OCTOBER 1978

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

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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. 65, 65 and 62 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 requests dated April 20, 1978 and June 26, 1978, as supplemented April 27, August 21 and 28, September 6, 22 and 26, 1978.

These amendments revise the Technical Specifications to support the operation of Oconee Unit No. 1 at full rated power during Cycle 5 after core reload, removal of the orifice rod assemblies from the core, and High Pressure Injection Pump operability and operating procedure requirements in the unlikely event of a small break loss-of-coolant accident. By letter dated October 19, 1978, you reported that two steam generator tube plugs were unaccounted for and should be assumed to be in the primary coolant system. Your letter included a safety analysis of this concern. We reviewed your submittal and Appendix A to our enclosed Safety Evaluation represents our evaluation of the safety consequences of these two plugs.

In accordance with your letter dated August 21, 1978, the Commission has also issued the enclosed Exemption for Oconee Unit No. 1 from the requirements of 10 CFR 50.46(a)(1) that Emergency Core Cooling System (ECCS) performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K to 10 CFR 50.

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SURNAME ➤						ccp
DATE ➤						

Copies of the Safety Evaluation/Environmental Impact Appraisal and the Notice of Issuance/Negative Declaration are also enclosed. A copy of the Exemption is also being filed with the Office of the Federal Register for publication.

Sincerely,

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

Enclosures:
1. Amendment No. to DPR-38
2. Amendment No. to DPR-47
3. Amendment No. to DPR-55
4. Exemption
5. Safety Evaluation/Environmental Impact Appraisal
6. Notice/Negative Declaration

cc w/enclosures: See next page

NRR

10/ /78

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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

J. Michael McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806-15th Street, NW.,
Washington, D.C. 20005

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

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

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

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

cc w/enclosures & incoming dtd:
4/20 & 6/26, 4/27, 8/21&28, 9/6,22&26
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. 65
License No. DPR-38

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

   A. The applications for amendment by Duke Power Company (the licensee) dated April 20, 1978, and June 26, 1978, as supplemented April 27, August 21, 28, September 6, 22 and 26, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 65 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: October 23, 1978

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. 65
License No. DPR-47

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

   A. The applications for amendment by Duke Power Company (the licensee) dated April 20, 1978, and June 26, 1978, as supplemented April 27, August 21, 28, September 6, 22 and 26, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 65 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: October 23, 1978

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. 62
License No. DPR-55

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

   A. The applications for amendment by Duke Power Company (the licensee) dated April 20, 1978, and June 26, 1978, as supplemented April 27, August 21, 28, September 6, 22 and 26, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 62 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: October 23, 1978

ATTACHMENTS TO LICENSE AMENDMENT

AMENDMENT NO. 65 TO DPR-38

AMENDMENT NO. 65 TO DPR-47

AMENDMENT NO. 62 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.

2.1-2

2.1-7 (Figure 2.1-2A)

2.1-10 (Figure 2.1-3A)

2.3-8 (Figure 2.3-2A)

3.2-2

3.3-3 & 3.3-4

3.5-9

3.5-10

3.5-11

3.5-11a

3.5-11b* (Table 3.5-1)

3.5-12 (Figure 3.5.2-1A1)

3.5-13 (Figure 3.5.2-1A2)

3.5-18 (Figure 3.5.2-2A1)

3.5-18a (Figure 3.5.2-2A2)

3.5-21 (Figure 3.5.2-3A1)

3.5-21a* (Figure 3.5.2-3A2)

3.5-23c (Figure 3.5.2-4A1)

3.5-23d (Figure 3.5.2-4A2)

4.1-1

6.4-1

Changes on the revised pages are identified by marginal lines.

*New Page

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 131.3×10^6 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.

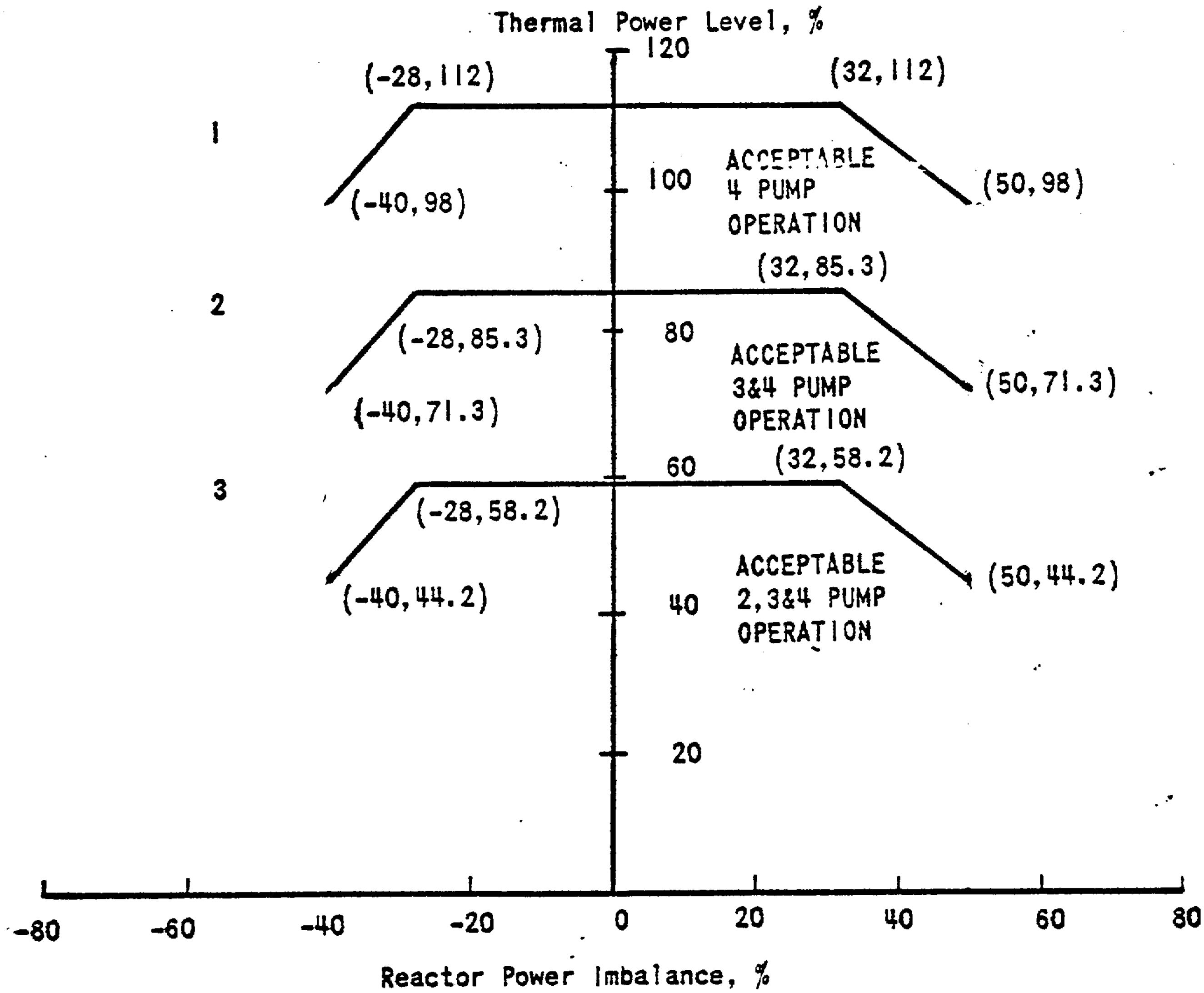
2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 1.

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 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055 = 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

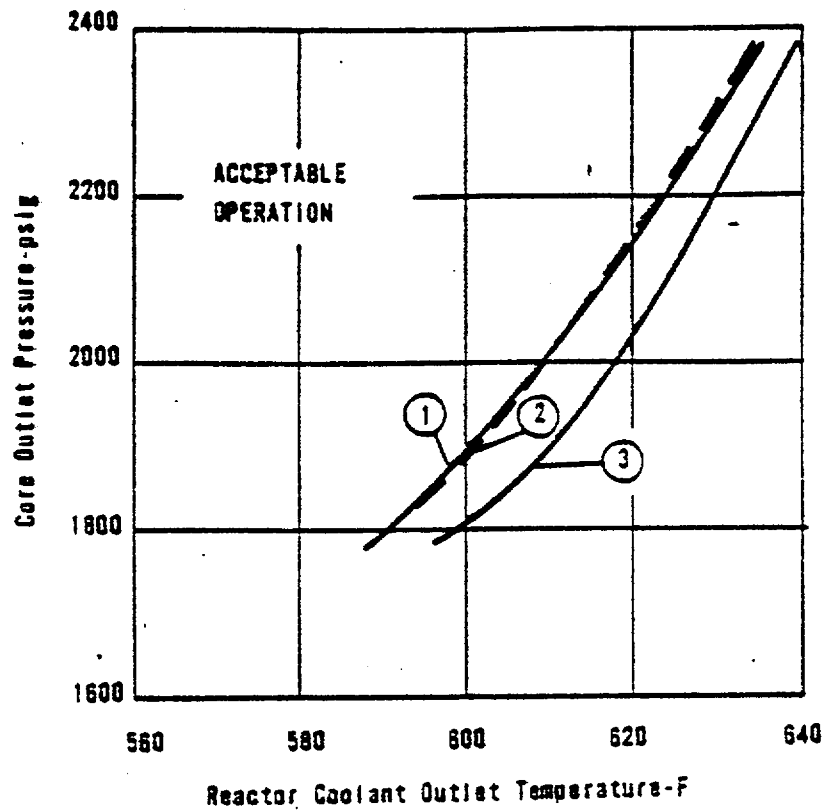


CURVE	RC FLOW (GPM)
1	374,880
2	280,035
3	183,690

CORE PROTECTION
SAFETY LIMITS
UNIT 1

OCONEE NUCLEAR STATION

Figure 2.1-2A

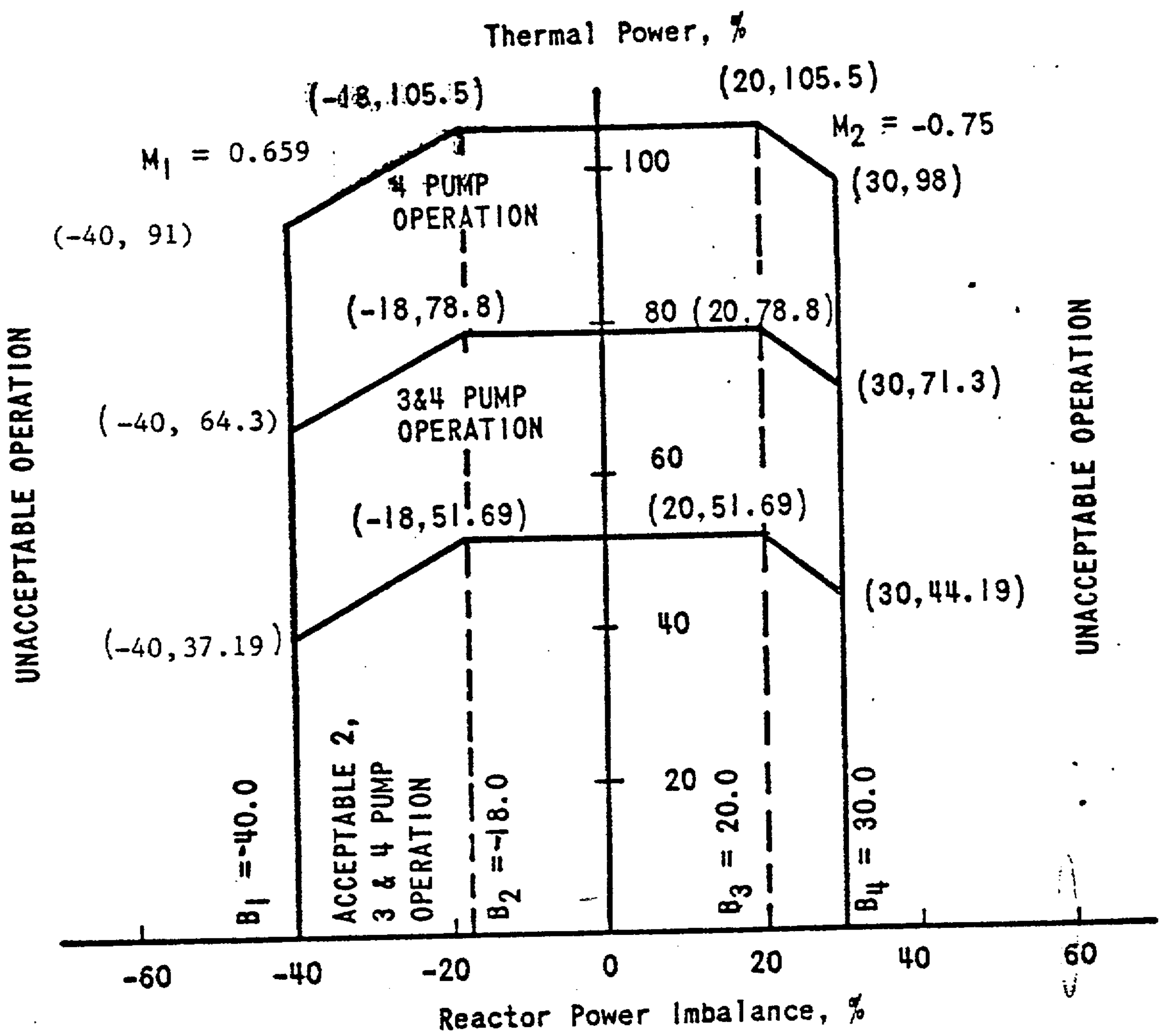


CURVE	REACTOR COOLANT FLOW (GPM)	POWER	PUMPS OPERATING	TYPE OF LIMIT
1	374880 (100%)*	112%	4	(DNBR)
2	280035 (74.7%)	85.3%	3	(DNBR)
3	183690 (49.0%)	58.2%	2	(QUALITY)

*106.5% of first core design flow

CORE PROTECTION
SAFETY LIMITS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.1-3A

Amendments Nos. 65, 65 & 62



PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 1

OCONEE NUCLEAR STATION

Figure 2.3-2A

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%Δk/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 5, Oconee 2, Cycle 3, and Oconee 3, Cycle 4 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 980 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

3.3.6 Exceptions to 3.3.5 shall be as follows:

(a) Both core flooding tanks shall be operational above 800 psig.

(b) Both motor-operated valves associated with the core flooding tanks shall be fully open above 800 psig.

(c) One pressure instrument channel and one level instrument channel per core flood tank shall be operable above 800 psig.

(d) One reactor building cooling fan and associated cooling unit shall be permitted to be out of service for seven days provided both reactor building spray pumps and associated spray nozzle headers are in service at the same time.

(e) If the requirements of Specification 3.3.1(f) are not met, the borated water storage tank shall be considered unavailable and action shall be initiated in accordance with Specification 3.2.

3.3.7 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to assure operability.

3.3.8 (a) Reactor power shall not be increased above 60% FP unless three HPI pumps and two HPI flow paths are operable.

(b) During power operation above 60% FP, tests or maintenance shall be allowed on any one HPI pump, provided two trains of the HPI system are operable. 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.

(c) If during power operation above 60% FP a high pressure injection flow path becomes inoperable, reactor power shall be reduced below 60% FP within 12 hours.

<u>Bases</u>

The requirements of Specification 3.3 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two high pressure injection pumps and two low pressure injection pumps are required (except as specified in Specification 3.3.8 and as discussed further on in these bases.) However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core. (1)

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. (2)

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 $70^{0}F$ 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.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.5 and 3.3.6 provided requirements in Specification 3.3.7 are met which assure operability of the duplicate components. Operability of the specified components 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.

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. Therefore, a maintenance period of seven days is acceptable for one reactor building cooling fan and its associated cooling unit. (3)

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop serverence, limit the peak clad temperature to less than $2,200^{0}F$ and the metal-water reaction to that representing less than 1 percent of the clad.

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.

The requirement to have three HPI pumps and two HPI flow paths operable during power operation above 60% FP (Specification 3.3.8) is based on considerations of a 0.04 square foot break 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. The analysis of this break 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.

REFERENCES

(1) FSAR, Section 14.2.2.3
(2) FSAR, Section 9.5.2
(3) FSAR, Supplement 13

3.5.2.5 Control Rod Positions

a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

b. Except for physics test, operating rod group overlap shall be 25% ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours or the reactor shall be placed in a hot shutdown condition within an additional 12 hours.

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on figures 3.5.2-1A1 and 3.5.2-1A2 (Unit 1); 3.5.2-1B1, 3.5.2-1B2 and 3.5.2-1B3 (Unit 2); 3.5.2-1C1, 3.5.2-1C2 and 3.5.2-1C3 (Unit 3) for four pump operation, and on figures 3.5.2-2A1 and 3.5.2-2A2 (Unit 1); 3.5.3-2B1, 3.5.2-2B2 and 3.5.2-2B3 (Unit 2); 3.5.2-2C1, 3.5.2-2C2 and 3.5.2-2C3 (Unit 3) for two or three pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-4A1, and 3.5.2-4A2 (Unit 1); 3.5.2-4B1, 3.5.2-4B2, and 3.5.2-4B3 (Unit 2); 3.5.2-4C1, 3.5.2-4C2, and 3.5.2-4C3 (Unit 3).

If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times.

3.5.2.6 Xenon Reactivity

a. Except for physics tests, reactor power in Unit 1 shall not be increased above the power-level-cutoff shown in Figures 3.5.2-1A1, and 3.5.2-1A2 unless one of the following conditions is satisfied:

1. Xenon reactivity did not deviate more than 10 percent from the equilibrium value for operation at steady state power.

2. Xenon reactivity deviated more than 10 percent but is now within 10 percent of the equilibrium value for operation at steady state rated power and has passed its final maximum or minimum peak during its approach to its equilibrium value for operation at the power level cutoff.

3. Except for xenon free startup (when 2. applies), the reactor has operated within a range of 87 to 92 percent of rated thermal power for a period exceeding 2 hours in the soluble poison control mode.

b. Except for physics tests, reactor power in Units 2 and 3 shall not be increased above the power level cutoff shown in Figures 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 (Unit 2); 3.5.2-1C1, 3.5.2-1C2, 3.5.2-1C3 (Unit 3); unless the following requirements are met:

1. The xenon reactivity shall be within 10 percent of the value for operation at steady-state rated power.

2. The xenon reactivity worth has passed its final maximum or minimum peak during its approach to its equilibrium value for operation at the power level cutoff.

3.5.2.7 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.8 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.

Bases

The power-imbalance envelope defined in Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2 and 3.5.2-3C3 is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification power spike factors (Units 1 and 2 only)
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping bank)

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2,3, 4,5) of hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

** Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5,6, and 7 are overlapped 25 percent. The normal position at power is for Groups 6 and 7 to be partially inserted.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding
7.50% for Unit 1. The limits shown in Specification 3.5.2.4
5.10% for Unit 2
7.50% for Unit 3
are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.7, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions are included in Technical Specification 3.5.2.6 to prevent excessive power peaking by transient xenon. The xenon reactivity must be beyond its final maximum or minimum peak and approaching its equilibrium value at the power level cutoff.

REFERENCES

[1]FSAR, Section 3.2.2.1.2

[2]FSAR, Section 14.2.2.2

[3]FSAR, SUPPLEMENT 9

[4]B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

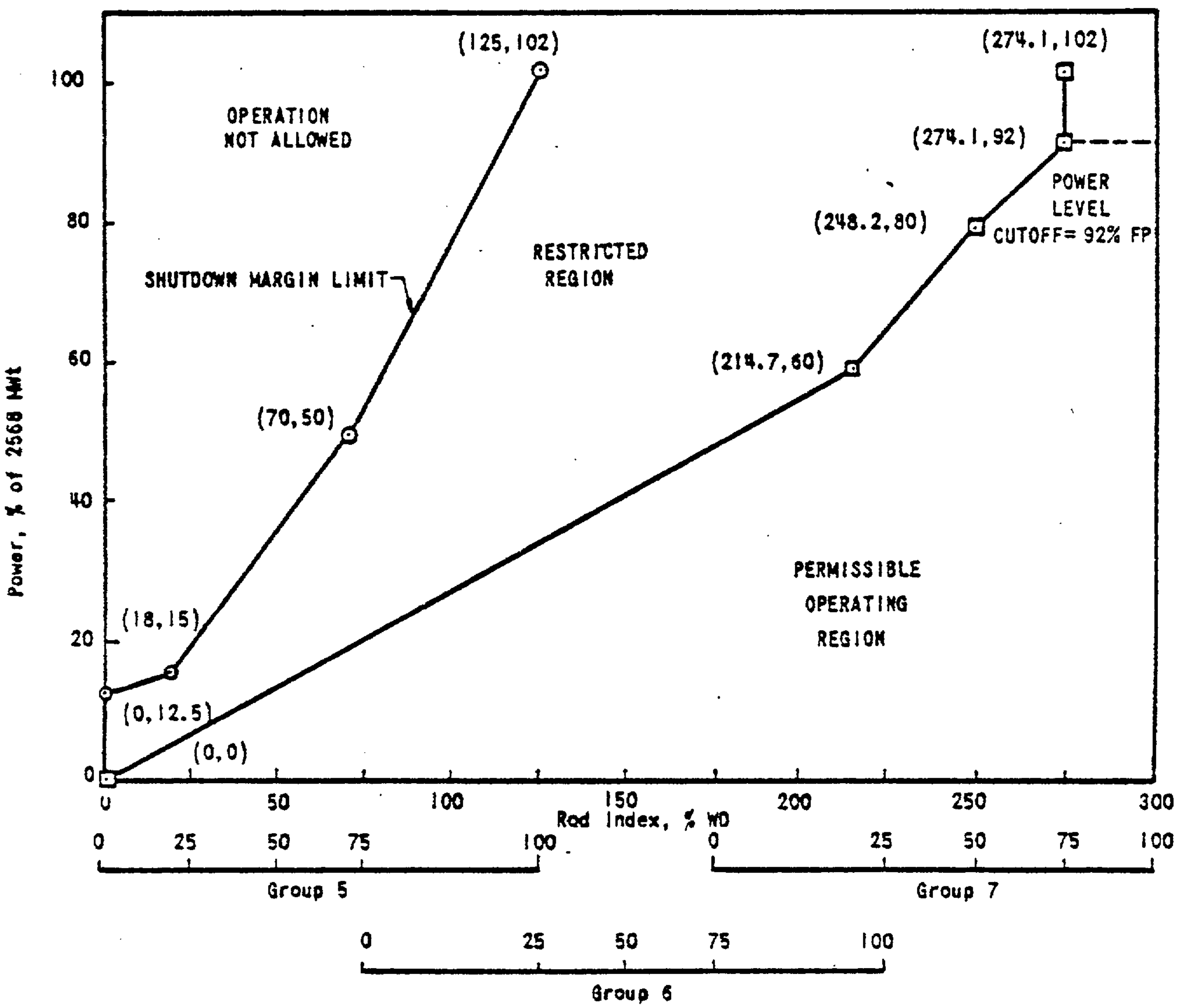
BAW-1400 (UNIT 3)

[5]Oconee 1, Cycle 4 - Reload Report - BAW-1447, March 1977, Section 7.11.

TABLE 3.5-1

Quadrant Power Tilt Limits

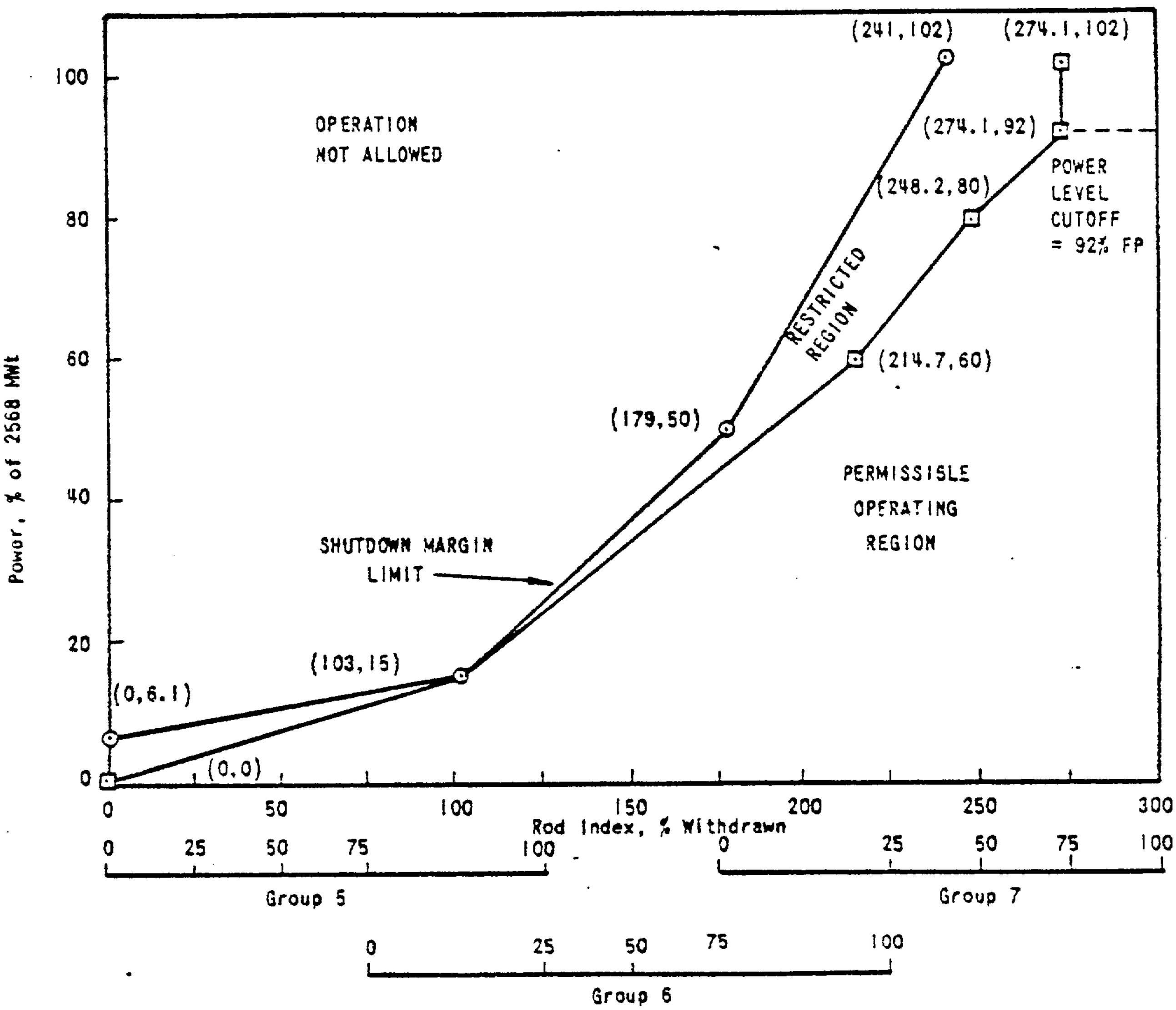
	Steady State Limit	Transient Limit	Maximum Limit
Unit 1	5.00	9.44	20.0
Unit 2	3.41	9.44	20.0
Unit 3	5.00	9.44	20.0



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 TO 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

