June 1, 2004

Mr. Ronald A. Jones, Vice President
Oconee Nuclear Station
Duke Energy Corporation
7800 Rochester Highway
Seneca, South Carolina 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3537, MB3538, AND MB3539)

Dear Mr. Jones:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment Nos. 338, 339, and 339 to Renewed 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 (TSs) in response to your application dated October 16, 2001; as supplemented by letters dated May 20, September 12, and November 21, 2002; September 22 and November 20, 2003; and February 18 and April 14, 2004.

The amendments revise the TSs to incorporate changes resulting from use of an alternate source term.

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,

/RA/

Leonard N. Olshan, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 338 to DPR-38
2. Amendment No. 339 to DPR-47
3. Amendment No. 339 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

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DUKE ENERGY CORPORATION

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 338
Renewed 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) Renewed Facility Operating License No. DPR-38 filed by the Duke Energy Corporation (the licensee) dated October 16, 2001; as supplemented by letters dated May 20, September 12, and November 21, 2002; September 22 and November 20, 2003; and February 18 and April 14, 2004; 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 Renewed Facility Operating License No. DPR-38 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 338, 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 in accordance with the schedule provided in the licensee's letter dated February 18, 2004.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephanie M. Coffin, Acting Section Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Technical Specification Changes

Date of Issuance: June 1, 2004

DUKE ENERGY CORPORATION

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 339
Renewed 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) Renewed Facility Operating License No. DPR-47 filed by the Duke Energy Corporation (the licensee) dated October 16, 2001; as supplemented by letters dated May 20, September 12, and November 21, 2002; September 22 and November 20, 2003; and February 18 and April 14, 2004; 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 Renewed Facility Operating License No. DPR-47 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 339, 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 in accordance with the schedule provided in the licensee's letter dated February 18, 2004.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephanie M. Coffin, Acting Section Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Technical Specification Changes

Date of Issuance: June 1, 2004

DUKE ENERGY CORPORATION

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 339
Renewed 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) Renewed Facility Operating License No. DPR-55 filed by the Duke Energy Corporation (the licensee) dated October 16, 2001; as supplemented by letters dated May 20, September 12, and November 21, 2002; September 22 and November 20, 2003; and February 18 and April 14, 2004; 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 Renewed Facility Operating License No. DPR-55 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 339, 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 in accordance with the schedule provided in the licensee's letter dated February 18, 2004.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stephanie M. Coffin, Acting Section Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Technical Specification Changes

Date of Issuance: June 1, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 338

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 339

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 339

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
iii	iii
3.3.6-1	3.3.6-1
3.3.16-1	3.3.16-1
3.3.16-2	3.3.16-2
3.7.9-1	3.7.9-1
3.7.9-2	3.7.9-2
3.7.9-3	3.7.9-3
3.7.10-1	3.7.10-1
3.7.10-2	-
3.7.16-1	3.7.16-1
3.7.16-2	3.7.16-2
3.7.16-3	3.7.16-3
3.7.17-1	3.7.17-1
3.7.17-2	3.7.17-2
3.8.2-2	3.8.2-2
3.8.2-3	3.8.2-3
3.8.4-1	3.8.4-1
3.8.7-1	3.8.7-1
3.8.9-1	3.8.9-1
3.9.3-1	3.9.3-1
3.9.3-2	3.9.3-2
3.9.6-1	3.9.6-1

