Duke Power Company
Oconee Nuclear Station

Attachment 1

| Remove Page | Insert Page |
|---|---|
| vi, vii | vi, vii |
| 2.1-1, -2, -3, -4, -5 | 2.1-1, -2 |
| 2.3-1, -2, -3, -4 | 2.3-1, -2, -3, -4 |
| 2.3-5, -6,-7 | 2.3-5 |
| 3.5-7 thru 3.5-14 | 3.5-7 thru 3.5-14 |
| 5.3-1 | 5.3-1 |

## LIST OF TABLES

## LIST OF FIGURES

# 2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

## 2.1 SAFETY LIMITS, REACTOR CORE

### Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

### Objective

To maintain the integrity of the fuel cladding.

### Specification

The maximum local fuel pin centerline temperature shall be less than 5080 - $(6.5 \times 10^{-3}) \times$( Burnup, MWD/MTU) $^{o}$F. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report and preserved by the RPS setpoints in Specification 2.3.

The DNBR shall be maintained greater than the correlation limits of 1.3 for BAW-2 and 1.18 for BWC. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits and variable low RCS pressure limits as specified in the Core Operating Limits Report and preserved by the RPS setpoints in Specification 2.3.

### Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions and anticipated transients. This is accomplished by operating within the nuclear boiling heat transfer regime where the heat transfer coefficient is large and the cladding temperature is only slightly greater than the coolant temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation, but neutron power and reactor coolant pressure and temperature can be related to DNB using a critical heat flux (CHF) correlation. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB.

The BAW-2 and BWC CHF correlations (1,2) have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The curve presented in Figure 1.1 of the COLR 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 provided in the Core Operating Limits Report.

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 1.2 of the Core Operating Limits Report, 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 provided in Figure 1.2 of the Core Operating Limits Report.

References

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

(2) Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143P, Part 2, August 1981.

## 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.

The pump monitors shall produce a reactor trip when a loss of two pumps occurs and the reactor is at power operation greater than 2.0% 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 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. For example, typical power level and flow rate combinations for different pump situations are as follows (actual values are given in the Core Operating Limits Report):

1. Assuming a flux/flow ratio of 1.07, a reactor 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 1.3 of the Core Operating Limits Report. The flux/flow ratio reduces the power level trip and associated power-imbalance boundaries to account for any reduction in RCS flow. The power-imbalance boundaries shown in Figure 1.3 of the COLR 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 trip setpoints shown in Figure 1.4 of the Core Operating Limits Report ensure that the minimum DNBR is greater than or equal to the minimum allowable DNBR for those accidents that result in a reduction in pressure. (3,4) The limits shown in Figure 1.4 of the Core Operating Limits Report bound the pressure-temperature curves calculated for 4 and 3 pump operation.

The safety analyses use a variable low RCS pressure trip setpoint which accounts for calibration and instrumentation uncertainties.

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618°F) shown in Figure 1.4 of the Core Operating Limits Report has been established to prevent excessive core coolant temperatures. Accounting for calibration and instrumentation errors, the safety analyses use 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 ≤ 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

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 | Figure 1.3 of the Core Operating Limits Report | Bypassed |
| 3. | Pump Monitors | At power operation >2.0% Rated Power and 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 | Figure 1.4 of the Core Operating Limits Report | 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.

c. If a control rod is declared inoperable by being immovable due to excessive friction or mechanical interference or known to be untrippable then:

1. Within 1 hour verify that the shutdown margin requirement of Specification 3.5.2.1 is satisfied and,

2. Within 12 hours place the reactor in the hot standby condition.

d. If a control rod is declared inoperable due to causes other than addressed in 3.5.2.2.c above then:

1. Within 1 hour either restore the rod to operable status or,

2. Continue power operation with the control rod declared inoperable and

a. Within 1 hour verify the shutdown margin requirement of Specification 3.5.2.1 with an additional allowance for the withdrawn worth of the inoperable rod and,

b. Either reactor thermal power shall be reduced to less than 60% of the allowable power for the reactor coolant pump combination within 1 hour and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow/imbalance, shall be reduced within the next 4 hours to 65.5% of thermal power value allowable for the reactor coolant pump combination or,

c. Position the remaining rods in the affected group such that the inoperable rod is maintained within allowable group average limits of Specification 3.5.2.2.a and within acceptable operating rod position withdrawal/insertion limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT.

e. If more than one control rod is inoperable or misaligned, the reactor shall be shut down to the hot standby condition within 12 hours.

