is higher than those considered in Reference 10, the conclusions in that reference are still valid.

A generic loss of coolant accident (LOCA) analysis for the B&W 177-FA, lowered loop nuclear steam system supply has been performed using the final acceptance criteria emergency core cooling system (ECCS) evaluation model (Reference 15). That analysis used the limiting values of key parameters for all plants in the 177-FA lowered loop category, and therefore is bounding for the Oconee 1 Cycle 6 operation.

It is concluded from the examination of Cycle 6 core thermal and kinetic properties, with respect to acceptable previous cycle values and with respect to the FSAR values, that this core reload will not adversely affect the Oconee 1 plant's ability to operate safely during Cycle 6.

4.0 Emergency Core Cooling System

An Exemption was granted on October 23, 1978 to 10 CFR 50.46(a), "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." The Exemption provided for its own termination upon completion of the modifications required by the Exemption and for prior NRC staff approval of the design. By letter dated December 13, 1978 (Reference 17), we found the design of the modifications to be acceptable. DPC has installed the modifications at Oconee 1 (Reference 18) and prepared acceptable operating procedures; thus, we conclude that the as-modified ECCS required by the Exemption of October 23, 1978 is acceptable.

5.0 Startup Test Program

Startup tests have been proposed by DPC to provide assurance that Oconee 1 has been loaded as intended. This test program is similar to that used at Oconee Nuclear Station and other B&W reactors. We have reviewed the test program and consider it acceptable.

6.0 Control Rod Guide Tube Wear

By letter dated November 23, 1979, we requested DPC to provide detailed information on the wear characteristics of the control rods on the guide tubes in FAs at the Oconee Nuclear Station. In response, DPC engaged B&W to perform confirmatory inspections on selected control rod guide tubes. Results from the preliminary inspections are discussed in our Safety Evaluation issued on January 4, 1980. We have determined that operation of Oconee 1 for Cycle 6 is acceptable.

Our approval for operation of Oconee 1 for Cycle 6, based on the preliminary Eddy Current Test (ECT) inspections, has considered the following:

1) guide tube wear is a time-dependent process,

2) available evidence indicates that a sufficient margin exists between guide tube wear observed to date in B&W plants and design limits, and

3) confirmatory inspections are planned for February 1980.

Our evaluation of guide tube wear is currently based on preliminary information. Future information may show that additional surveillance or restrictions are required, particularly for the last cycle of extended cycles.

7.0 Technical Specification Changes

Proposed modifications to the Oconee 1 TSs are described below (Reference 1):

(1) The LHR limit to fuel control melting has been decreased to 20.05 kw/ft.

(2) Primarily due to the decrease in core bypass flow the power to flow trip ratio has been increased to 1.08. Reactor trip setpoints of power based on flow have been correspondingly changed.

(3) The following limits have been changed:

a. Reactor power/Reactor power imbalance safety limits and trip setpoints for 2, 3 and 4 reactor coolant (RC) pump operation.

b. Operational power imbalance envelope for less than 200 EFPDs and for more than 200 EFPDs.

c. Rod position limits for 2, 3 and 4 RC pump operation for less than 200 EFPDs and for more than 200 EFPDs during the cycle life.

d. Axial power shaping rod (APSR) position limits for less than 200 EFPDs and for more than 200 EFPDs.

We have evaluated the reload report for the Oconee 1 Cycle 6 operation and the proposed TS changes that reflect the changed parameters for the new cycle and find the revised TSs acceptable.

We have also evaluated the proposed revision of TS 3.3, Emergency Core Cooling, Reactor Building Cooling, Reactor Building Spray, and Low Pressure Service Water Systems. This revision consists of a reorganization of the TS into system headings; the present format has the four systems all listed together. The new format improves clarity and is an editorial improvement. The change also incorporates the needed limiting conditions for operation for the recently installed high pressure injection system cross-connect in Oconee 1 and 3, which we approved in our December 13, 1978 letter to DPC (Reference 17). This cross-connect will be installed in Oconee 2 during its next scheduled reload or extended maintenance shutdown, whichever comes first.

8.0 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.

9.0 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 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: February 22, 1980

References

1. Letter, W. O. Parker (DPC) to H. R. Denton (NRC), dated August 6, 1979, with attachment, Oconee Unit 1, Cycle 6 Reload Report, BAW-1552, July 1979.
2. Letter, W. O. Parker (DPC) to H. R. Denton (NRC), dated August 22, 1979.
3. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084, Rev. 2, October 1978.
4. C. D. Morgan and H. S. Kao, TAFY - Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, May 1972.
5. B&W Version of PDQ07 Code, BAW-10117A, January 1977.
6. Core Calculational Techniques and Procedures, BAW-10118, October 1977.
7. Assembly Calculations and Fitted Nuclear Data, BAW-10116A, May 1977.
8. Oconee Nuclear Station, Units 1, 2 and 3 - Final Safety Analysis Reports.
9. Oconee Unit 1, Cycle 5 Reload Report, BAW-1493, Rev. 2, September 1978.
10. Oconee 1 Fuel Densification Report, BAW-1388, Rev. 1, July 1973.
11. BPRA Retainer Design Report, BAW-1496, May 1978.
12. Mark-BZ Demonstration Assemblies Licensing Report, BAW-1533, April 1979.
13. J. H. Taylor (B&W) to D. B. Vassallo (NRC), letter, "Determination of the Fuel Rod Bow DNB Penalty," dated December 13, 1978.
14. J. H. Taylor (B&W) to S. A. Varga (NRC), letter, June 22, 1979.
15. ECCS Analysis of B&W's 177-FA Lowered Loop NSS, BAW-10103, Rev. 1, September 1975.
16. Letter, W. O. Parker (DPC) to H. R. Denton (NRC) dated December 31, 1979.
17. Letter, Robert W. Reid (NRC) to W. O. Parker (DPC) dated Decwmber 13, 1978.
18. Letter, W. O. Parker (DPC) to H. R. Denton (NRC) dated January 28, 1980.
19. BPRA Retainer Design Report, BAW-1496, B&W, May 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION 7590-01

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. 81, 81, and 78 to Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55, respectively, issued to Duke Power Company (the licensee), which revised the Station's common Technical Specifications 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 Technical Specifications to support the operation of Oconee Unit No. 1 at full rated power during Cycle 6. The amendments also revise the Technical Specifications for Units Nos. 1, 2, and 3 in regard to engineered safety features.

The application for the amendments complies 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 and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

Over

For further details with respect to this action, see (1) the application for amendments dated August 6, 1979, as supplemented August 22, 1979, December 31, 1979, and January 28, 1980, (2) Amendments Nos. 81 , 81, and 78 to Licenses Nos. DPR-38, DPR-47, and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C., and at the Oconee County Library, 201 South Spring Street, Walhalla, South Carolina. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 22nd day of February 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors