January 13, 1994

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

DISTRIBUTION	JNorberg, 7E23
Docket File	D.Hagan MNBB4702
NRC/Local PDRs	G.Hill(4) P1-37
PDII-3 RF	C.Grimes 11F23
S.Varga	ACRS(10) P-135
D.Matthews	PA 17F2
L.Wiens	OC/LFMB MNBB4702
L.Berry	E.Merschoff, RII
OGC 15B18	R.Jones, 8E23
RBarrett, 8H7	CMcCracken, 8D1

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. M86729, M86730, AND M86731)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 203 , 203 , and 200 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 May 3, 1993, as supplemented August 11, 1993.

The amendments revise the limiting conditions for operation and surveillance requirements related to the Low Pressure Service Water 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,

ORIGINAL SIGNED BY:

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

Enclosures:
1. Amendment No.203 to DPR-38
2. Amendment No.203 to DPR-47
3. Amendment No.200 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

BC:EMEB
J.NORBERG
12/28/93

OFFICE	PDII-3/LA	PDII-3/PM	BC:SRXB	SCSB	SPLB	OGC	PDII-3/D
NAME	L.BERRY	L.WIENS	R.JONES	R.BARRETT	C.McCRACKEN	S. Hom	L. Plisco ~~R.HERMANN~~
DATE	11/1/93	11/10/93	11/15/93	11/22/93	12/29/93	1/07/94	1/13/94

Document Name: OC086729.AMD

140019

9401180064 940113
PDR ADOCK 05000269
P PDR

NRC FILE CENTER COPY

Mr. J. W. Hampton
Duke Power Company

Oconee Nuclear Station

cc:
Mr. A. V. Carr, Esquire
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242-0001

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Division
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

Max Batavia, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Office of Intergovernmental
Relations
116 West Jones Street
Raleigh, North Carolina 27603

County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. M. E. Patrick
Compliance
Duke Power Company
Oconee Nuclear Site
P. O. Box 1439
Seneca, South Carolina 29679

Mr. Alan R. Herdt, Chief
Project Branch #3
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. G. A. Copp
Licensing - EC050
Duke Power Company
P. O. Box 1006
Charlotte, North Carolina 28201-1006

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. 203
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 3, 1993, as supplemented August 11, 1993, 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 203, 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

Loren R. Plisco

Loren R. Plisco, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: **January 13, 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. 203
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 3, 1993, as supplemented August 11, 1993, 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. 203 , 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

Loren R Plisco

Loren R. Plisco, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: **January 13, 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. 200
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 3, 1993, as supplemented August 11, 1993, 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. 200, 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

Loren R. Plisco

Loren R. Plisco, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification Changes

Date of Issuance: **January 13, 1994**

ATTACHMENT TO LICENSE AMENDMENT NO. 203

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 203

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 200

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
iv	iv
viii	viii
ix	-
x	-
xi	-
3.3-2	3.3-2
3.3-3	3.3-3
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	3.3-7
4.5-2 - 4.5-12	4.5-2 - 4.5-9

Section		Page

INTRODUCTION

These Technical Specifications apply to the Oconee Nuclear Station, Units 1, 2, and 3 and are in accordance with the requirements of 10CFR50, Section 50.36. The bases, which provide technical support or reference the pertinent FSAR section for technical support of the individual specifications, are included for informational purposes and to clarify the intent of the specification. These bases are not part of the Technical Specifications, and they do not constitute limitations or requirements for the licensee. The Technical Specifications while applying to Units 1, 2, and 3 are written on a single unit basis; exceptions to this are identified.

within 72 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.1.b(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS temperature below 350° F within an additional 24 hours.

c. For all Units, when reactor power is greater than 60% FP:

(1) In addition to the requirements of Specification 3.3.1.a(1) and 3.3.1.b(1) above, the remaining HPI pump and valves HP-409 and HP-410 shall be operable and valves HP-99 and HP-100 shall be open.

(2) Tests or maintenance shall be allowed on any component of the HPI system, provided two trains of HPI system are operable. If the inoperable component is not restored to operable status within 72 hours, reactor power shall be reduced below 60% FP within an additional 12 hours.

3.3.2 Low Pressure Injection (LPI) System

a. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

(1) Two independent LPI trains, each comprised of an LPI pump and a flowpath capable of taking suction from the borated water storage tank and discharging into the RCS automatically upon ESPS actuation (LPI segment), together with two LPI coolers and two reactor building emergency sump isolation valves (manual or remote-manual) shall be operable.

