



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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated May 7, 1991, as supplemented May 13, August 1, and August 15, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this license 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 hereby amended by page 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:

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Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 191, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 16, 1991



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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated May 7, 1991, as supplemented May 13, August 1, and August 15, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-47 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 191, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 16, 1991



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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated May 7, 1991, as supplemented May 13, August 1, and August 15, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this license 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 hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-55 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 188, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 16, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

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5.3-1
6.9-1

Insert Pages

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2.1-3
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5.3-1
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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 $5080 - (6.5 \times 10^{-3}) \times (\text{Burnup, MWD/MTU})$ °F. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits as specified in Figure 2.1-2.

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 Figures 2.1-2 and 2.1-1 respectively.

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 2.1-1 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 2.1-2, 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 2.1-2.

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.

Figure 2.1-1
 Variable Low Pressure
 Protective Limits

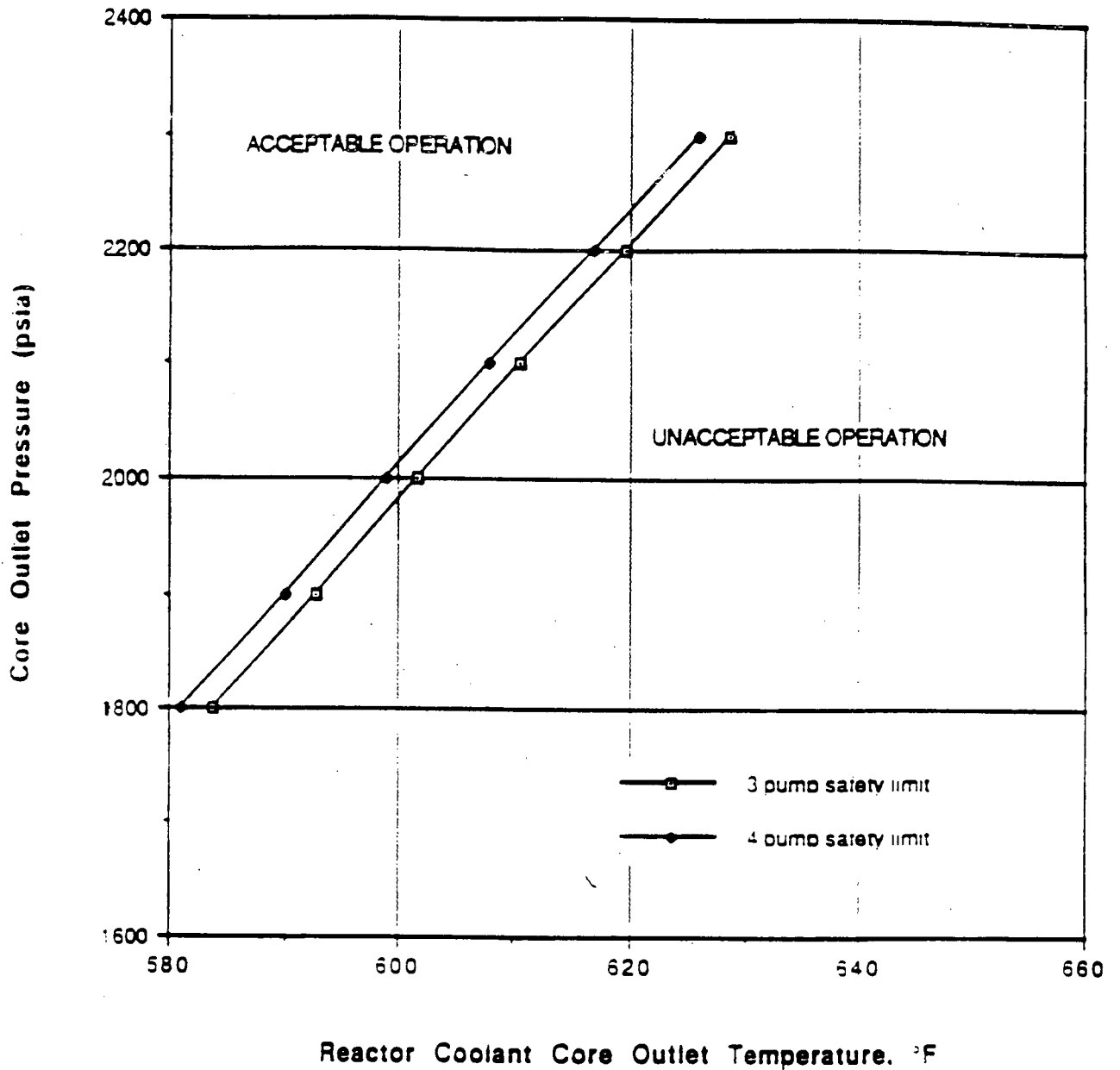
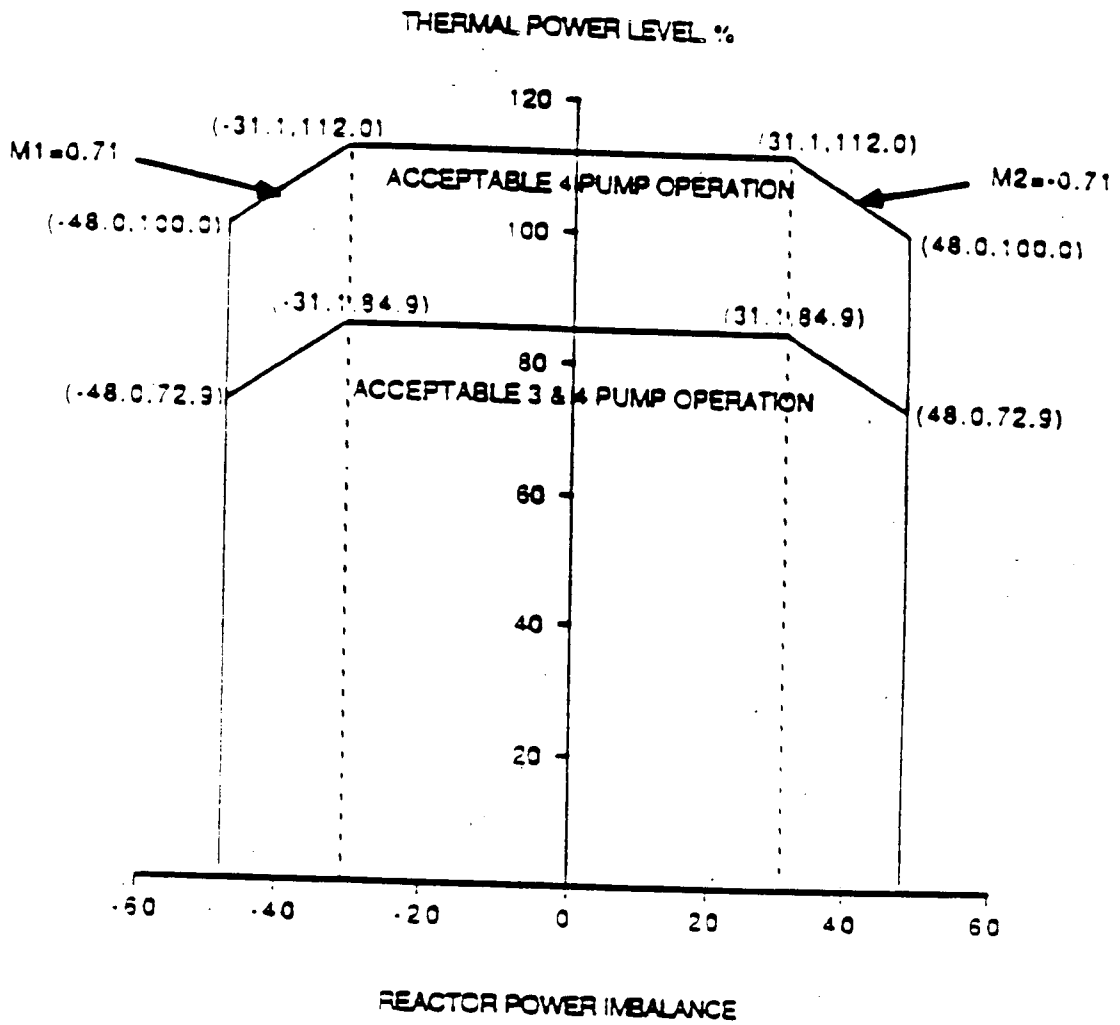


Figure 2.1-2
 Axial Power
 Imbalance Protective Limits



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 $\leq 5.0\%$ 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

TABLE 2.3-1

Reactor Protective System Trip Setting Limits

| | <u>RPS Trip</u> | <u>RPS Trip Setpoint</u> | <u>Shutdown Bypass</u> |
|----|----------------------------------------------------|------------------------------------------------------------------|------------------------------------|
| 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% \pm 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:

| <u>Group</u> | <u>Function</u> |
|--------------|--------------------------------|
| 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 given in Specification 6.9.2.

REFERENCES

- (1) FSAR, Section 3.2.2.1.2
- (2) FSAR, Section 15.12

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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 ft³.

REFERENCES

- (1) FSAR Section 4.2.2
- (2) FSAR Section 5.2.3.1
- (3) FSAR Section 5.2.1

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) Reactor Protective System Trip Setting Limits for the Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure trip function in Specification 2.3.
- (2) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.1.11, 3.5.2.1.b, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.
- (3) Quadrant Power Tilt Limits for Specification 3.5.2.4.a, 3.5.2.4.b, 3.5.2.4.d, 3.5.2.4.e, and 3.5.2.4.f.
- (4) 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.