3.5.2.3 The worths of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the control rod position limits provided in the CORE OPERATING LIMITS REPORT.

3.5.2.4 Quadrant Power Tilt

a. Except for physics tests, the maximum positive quadrant power tilt shall not exceed the Steady State Limit provided in the Core Operating Limits Report during power operation above 15% full power.

b. ◊ If the maximum positive quadrant power tilt exceeds the Steady State Limit but is less than or equal to the Transient Limit provided in the Core Operating Limits Report, then:

1. Either the quadrant power tilt shall be reduced within 2 hours to within its Steady State Limit or,

2. The reactor thermal power shall be reduced below 100% full power by 2% thermal power for each 1% of quadrant power tilt in excess of the Steady State Limit, and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within 4 hours by 2% thermal power for each 1% tilt in excess of the Steady State Limit. If less than four reactor coolant pumps are in operation, the allowable thermal power for the reactor coolant pump combination shall be reduced by 2% for each 1% excess tilt.

c. Quadrant power tilt shall be reduced within 24 hours to within its Steady State Limit or,

1. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

d. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit provided in the Core Operating Limits Report and if there is a simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 30 minutes at least 2% for each 1% of the quadrant power tilt in excess of the Steady State Limit.

2. Either quadrant power tilt shall be reduced within 2 hours to within its Transient Limit or,

3. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

e. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit provided in the Core Operating Limits Report, due to causes other than simultaneous indication of a misaligned control rod then:

1. Reactor thermal power shall be reduced within 2 hours to less than 60% of the allowable power for the reactor 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 provided in the Core Operating Limits Report, 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% $\pm$ 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.

If the imbalance is not within the acceptable envelope, 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.

3.5.2.7 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

Operation at power with an inoperable control rod is permitted within the limits provided. These limits assure that an acceptable power distribution is maintained and that the potential effects of rod misalignment on associated accident analyses are minimized. For a rod declared inoperable due to misalignment, the rod with the greatest misalignment shall be evaluated first. Additionally, the position of the rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments. When a control rod is declared inoperable, boration may be initiated to achieve the existence of 1% $\Delta k/k$ hot shutdown margin.

The power-imbalance envelope obtained in accordance with the approved methodology is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-16) 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. Hot rod manufacturing tolerance 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 rod) |

** 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.

The rod position limits obtained in accordance with the approved methodology 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) 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.

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 Group 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 the values assumed in the reload design analyses. The limits in Specification 3.5.2.4 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 3.5.2.6, 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 resulting from xenon transients and power maneuvers are inherently included in the limits determined in accordance with the approved methodology.

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 15.12

Page 3.5-14 Not Used

## 5.3 REACTOR

Specification

### 5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 93 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies, all of which are prepressurized with Helium. (1)

5.3.1.2 The fuel assemblies shall form an essentially cylindrical lattice with an active height range of 140.5 in. to 142 in. and an equivalent diameter of 128.9 in. (1)

5.3.1.3 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSR) distributed in the reactor core as shown in FSAR Figure 4.3-3. The full-length CRA and the APSR shall conform to the design described in the FSAR or reload report. (1)

5.3.1.4 Initial core and reload fuel assemblies and rods shall conform to design and evaluation described in the FSAR.

### 5.3.2 Reactor Coolant System

5.3.2.1 The design of the pressure components in the reactor coolant system shall be in accordance with the code requirements. (2)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670°F. (3)

