October 31, 1994

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Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, South Carolina 29679

Dear Mr. Hampton:

SUBJECT: ISSUANCE OF AMENDMENTS - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M88406, M88407, AND M88409)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 207, 207, and 204 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated December 8, 1993, as supplemented April 20, September 8, and October 25, 1994.

The amendments revise TS 3.4 to address the need to bypass automatic initiation of the Emergency Feedwater system when the main feedwater pump discharge pressure is below actuation setpoint during startup and shutdown in order to prevent inadvertent actuation. The amendments would also delete operability requirements for the Emergency Condenser Cooling Water (ECCW) system.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Leonard A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270
and 50-287

Enclosures:
1. Amendment No. 207 to DPR-38
2. Amendment No. 207 to DPR-47
3. Amendment No. 204 to DPR-55
4. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 31, 1994

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, South Carolina 29679

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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270
and 50-287

Enclosures:
1. Amendment No. 207 to DPR-38
2. Amendment No. 207 to DPR-47
3. Amendment No. 204 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

Mr. J. W. Hampton
Duke Power Company

Oconee Nuclear Station

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
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 December 8, 1993, as supplemented April 20, 1994, September 8, 1994, and October 25, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 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. 207, 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 31, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
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 December 8, 1993, as supplemented April 20, 1994, September 8, 1994, and October 25, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 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. 207, 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 31, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 204
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 December 8, 1993, as supplemented April 20, 1994, September 8, 1994, and October 25, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 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. 204, 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 31, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 207

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 204

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.

Remove Pages	Insert Pages
3.4-1	3.4-1
3.4-2	3.4-2
3.4-3	3.4-3
3.4-4	3.4-4
4.1-9	4.1-9

3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

Applicability

Applies to the secondary system requirements for removal of reactor decay heat.

Objective

To specify minimum conditions necessary to assure the capability to remove decay heat from the reactor core.

Specification

3.4.1 The reactor shall not be heated above 250°F unless the following conditions are met:

a. Three emergency feedwater pumps (one steam driven pump capable of being driven from an operable steam supply system and two motor driven pumps) and associated manual initiation circuitry shall be operable.

b. Two 100% emergency feedwater flow paths shall be operable. Each flow path shall have at least one flow indicator operable.

3.4.2 In addition to the requirements of 3.4.1, prior to criticality, the automatic initiation circuitry associated with loss of main feedwater pumps as sensed by low hydraulic oil pressure or low discharge pressure shall be operable.

3.4.3 During operation greater than 250°F, the provisions of 3.4.1 and 3.4.2 may be modified to permit the following conditions:

a. One motor driven emergency feedwater pump may be inoperable for a period of up to seven days. If the inoperable pump is not restored to operable status within 7 days, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

b. One turbine driven emergency feedwater pump or one emergency feedwater flow path may be inoperable for a period of up to 72 hours. If the inoperable pump or flow path is not restored to operable status within 72 hours the unit will be at hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

c. Two motor driven emergency feedwater pumps may be inoperable for a period of up to 12 hours. If at least one pump is not restored to operable status within 12 hours, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

d. With three emergency feedwater pumps and/or both emergency feedwater flow paths inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump and associated emergency feedwater flowpath to operable status. The unit shall be at hot shutdown within

12 hours and below 250°F in another 12 hours if one emergency feedwater pump and associated flowpath are not restored to operable status.

e. If an emergency feedwater pump is inoperable due only to automatic initiation circuitry as specified by 3.4.2, the additional provisions of 3.4.3 a, b, c, and d which require cooldown of the RCS do not apply.

3.4.4 The 16 main steam safety relief valves shall be operable.

3.4.5 A minimum of 72,000 gallons of water per operating unit shall be available in the upper surge tank, condensate storage tank, and hot-well. A minimum of 6 ft. (=30,000 gal.) shall be available in the upper surge tank.

3.4.6 The controls of the Emergency Feedwater System shall be independent of the Integrated Control System.

Bases

The Main Feedwater System and the Turbine Bypass System are normally used for decay heat removal and cooldown above 250°F. Feedwater makeup is supplied by operation of a hotwell pump, condensate booster pump, and a main feedwater pump.

Operability of the Emergency Feedwater System (EFW) assures the capability to remove decay heat and cool down the Reactor Coolant System to the operating conditions for switch over to decay heat removal by the Decay Heat Removal System, in the event that the Main Feedwater System is inoperable. The EFW system consists of a turbine driven pump (880 gpm), two motor driven pumps (450 gpm each), and associated flow paths to the steam generators.

The limiting transient requiring maximum EFW flow is the loss of main feedwater with offsite power available. For this transient, a minimum EFW flow rate equivalent to 400 gpm at 1050 psia and no more than 130°F is adequate. Each of the three EFW pumps is capable of delivering this flow.

