Duke Power Company

Oconee Nuclear Station

Attachment 1

Remove Pages	Insert Pages
2.1-2	2.1-2
2.1-3	2.1-3
2.1-5	2.1-5
2.3-1 through 2.3-4	2.3-1 through 2.3-4
2.3-6	2.3-6
2.3-7	2.3-7
3.1-1	3.1-1
3.1-2	3.1-2
3.5-9	3.5-9
6.9-1	6.9-1

The curve presented in Figure 2.1-1(3) represents the conditions at which the minimum allowable DNBR is predicted to occur for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based upon the design nuclear peaking factors (4,6,7):

$$F^N_{\Delta H} = 1.714$$

$$F^N_Z = 1.50$$

Since power peaking is not a directly measurable quantity, DNBR limited power peaks and fuel melt limited power peaks are separately correlated to measurable reactor power and power imbalance. The reactor power imbalance limits,

Figure 2.1-2(5), define the values of reactor power as a function of axial imbalance that correspond to the more restrictive of two thermal limits - MDNBR equal to the DNBR limit or the linear heat rate equal to the centerline fuel melt limit.

The core protection safety limits are based on an RCS flow less than or equal to 385,440 gpm (4 pump operation). Three pump operation is analyzed assuming 74.7 percent of four pump flow. The maximum thermal power for three pump operation is 84.9 percent (Figure 2.1-2) due to a power level trip produced by the flux/flow ratio (74.7 percent flow x 1.07 = 79.9 percent power = 84.9 percent power adding the maximum calibration and instrument error).

REFERENCES

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

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

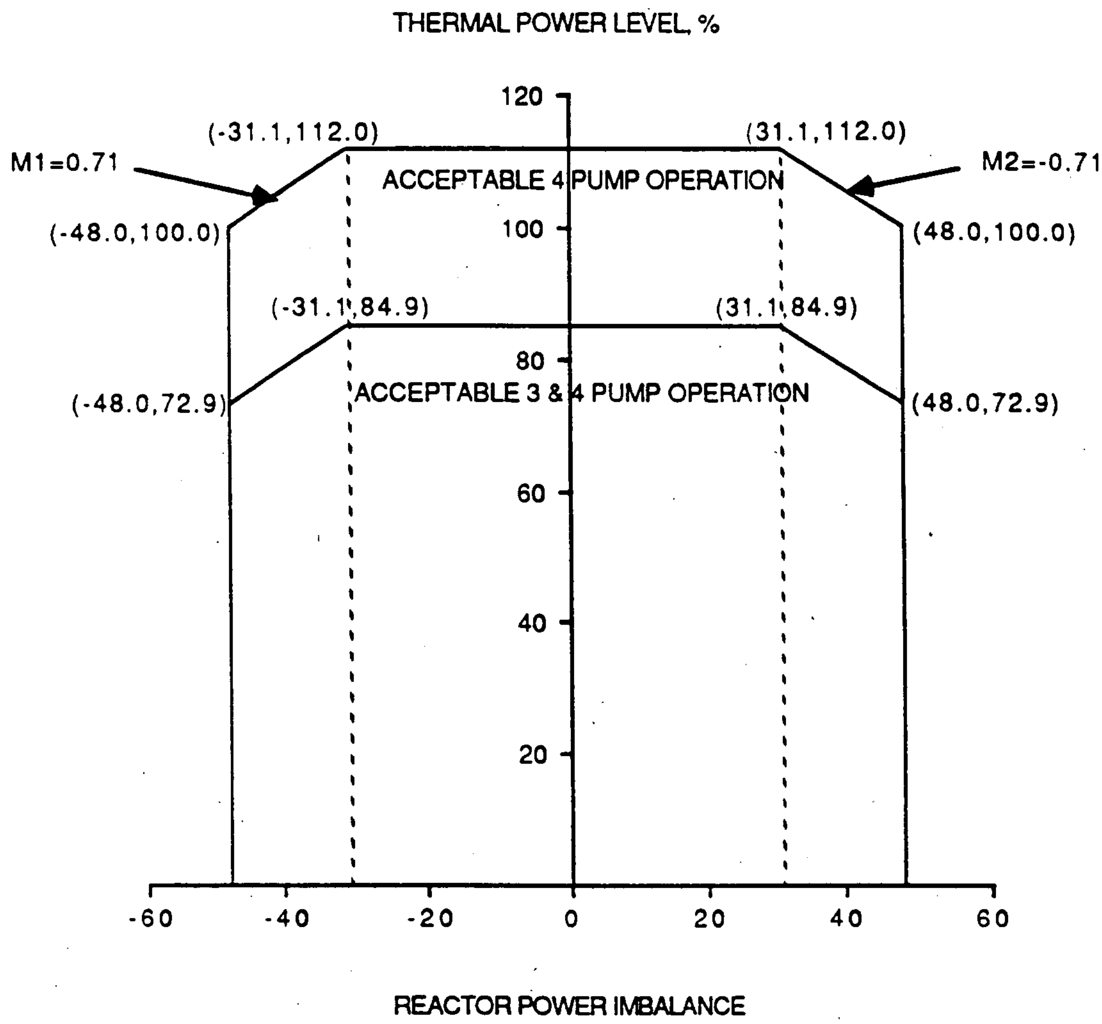
(3) Oconee Unit 3, Cycle 7 - Reload Report, DPC-RD-2001, Rev. 1, Duke Power Company, July 1982.