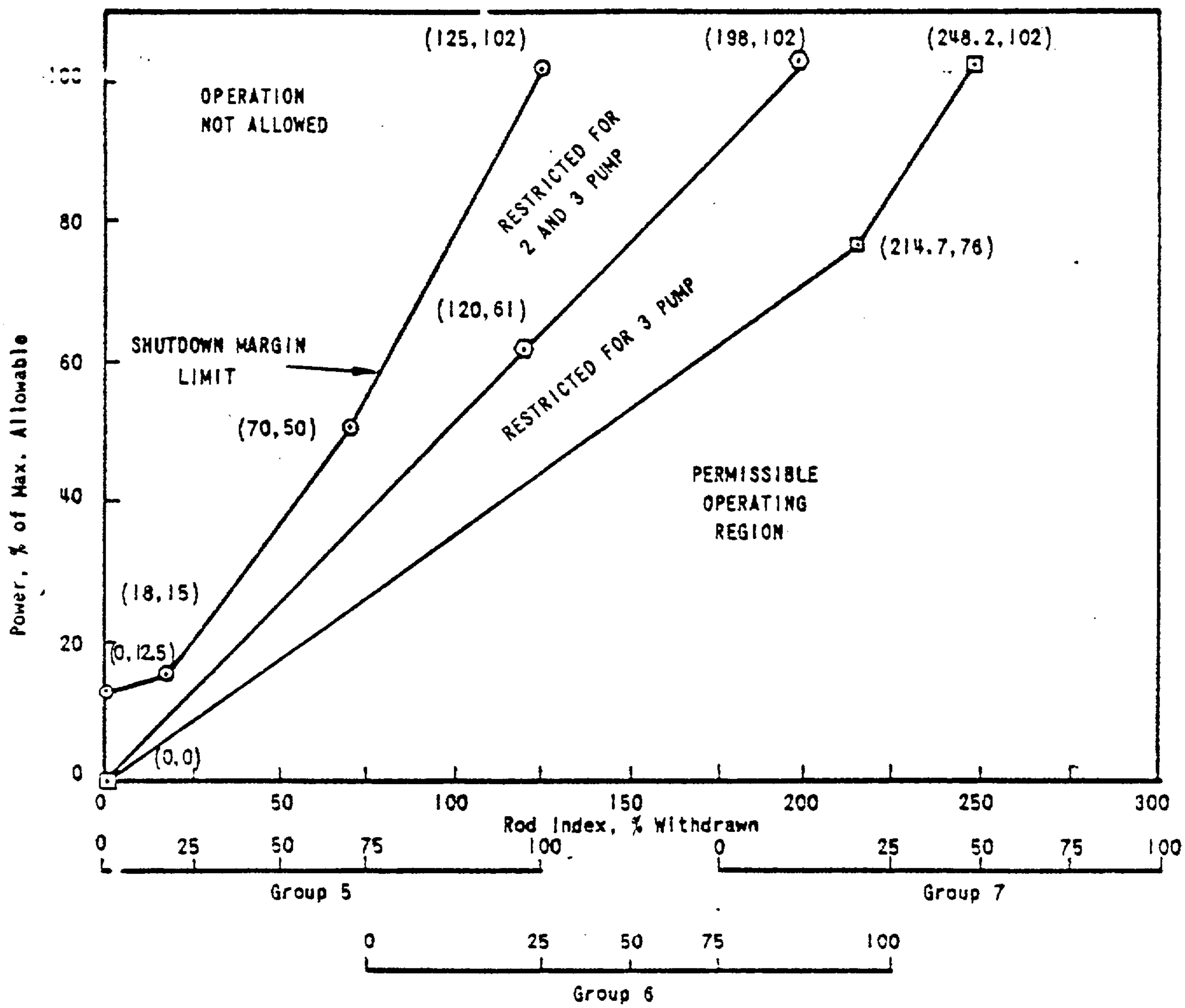
Figure 3.5.2-1A1



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-1A2

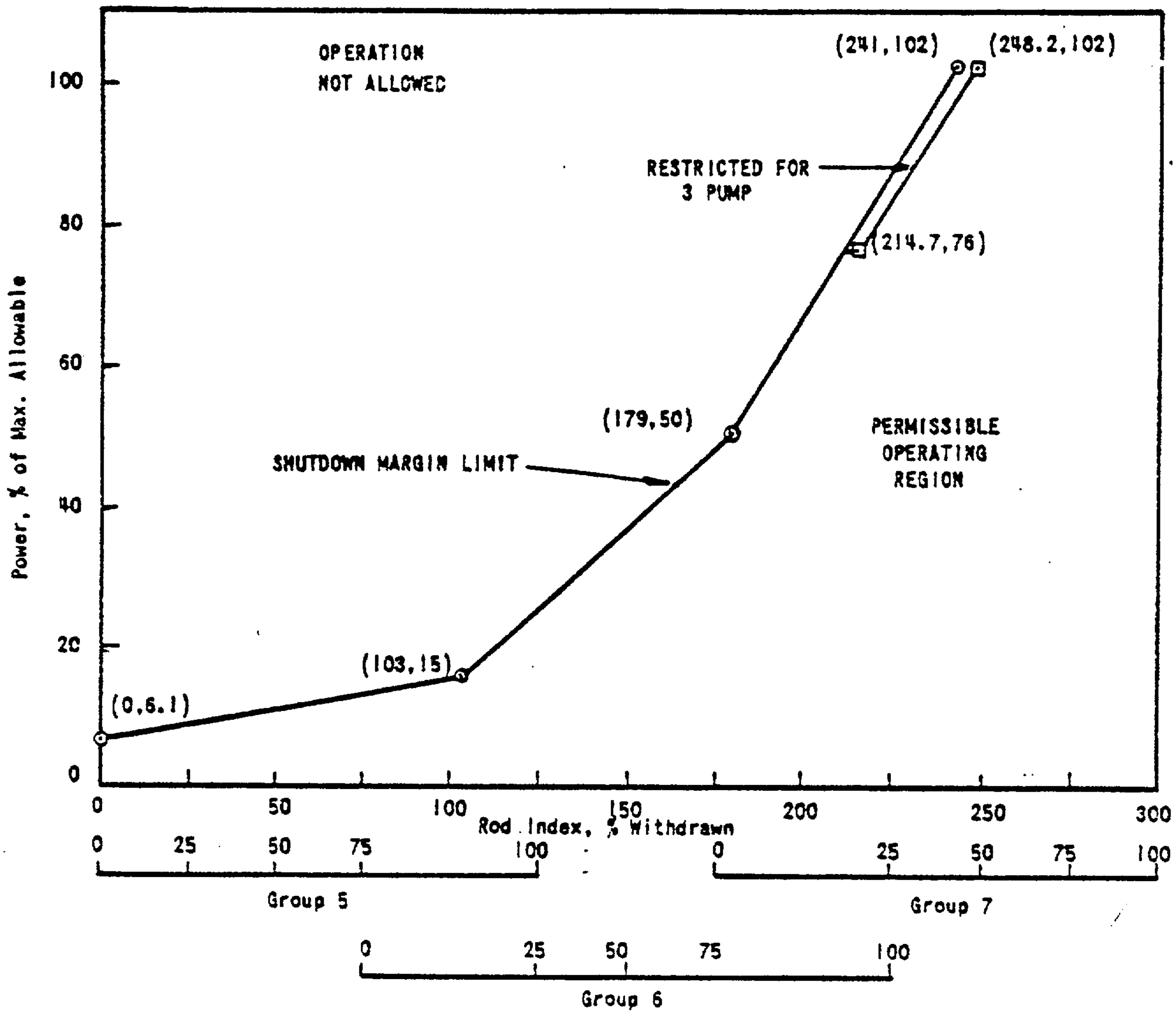


ROD POSITION LIMITS
FOR TWO AND THREE PUMP OPERATION
FROM 0 TO 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-2A1

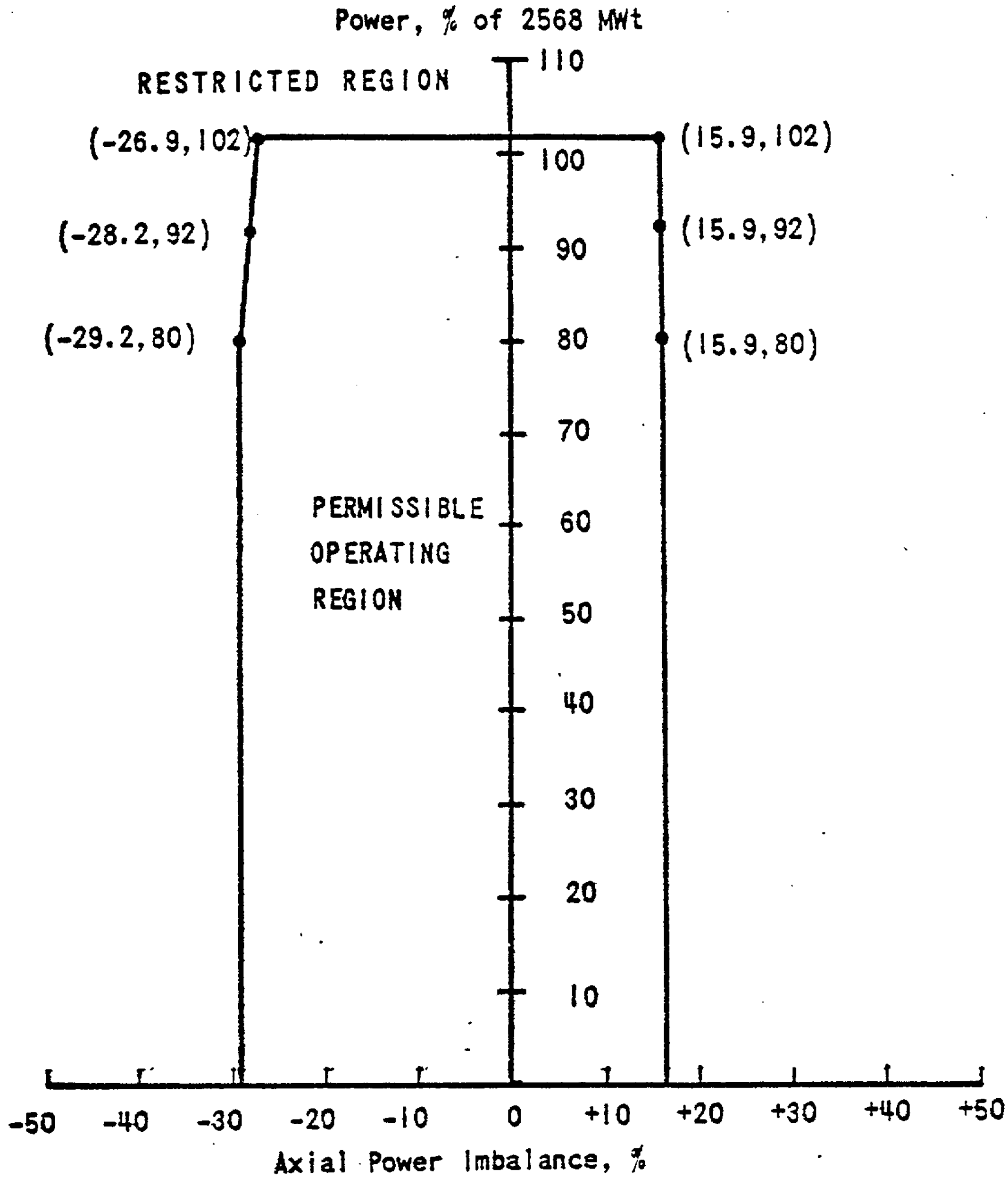
Amendments Nos. 65, 65 & 62



ROD POSITION LIMITS
FOR TWO AND THREE PUMP OPERATION
AFTER 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

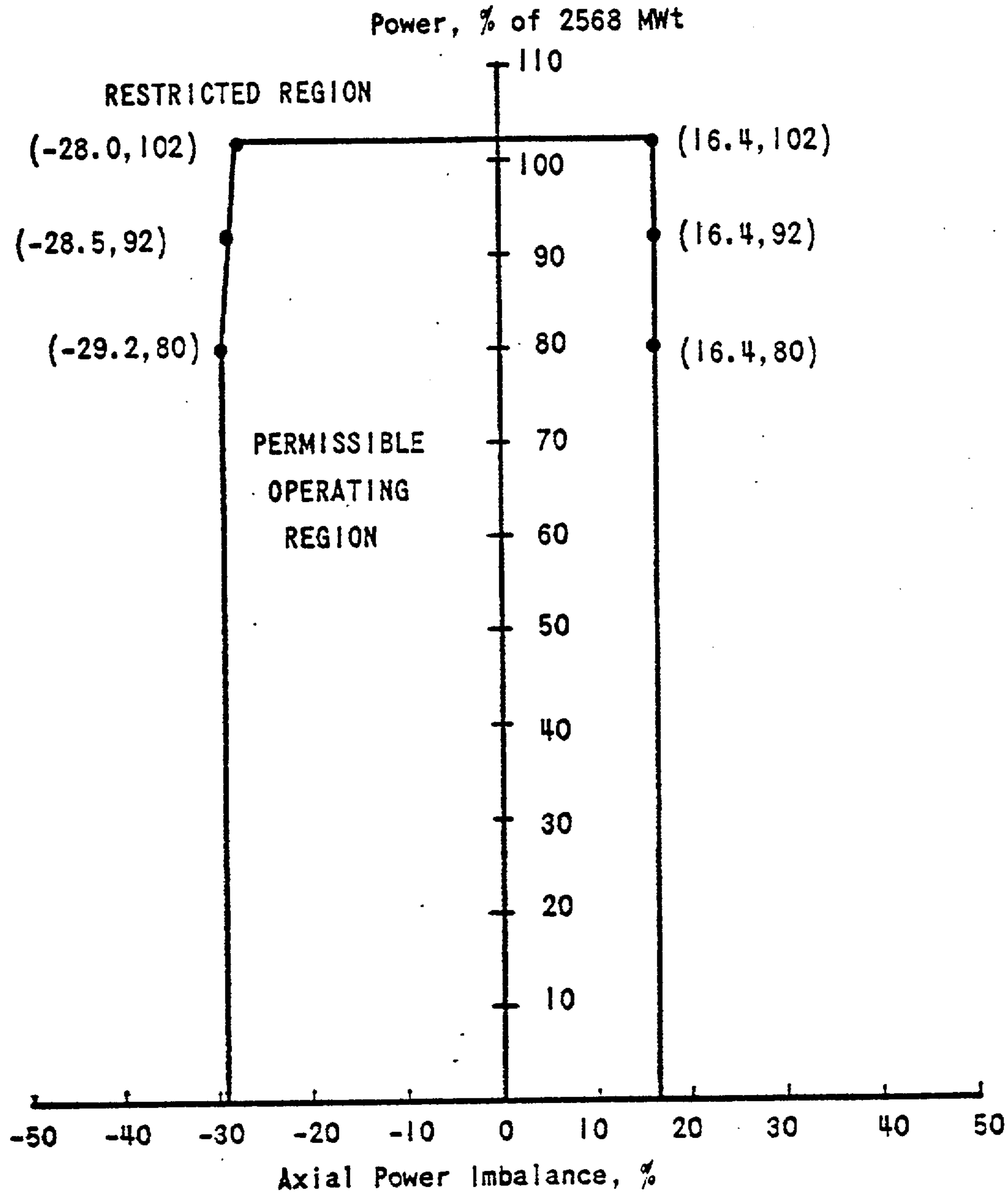
Figure 3.5.2-2A2



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 0 to 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

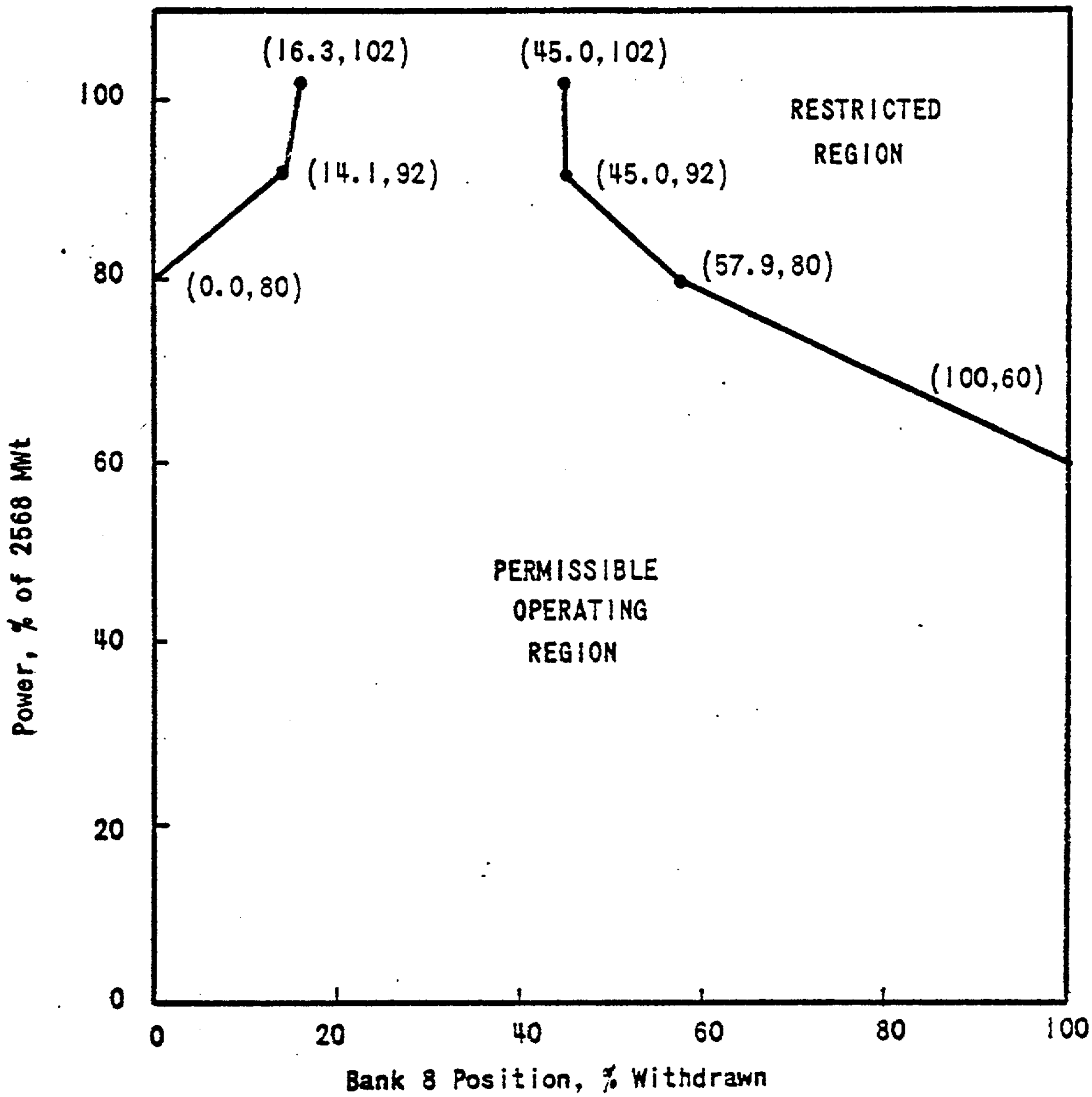
Figure 3.5.2-3A1



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 100 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