(2) Tests or maintenance shall be allowed on any component of the LPI system provided the redundant train of the LPI system is operable. If the LPI system is not restored to meet the requirements of Specification 3.3.2.a(1) above within 72 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.2.a(1) are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

3.3.3 Core Flood Tank (CFT) System

When the RCS is in a condition with pressure above 800 psig both CFT's shall be operable with the electrically operated discharge valves open and breakers locked open and tagged; a minimum level of 13 ± .44 feet (1040 ± 30 $ft.^3$) and one level instrument channel per CFT; a minimum boron concentration within the limit specified in the Core Operating Limits Report in each CFT; and pressure at 600 ± 25 psig with one pressure instrument channel per CFT.

3.3.4 Borated Water Storage Tank (BWST)

When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

a. The BWST shall have operable two level instrument channels.

(1) Tests or maintenance shall be allowed on one channel of BWST level instrumentation provided the other channel is operable.

(2) If the BWST level instrumentation is not restored to meet the requirements of Specification 3.3.4.a above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.4.a are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

b. The BWST shall contain a minimum level of 46 feet of water having a minimum concentration of boron within the limit specified in the Core Operating Limits Report at a minimum temperature of 50°F. The manual valve, LP-28, on the discharge line shall be locked open. If these requirements are not met, the BWST shall be considered unavailable and action initiated in accordance with Specification 3.2.

3.3.5 Reactor Building Cooling (RBC) System

a. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical:

(1) Two independent RBC trains, each comprised of an RBC fan, associated cooling unit, and associated ESF valves shall be operable. Valve LPSW-108 shall be locked open.

(2) Tests or maintenance shall be allowed on any component of the RBC system provided one train of the RBC and one train of the RBS are operable. If the RBC system is not restored to meet the requirements of Specification 3.3.5.a(1) above within 24 hours, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

b. When the reactor is critical:

(1) In addition to the requirements of Specifications 3.3.6.a(1) above, the other RBS train comprised of an RBS pump and a flowpath capable of taking suction of the LPI system and discharging through the spray nozzle header automatically upon ESPS actuation (RBS segment) shall be operable.

(2) Tests or maintenance shall be allowed on one RBS train under either of the following conditions:

(a) One RBS train may be out of service for 24 hours.

(b) One RBS train may be out of service for 7 days provided all three RBC trains are operable.

(c) If the inoperable RBS train is not restored to meet the requirements of Specification 3.3.6.b(1) above within the time permitted by Specification 3.3.6.b(2) (a) or (b), the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.6.b(1) are not met within an additional 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

3.3.7 Low Pressure Service Water (LPSW)

a. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

(1) Three LPSW pumps for the shared Unit 1, 2 LPSW system shall be operable, except as provided in (2) below.

(2) Two LPSW pumps for the shared Unit 1, 2 LPSW system shall be operable if Unit 1 or Unit 2 has been defueled and one LPSW pump is capable of mitigating the consequences of a design basis accident in the remaining Unit.

(3) Two pumps for the Unit 3 LPSW system shall be operable.

b. Tests or maintenance shall be allowed on any component of the LPSW system provided the redundant train of the LPSW system is operable. If the LPSW system is not restored to meet the requirements of Specification 3.3.7.a above within 72 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.7.a are not met within 24 hours following hot shutdown, the reactor shall be placed in condition with RCS pressure below 350 psig and RCS temperature below 250° within an additional 24 hours.

Bases

Specification 3.3 assures that, for whatever condition the reactor coolant system is in, adequate engineered safety feature equipment is operable.

For operation up to 60% FP, two high pressure injection pumps are specified. Also, two low pressure injection pumps and both core flood tanks are required. In the event that the need for emergency core cooling should occur, functioning of one high pressure injection pump, one low pressure injection pump, and both core flood tanks will protect the core, and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad. (1) Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.

The requirement to have three HPI pumps and two HPI flowpaths operable during power operation above 60% FP is based on considerations of potential small breaks at the reactor coolant pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling.(2) The analysis of these breaks indicates that for operation at or below 60% FP only a single train of the HPI system is needed to provide the necessary core cooling.

The requirement for a flowpath from LPI discharge to HPI pump suction is provided to assure availability of long term core cooling following a small break LOCA in which the BWST is depleted and RCS pressure remains above the shutoff head of the LPI pumps.