(4) Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002A, Duke Power Company, October 1985.

(5) Oconee Unit 2, Cycle 7 - Reload Report, DPC-RD-2002, Duke Power Company, September 1983.

(6) Oconee Nuclear Station Core Thermal Hydraulic Methodology using VIPRE-01, DPC-NE-2003A, Duke Power Company, July 1989.

(7) Oconee Nuclear Station Reload Design Methodology, NFS-1001A, Duke Power Company, April 1984.



CORE PROTECTION SAFETY LIMITS UNITS 1, 2, AND 3

Figure 2.1-2

OCONEE NUCLEAR STATION

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protective action to prevent any combination of process variables from exceeding a safety limit.

Specification

The reactor protective system trip setpoints and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and Figure 2.3-2.

The pump monitors shall produce a reactor trip when a loss of two pumps occurs and reactor power level is greater than 0.0% of rated power.

Bases

The reactor trip setpoints for reactor protective system (RPS) instrumentation are given in Table 2.3-1. The trip setpoints have been selected to ensure that the core and reactor coolant system are prevented from exceeding their safety limits. The various reactor trip circuits automatically open the reactor trip breakers whenever a parameter monitored by the RPS deviates from an allowed range. The RPS consists of four instrument channels for redundancy. The plant safety analyses are based on the trip setpoints given in Table 2.3-1 plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, a reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in the trip setpoint due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis. (1)

Overpower Trip Based on Flow and Imbalance

Following the loss of one or more reactor coolant pumps, the core is prevented from violating the minimum DNBR criterion by a reactor trip initiated by exceeding the allowable reactor power to reactor coolant flow (flux/flow) ratio setpoint. Loss of one or more reactor coolant pumps is also detected by the pump monitors. The power level trip produced by the flux/flow ratio provides DNB protection for all modes of pump operation.

The power level trip setpoint produced by the flux/flow ratio provides both high power level and low flow protection. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible flow rate. Typical power level and flow rate combinations for different pump situations are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.46% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.9% and reactor flow rate is 74.7% or flow rate is 70.09% and power level is 75%.

   The analysis to determine the flux/flow setpoint accounts for calibration and instrument errors and the variation in RC flow in such a manner as to ensure a conservative setpoint. Statistical methods are used to determine the combined effects of calibration and instrument uncertainties with the final string uncertainties used in the analysis corresponding to the 95/95 tolerance limits.

The reactor power imbalance (power in the top half of the core minus the power in the bottom half) reduces the power level trip produced by the flux/flow ratio as shown in Figure 2.3-2. The flux/flow ratio reduces the power level trip and associated power-imbalance boundaries by 1.07% for a 1% flow reduction. The power-imbalance boundaries shown in Figure 2.3-2 are established to prevent fuel thermal limits, DNBR and centerline fuel melt limits, from being exceeded.

Pump Monitors

The pump monitors trip the reactor due to the loss of reactor coolant pump(s) to ensure the DNBR remains above the minimum allowable DNBR. The pump monitors provide redundant trip protection of DNB; tripping the reactor on a signal diverse from that of the flux/flow trip. The pump monitors also restrict the power level depending on the number of operating reactor coolant pumps.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdraw from high power, the reactor coolant system (RCS) high pressure setpoint is reached before the nuclear overpower trip setpoint. The high RCS pressure trip setpoint (2355 psig) ensures that the pressure remains below the safety limit (2750 psig) for any design transient. (2) The low pressure (1800 psig) and variable low pressure ($11.14\ T_{out} - 4706$) trip setpoints shown in Figure 2.3-1 ensure that the minimum DNBR is greater than or equal to minimum allowable DNBR for those accidents that result in a reduction in pressure. (3,4) The limits shown in Figure 2.3-1 bound the pressure-temperature curves calculated for 4 and 3 pump operation.