A 100% flowpath is defined as: The flowpath to either steam generator including associated valves and piping capable of being supplied by either the turbine driven pump or the associated motor driven pump.

One flow indicator or steam generator level indicator per steam generator is sufficient to provide indication of emergency feedwater flow to the steam generators and to confirm Emergency Feedwater System operation. In the event that at least one indicator per steam generator is not available, then the flowpath to this steam generator is considered to be inoperable.

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps as sensed by low hydraulic oil pressure or low feedwater pump discharge pressure. This specific automatic initiation logic is placed in service prior to criticality and may be bypassed when shutdown to prevent inadvertent actuation during startup and shutdown. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

Normally, decay heat is removed by steam relief through the Turbine Bypass System to the condenser. Decay heat can also be removed from the steam generators by steam relief through the main steam safety relief valves. The total relief capacity of the 16 main steam safety relief valves is 13,105,000 lbs./hr. In this case the minimum amount of water in the upper surge tank, condensate storage tank, and hotwell is sufficient to remove decay heat for at least 4 hours at hot shutdown conditions. This provides adequate time to establish normal flow through the condenser by restarting a Condenser Cooling Water (CCW) pump in a loss of station power events. The turbine bypass valves can then be utilized to relieve steam to the condenser and commence a cooldown of the RCS.

A 6 foot level in the upper surge tank will ensure that 30,000 gallons of water are available to the EFW pumps from that source. The 6 foot level setpoint includes an allowance for instrument error and for the depletion of inventory while switching to an alternate suction source.

REFERENCES

1. FSAR, Section 10.
2. Selected Licensee Commitments, Section 16.7

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rod Movement [1]	Movement of Each Rod	Monthly
2.	Pressurizer Safety Valves	Setpoint	Each Refueling[4]
3.	Main Steam Safety Valves	Setpoint	Each Refueling[4]
4.	Refueling System Interlocks[5]	Functional	Prior to Refueling
5.	Main Steam Stop Valves [1]	Movement of Each Stop Valve	Monthly
6.	Reactor Coolant System [2] Leakage	Evaluate	Daily
7.	Condenser Circulating Water [6] Flow Test	Functional	Each Refueling
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9.	Spent Fuel Cooling System	Functional	Prior to Refueling
10.	High Pressure and Low [3] Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	Each Refueling

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

(4) Number of safety valves to be tested each refueling shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

(5) Applicable only to the interlocks associated with the Reactor Building Purge System.

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 204 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By letter dated December 8, 1993, as supplemented April 20, September 8, and October 25, 1994, Duke Power Company, et al. (the licensee), submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3, Technical Specifications (TS). The requested changes would make two revisions to Technical Specification (TS) 3.4, "Secondary System Decay Heat Removal" of the Oconee Nuclear Station, Units 1, 2 and 3, plant technical specifications. The first revision proposes to revise TS 3.4.1 "Emergency Feedwater System" to address the need to bypass the loss of main feedwater (MFW) automatic initiation signal to the emergency feedwater (EFW) system when the (MFW) discharge pressure is below the EFW actuation setpoint. The basis for this revision is to prevent inadvertent actuation of the EFW system during startups and shutdowns. The present TS could also be interpreted to require the licensee to enter a limiting condition for operation (LCO) while following normal startup and shutdown procedures, therefore, clarification was needed. In the second revision, Duke proposed to remove the operability requirements and surveillance requirements (SR) for the emergency condenser circulating water (ECCW) system as currently specified in TS 3.4.5 and TS Table 4.1-2, respectively. The licensee's basis for removal of the ECCW system from the TSs is that it is not required for decay heat removal to mitigate any design basis events. In a telephone call on October 25, 1994, the licensee agreed to retain surveillance requirements for the portion of ECCW required to support operability of the low pressure service water (LPSW) system.

The EFW system consists of one turbine-driven and two motor-driven EFW pumps with associated flow paths to the steam generators. Each of the motor-driven pumps will automatically start on a low steam generator (SG) level in both SGs or a loss of both MFW pumps. The turbine driven-pump only receives an automatic start signal on a loss of both MFW pumps. The loss of both MFW pumps is sensed by either low MFW turbine control oil pressure or low MFW discharge header pressure. The proposed change would bypass only the automatic start on loss of both MWF pumps signal when the MFW pump discharge pressure is below the actuation trip setpoint. The automatic start of the motor-driven pumps on low level in both SGs would still be available.