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core. The minimum boron concentration is specified in the Core Operating Limits Report.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the Reactor Building design pressure will not be exceeded with one spray and two coolers operable. (4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor Building spray systems are operable for seven days or one Reactor Building spray system provided all three Reactor Building cooling units are operable. Valve LPSW-108 is the LPSW isolation valve on the discharge side of each Unit's RBCUs. This valve is required to be locked open in order to assure the LPSW flowpath for the RBCUs is available.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Unit 1, 2, and 3. One low pressure service water pump per unit is required for normal operation.

The Unit 1 and 2 LPSW system requires two pumps to meet the single failure criterion provided that one of the Units has been defueled and the following LPSW system loads on the defueled Unit are isolated: RBCUs, Component Cooling, main turbine oil tank, RC pumps, and LPI coolers. In this configuration, if two of the three LPSW pumps are inoperable, 72 hours are permitted by TS 3.3.7.b to restore two of the three LPSW pumps to operable status. At all other times when the RCS of Unit 1 or 2 is ≥ 350 psig or ≥ 250°F, all three LPSW pumps are required to meet the single failure criterion. When all three LPSW pumps are required to be operable and one of the three pumps is inoperable, 72 hours are permitted by TS 3.3.7.b to restore the pump to operable status.

The operability of redundant equipment(s) is determined based on the results of inservice inspection and testing as required by Technical Specification 4.5 and ASME Section XI.

REFERENCES

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) FSAR, Section 9.3.3.2

(4) FSAR, Section 15.14.5

System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests

4.5.1.2.1 Valves - Power Operated

a. Valves LP-17, -18, shall only be tested every cold shutdown unless previously tested during the current quarter.

b. During each refueling outage the following LPI system valves shall be cycled manually to verify the manual operability of these power operated valves:

(1) LPI pump discharge (ES) LP-17,-18
(2) LPI discharge throttling LP-12,-14
(3) LPI discharge header crossover LP-9,-10
(4) LPI discharge to HPI/RBS LP-15,-16

4.5.1.2.2 Check Valves

Periodic individual leakage testing[a] of valves CF-12, CF-14, LP-47 and LP-48 shall be accomplished prior to power operation after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed. Whenever integrity of these valves cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily. For the allowable leakage rates and limiting conditions for operation, see Technical Specification 3.1.6.10.

Bases

The Emergency Core Cooling Systems are the principle reactor safety features in the event of loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. The HPI system test required by Specification 4.5.1.1.1 verifies that the HPI system responds as required to actuation of ES channels 1 and 2.

The LPI system test required by Specification 4.5.1.1.2 verifies that the LPI system responds as required to actuation of ES channels 3 and 4. In addition, this test verifies that the LPSW pumps and LPSW-4 and -5 (LPSW supply to LPI coolers) respond as required to actuation of ES channels 3 and 4. The test required by Specification 4.5.3 verifies the containment heat removal capability of the LPI coolers (in conjunction with the RBCUs and RB Spray system).

---

[a] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

Testing the manual operability of power-operated valves in the Low Pressure Injection System gives assurance that flow can be established in a timely manner even if the capability to operate a valve from the control room is lost.

With the reactor shut down, the valves in each core flooding line are checked for operability by reducing the Reactor Coolant System Pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

Power Operated Valves LP-17 and LP-18, are boundary valves between high pressure and low pressure design piping. As such, functional testing of these valves is performed during cold shutdown conditions when the Reactor Coolant System pressure is below the design pressure of the Low Pressure Injection System piping and the potential for over-pressurization of the low pressure system is eliminated. Check Valves CF-12, CF-14, LP-47, and LP-48 are located on the high pressure piping and therefore can be leak tested with the Reactor Coolant System at hot shutdown conditions.

REFERENCE

(1) FSAR, Section 6

4.5.2 Reactor Building Cooling Systems

Applicability

Applies to testing of the Reactor Building Cooling Systems.

Objective

To verify that the Reactor Building Cooling Systems are operable.

Specification

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

a. (1) During each refueling outage, a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the Reactor Building Spray System.

(2) The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly; the appropriate pump breakers shall have closed, and all valves shall have completed their travel.

b. Station compressed air will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every ten years.

4.5.2.1.2 Reactor Building Cooling System

a. During each refueling outage, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the Reactor Building Cooling System for reactor building cooling operation.