5.3.2.3 The maximum reactor coolant system volume shall be 12,200 $ft^3$.

REFERENCES

(1) FSAR Section 4.2.2

(2) FSAR Section 5.2.3.1

(3) FSAR Section 5.2.1

Duke Power Company
Oconee Nuclear Station

Attachment 2

No Significant Hazards Consideration Evaluation

Duke Power Company
Oconee Nuclear Station

Attachment 2

No Significant Hazards Consideration Evaluation

Introduction

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

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

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

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

The proposed technical specifications concern the deletion of the existing cycle-dependent core operating limits from the Oconee Technical Specifications. Specifically, the present Oconee Technical Specifications are revised for cycle reloads that are affected by changes in cycle specific variables which are obtained from application of acceptable methodologies. This is an administrative burden to both the NRC staff and Duke Power Company (Duke). The deletion of cycle specific core operating limits from the Technical Specifications would facilitate 10 CFR 50, Part 50.59 reviews for future core reloads for all Oconee Units. The proposed revisions are required to support operation of Unit 1 at full power during Cycle 14.

Background

By letter dated November 14, 1986 the NRC staff provided guidance for removal of certain cycle-dependent core operating limits from the Oconee Technical Specifications. This would allow for changes to the values of core operating limits without prior approval (i.e., license amendment) by the NRC so long as an approved methodology for core reload design is followed. Subsequently, future changes in core operating limits at Oconee would only involve a safety review in accordance with the requirements of 10 CFR 50, Part 50.59.

Based on guidance provided by the NRC letter dated November 14, 1986, Duke Power Company submitted proposed technical specifications on September 3, 1987 which removed certain cycle dependent core operating limits from the technical specification. Subsequently, the NRC issued Amendment Nos. 172, 172, and 169 on January 26, 1989 for Oconee Units 1, 2 and 3, respectively, revising the ONS Technical Specifications by replacing the values of certain cycle specific parameter limits with a reference to the Core Operating Limits Report (COLR) which contains the values of these limits. However, other cycle specific core operating limits in the ONS Technical Specifications remained unchanged. Changes to these remaining limits will be necessary in support of Unit 1, Cycle 14 operation.

In addition, recognizing the burden on licensee and NRC resources associated with changes to Technical Specifications, on October 4, 1988 the NRC issued Generic Letter 88-16. This generic letter encourages licensees to propose changes to Technical Specifications that are consistent with the guidance provided in its enclosure. The enclosure provided guidance for the preparation of a license amendment request to modify Technical Specifications that have cycle specific parameter limits. With the implementation of this alternative, the NRC concluded that reload license amendments for the sole purpose of updating cycle-specific parameter limits would be unnecessary.

The proposed revisions provided in Attachment 1 would delete the remaining cycle specific core operating limits from the Oconee Technical Specifications in accordance with the guidance provided in the enclosure to Generic Letter 88-16. The cycle specific core operating limits calculated in accordance with the approved methodologies will be included in the COLR and provided to the NRC upon issuance as required by Technical Specification 6.9.3. A description of all changes is provided in the following paragraphs.

Discussion of Proposed Revisions

The following is a summary of the proposed Technical Specification revisions included in Attachment 1.

1. The current Technical Specification 2.1 establishes safety limits requirements for the combination of the reactor system pressure and temperature in Figure 2.1.1 and for the reactor power imbalance in Figure 2.1.2. These figures have been deleted and relocated to the COLR (Attachment 4) and replaced with the non-cycle specific safety limits of fuel centerline temperature and DNBR in the proposed Technical Specification 2.1. The flux/flow/imbalance and variable low pressure trip setpoints depend upon the design peaking factors that are assumed. Since Oconee is transitioning to longer cycle lengths, it is expected that peaking margin will decrease. In order to provide flexibility in performing this work, it is desirable to place cycle dependent Technical Specifications in the COLR. The two RPS trip functions which are dependent on peaking factors are replaced by the actual safety limits. For example, the flux/flow/imbalance trip defines the range of imbalance at which the reactor must be tripped to avoid exceeding the centerline fuel melt or DNB safety limits. Depending upon the peaks predicted for a fuel cycle, the imbalance breakpoints may change. However, the safety limit itself does not change unless a new fuel type is used. Any future changes to the safety limits would require NRC approval prior to implementation. Therefore, Technical Specification 2.1 on Page 2.1-1 is revised to replace the imbalance and pressure/temperature safety limits with the actual DNB and centerline temperature safety limits.