Accounting for calibration and instrumentation errors, the safety analyses used a variable low RCS pressure trip setpoint of ($11.14\ T_{out} - 4756$).

Coolant Outlet Temperature

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

Reactor Building Pressure

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

Shutdown Bypass

In order to startup the reactor and to be able to perform control rod drive tests and zero power physics tests (see Technical Specification 3.1.9), there is provision for bypassing certain segments of the reactor protective system (RPS). The RPS segments which can be bypassed are given in Table 2.3-1. Two conditions are imposed when the RPS is bypassed:

1. By administrative control the nuclear overpower trip setpoint is reduced to a value of $\leq$ 5.0% of rated power.

2. The high reactor coolant system pressure trip setpoint is automatically lowered to 1720 psig.

The high RCS pressure trip setpoint is reduced to prevent normal operation with part of the RPS bypassed. The reactor must be tripped before the bypass is initiated since the high pressure trip setpoint is lower than the normal low pressure trip setpoint (1800 psig).

The overpower trip setpoint of $\leq$ 5.0% prevents any significant reactor power from being produced when performing physics tests. If no reactor coolant pumps are operating, sufficient natural circulation would be available to remove 5.0% of rated power.(5)

REFERENCES

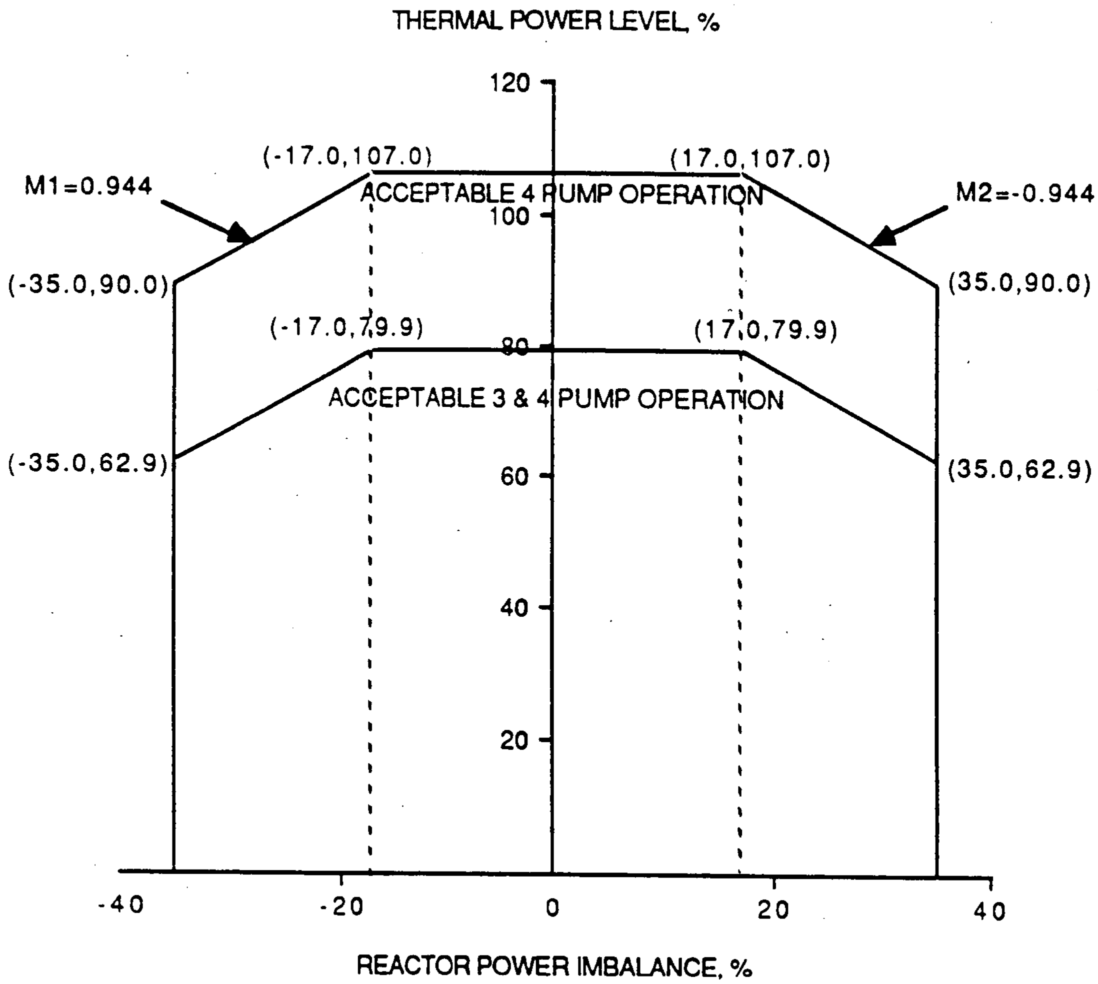
(1) FSAR, Section 15.3

(2) FSAR, Section 15.2

(3) FSAR, Section 15.7

(4) FSAR, Section 15.8

(5) FSAR, Section 15.6



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS UNITS 1, 2, AND 3

Figure 2.3-2

TABLE 2.3-1

Reactor Protective System Trip Setting Limits

	RPS Trip	RPS Trip Setpoint	Shutdown Bypass
1.	Nuclear Overpower	105.5% Rated Power	5.0% Rated Power (1)
2.	Flux/Flow/Imbalance	1.07	Bypassed
3.	Pump Monitors	> 0% Rated Power loss of two pumps	Bypassed
4.	High Reactor Coolant System Pressure	2355 psig	1720(2)
5.	Low Reactor Coolant System Pressure	1800 psig	Bypassed
6.	Variable Low Reactor Coolant System Pressure	$P (psig) = (11.14 T_{out} - 4706)$(3)	Bypassed
7.	High Reactor Coolant Temperature	618°F	618°F
8.	High Reactor Building Pressure	4 psig	4 psig

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

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

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

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system components which must be met to ensure safe reactor operation.

Specification

3.1.1 Operational Components

a. Reactor Coolant Pumps

1. Whenever the reactor is critical, one and two pump operation shall be prohibited, single-loop operation shall be restricted to testing, and other pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

2. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one low pressure injection pump is circulating reactor coolant.

b. Steam Generator

1. One steam generator shall be operable whenever the reactor coolant average temperature is above 250°F.

c. Pressurizer Safety Valves

1. All pressurizer code safety valves shall be operable whenever the reactor is critical.

2. At least one pressurizer code safety valve shall be operable whenever all reactor coolant system openings are closed, except for hydrostatic tests in accordance with the ASME Section III Boiler and Pressure Vessel Code.

Bases

A reactor coolant pump or low pressure injection pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One low pressure injection pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less. (1)

The low pressure injection system suction piping is designed for 300°F and 370 psig; thus the system with its redundant components can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident at hot shutdown. (5) The pressurizer code safety valve lift setpoint shall be set at 2500 psig ±1% allowance for error and each valve shall be capable of relieving 300,000 lb/hr of saturated steam at a pressure no greater than 3% above the set pressure.

REFERENCES

(1) FSAR, Section 6.3.3.2, and Tables 5.3-1, 5.4-2, 5.4-3, 5.4-6, 5.4-7, 5.4-8 and 6.3-2.

(2) FSAR, Sections 5.4.7-1 and 9.3.3.2.3.

(3) FSAR, Sections 5.4.7.4 and 6.3.3.2

(4) FSAR, Sections 5.2.3.10.4 and 5.4.6.

(5) FSAR, Sections 5.2.3.7 and 15.2.3.

coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 2 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

f. If the maximum positive quadrant power tilt exceeds the Maximum Limit of Table 3.5-1, the reactor shall be shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.

g. Quadrant power tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% full power.

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 tests, 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 shall be maintained within acceptable operating limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT for the particular number of operating reactor coolant pumps (4,3).

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 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 acceptable operating limits for reactor power imbalance provided in the CORE OPERATING LIMITS REPORT.

6.9 CORE OPERATING LIMITS REPORT

Specification

6.9.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle, for the following:

(1) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.

(2) Power Imbalance Limits for Specification 3.5.2.6

and shall be documented in the CORE OPERATING LIMITS REPORTS.

6.9.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically:

(1) DPC-NE-1002A, Reload Design Methodology II, October 1985.

(2) NFS-1001A, Reload Design Methodology, April 1984.

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989.