(2) Verification of the engineered safety features function of the Low Pressure Service Water System which supplies coolant to the reactor building coolers shall be made to demonstrate operability of the coolers.

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate valves have completed their travel, and fans are running at half speed.

Bases

The Reactor Building Cooling System and Reactor Building Spray System are designed to remove heat in the containment atmosphere to control the rate of depressurization in the containment. The peak transient pressure in the containment is not affected by the two heat removal systems.

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or fog can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The RB Spray system test required by Specification 4.5.2.1.1 verifies that the RB Spray pumps and valves respond as required to actuation of ES channels 7 and 8. In addition, this test verifies that LP-21, and LP-22 (BWST supply to the RB Spray pumps) respond as required to actuation of ES channels 7 and 8. The test required by Specification 4.5.3 verifies the containment heat removal capability of the RB Spray system (in conjunction with the LPI coolers and RBCUs).

The equipment, piping, valves, and instrumentation of the Reactor Building Cooling System are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment. The service water piping and valves out-side the Reactor Building are inspectable at all times. The reactor building fans are normally operated periodically, constituting the test that these fans are operable.

The RBCU system test required by Specification 4.5.2.1.2 verifies that the RBCU fans respond as required to actuation of ES channels 5 and 6. In addition, this test verifies that LPSW-18 (LPSW for "A" RBCU), LPSW-21, LPSW-565, and LPSW-566 (LPSW for "B" RBCU), and LPSW-24 (LPSW for "C" RBCU) respond as required to actuation of ES channels 5 and 6. The LPI system test required by Specification 4.5.1.1.2 verifies that the LPSW pumps respond as required to actuation of ES channels 3 and 4. The test required by Specification 4.5.3 verifies the containment heat removal capability of the RBCUs (in conjunction with the LPI coolers and RB Spray system).

REFERENCE

(1) FSAR, Section 6

4.5.3 Containment Heat Removal Capability

Applicability

Applies to verification of adequate containment heat removal capability.

Objective

To verify that containment heat removal capability is sufficient to maintain post accident conditions within design limits.

Specification

4.5.3.1 Containment Heat Removal Capability

a. On a refueling frequency, containment heat removal capability shall be verified to be sufficient to maintain post accident conditions within design limits.

b. In addition to the requirements of 4.5.3.1.a, on a frequency consistent with the LPI cooler and RBCU fouling rate, containment heat removal capability shall be verified to be sufficient to maintain post accident conditions within design limits.

Bases

The safety functions of the LPI system, RB Spray system, and RBCUs include maintaining containment pressure and temperature below design limits following an accident. This surveillance assures that containment heat removal capability is adequate assuming a worst case single failure. Specification 4.5.3.1.a requires that at a minimum the surveillance be performed on a refueling frequency. In addition, since service induced fouling can reduce containment heat removal capability, Specification 4.5.3.1.b requires that a fouling rate be determined in order to establish a more frequent test interval if required.

REFERENCES:

FSAR Section 6.2
FSAR Section 15.14

4.5.4 Penetration Room Ventilation System

Applicability

Applies to testing of the Penetration Room Ventilation System

Objective

To verify that the Penetration Room Ventilation System is operable.

Specification

4.5.4.1 Operational and Performance Testing

a. Monthly, each train of the Penetration Room Ventilation System shall be operated for at least 15 minutes at design flow ±10%.

b. During each refueling outage, it shall be demonstrated that:

1. The Penetration Room Ventilation System fans operate at design flow (± 10%) when tested in accordance with ANSI N510-1975.

2. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than six inches of water at the system design flow rate (± 10%).

3. Each branch of the Penetration Room Ventilation System is capable of automatic initiation.

4. The bypass valve for filter cooling is manually operable.

c. Leak tests using DOP or halogenated hydrocarbon, as appropriate shall be performed on the Penetration Room purge filters:

1. During each refueling outage;

2. After each complete or partial replacement of a HEPA filter bank or charcoal adsorber bank;

3. After any structural maintenance on the system housing;

4. After painting, fire, or chemical release in any ventilation zone communicating with the system.

d. The results of the DOP and halogenated hydrocarbon tests on HEPA filters and charcoal adsorber banks shall show ≥99% DOP removal and ≥99% halogenated hydrocarbon removal, respectively, when tested in accordance with ANSI N510-1975.

e. During each refueling outage, following 720 hours of system operation, or after painting, fire, or chemical release in any ventilation zone communicating with the system, a carbon sample shall be removed from the Reactor Building purge filters for laboratory analysis. Within 31 days of removal, this sample shall be verified to show ≥90% radioactive methyl iodide removal when tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). Otherwise, the filter system shall be declared inoperable.