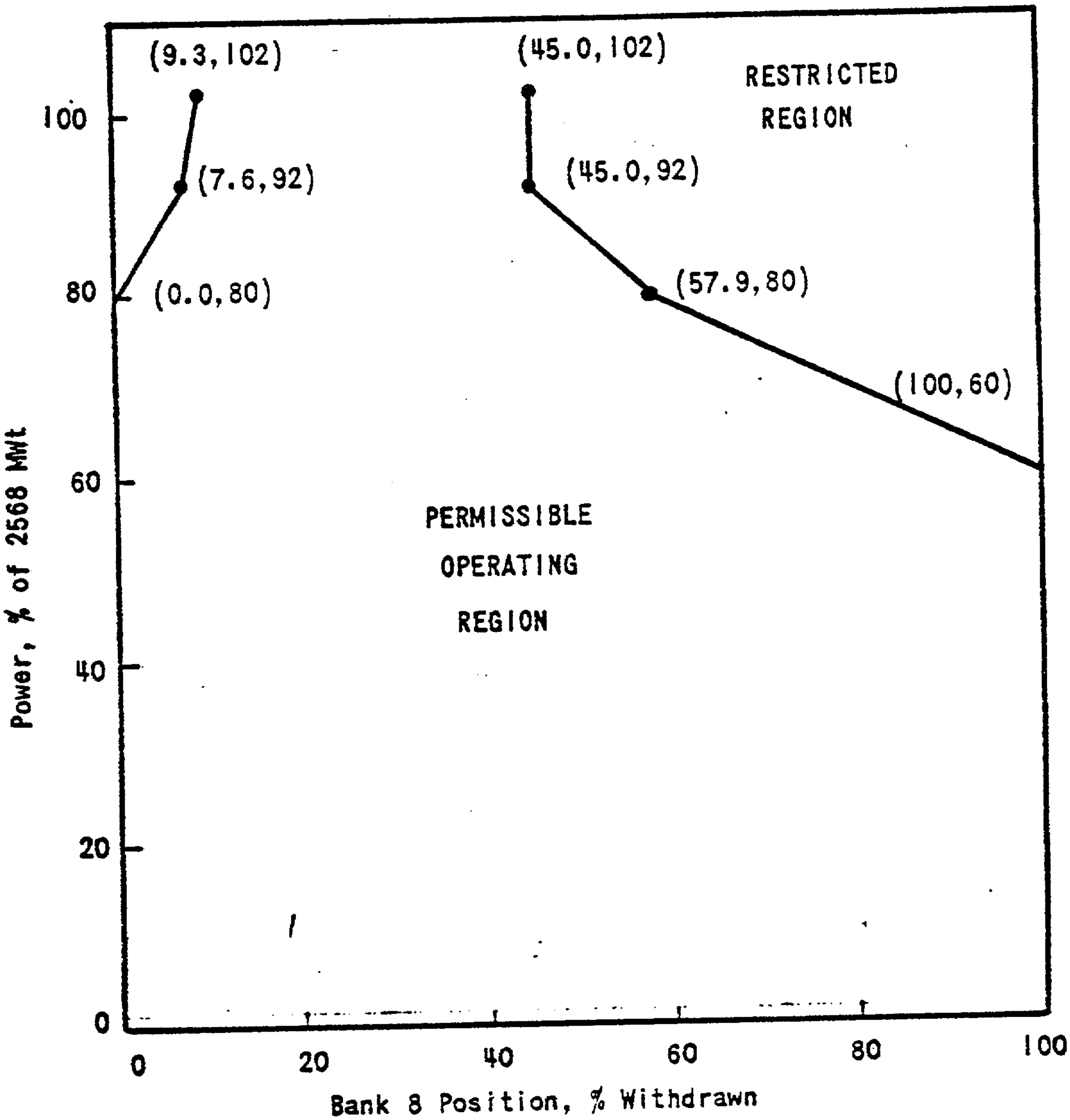
Figure 3.5.2-3A2



APSR POSITION LIMITS
FOR OPERATION FROM 0 to 100 $\pm$ 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-4A1



APSR POSITION LIMITS
FOR OPERATION AFTER 100 ± 10 EFPD
Unit 1

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.

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.

6.4 STATION OPERATING PROCEDURES

Specification

6.4.1 The station shall be operated and maintained in accordance with approved procedures. Written procedures with appropriate check-off lists and instructions shall be provided for the following conditions:

a. Normal startup, operation and shutdown of the complete facility and of all systems and components involving nuclear safety of the facility.

b. Refueling operations.

c. Actions taken to correct specific and foreseen potential malfunctions of systems or components involving nuclear safety and radiation levels, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.

d. Emergency procedures involving potential or actual release of radioactivity.

e. Preventive or corrective maintenance which could affect nuclear safety or radiation exposure to personnel.

f. Station survey following an earthquake.

g. Radiation control procedures.

h. Operation of radioactive waste management systems.

i. Control of pH in recirculated coolant after loss-of-coolant accident. Procedure shall state that pH will be measured and the addition of appropriate caustic to coolant will commence within 30 minutes after switchover to recirculation mode of core cooling to adjust the pH to a range of 7.0 to 8.0 within 24 hours.

j. Nuclear safety-related periodic test procedures.

k. Long-term emergency core cooling systems. Procedures shall include provision for remote or local operation of system components necessary to establish high and low pressure injection within 15 minutes after a line break.

l. Fire Protection Program implementation.

6.4.2 Quarterly selected drills shall be conducted on site emergency procedures including assembly preparatory to evacuation off site and a check of the adequacy of communications with off-site support groups.

6.4.3 A respiratory protective program approved by the Commission shall be in force.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Duke Power Company	)	DOCKET NO. 50-269
	)	
Oconee Nuclear Station Unit No. 1	)	

EXEMPTION

I.

Duke Power Company (the licensee) is the holder of Facility Operating License No. DPR-38 which authorizes the operation of the nuclear power reactor known as Oconee Nuclear Station, Unit No. 1 (the facility), at steady reactor power levels not in excess of 2568 megawatts thermal (rated power). The facility consists of a Babcock & Wilcox (B&W) designed pressurized water reactor (PWR) located at the licensee's site in Oconee County, South Carolina.

II.

In accordance with the requirements of the Commission's Emergency Core Cooling System (ECCS) Acceptance Criteria, 10 CFR 50.46, the licensee submitted on July 9, 1975 an ECCS evaluation for the facility. The ECCS performance submitted by the licensee was based upon an ECCS Evaluation Model developed by B&W, the designer of the Nuclear Steam Supply System for this facility. The B&W ECCS Evaluation Model had been previously found to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50.46, and Appendix K. The evaluation indicated that with the limits set forth in the facility's Technical Specifications, the ECCS cooling performance for the facility would conform with the criteria contained in 10 CFR 50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

On April 12, 1978, B&W informed the NRC that it had determined that in the event of a small break Loss of Coolant Accident (LOCA) on the discharge side of a reactor coolant pump, high pressure injection (HPI) flow to the core could be reduced somewhat. Subsequent calculations indicated that in such a case the calculated peak clad temperature might exceed 2200°F.

Previous small break analyses for B&W 177 fuel assembly (FA) lowered loop plants had identified the limiting small break to be in the suction line of the reactor coolant pump. Recent analyses have shown that the discharge line break is more limiting than the suction line break.

The Oconee Nuclear Station Unit No. 1 has an ECCS configuration which consists of two HPI trains which are supplied by three HPI pumps. Each train injects into two of the four reactor coolant system (RCS) cold legs on the discharge side of the RCS pump. The two parallel HPI trains are connected but are kept isolated by manual valves (known as the cross-over valves) that are normally closed.

Duke Power has proposed by letter dated April 21, 1978, to maintain all three pumps in an operable status. The Oconee emergency power system is designed with sufficient capacity for this mode of operation. Upon receiving a safety injection signal the HPI pumps are started and valves in the injection lines are opened. Assuming loss of offsite power and the worst single failure (the HPI pump C or the HPI valve HP26), two HPI pumps would still be available and only one of the two injection valves would fail to open.

If a small break is postulated to occur in the RCS piping between the RCS pump discharge and the reactor vessel, the high pressure injection flow injected into this line (about 50% of the output of two high pressure pumps) could flow out the break. Therefore, for the worst combination of break location and single failure, 50% of the flow rate of two high pressure ECCS pumps would contribute to maintaining the coolant inventory in the reactor vessel. This situation had not been previously analyzed and B&W had indicated that the limits specified in 10 CFR 50.46 may be exceeded.

B&W has stated that they have analyzed a spectrum of small breaks in the pump discharge line and have determined that to meet the limits of 10 CFR 50.46(b), operator action is required to open the two manual operated crossover valves and to manually align the motor driven isolation valve which had failed to open. This would allow the flow from the two HPI pumps to feed all four reactor coolant legs. B&W has assumed that 30% of the flow would be lost through the break and 70% would enter the core. The licensee has committed to provide for the necessary operator actions within the required time frame. That is, in the event of a small break and a limiting single failure, manual action will be taken to begin opening these valves within five minutes and have them fully opened and an adequate flow split obtained within the following 10 minutes. The analyses performed by B&W assumed that the flow split was established at 650 seconds by operator action. We conclude that the analyses are a reasonable approximation of the operator action that actually will be taken, provided specific procedures are prepared and followed to assure such action.

B&W has prepared a summary entitled "Analysis of Small Breaks in the Reactor Coolant Pump Discharge Piping for the B&W Lowered Loop 177 FA Plants," April 24, 1978 (the B&W Summary), which describes the methods used and the results obtained in the above analysis. The analysis models operator action by assuming a step increase in flow to the reactor vessel (with balanced flow in the three intact loops) ten minutes after the LOCA reactor protection system trip signal occurs.

On April 26, 1978, the Commission issued an Order for Modification of License which amended the license for Oconee Unit 1 requiring (1) submission of a reevaluation of the emergency core cooling system calculated in accordance with the B&W Evaluation Model for operation with operating procedures described in the licensee's letter of April 21, 1978 and (2) operation in accordance with the procedures described in the licensee's letter of April 21, 1978.

By letter dated May 16, 1978, the licensee submitted a copy of the B&W Summary for our review. In their submittal the licensee stated that the analysis indicates that the ECCS cooling performance calculated in accordance with the B&W Evaluation Model for operation of Oconee units at the rated core thermal power of 2568 Mwt with operating procedures described in their letter of April 21, 1978, is wholly in conformance with the provisions of 10 CFR 50.46. We have reviewed the B&W Summary and find that the methods of analysis meet the requirements of 10 CFR Part 50.46.

By letter dated April 20, 1978 and as supplemented on April 27, 1978, the licensee submitted proposed Technical Specifications to implement the operating procedures and maintenance of all three HPI pumps in an operable status as described in the licensee's April 21, 1978 letter. We are issuing these Technical Specifications in the license amendment accompanying this Exemption.

On August 21, 1978, the licensee requested an exemption from the provisions of 50.46.

In the licensee's submittal of June 8, 1978, it was stated that to meet the limits of 10 CFR 50.46, operator action at the valve locations is required to open High Pressure Injection (HPI) Pump B-C discharge header cross over valves (HP-116 and HP-117) and the HPI injection line A engineering safeguards valve (HP-26) within 10 minutes.

Reliance on local operation of valves this soon after the onset of a loss-of-coolant accident is not desirable on a permanent basis. The licensee has requested an exemption from the requirements of 10 CFR 50.46 for operation at Oconee 1 during Cycle 5 until such time as a permanent solution to this problem can be implemented.

The original concern derived from an unexpected but nevertheless inadequate assessment of a spectrum of breaks. This deviation from 10 CFR 50.46 has been ameliorated on a temporary basis by the actions discussed herein. However, combined reliance on prompt operator action to perform the required steps to assure plant safety over a period of years into the future is undesirable and should be replaced as promptly as possible by returning the system to automatic or control room actuation. To this extent, the original defect still remains until the modifications are made to eliminate the reliance on prompt operator actions.

We have reviewed the effects of changes made to the facility during the current refueling outage and have concluded that operation of Oconee Unit 1 at power levels of up to 2568 Mwt and in accordance with the Technical Specifications will assure that the ECCS system will conform to the performance criteria of 10 CFR 50.46. Accordingly, until modifications are completed to achieve full compliance with 10 CFR 50.46, operation of the facility at power levels up to 2568 Mwt with appropriate operating procedures will not endanger life or property or the common defense and security.