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The ECCW system is actually the condenser circulating water (CCW) system relying on syphon flow in lieu of the CCW pumps and with discharge flow diverted via an emergency discharge line to the Keowee Hydro Plant tailrace. Normally, the CCW pumps take suction from Lake Keowee and the discharge from the condenser is returned to the Keowee River Arm of Lake Keowee. The CCW system is designed to take advantage of an inherent system syphon effect so that the CCW pumps are required only to overcome pipe and condenser friction losses. Under a loss of power situation, the emergency discharge line (to the tailrace) will automatically open and the CCW system will continue to operate as an unassisted siphon system supplying sufficient water to the condenser for decay heat removal. This "siphon system" is the ECCW system and is divided into two distinct parts. The "first siphon" supplies suction to the LPSW system and the "second siphon" provides flow through the main condenser to remove decay heat. The operation of the first siphon is unaffected by the operation of the second siphon. However, operation of the second siphon relies in part on the operation of the first siphon. The distinction between these two "siphons" is important because this distinction essentially forms the basis for the licensee's proposal to remove the ECCW system from the TSs. The ECCW system includes operation of both siphons with flow through the main condenser. To support removal of the ECCW TS, the licensee states that only the first siphon is required for safe plant shutdown (provides net positive suction head [NPSH] to the LPSW pumps) and is already controlled via FSAR Chapter 16 and the LPSW system TS.

The April 20, September 8, and October 25, 1994 supplements provided additional information that did not change the scope of the December 8, 1994, application and the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Emergency Feedwater System

During a normal plant startup, reactor coolant temperature could be above 250°F while actual MFW discharge pressure is still below the EFW initiation setpoint or reset point (about 100 pounds above setpoint) even though the MFW pumps are running and feeding the steam generators. Under these conditions, an unwanted EFW initiation signal would be generated and EFW would be delivered to the steam generators. The plant startup procedures presently bypass this initiation signal until the actual MFW discharge pressure is above the EFW actuation reset pressure to prevent inadvertent EFW initiation. TS 3.4.1a presently requires three EFW pumps and associated initiation circuitry to be operable before the reactor can be heated above 250 degrees fahrenheit (°F). The requirement does not define the initiation circuitry to be automatic or manual, however, the Bases for 3.4.1 do describe the automatic initiation logic. Therefore, the present TS could be interpreted to be violated whenever the actuation logic is bypassed when reactor coolant temperature is above 250°F.

To correct these conflicting requirements the proposed changes separately address the operability requirements for the manual and automatic functions and revised the TS as follows:

1 TS 3.4.1a will now require that the manual initiation circuitry be operable whenever the EFW pumps are required to be operable, i.e., whenever the reactor coolant system is above 250°F. This is basically the same as the existing TS.

2 A new TS 3.4.2 has been added to require the automatic initiation circuitry associated with the loss of the MFW pumps be operable prior to criticality. This eliminates the confusion as to when the automatic function must be operable.

3 Existing TS 3.4.2 has been renumbered to 3.4.3 and revised (TS 3.4.3e) to clarify the required action if the EFW automatic initiation function becomes inoperable. Specifically, the actions are identical to the actions for an inoperable EFW pump, except that cooldown to 250°F is not required since shutdown to hot shutdown will exit the applicability requirements for the automatic initiation circuitry.

Other minor changes and renumbering have also been made to the EFW system TS for clarification and consistency. The staff has reviewed these minor changes and found them acceptable as they do not have any impact on safety and more accurately reflect the actual EFW system requirements and design.

The change to TS 3.4.1a which addresses only the manual initiation circuitry is acceptable because it basically reflects the existing requirements for the manual initiation circuitry to be operable whenever the EFW pumps are required to be operable. Therefore, this is essentially no change.

The change to add TS 3.4.2 which addresses the automatic initiation function as sensed by the loss of MFW pumps logic is the main thrust of this TS revision. This new TS is also considered to be basically the same as the existing TS except that the operability requirements specify "prior to criticality" rather than "prior to the reactor being heated above 250°F." However, the existing TS do not adequately reflect actual plant operating conditions during startup and, therefore, had to be changed to prevent inadvertent EFW system initiation and to prevent violating TS requirements during normal plant startups or shutdowns. The proposed change would require the automatic initiation function as sensed by loss of both MFW pumps to be operable prior to criticality in lieu of prior to reaching 250°F. The bypassing of the automatic initiation function during startup and shutdown is in accordance with the plant design basis as reflected in FSAR Sections 10.4.7 and 16.7.3 which specify that this function will be bypassed whenever the operating MFW pump(s) discharge pressure approaches the actuation setpoint. FSAR Section 16.7.3 also requires the EFW automatic initiation low steam generator level function (motor-driven pumps only) to be operable whenever the reactor coolant system is greater than 250°F. FSAR Section 16.7.3 further states that adequate time is available for manual initiation of the EFW system whenever plant conditions are such that the automatic initiation function on loss of MFW pumps must be bypassed. The staff agrees that the proposed change, as it reflects the actual plant design basis, is consistent with the FSAR accident analyses which assume power operation as an initial plant condition, and is necessary to prevent inadvertent initiation of the EFW system.