Bases

Pressure drop across the combined high efficiency particulate air (HEPA) filters and charcoal adsorbers of less than six inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per year operating cycle establishes performance capability.

(HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system every month will demonstrate operability of the filters and adsorber system. Operation for 15 minutes demonstrates operability and minimizes the moisture build up during testing.

If painting, fire or chemical release occurs during system operation such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

4.5.5 Low Pressure Injection System Leakage

Applicability

Applies to Low Pressure Injection System leakage.

Objective

To maintain a preventive leakage rate for the Low Pressure Injection System which will prevent significant off-site exposures.

Specification

4.5.5.1 Acceptance Limit

The maximum allowable leakage from the Low Pressure Injection System components (which includes valve stems, flanges and pump seals) shall not exceed two gallons per hour.

4.5.5.2 Test

During each refueling outage, the following tests of the Low Pressure Injection System shall be conducted to determine leakage:

a. The portion of the Low Pressure Injection System, except as specified in (b), that is outside the containment shall be tested either by use in normal operation or by hydrostatically testing at 350 psig.

b. Piping from the containment emergency sump to the low pressure injection pump suction isolation valve shall be pressure tested at no less than 59 psig.

c. Visual inspection shall be made for excessive leakage from components of the system. Any excessive leakage shall be measured by collection and weighing or by another equivalent method.

Bases

The leakage rate limit for the Low Pressure Injection System is a judgement value based on assuring that the components can be expected to operate with-out mechanical failure for a period on the order of 200 days after a loss of coolant accident. The test pressure (350 psig) achieved either by normal system operation or by hydrostatically testing, gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the pressure test for the return lines from the containment to the Low Pressure Injection System (59 psig) is equivalent to the design pressure of the containment. The dose to the thyroid calculated as a result of this leakage is 0.76 rem for a two-hour exposure at the site boundary.

REFERENCE

FSAR, Section 15.15.4, and 6.3.3.2.2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.203 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO.200 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 May 3, 1993, as supplemented August 11, 1993, Duke Power Company (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3 Technical Specifications (TS). The requested changes would revise the limiting conditions of operation and surveillance requirements related to the Low Pressure Service Water (LPSW), the Low Pressure Injection (LPI), the High Pressure Injection (HPI), the Reactor Building Cooling Unit (RBCU), and the Reactor Building Spray (RBS) systems. Administrative changes are included to delete redundant requirements, correct a misspelling and update the table of contents.

2.0 EVALUATION

The staff's review and evaluation of each proposed change is given below.

Administrative changes to the table of contents and introduction (pages iv, viii, ix, x, and xi):

The licensee proposes to revise the table of contents to add new surveillance requirement TS 4.5.3, "Containment Heat Removal Capability," delete references to Figures 4.5.1-1, 4.5.1-2, and 4.5.2-1, delete blank pages and revise associated page and section numbers. These changes to the table of contents are purely administrative in nature, do not affect the substance of the TS changes themselves, and are acceptable.

TS 3.3.2 - Extension of the allowable outage time for one Low Pressure Injection (LPI) train inoperable from 24 hours to 72 hours:

The licensee proposes to revise TS 3.3.2.a(2) to extend the allowable outage time for one LPI train inoperable from 24 hours to 72 hours. During this time period, the remaining operable LPI train would be capable of mitigating the consequences of a design basis accident. This change is consistent with the requirements of NUREG-1430, "Standard Technical Specifications for B&W Plants," LCO 3.5.2 (ECCS - Operating), Required Action A.1. The 72-hour completion time is reasonable, based on the redundant capabilities afforded by the operable LPI train, and the low probability of a Design Basis Accident (DBA) occurring during this period.