While Oconee Unit No. 1 does not comply with our requirements for ECCS, appropriate actions, as previously described, have been taken to mitigate the consequences of any accidents at this plant. The Technical Specifications will provide protection against the subject small break LOCA and will bring plant operation wholly in conformance with 10 CFR 50.46. These Technical Specifications will be in force only for the brief interval of time until the proposed modifications of the ECCS are completed. The public interest is served in that by issuing this exemption for Unit No. 1 a significant power reduction with no concommitant increase in safety is avoided. Such a power reduction could affect system reliability, cause unemployment and increase consumer power costs in the area.

III.

Copies of the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, Washington, D.C. 20555, and are being placed in the Commission's local public document room at the Oconee County Library, 201 South Spring, Walhalla, South Carolina.

(1) The application for exemption dated August 21, 1978, and

(2) This Exemption in the matter of Duke Power Company, Oconee Nuclear Station, Unit No. 1.

IV.

WHEREFORE, in accordance with the Commission's regulations as set forth in 10 CFR 50.12, the licensee is hereby granted an exemption from the provisions of 10 CFR Part 50, Paragraph 50.46(a). With respect to Oconee Unit 1 this exemption supersedes the conditions of the Commission's Order for Modification of License dated April 26, 1978, and is conditioned as follows:

(1) The licensee has submitted the plans and schedules to modify the facility to eliminate reliance on prompt operator action described herein. Additional guidance in these areas has been provided by the NRC letter of September 26, 1978 to Duke Power Company.

(2) Upon approval by the staff the licensee shall undertake such modifications in accordance with the approved schedule.

(3) This exemption shall be terminated upon completion of the modifications in accordance with this exemption or upon shutdown for the next scheduled refueling outage, whichever occurs first.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this 23rd day of October 1978.

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

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 62 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

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

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

Introduction

By the applications dated April 20, 1978(13) and June 26, 1978(1), as supplemented April 27, 1978(14), August 21, 1978(12), August 28, 1978(2), September 6, 1978(3), September 22, 1978(4), and September 26, 1978(11), Duke Power Company (the licensee) proposed to change the common Technical Specifications (TS) for the Oconee Nuclear Station, Units Nos. 1, 2 and 3 in connection with the refueling of Unit No. 1 for Cycle 5 operation. The refueling consists of the replacement of 61 burned fuel assemblies by 56 fresh assemblies and five previously burned assemblies. The five previously burned assemblies were last irradiated in Cycle 4 of Oconee Unit No. 1. These assemblies will be irradiated for a fourth cycle as part of a joint Duke Power/Babcock & Wilcox (B&W)/Department of Energy program to demonstrate reliable fuel performance at extended burnups and to obtain post-irradiation data.

Because of performance anomalies observed at other B&W plants, orifice rod assemblies will not be used in Cycle 5.

Cycle 5 will nominally extend for one year. The design cycle length is 320 effective full power days (EFPD). The mode of operation will be feed-and-bleed. Operation of the reactor was converted from the rodded mode to feed-and-bleed to increase operating margin because of a quadrant tilt problem in Cycle 4. The Cycle 5 fuel shuffle pattern was designed to minimize the effects of any power tilt present in Cycle 4.

Reactivity control during Cycle 5 will be accomplished using the 61 full-length Ag-I_n-Cd control rods and by soluble boron shim. In addition to the full-length control rods, eight axial power shaping rods are provided for additional control of the axial power distribution. Neither control rod interchange nor burnable poison rods are necessary for Cycle 5.

Analyses performed for the Cycle 5 reload core design were based on the following assumptions:

1) Cycle 4 operation is terminated at 250 EFPD.

2) Cycle 5 operation will not exceed 320 EFPD.

The licensee has proposed the following changes to the Technical Specifications for Unit 1. These changes are in accord with the analysis used to support Cycle 5 operation.

1) Revise the protective system maximum allowable setpoint contained in Specifications 2.1 and 2.3 respectively.

2) Revise the xenon reactivity hold Specification.

3) Change the steady state quadrant power tilt limit to 5.00%.

4) Change Specification Figures 3.5.2-1A1, 3.5.2-1A2, 3.5.2-2A1, 3.5.2-2A2, Rod Position Limits, Figures 3.5.2-3A1, 3.5.2-3A2, Power Imbalance Limits, and Figures 3.5.2-4A1, 3.5.2-4A2, Axial Power Shaping Rod (APSR) Position Limits.

5) Add requirements on High Pressure Injection pump operability and operating procedures.

I. Safety Evaluation

Fuel Mechanical Design

The batch 7 fresh fuel uses the Mark B4 fuel assembly design reviewed and accepted by us for use during Cycle 3. Also, these types of fuel assemblies are currently operating in Oconee 3 and Arkansas Nuclear One, Unit No. 1 (ANO-1).

The batch 7 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previous successful operations with equivalent fuel, we conclude that the fuel mechanical design of this fuel is acceptable and does not decrease the safety margin.

Five batch 4D Mark B3 assemblies will remain in the core for their fourth cycle of irradiation and will experience burnups up to approximately 41,000 MWD/MTU. This is part of a joint Duke Power/B&W/ Department of Energy program to demonstrate extended burnup feasibility in light water reactors (LWR's). The fuel is predicted to maintain its structural integrity with these burnups. The licensee states that the fuel parameters most affected by amount of irradiation are fuel rod and assembly growth and fuel swelling. These parameters will remain within the original batch 4D design limits during the Cycle 5 irradiation, as the Final Safety Analysis Report (FSAR) design basis burnup is 44,000 MWD/MTU, significantly greater than the planned 41,000 MWD/MTU exposure. The licensee's evaluation of post irradiation data from two cycles of operation in the Oconee 1 reactor indicate the fuel holddown spring force, which is affected by residence time as well as burnup, will meet performance requirements through the fourth cycle of irradiation.

Creep collapse time of the cladding was calculated to be in excess of 30,000 effective full power hours (EFPH) which is longer than the maximum fuel design exposure for Cycle 5 of 28,469 EFPH for batch 4D fuel. The calculation of creep collapse time was performed using the power history of the limiting fuel assembly. As was done in previous analyses, the CROV computer code was used to predict the collapse time(5). The licensee stated(6) and we agree that the CROV code conservatively predicts cladding collapse.

Additional conservatisms used in the CROV calculations were that no credit was taken for fission gas release; the cladding thickness used in CROV was the lower tolerance limit (LTL) of the as-built measurements; and the lowest as-fabricated pellet densities were assumed to be located in the worst case power region of the core.

The fuel cladding strain analysis was performed using a number of conservative assumptions: maximum allowable fuel pellet diameter and density; lowest permitted tolerance for the cladding inner diameter; conservatively high local pellet burnup; and conservatively high heat generation rate. This insures that the 1.0% limit on cladding plastic circumferential strain is not violated.

We find that the licensee's evaluation of the batch 4D fuel assemblies provides reasonable assurance that the fuel can safely be irradiated for a fourth cycle. Furthermore, coolant activity TS are based upon the equivalence of 1% failed fuel in the reactor. This specification would halt operation of the reactor in the unlikely event that predictions of a low failure rate for the batch 4D fuel are grossly in error. Since the activity corresponding to failure of 1% of the fuel remains a limiting condition for operation of the reactor, irradiation of the five batch 4D fuel assemblies does not result in a reduction of safety margin for Unit 1 Cycle 5 operation.

Fuel Thermal Design

The batch 7 fuel produces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. As was done in earlier Oconee reload calculations, the linear heat rate (LHR) capability was calculated using the TAFY-3 computer code[7]. The nominal LHR for Cycle 5 is 5.80 kw/ft and the LHR capability is 20.15 kw/ft.

During the last several years, data have become available that indicate the fission gas release rate from LWR fuel pellets increases with burnup. The effect of enhanced fission gas release on Emergency Core Cooling System (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 Loss of Coolant Accident (LOCA) analyses. These inputs were calculated for the Oconee 1[5] reload using the TAFY-3[7] 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 also 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/T fuel rod burnup. We have reviewed this application of the TACO code and concur in the results. The limiting LOCA calculation for this cycle of Oconee 1 occurs at a burnup within this range. Thus, the use of TAFY-3 to calculate the fuel rod pressure and volumetric average temperature input for the LOCA analysis conservatively bounds the effects of enhanced fission gas release.

Nuclear Analysis

The reactor core physics parameters for Oconee 1 Cycle 5 operation were calculated using a PDQ computer code. Since the core has not yet reached an equilibrium cycle, there were minor differences in the physics parameters between the Cycle 5 and Cycle 4 cores.

The licensee proposed a change in the plant Technical Specifications increasing the allowable steady state quadrant tilt from 3.41% to 5.00%. This tilt allowance was appropriately accounted for in the licensee's derivation of rod position, axial shape index, and minimum reactor trip setpoint analyses, and is therefore acceptable.

There was a quadrant flux tilt present in the Oconee 1 reactor during Cycle 4[8]. This tilt was 2.4% when full power operation was achieved, and burned out to an insignificant level during the cycle. The shuffle pattern for Cycle 5 was designed to minimize the carry over of any Cycle 4 tilt to Cycle 5. In response to our questions, the licensee provided[2] details of the new shuffle pattern. We have reviewed this information and agree the shuffle pattern will effectively minimize tilt carry over effects from one cycle to the next.

The original Technical Specification tilt limit for Cycle 4 was 3.41%. Early in Cycle 4, a tilt anomaly occurred resulting in the core quadrant tilt exceeding the then current limit. The staff reviewed and approved an increased tilt limit of 6%, with concommittant compensating changes to the Technical Specifications. As the cycle proceeded the tilt decreased. After extensive discussion and study, the licensee proposed and the staff accepted a reduction of the limit to its original value of 3.41% with again concommittant compensating changes to the Technical Specifications. The licensee has established and verified the cause of the Cycle 4 tilt. The tilt was attributed to asymmetry at the end of Cycle 3 burnup distribution which was accentuated by the core loading pattern for Cycle 4. The licensee has revised his methods of selection of the core loading scheme in order to reduce the future potential for core isotopic asymmetries and resultant quadrant tilt. We have reviewed the licensee's analysis of this situation and find it acceptable. The Cycle 4 core exhibited a decreasing tilt with increasing core burnup. During the latter part of Cycle 4, the tilt magnitude was in the order of less than or equal to 1% (the normal range of expected measured tilt).

The licensee has now proposed to increase the current quadrant tilt Technical Specification limit to 5%. The quadrant tilt Technical Specification in conjunction with the control rod insertion limit and power imbalance limit Technical Specifications ensure that plant limiting conditions for operation are not exceeded. These conditions ensure that limiting values of linear heat generation rate and peak enthalpy rise assumed in the safety analysis are not exceeded. These limiting values are not altered by the proposed Technical Specification change. The margin to safety and operating limits have not been altered; hence the Oconee 1, Cycle 5, core is not anticipated to exhibit future anomalous tilt behavior. The change does not alter the probability that the core will exhibit anomalous behavior. Hence, the change is acceptable. The increased tilt limit permits greater operating flexibility with no decrease in safety margin.

The licensee proposed a change to TS 3.5.2.6, Xenon Reactivity. This specification will limit potential Xenon reactivity transients and the associated change in transient power distribution during power operation by restricting the nonequilibrium Xenon reactivity. During steady state operation and power maneuvers at or near rated power, transient Xenon power distribution effects would be compensated for by a proposed 5% allowance in the power imbalance analyses, TS 3.5.2.7, and in the control rod position limit analyses, TS 3.2.2.5. In response to staff questions the licensee has shown the adequacy of the 5% allowance(2). The magnitude of the nonequilibrium Xenon reactivity is calculated by the reactor operator as a function of fuel burnup, core power and power history.

This TS is common to reactors that use the feed and bleed operational mode such as Oconee 1. This change is intended to limit transient Xenon reactivity. Section 3 of the TS limits power operation below the power level cut-off point until "the reactor has operated within a range of 87 to 92% of rated thermal power for a period exceeding two hours in the soluable poison control mode." This TS ensures that plant operation will be in conformance with the assumptions of the analyses described above. Based on the licensee response in Reference 2 and on the fact that this specification has been accepted for use, for the discussed purpose, at other operating reactors (e.g., Rancho Seco), the staff finds this change acceptable.