The proposed addition of TS 3.4.3e is the action statement counterpart to the new TS 3.4.2. In the event the automatic actuation function becomes inoperable, the plant must be brought to hot shutdown in lieu of being cooled to below 250°F as presently required by the existing specifications. The staff concurs with this proposed change because it makes the TS action statement consistent with the new operability requirements of TS 3.4.2 as justified above. By bringing the plant to hot shutdown conditions, the operability requirements are no longer applicable because the reactor is no longer critical and, therefore, no further action is required.

The staff has also reviewed the proposed revised Bases for the above TS changes and concludes that they adequately reflect the design basis and operation of the EFW system and are also consistent with the proposed corresponding changes to the existing technical specifications. The proposed new bases are, therefore, acceptable to the staff.

2.2 Emergency Condenser Cooling Water System

The licensee has proposed to eliminate TS 3.4.5 (and its Bases) which presently requires the ECCW system to be operable whenever the reactor coolant system temperature is greater than 250°F. An allowed outage time of seven days is specified for loss of system function. The licensee contends that the original design basis for the ECCW system was to provide decay heat removal through the condenser in the event of a complete loss of offsite and onsite power. Section 10.4 of the staff's original Safety Evaluation Report (SER), dated December 29, 1970, supports the licensee's contention that the design basis for the ECCW system was to accommodate a complete loss of all ac power. This was a design basis that was unique to Oconee at the time of licensing. The staff confirmed this licensing basis in an April 30, 1984 Safety Evaluation related to a TS change for the ECCW system. Subsequently, the licensee provided other means to cope with a complete loss of ac power in order to meet the Station Blackout (SBO) Rule, 10 CFR 50.63.

Following an SBO, the preferred method of coping would rely on the ECCW system and the turbine-driven emergency feedwater (EFW) pump to conserve makeup inventory and to maintain chemistry in the secondary system. However, because of certain limitations, this method is not credited to meet the SBO rule. The credited method, and therefore the method required to be available for coping with SBO, is the standby shutdown facility (SSF). The SSF auxiliary service water (ASW) system is capable of providing lake water to the steam generators under SBO conditions and meets all the requirements of the SBO rule. In the December 3, 1992 Supplemental Safety Evaluation for SBO at Oconee, the staff concluded that the SSF was acceptable with respect to coping with an SBO. The staff, therefore, concludes that the ECCW system is no longer required to be in the TS because it is not required to cope with a complete loss of ac power.

During its review, the staff questioned how the operability of the LPSW pumps would be assured without reliance on the ECCW system to provide a NPSH following design basis events involving the loss of the CCW pumps. In its April 20, 1994 response, the licensee clarified the operation of the "first" siphon and demonstrated that it would not be affected by removal of the ECCW system from the TS. Administrative controls are provided in FSAR Chapter 16

system from the TS. Administrative controls are provided in FSAR Chapter 16 where selected licensee commitments are identified. Section 16.9.7, Keowee Lake Level, specifies a number of required actions in the event the lake level falls below certain levels. The actions associated with the first siphon flow puts the plant in a 24-hour LCO via the LPSW TS 3.3.7 if only one siphon source is available (LPSW does not meet the single failure criterion) and requires a plant shutdown (via TS 3.0) if no siphon source is available (LPSW system is declared inoperable). In addition, the licensee stated in the September 8, 1994 supplement that the "first" siphon mode of operation is tested during the LPSW System Flow Test. In the October 25, 1994 supplement, the licensee provided a clarification of the testing performed and stated that LPSW pump suction would not be aligned to this siphon during the flow test, but adequate suction flow for the LPSW system would be verified. In a telephone call on, October 25, 1994, the licensee agreed to include surveillance test requirements in TS Table 4-1-2 for the "first" siphon. Based on the inclusion of the requirements in FSAR Chapter 16, and the surveillance requirements for the "first" siphon, the staff concludes that adequate controls are in place to verify LPSW system operability and the ECCW system TS is, therefore, not required for LPSW operability.

Based on its review the staff concludes that the proposed changes to the EFW system TS are consistent with the system design basis, plant operating procedures, and the FSAR accident analysis. The proposed changes are also necessary to avoid confusion in interpreting the EFW TS and prevent inadvertent initiation of the EFW system during startups and shutdowns. The staff also concludes that the proposed change to eliminate the ECCW system from the TS is consistent with the current licensing basis which does not rely on operation of the ECCW system for any design basis events, including a complete loss of ac power, i.e., SBO. The staff, therefore, concludes that the proposed changes to TS 3.4 related to the EFW and ECCW systems are acceptable and should be approved.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (59 FR 39584 dated August 3, 1994). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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