TS 3.3.5 - Relocation of TS requirement to lock open valve LPSW-108:

Valve LPSW-108 is the LPSW isolation valve on the discharge side of the cooler in each Oconee unit's Reactor Building Cooling Unit (RBCU). Currently, TS 3.3.7 "Low Pressure Service Water System" requires that valve LPSW-108 be locked open. In the event this valve were to close, the associated RBCU would be inoperable. However, the operability of the entire LPSW system would not be affected. Therefore, since the requirement to lock open valve LPSW-108 pertains more directly to the RBCU system than to the LPSW system, the licensee proposes to relocate the requirement to TS 3.3.5 "Reactor Building Cooling (RBC) System." The Bases on page 3.3-6 have been revised to describe this change.

We find that the proposed relocation of the requirement to lock open valve LPSW-108 from TS 3.3.7 to TS 3.3.5 is administrative in nature and does not change the substance or effect of the requirement. Therefore, the change is acceptable.

TS 3.3.7.a - Requirement for the third LPSW pump in the shared Unit 1 and Unit 2 LPSW system to be operable:

For the shared Unit 1 and Unit 2 LPSW system, TS 3.3.7a currently requires only two of the three LPSW pumps to be operable at all times. However, calculations of the consequences of severe Loss-of-Coolant Accidents (LOCAs) are based on the assumption that two LPSW pumps are operating. Two operating LPSW pumps are required to provide adequate post-LOCA cooling of the reactor building. If only two LPSW pumps are operable and one of them fails, then the requirement for two operating pumps could not be met. To provide for the possible single failure of one LPSW pump, all three LPSW pumps must be operable. Therefore, the proposed revision of TS 3.3.7.a to require all three LPSW pumps to be operable in the shared Unit 1 and Unit 2 LPSW system is acceptable.

TS 3.3.7.b - Extension of the allowable outage time for one LPSW train inoperable from 24 hours to 72 hours:

For Unit 3, the LPSW is supplied by either of the two LPSW trains, each containing a LPSW pump required to be operable by TS 3.3.7.a. In normal or post-accident operation, one operating pump supplies the service water needs of the unit. In the event of a failure of the operating pump or its associated train, the redundant operable LPSW pump and train would be capable of providing the required service water.

For Units 1 and 2, the LPSW is supplied to each unit by either of the two LPSW trains from a shared LPSW system containing three LPSW pumps required to be operable by TS 3.3.7.a. In normal or accident operation, two of the three pumps are operating, so that the flow of one pump is available to supply the service water needs of each unit. The third operable LPSW pump could supply these needs in the event of a failure of one of the operating pumps or a component of its train.

The licensee proposes to extend the allowable outage time for one LPSW train inoperable from 24 hours to 72 hours. During this time period, the remaining operable LPSW train would be capable of mitigating the consequences of a design basis accident. This change is consistent with the requirements of NUREG-1430, "Standard Technical Specifications for B&W Plants," LCO 3.7.7 (Component Cooling Water System), Required Action A.1. The 72-hour completion time is reasonable, based on the redundant capabilities afforded by the operable LPSW train, and the low probability of a DBA occurring during this period. The extended 72-hour outage also provides a more adequate time period for the repair or replacement of an inoperable LPSW system component. Therefore, extending the allowable outage time for one LPSW train inoperable to 72 hours is acceptable. Therefore, this TS change is acceptable.

TS 4.5.1.1.1 Bases - Revision to High Pressure Injection (HPI) testing Bases:

The licensee proposes to revise the Bases associated with TS 4.5.1.1.1 to make it clear that the intent of this HPI testing requirement is to verify proper response to actuation of Engineered Safeguards (ES) channels 1 and 2 by the HPI system, as indicated by control room instrumentation. The test is not intended to verify HPI pump performance, which is tested in accordance with American Society of Mechanical Engineers (ASME) Code requirements (Section XI IWP) as required by 10 CFR Part 50.55a(f).

This revision to the Bases of TS 4.5.1.1.1 is acceptable because it merely clarifies the intent of the specification.

TS 4.5.1.1.2 - Revision to Bases for testing Low Pressure Injection (LPI) and related LPSW Systems:

The Bases associated with TS 4.5.1.1.2 have been revised to clarify the intent of the LPI testing requirements of TS 4.5.1.1.2. The intent is to verify proper response, as indicated by control room instrumentation, to actuation of Engineered Safeguards (ES) channels 3 and 4 by the LPI system, as well as by LPSW system components which support the LPI system (e.g., valves LPSW-4 and LPSW-5). The test is not intended to verify containment heat removal capability of the LPI coolers; this is accomplished by testing in accordance with proposed Specification 4.5.3. The test is not intended to verify LPSW pump performance, which is tested in accordance with ASME Section XI IWP requirements.