We find that, based on our review of the licensee's nuclear analysis techniques and their commitment to perform acceptable physics startup testing, the Oconee 1 nuclear analysis is acceptable. The proposed Technical Specifications of APSR position limits and the usual regulating control rod and imbalance limits, which assure that the LOCA LHR limits are not exceeded, are acceptable because the licensee has determined these limits using appropriate parameters for Cycle 5 and analysis techniques approved for earlier cycles of the Oconee reactors.

Thermal-Hydraulic Analyses

The licensee is proposing to remove all the Orifice Rod Assemblies (ORA) and has revised the thermal-hydraulic analysis accordingly[3]. The core bypass flow has increased to 10.4% (106 ORAs removed) from the 8.34% value used for Cycle 4 analysis (44 ORAs removed).

To offset the increase in core bypass flow, the reference design radial times local peaking factor ($F_{\Delta h}$) has been reduced from 1.78 to 1.71. The most limiting transient, the loss of two reactor coolant pumps, has been reanalyzed with an $F_{\Delta h}$ of 1.71 and the minimum Departure from Nucleate Boiling Ratio (DNBR) remains above 1.3, with the trip setpoints previously established for Cycles 3 and 4. The ORAs were also removed from Oconee Unit 3. This was recently approved for the Unit 3 reload[9].

We have reviewed the licensee's analyses and conclude that the thermal hydraulic analyses for Oconee 1 cycle 5 are acceptable.

Accident and Transient Analysis

The accident and transient analysis provided by the licensee demonstrates that the Oconee FSAR analyses conservatively bound the predicted conditions of the Oconee Unit 1 Cycle 5 core and are, therefore, acceptable. Each FSAR accident analysis has been examined, with respect to changes in Cycle 5 parameters, to determine the effects of the reload and to insure that performance is not degraded during hypothetical transients. The core thermal parameters used in the FSAR accident analysis were design operating values based on calculated values plus uncertainties. FSAR values of core thermal parameters were compared with those used in the Cycle 5 analysis. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in the Oconee Unit 1 fuel densification report(10). Since Cycle 5 reload fuel assemblies contain fuel rods with theoretical density higher than those considered there, the conclusions derived in that report are valid for Oconee Unit 1 Cycle 5. The limited conditions of the analyses for transients in Cycle 5 are bounded by the initial conditions for previous analyses performed in either the FSAR, the fuel densification report or previous reload submittals. Calculational techniques and methods for Cycle 5 analyses remain consistent with those used for the FSAR. No new dose calculations were performed for Cycle 5 operation. The dose considerations in the FSAR are based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

ECCS Analysis

This matter has been separately considered by the staff and is discussed in the NRC's Order in the captioned matter dated April 26, 1978, and in the NRC's Exemption in the captioned matter dated October 23, 1978, which accompanies this Safety Evaluation.

Physics Startup Tests

The physics startup test program for Cycle 5 as stated in Section 9 of the reload submittal has been reviewed. The physics startup test program includes zero power measurements of critical boron concentration, temperature coefficients, ejected control rod worth and control rod group reactivity worth. Power distribution, temperature coefficient and power coefficient measurements will be made at higher powers. The acceptance criteria and the actions to be taken if the acceptance criteria are not met were reviewed as well as the tests. The licensee has stated(11) that the action to be taken if the sum of the worth of groups 5, 6 & 7 differs from the predicted by more than ±10%, is to measure group 4 and that if the sum of the worths of groups 4, 5, 6 and 7 differs from the predicted by more than ±10%, additional measurements, as well as evaluation of the discrepancy, will be made.

A summary of the results of this test program will be submitted to the NRC. This entire program has been reviewed by the NRC staff and found to be acceptable.

Effects of Fuel Demonstration Program on Accident Analysis

Irradiating the entire core to extended burnups of about 41,000 MWD/MTU, not just the five demonstration fuel assemblies, would increase the amount of long-lived fission products in the core. The only significant long-lived radioisotope of concern with respect to the potential consequences of the postulated design basis accidents is the noble gas

Krypton 85. Even if the entire core burnup were extended to 44,000 MWD/MTU, the FSAR assumption for Design Basis Accidents, the amount of Krytpon 85 generated would not show an increase; therefore, the potential consequences of the postulated design basis accidents given in our Safety Evaluation (SE) dated December 29, 1970, for Oconee Unit 1 will not change because fuel assemblies in the core will be irradiated to burnups of only 41,000 MWD/MTU, and only for five fuel assemblies not an entire core of 177 fuel assemblies.

Conclusion on Safety

Based on our evaluation of the reload application and available information, we conclude that it is acceptable for the licensee to proceed with Cycle 5 operation of Oconee 1 in the manner proposed.

We have reviewed the proposed changes to the Technical Specifications and find them acceptable. These consist of all the changes requested by the licensee in his letter of June 26, 1978[1], except for Figure 2.3-2A which was acceptably revised in the supplement of September 6, 1978[3], and the submittal of April 20, 1978[13], as supplemented April 27, 1978[14], which provides both for timely operator action and maintenance of all the High Pressure Injection pumps in an operable condition in the unlikely event of a small break LOCA during plant operation. The TS for the Oconee Nuclear Station, in terms of radioactivity in the primary coolant and radioactivity releases from the station need not be revised for the five batch 4D Mark B3 demonstration fuel assemblies. These TS are based on a 44,000 MWD/MTU burnup, while the demonstration assemblies will experience only about 41,000 MWD/MTU burnup.

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

II. Environmental Conclusion Regarding Cycle 5 Reload Excluding Fuel Demonstration Program

We have determined that this action does 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 this change involves 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 this change.

III. Environmental Considerations of Fuel Demonstration Program

By letter dated October 23, 1978[15], the Department of Energy (DOE) the cognizant Federal agency for the fuel demonstration program, of which Oconee Unit No. 1 is a small portion, stated that an environmental review of possible future DOE funded extended fuel burnup work and widespread utilization of the process is not required at this time.

We have considered the effect of irradiating five fuel assemblies to extend burnups in Oconee 1 on the environmental impacts from the uranium fuel cycle and from shipping fuel and waste to and from Oconee Unit 1. We conclude that these five assemblies will have no significant effect on these environmental impacts over the operating lifetime of the plant. The licensee is not expecting at this time to change the amount of uranium or the number of fuel assemblies shipped to and from the plant by irradiating the five assemblies to extended burnups. The licensee will add five fewer new fuel assemblies than normal to the core for Cycle 5 and will add five more new fuel assemblies than normal to the core for Cycle 6. The remaining cycles, as now planned, will have the normal number of new assemblies. Irradiating these five fuel assemblies to extended burnups does not increase the number of fissions in any fuel cycle for Oconee Unit 1 or over the operating lifetime of the plant, therefore, the amount of fission products generated by Oconee Unit 1 over its operating lifetime does not change. There will be more than the normal amount of long-lived fission products in the core during Cycle 5 and fewer during Cycles 6 and 7. Therefore, on the average, each fuel assembly will have the same magnitude of fission products as if these five assemblies were not irradiated to extended burnups.

The proposed action will therefore not significantly increase normal radiological effluents from the plant. It will also not allow the licensee to discharge concentrations greater than the maximum allowed nor to discharge more activity in a year than the maximum allowed. Compliance with the present TS will adequately control releases such that there will be no appreciable effect on the environment due to operation under these proposed changes.

Conclusion and Basis for Negative Declaration

On the basis of the NRC evaluation and information supplied by the licensee, it is concluded that the proposed action will have no appreciable impact on the environment due to radiological effluents from the plant and will not affect the cost/benefit balance.

Having reached these conclusions, the Commission has determined that an environmental impact statement need not be prepared for this proposed change and that a Negative Declaration to that effect should be issued.

Dated: October 23, 1978

REFERENCES

1. Ltr. from William O. Parker, Jr., Duke Power Company (DPC) to R. Reid, U. S. Nuclear Regulatory Commission (NRC), 6/26/78, forwarding the Oconee Nuclear Station, Unit No. 1, Cycle 5 Reload Report, BAW-1493.

2. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 8/28/78, forwarding additional information.

3. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 9/6/78, forwarding Revision 1 to BAW-1493.

4. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 9/22/78, forwarding additional information.

5. BAW-1493, "Oconee Unit 1, Cycle 5, Reload Report," 7/78.

6. BAW-10084, Rev. 1, "Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse," 11/76.

7. BAW-1004, "TAFY-Fuel Pin Temperature and Gas Pressure Analysis," 5/72.

8. BAW-1477, "Oconee 1 Cycle 4 Quadrant Flux Tilt," 1/78.

9. Ltr. from R. Reid (NRC) to William O. Parker, Jr., (DPC), 7/6/78.

10. BAW-1388, "Oconee 1 Fuel Densification Report, Revision 1," 7/73.

11. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 9/26/78.

12. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 8/21/78, ORA Removal and Exemption Request.

13. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 4/20/78, HPI Pump Operability.

14. Ltr. from William O. Parker, Jr., (DPC) to R. Reid (NRC), 4/27/78, Operating Procedures Related to Small Break LOCA.

15. Ltr. from Peter Lang, Department of Energy to R. Reid (NRC), 10/23/78, Environmental Considerations of Fuel Demonstration Program.

APPENDIX A

LOST STEAM GENERATOR TUBE PLUGS AT OCONEE UNIT 1
SAFETY EVALUATION REPORT
ENGINEERING BRANCH, DIVISION OF OPERATING REACTORS

I. BACKGROUND

On Thursday, October 19 Duke Power Company (the licensee) informed the NRC that two steam generator tube plugs had been lost at the Oconee Unit 1 Nuclear Power Plant and were believed to be loose in the primary coolant system. The two plugs were lost during tube plugging operations in the Unit 1, B steam generator.

One plug being installed in the top of a tube in the "1B" steam generator failed to detonate. In order to remove the faulty plug the tube was pressurized from the unplugged end and the plug was forced out of the tube. However, when an attempt was made to retrieve the plug it could not be located in the upper head region and it is therefore believed that the plug may have entered the hot leg pipe from the reactor vessel. The hot leg pipe runs horizontally from the reactor pressure vessel and then vertically through the "candy cane" configuration into the upper steam generator head.

A second plug was discovered missing during review of photos of the lower tube sheet which are taken to confirm that the tube plugging operation has been properly completed. One tube which was thought to have been plugged was determined to be unplugged from the photograph. The licensee has suggested that a tube might have been double plugged. This means that the "jumper" who inserts the tube plugs during the plugging process could have possibly inserted a second plug in a previously plugged tube rather than the tube intended for plugging. Since the plugs are inserted deep into the tube sheet this is a possibility. However, since there is no way to confirm this scenario, it must be assumed that the plug has been lost in the lower steam generator head or in the cold leg of the reactor coolant system. However, the licensee has indicated that all plugs that were placed in lower tube sheet had properly detonated. Thus it is believed that the plug lost in the lower head was detonated. The lost plugs are approximately 3 inches in length, one half inch in diameter and 1/2 pound in weight.

In their October 19, 1978 submittal and in telephone conversations on October 19 and October 20 the licensee has addressed concerns regarding (1) potential for detonation of the undetonated plug (2) consequences of plug detonation, and (3) the consequences of loose parts in the primary coolant system.