6.9.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

Attachment 2

No Significant Hazards Consideration Evaluation

No Significant Hazards Consideration Evaluation

Duke Power Company has determined that the present amendment request poses no significant hazards as defined by NRC regulations in 10CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

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

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

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

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48FR14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (iii) relates to nuclear power reactor core reloads.

Example (iii) of amendments not likely to involve significant hazards considerations expressly applies in that:

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

2) no significant changes have been made to the acceptance criteria for the Technical Specifications;

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

4) The NRC has previously found such methods acceptable.

The Batch 14 MK-B8 fuel assemblies are similar in design to the MK-B7 fuel previously reloaded into Oconee 3 Cycle 11. Like the MK-B7 fuel design, the MK-B8 features intermediate zircaloy grids, a skirtless inconnel upper end grid (dimensionally equivalent to the intermediate zircaloy grids), and a removable upper end fitting. A new feature for the MK-B8 is a debris fretting resistant fuel rod design, which utilizes a lengthened solid lower end plug extending below the bottom end grid. The fuel rods' prepressurization level was also slightly reduced to compensate for the reduction in plenum void volume.

The Oconee 3, Cycle 12 core will have 44 BPRAs inserted in the Batch 14 fuel assemblies. 36 of the BPRAs will be new, and the remaining 8 will be reinserted from the Cycle 11 core (once burned).

The Oconee 3, Cycle 12 Reload Report (Attachment 3) justifies the operation of the Cycle 12 at the rated core power of 2568 MWt. Included are the required analyses as outlined in the US NRC document "Guidance for Proposed License Amendments Relating to Refueling," June 1975. The Reload Report employs analytical techniques and design bases established in reports submitted for previous reloads which were accepted by the NRC and its predecessor. These techniques are described in the Reload Report references.

As discussed in Section 7 of the Reload Report, a generic LOCA analysis for the B&W 177-FA, lowered loop NSSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model (as reported in BAW-10103, Rev. 3). The LOCA-Limited Maximum Allowable Linear Heat Rate given in BAW-10103 has been impacted by TACO2, NUREG-0630, and FLECSET. The net effect of these factors is summarized by the LOCA kw/ft limits in BAW-1915P. The LOCA kw/ft limits given in BAW-1915P were used in the design of Oconee 3, Cycle 12.

Revisions to Technical Specifications included in this amendment request account for minor changes in power peaking and control rod worths, include use of the VIPRE thermal-hydraulic code and remove all specifications associated with two reactor coolant pump operation.

With supporting reference to previously performed analyses, the following evaluation measures aspects of this amendment request against the Part 50.92(c) requirements to demonstrate that all three standards are satisfied.

First Standard

(Amendment would not) involve a significant increase in the probability or consequences of an accident previously evaluated.

> Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in Cycle 11 parameters to determine the effect of the Cycle 12 reload and to ensure the thermal performance during hypothetical transients is not degraded. The transient evaluation of Cycle 12 is considered to be bounded by previously accepted analyses. Section 7 of the Reload Report addresses "Accident and Transient Analysis" for this core reload. This analysis ensures that the proposed reload will not involve a significant increase in the probability or consequences of any accident previously evaluated.

Second Standard

(Amendment would not) create the possibility of a new or different kind of accident from any accident previously evaluated.

The analyses performed in support of this reload are in accordance with the US NRC document "Guidance for Proposed License Amendments Relating to Refueling," June 1975. The transient evaluation of Cycle 12 is considered to be bounded by previously accepted analyses (reference Reload Report Section 7), as such the proposed reload does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

(Amendment would not) involve a significant reduction in a margin of safety.

The issue of margin of safety for a reload modification involves the following areas:

1. Fuel System Design considerations,

2. Nuclear Design considerations, and

3. Thermal-Hydraulic Design considerations.

Sections 4, 5, and 6 respectively of the Oconee Unit 3, Cycle 12 Reload Report addresses the above areas. The value limits and margins discussed in these areas are well within the allowable limits and requirements, and reflect no significant reductions to any margins of safety. By examining these Sections of the Reload Report and the Cycle 12 core thermal and kinetic properties (with respect to previous cycle values), it can be concluded that this core reload will not reduce the ability of Oconee Unit 3 to operate safety during Cycle 12.

The above evaluation, with its accompanying references shows that the three Part 50.92(c) standards are satisfied. In summary, Duke has determined and submits that the proposed reload described herein does not represent any significant hazards.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

Attachment 3

Oconee Unit 3, Cycle 12 Reload Report