This revision to the Bases of TS 4.5.1.1.2 is acceptable because it serves to clarify the intent of the specification.

TS 4.5.1.2.1 - Deletion of redundant testing requirements:

Currently, TS 4.5.1.2.1 specifies that the HPI and LPI pumps be tested in accordance with TS 4.0.4. In addition, this TS requires verification of initial pump startup and operation for 15 minutes with discharge pressure and flow within ± 10 percent of a point on the generic pump head curves in Figures 4.5.1-1 and 4.5.1-2 as the acceptance criteria. This requirement is redundant to both TS 4.0.4 and 10 CFR 50.55a, which specify testing of safety-related pumps in accordance with the ASME Boiler and Pressure Vessel Code. The Code requires measurement of pressure head and flow at a single selected point of operation. When compared with previous measurements of flow and pressure head, this testing will monitor pump degradation and will indicate a shift in the pump head-flow curve. Pump curves are available in the Final Safety Analysis Report for indicating design basis/safety limits for these pumps.

The deletion of TS 4.5.1.2.1 is acceptable because its requirements are redundant to those of TS 4.0.4 and 10 CFR 50.55a. The proposed deletion of Figures 4.5.1-1 and 4.5.1-2 is acceptable because these pump curves are available in the Final Safety Analysis Report.

TS 4.5.2.1.1 - Extension of the test interval for the Reactor Building Spray (RBS) system spray nozzle flow test from 5 years to 10 years:

Currently, TS 4.5.2.1. requires testing for obstruction of the spray nozzles every five years by blowing compressed air or fog through the spray headers and nozzles. After additional review of the required frequency of spray nozzle testing, and consideration of the passive nature of the design of the spray nozzles, the NRC staff concluded that a test at 10-year intervals is adequate to detect obstruction of the spray nozzles. The 10-year test interval is specified in Surveillance Requirement 3.6.6.8. in NUREG-1430, "Standard Technical Specifications for Babcock and Wilcox (B&W) Plants," September 28, 1992. Also, in NUREG-1366, "Improvements to Technical Specification Surveillance Requirements," September 28, 1992, the NRC staff recommends that the test interval for testing spray nozzles be extended to 10 years. The 10-year surveillance interval for nozzle plugging in spray systems constructed of stainless steel tubing was also included in Generic Letter (GL) 93-05, "Line-Item TS Improvements to Reduce Surveillance Requirements Testing during Power Operation," dated September 27, 1993. For spray system piping constructed of carbon steel, any change of the surveillance interval must be justified. By telephone on November 1, 1993, the licensee confirmed that the spray system piping in the three Oconee units is constructed of stainless steel.

In light of these generic recommendations for B&W plants, we find the proposed revision to TS 4.5.2.1.1, extending the test interval to 10 years, acceptable.

The licensee has relocated TS 4.5.2.1.1.c to become a subsection of TS 4.5.2.1.1.a to make it clear that these test acceptance criteria (visual observation and control board indication of response to the actuation signal) apply to the test of TS 4.5.2.1.1.a but do not apply to the nozzle air flow test of TS 4.5.2.1.1.b. We find this revision is administrative in nature and clarifies the intent of TS 5.4.5.2.1.1; therefore, the revision is acceptable.

TS 4.5.2.1.1.a Bases - Revision to Bases for testing the RBS System:

The Bases of TS 4.5.2.1.1.a have been revised to provide additional clarification that the intent of revised TS 4.5.2.1.1.a is to verify proper response to actuation of Engineered Safeguards (ES) channels 7 and 8 by the Reactor Building Spray system.

Also, the reference in the Bases to a spray pump flow acceptance criterion (1000 gpm) has been deleted, as well as a reference to the monthly rotation of LPSW pumps. These revisions to the Bases are acceptable because the LPSW pumps are tested per the requirements of the ASME Code and 10 CFR Part 50.55a. These tests do not verify the capability of the pumps to meet design basis but do monitor the pumps for degradation.

TS 4.5.2.1.2.b - Revision of RBCU testing requirements to show acceptance criteria in terms of response to an ES signal:

The current RBCU test acceptance criteria in TS 4.5.2.1.2.b include specifications of LPSW flow through each cooler (greater than 1400 gpm) and air flow through each fan (greater than 40,000 CFM). The licensee proposes to revise the surveillance to delete the LPSW flow and air flow test requirements, and to state acceptance criteria in terms of component response to an ES signal. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate valves have completed their travel and fans are running at half speed.