2. Other changes under Section 2 of the Technical Specifications include revisions to the "Bases" of Technical Specification 2.1 to incorporate references to the Core Operating Limits Reports (Attachment 4). In addition, the design radial and axial peaking factors are removed from the bases of the Technical Specification and placed in the COLR. This change is being made in anticipation

that the design peaking factors may change in future cycles. Also, the maximum power level for three pump operation as defined by the flux/flow/imbalance trip is being removed from the bases and placed in the COLR. This is because the maximum 3 pump power level safety limit is dependent on the flux/flow ratio, which may change as peaking factors change.

In addition, the list of references on page 2.1.3, is revised to delete references 3 and 5 from the bases since they are no longer applicable. References 4, 6 and 7 are also deleted since they are included in Technical Specification 6.9 for the COLR.

Since the "Bases" are not considered as part of the Technical Specifications, the revised "Bases" is not part of the amendment request. However, the revised "Bases" are provided in this package for completeness and information purposes.

The proposed new Technical Specification 2.1 requirements are consistent with the draft of standard technical specification 2.1 for B&W plants contained in NUREG-1430, Vol. 1 which has been issued by the NRC for comments.

3. Technical Specification 2.3 and its "Bases" are revised to delete Figures 2.3-1 and 2.3-2 which provide allowable protective setpoints. Instead, these figures are now located in the COLR (Attachment 4) which is referenced in the proposed Technical Specification 2.3, accordingly. In addition, Table 2.3-1 is revised to reference the COLR for the values of the Core Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure trip functions. The revised Table 2.3-1 provides reactor protective system (RPS) trip setting limits as part of the Technical Specifications. Therefore, Figures 2.3-1 and 2.3-2 which will likely be updated for future reload designs are relocated to the Core Operating Limits Report.

4. Table 3.5-1, which includes limits for cycle-specific quadrant power tilt for all 3 Oconee Units, is deleted from Technical Specifications and included in the COLR. The tilt setpoints also depend on the peaking factors. Historically, these numbers have not changed much at Oconee. However, since peaks may increase in the future, these setpoints may need to be revised. Thus, these limits are being removed from the Technical Specifications and placed in the COLR. These changes are very similar to the rod insertion and imbalance limits placed in the COLR as approved in Amendment Nos. 172, 172 and 169. Accordingly, Technical Specification 3.5.2.4 is revised in several places to reference the COLR. Also, the "Bases" of Technical Specification 3.5.2 on Page 3.5-12 concerning the linear heat rate peaking increase has been revised to reference the value assumed in the reload design analyses.

In addition, references to fuel densification power spike factors (Item c), and fuel bowing power spike factors (Item e) of the bases on page 3.5-11 have been removed. As explained in Section 6 of the attached reload report, these are no longer applicable. The list of references on page 3.5-13 is also revised to reference appropriate sections of the FSAR.

The change to the "Bases" is not considered part of the amendment request, however, it is provided for completeness and consistency.