II. DISCUSSION

A. Significance of Undetonated Plug

1. Potential for Detonation of Undetonated Plug

Babcock and Wilcox has run several tests to assess the potential for detonation of the undetonated lost plug. These tests were conducted with the same type of plugs that were lost at Oconee Unit 1 which are delivered pre-assembled by the manufacturer. Two plugs were heated in pressurized (2250 psi) reactor coolant grade water (6000 ppm H_3BO_3, 1.0 ppm LiOH) to 620°F at approximately 64°F per hour. This testing indicated no evidence of detonation and examination at the conclusion of the testing indicated that the explosives had dissolved from the plug and no solids remained in the plugs. Babcock and Wilcox consultation with duPont Explosive Products Division and military explosive personnel confirmed that decomposition of the chemical explosives will occur when the plugs are heated at temperatures and rates comparable to those existing during reactor coolant system heatup. Four plugs were also heated in air as high as 980°F with no evidence of detonation. The explosive vaporizes at 290°F and therefore would not be in an explosive geometry beyond this temperature.

A second set of testing included impact testing with dry and wet plugs. Impact testing with dry material indicated that detonation could occur at an impact energy of 25 Ft-lbs. Under wet conditions impact energy as high as 185 Ft-lbs. did not cause detonation. Calculations by Babcock and Wilcox indicate that 185 ft-lbs. bounds the impact energy which a plug could be subjected to in the RCS.

2. Consequences of Detonation

Although the licensee maintains that the probability of the unrecovered, undetonated plug not decomposing and subsequently detonating during operation is negligible, they addressed the consequences of such an event in the October 19 submittal. We have been informed that a tube plug was detonated in air by the licensee. As a result of the detonation the walls of the hollow plug flared open in three sections. No shrapnel effects were observed. If a plug detonated within a steam generator tube outside of the tubesheet area, the affected tube and approximately ten surrounding tubes could be affected. The basis of this scenario is that the tube containing the plug might burst and that it could then cause damage to the immediately adjacent tubes. The primary to secondary leak which might result would be promptly detected and the unit brought to cold shutdown.

If the plug is postulated to detonate in the vicinty of the fuel assemblies, several rods could be affected. It is not believed that the explosive energy of the plug would be sufficient to damage more than a limited number of rods.

B. Significance of Detonated and Undetonated Plugs As Loose Parts

1. Loose Parts Monitoring Capability

The Loose-Parts Monitoring System (LPMS) installed at Oconee 1 is an early model of the system marketed by B&W. The LPMS uses piezoelectric crystal accelerometers to detect the sounds or vibrations associated with a loose part impacting in the primary system. The B&W system differs from that of other vendors in that the low frequency natural resonances of the pressure vessesl ("bell" frequencies) are utilized for detection, whereas other LPMS vendors use much higher, ultrasonic frequencies.

The design of the system assumes that debris in the primary coolant loop will rapidly migrate to natural collection areas, in this case the inlet plena of the reactor vessel and the two steam generators. Therefore, only these areas are instrumented with LPMS sensors. However, actual experience has shown that impacts at a considerable distance from the sensors can still be detected, although with somewhat reduced sensitivity. For example, an identical LPMS on Oconee 2 was able to detect a loose surveillance capsule tube in 1976.

The sensitivity of the LPMS is limited by the false alarm rate. At the alarm levels now in use at Oconee 1, false alarms occur at the rate of one or two per day. However, by checking loose part alarms against known events such as control rod stepping, most of these alarms can be discounted by the operations personnel. The remainder are investigated by manual monitoring using headphones or a loudspeaker. The licensee has been using this system for nearly six years, and has become quite skilled in its use. The LPMS on Unit 2 was successfully used to detect loose parts in 1974 and 1976. There was one incident on Unit 3 in 1976 where the LPMS failed to detect two small objects. However, the two objects were found lodged in place, and therefore would not be expected to trigger an LPMS alarm.

It should also be noted that a similar LPMS was used in 1978 to detect ejected burnable poison assemblies in the Crystal River 3 reactor. Since the Crystal River incident, B&W has recommended to its customers that extra attention be given to the LPMS.

The question of greatest interest for Oconee 1 is: will the LPMS detect a loose steam generator plug? Regulatory Guide 1.133 requires new plants to install systems capable of detecting impacts of energies of 1/2 ft-lb within 3 feet of a sensor. LPMS manufacturers claim no difficulty with this sensitivity, provided the background noise of the reactor is sufficiently low. Although detailed data on the Oconee system's signal to noise ratio is not readily available, it is expected that the system sensitivity is of this order. Therefore, the LPMS is probably capable of detecting a loose plug wandering randomly in an inlet plenum, since that is where the detectors are. More importantly, the system is almost certain to detect impacts energetic enough to cause damage provided some of these impacts involve the outer vessel wall or some other component with a direct acoustic path to a sensor.

2. Consequences to Reactor Internals

a. Mechanical damage

A steam generator plug weighs approximately 1/2 lb. If it is moving with the coolant (≈15 ft/sec.), it will have a kinetic energy on the order of 1 3/4 ft-lb. No data on the threshold for impact damage is available for B&W fuel. However, another reactor vendor has found that one fuel rod can absorb either one ft-lb. of bending energy, or about 250 ft-lbs. of compression loading before cladding failure. The B&W fuel rod should not be greatly different in behavior. It is not credible that a steam generator plug could enter the fuel lattice and still posess enough transverse velocity to apply 1 ft-lb. to bend a fuel rod. Nor is it credible that the plug could hit the end of a rod with sufficient velocity to cause failure due to compression loading. This does not take credit for the additional protection supplied by the grid spacers and upper and lower tie plates.

Damage to the control rods is also not credible. The control rods are protected by guide tubes when withdrawn, and are better protected than the fuel rods when inserted. It is instructive to note that the control rod guide tubes successfully protected the control rods from the considerably more massive burnable poison rod assemblies during the recent Crystal River incident.

The remainder of the internals should not be damaged by impacts of less than 2 ft-lbs. The steam generator plug should be able to travel freely about the plenum, thus there is no concern for fatigue due to repeated impacts at one location.

b. Flow blockage

Because of the small size of the steam generator plug and the relatively high cross flow within the core, it should not be possible for the plug to cause departure from nucleate boiling, even during a transient by blocking flow at the core inlet.

If the loose plug should enter the fuel lattice, which is quite improbable considering the size and weight of the object and the size of the openings in the lower tie plates, it still will probably not cause DNB. Safety analyses of such situations in the past (generally borrowed from fuel rod bowing calculations) have shown that the decreased neutron moderation caused by displacement of the moderator by the object will lower power in the immediate vicinity of the object and maintain margin to DNB. The steam generator plugs are hollow and therefore do not displace as much moderator as a solid object would.

In any case, the steam generator plug would have to travel to a high power area of the core to cause any concern with DNB, which would require the penetration of several but not all grid spacers. Moreover, only four rods would be affected. Therefore, it is concluded that flow blockage induced DNB is not a concern.

c. Mechanical interference.

The only moving parts within the reactor vessel are the control rods and the vent valves. Since the vent valves remain closed during normal operation and are needed only in the event of a LOCA, and since a loose part is not likely to remain in the upper plenum (and even less time in the downcomer), mechanical interference with the operation of the vent valves is not a problem. Interference with control rods is somewhat more serious in that control rods are moved more often, but is still not a problem because:

- interference should be detected by control rod exercise programs already in the Technical Specifications,

- The direction of flow at the slots in the control rod weldments is outward, making it difficult for a loose object to enter,

- even under the worst-case conditions of a steam line break at end-of-cycle when the reactivity defects are at their maximum, the safety analyses assume the worst rod stuck out of the core, and

- under anticipated transient conditions, it is known from calculations carried out for the ATWS investigations that the reactor will still scram even if 5 clustered rods fail to insert.

Therefore, it is concluded that mechanical interference with moving parts within the reactor vessel is not a problem.

3. Consequences to Steam Generator

If a plug is in the reactor outlet portion of the RCS, it may be carried into the upper head of the steam generator. Experience with loose objects in the steam generator upper head has shown that the plug would not become lodged but would continue to impact the upper tubesheet. Recent experience at Crystal River has shown that impacting by loose parts, much larger than a tube plug, did not result in significant damage to the twenty four inch thick tubesheet, tubesheet cladding or tube to tube sheet joints. Any significant impact would be detected by the installed Loose Parts Monitoring System and the unit would promptly be brought to shutdown condition for retrieval of the plug. Thus, any damage to the steam generator would be expected to be minimal. Furthermore, the 0.3 gpm steam generator primary to secondary leakage rate technical specification limit would require prompt corrective action in the improbable event of primary system degradation resulting from damage imparted by a loose tube plug.

4. Consequences To The Reactor Coolant Pump

The primary coolant recirculation pump is a single stage centrifugal type pump with a diffusser. The diameter of the impeller is approximately 30 inches. The manufacturer (Westinghouse) was contacted to determine what would happen to this pump if the steam generator tube plug could reach the suction and be ingested into the pump internals. They indicated that the diffusser and impeller vane passages have adequate clearance for the plug to flow through. If the tube plug were to impact the pump internals, minor damage would be incurred. He further indicated that if the plug were to be lodged within a vane passage of the impeller that there would be higher detectable vibrational levels within the pump, but that the pump would not catastrophically destruct since the pump was designed for unbalanced rotor operation.

In view of the above information, even if the plug were to flow within this pump there is reasonable assurance that pump pressure boundary integrity would be maintained and that major damage to pump internals would not occur. Furthermore, the loss of one reactor coolant pump is an event determined to be acceptable in the licensee's transient accident analysis.

5. Similar experiences at Westinghouse plants

Of the Westinghouse experience, the most similar event occurred at Turkey Point 4 in June, 1977. During a steam generator inspection and tube plugging operation, it was discovered that twelve of the steam generator tubes presumed to have been plugged during the previous outage were not plugged. A check of the plant records was unable to produce definite proof that the steam generator tube plugs had indeed been installed. The reactor was defueled and both the reactor and main coolant pipes were searched by TV cameras. No plugs were found. It was concluded that the plugs had never been installed, and the reactor was reassembled. At this point, an LPMS was installed. When the reactor coolant pumps were started, the LPMS detected a loose part impacting the lower vessel head. Subsequent testing indicated impacting only at less than full-flow conditions. During the testing, the impact indications stopped, presumably because the loose part had jammed or found a low-flow area. Analysis of the data tapes indicated that there was only one loose part moving randomly about the lower plenum. After further pump testing, which failed to dislodge the loose part and appropriate safety evaluations, the reactor was returned to service. The loose part is still in the vessel, and was heard on the LPMS during pump tests after refueling in September, 1978.

III. EVALUATION

Based on the above discussion the staff has reached the following conclusions:

1. Detonation of the undetonated plug is highly improbable. B&W has conducted sufficient testing to establish that the explosive in the plug will disintegrate in the primary coolant system environment.

2. The consequence associated with the unlikely event of the plug exploding are not unacceptable. Damage to the steam generator or reactor intervals would be minimal.

3. The significance of the tube plugs as loose parts is minimal. Loose plugs will not unacceptably affect the reactor internals, steam generators, or reactor coolant pumps. The licensee has an excellent LPMS for monitoring any activity of the loose parts.

4. Similar events in other plants have not resulted in unacceptable consequences.

It is therefore our conclusion that operation of Oconee Unit 1 with the loose plugs in the primary coolant system is acceptable.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 65, 65 and 62 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company, which revised 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.

The amendments (1) revise the Station's common Technical Specifications to support the operation of Oconee Unit No. 1 at full rated power during Cycle 5 after core reload, to add High Pressure Injection pump operability requirements, to add procedures for remote operation of the High Pressure Injection System, and to remove the orifice rod assemblies from the core; and (2) permit the use of five previously burned fuel assemblies which will be irradiated for a fourth cycle as part of a fuel demonstration program.

The applications 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 prepared an environmental impact appraisal for the fuel demonstration program (Item 2, above) and has concluded that an environmental impact statement for this particular action is not warranted because there will be no appreciable environmental impact attributable to this action.

The Commission has determined that the issuance of Item 1, above, 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 this action.

For further details with respect to this action, see (1) the applications for amendment dated April 20 and June 26, 1978, as supplemented April 27, August 21, 28, September 6, 22 and 26, 1978, (2) Amendments Nos.65,65 and 62, to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation and Environmental Impact Appraisal. 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 23rd day of October 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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