The testing of the RBCU system pumps and valves is performed in accordance with the ASME Code, which does not require the special LPSW flow and air flow tests. However, the cooling capability of the RBCU system will be directly verified by the requirements of the proposed new TS 4.5.3 (see below). Therefore, the deletion of the LPSW flow and air flow test requirements from TS 4.5.2.1.2.b is acceptable.

TS 4.5.2.1.2 Bases - Revision of RBCU/LPSW testing requirements:

The licensee proposes to revise the Bases of TS 4.5.2.1.2 to make clear that the intent of its testing requirements is to verify response to activation of Engineered Safeguards (ES) channels 5 and 6 by the RBCUs, as well as by LPSW system components which support the RBCUs. The intent is not to verify containment heat removal capability of the RBCUs; this is accomplished by additional testing in accordance with proposed new TS 4.5.3. This revision is acceptable because it helps to clarify the intent of TS 4.5.2.1.2.

TS 4.5.2.2 - Deletion of redundant testing requirements:

Currently, TS 4.5.2.2 specifies that RBS system pumps and valves be tested in accordance with TS 4.0.4. In addition, this TS references a generic pump head curve in Figure 4.5.2-1 as the acceptance criterion. This requirement is redundant to both TS 4.0.4 and 10 CFR 50.55a, which require testing of safety-related pumps and valves in accordance with the ASME Boiler and Pressure Vessel Code. The Code requirements specify testing at reference values of pressure head and flow. These tests do not verify the capability of the pumps to meet design basis but do monitor the pumps for degradation.

The proposed deletion of TS 4.5.2.2 is acceptable because the testing of RBS system pumps and valves in accordance with the ASME Code is already required by TS 4.0.4 and 10 CFR 50.55a. The deletion of Figure 4.5.2-1 is also acceptable because it is available in the Final Safety Analysis Report for indicating design basis/safety limits for these pumps.

TS 4.5.2 Bases - Editorial change:

In the first sentence of the Bases relating to TS 4.5.2, the phrase "Reactor Building Coolant Systems," has been corrected to read "Reactor Building Cooling Systems." This change revising incorrect terminology is acceptable.

New TS 4.5.3 - Containment Heat Removal Capability:

The TS relating to the LPSW, LPI, RBS, and RBCU testing requirements do not specifically verify that these systems are capable of performing the intended safety function of maintaining containment pressure and temperature below design limits following an accident. It was presumed that the LPSW flow and air flow through each RBCU specified in the current TS 4.5.2.1.2, for example, would ensure an adequate post-accident containment heat removal capability for each RBCU.

These specifications did not provide for the possible loss of heat removal capability by service-induced fouling of the heat exchangers (coolers) in the LPSW, LPI, and RBCU systems. Therefore, the licensee proposes to add the new TS 4.5.3 requiring the specific surveillance of containment heat removal capability on a refueling frequency (TS 4.5.3.1.a). In addition, TS 4.5.3.1.b requires the determination of the fouling rate of the LPI and RBCU coolers so that the frequency of surveillance may be modified, if required to ensure that containment heat removal capability remains sufficient to maintain post-accident conditions within design limits.

With the addition of the proposed new TS 4.5.3, an additional justification is provided for the deletion of the LPSW pump flow and RBCU fan air flow requirements of TS 4.5.2.1.2, from which containment heat removal capability is currently inferred. This capability is more reliably determined from the surveillance of the proposed TS 4.5.3. As discussed above, the surveillance in the proposed new TS 4.5.3 would provide increased assurance that the containment heat removal systems would maintain post-accident conditions in the containment within design limits. Therefore, the proposed new TS 4.5.3 is acceptable.

New Bases for the new TS 4.5.3 have been added to clarify the need for the new specification.

Administrative changes to renumber the technical specifications:

Current TS 4.5.3, "Penetration Room Ventilation System," and TS 4.5.4, "Low Pressure Injection System Leakage," have been renumbered due to the addition of new TS 4.5.3, "Containment Heat Removal Capability." These changes are purely administrative in nature, and are, therefore, acceptable.

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 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent 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 (58 FR 52983 dated October 13, 1993). 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.

Principal Contributor: S. Kirslis

Date: January 13, 1994