5. Technical Specification 5.3.1.2 establishes the dimensional requirements for the fuel assemblies, including an active fuel height of 142 inches. The Oconee Unit 1, Cycle 14 reload design includes a new batch of fuel assemblies with an active height of 140.5 inches. The justification and safety analyses in support of this design change are provided in the Unit 1, Cycle 14 reload report (Attachment 3). Therefore, to support the Unit 1, Cycle 14 reload, a revision to Technical Specification 5.3.1.2 will be necessary. The proposed Technical Specification 5.3.1.2 provides a range of active fuel height to accommodate both the old fuel design and the new batch design. Duke intends to use the new fuel design in all future Oconee reloads. Related justifications and safety analyses for the future reloads will be included in the respective reload reports. The references for Section 5.3 on page 5.3-1 are also revised to reference appropriate sections of the FSAR.

6. The List of Tables and List of Figures on Pages vi and vii, respectively, are revised to reflect the deletion of Table 3.5-1 and Figures 2.1-1, 2.1-2, 2.3-1 and 2.3-2.

7. All actual changes in the proposed Technical Specifications are indicated by vertical lines on the right margin of the pages. However, some of the pages without indication of any change are submitted due to shifting of portions of the text between pages. Except for those changes specified, the information on these pages is unchanged.

## No Significant Hazards Consideration Evaluation

The proposed technical specification amendment in Attachment 1 contains two major changes in support of the Oconee Unit 1, Cycle 14 reload. These changes include the deletion of Cycle specific core operating limits from technical specifications and relocating them in the COLR in accordance with the guidance provided in Generic Letter 88-16, and a change to the fuel assembly active height in Technical Specification 5.3.1.2 in support of the new fuel batch design for the Unit 1, Cycle 14 reload.

The following provides a discussion of how the proposed amendment satisfies each of the three standards of 10 CFR 50.92 (c) for these changes.

### First Standard

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

This amendment request removes the cycle specific core operating limits from the Technical Specifications and is consistent with the guidance provided in Generic Letter 88-16 in that the proposed Technical Specifications reference a formal report, COLR, which contains the cycle specific core operating limits. In addition, the associated reporting requirements are already included in Administrative Controls section, Section 6.9, and finally the COLR is referenced in place of the limits removed from the Technical Specifications.

The removal of cycle dependent variables from the Technical Specifications has no impact upon plant operation or safety, or on the probability of a Design Basis Accident (DBA) occurrence. The Technical Specifications will continue to require operation within the core operational limits for each cycle reload calculated by the approved reload design methodologies. Appropriate actions to be taken if limits are violated will also remain in the Technical Specifications.

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in cycle specific parameters, and the new active fuel height in Technical Specification 5.3.2.1, in accordance with the approved reload methodologies to ensure that the transient evaluation for Unit 1, Cycle 14 reload with the new parameters is bounded by previously accepted analysis. The results of this examination are provided in the Unit 1, Cycle 14 reload report, Attachment 3. Future reloads will be evaluated per the requirements of 10 CFR 50.59 to assure that these reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated.

### Second Standard

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

As stated earlier, the removal of the cycle specific variables has no influence, impact nor does it contribute in any way to the probability or consequences of an accident. The cycle specific variables are calculated using the NRC approved methods. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken if

limits are exceeded. The revised Technical Specification 5.3.2.1 includes a new active fuel height which is in support of the new fuel assembly design implemented for the Unit 1, Cycle 14 reload. This new fuel design which will also be utilized in future reloads has been analyzed in accordance with the NRC approved reload methodologies. The analyses supporting this change are included in the attached reload report.

The proposed amendment 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 margin of safety is not affected by the removal of cycle specific core operating limits from the Technical Specifications nor by the change in the height of the active fuel in the proposed Technical Specification 5.3.2.1. The margin of safety presently provided by current Technical Specifications remains unchanged. The proposed amendment still requires operation within the core limits as obtained from the NRC approved reload design methodologies and appropriate actions to be taken when or if limits are violated remain unchanged.

The development of the limits for future reloads will continue to conform to those methods described in NRC approved documentation. In addition, each future reload will involve a Part 50.59 safety review to assure that operation of the unit within the cycle specific limits will not involve a significant reduction in a margin of safety.

In view of the preceding, Duke Power Company has determined that the proposed amendment does not involve significant hazards considerations.