Remove	Insert
5.0-8	5.0-8
5.0-20	5.0-20
5.0-21	5.0-21
iv	iv
B 3.3.5-1	B 3.3.5-1
B 3.3.5-2	B 3.3.5-2
B 3.3.5-6	B 3.3.5-6
B 3.3.6-2	B 3.3.6-2
B 3.3.7-2	B 3.3.7-2
B 3.3.16-1	B 3.3.16-1
B 3.3.16-2	B 3.3.16-2
B 3.3.16-3	B 3.3.16-3
B 3.7.9-2	B 3.7.9-2
B 3.7.9-3	B 3.7.9-3
B 3.7.9-4	B 3.7.9-4
B 3.7.9-5	B 3.7.9-5
B 3.7.10-1	B 3.7.10-1
B 3.7.10-2	-
B 3.7.10-3	-
B 3.7.10-4	-
B 3.7.16-4	B.3.7.16-4
B 3.7.16-6	B 3.7.16-6
B 3.7.16-7	B 3.7.16-7
B 3.7.17-1	B 3.7.17-1
B 3.7.17-2	B 3.7.17-2
B 3.7.17-3	B 3.7.17-3
B 3.8.2-1	B 3.8.2-1
B 3.8.2-2	B 3.8.2-2
B 3.8.2-3	B 3.8.2-3
B 3.8.2-4	B 3.8.2-4
B 3.8.2-5	B 3.8.2-5
B 3.8.4-1	B 3.8.4-1
B 3.8.4-2	B 3.8.4-2
B 3.8.4-3	B 3.8.4-3
B 3.8.7-1	B 3.8.7-1
B 3.8.7-2	B 3.8.7-2
B 3.8.7-3	B 3.8.7-3
B 3.8.9-1	B 3.8.9-1
B 3.8.9-2	B 3.8.9-2
B 3.8.9-3	B 3.8.9-3
B 3.9.3-1	B 3.9.3-1
B 3.9.3-2	B 3.9.3-2
B 3.9.3-3	B 3.9.3-3
B 3.9.3-4	B 3.9.3-4
B 3.9.3-5	B 3.9.3-5
B 3.9.6-1	B 3.9.6-1
B 3.9.6-2	B 3.9.6-2
B 3.9.6-3	B 3.9.6-3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 338 TO RENEWED FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 339 TO RENEWED FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 339 TO RENEWED FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC) dated October 16, 2001, as supplemented by letters dated May 20, September 12, and November 21, 2002, September 22, and November 20, 2003, and February 18 and April 14, 2004, Duke Energy Corporation (the licensee) submitted a request for full implementation of the alternative source term (AST) for changes to the Oconee Nuclear Station, Units 1, 2, and 3, technical specifications (TSs). The requested change would incorporate revisions to the TSs resulting from the use of an AST that revises the radiological consequence analyses for two design-basis accidents (DBAs) based on the AST, the loss-of-coolant accident (LOCA), and the fuel-handling accident (FHA). The supplements dated May 20, September 12, and November 21, 2002, and February 18 and April 14, 2004, provided clarifying information that did not change the scope of the October 16, 2001, submittal.

In the October 16, 2001, submittal, the licensee requested, among other things, to remove the operational requirements of the penetration room ventilation system (PRVS) and spent fuel pool ventilation system (SFPVS) from the TSs. The proposed no significant hazards consideration (NSHC) was published in the *Federal Register* (FR) on January 22, 2002 (67 FR 2922). The supplement dated September 22, 2003, changed the scope of the October 16, 2001, submittal based on discussions with the NRC staff. The revised proposed NSHC was published in the FR on October 14, 2003 (68 FR 59215). In the supplement dated September 22, 2003, the licensee requested to (1) retain the PRVS and SFPVS in the TSs without crediting the removal of fission products by these systems in the radiological consequence analyses for the site boundaries and for the control room, and (2) adopt Technical Specification Task Force (TSTF) Change Traveler TSTF-51, Revision 2.

On November 20, 2003, based on further discussions with the NRC staff, the licensee resubmitted the request to remove the PRVS from the TSs and to adopt TSTF-51 for the

SFPVS with the addition of one surveillance requirement change concerning operational testing of the SFPVS trains. The revised proposed NSHC was published in the FR on December 9, 2003 (68 FR 68660).

Specific TS changes requested are listed in Section 4.0, “Technical Specification Changes,” of this Safety Evaluation (SE).

2.0 REGULATORY EVALUATION

The NRC staff considered the following regulations, design criteria, and guides to evaluate the licensee’s request:

The current radiological consequence analyses for the DBA for Oconee are based on the TID-14844 accident source term. In this license amendment request, the licensee requested a full-scope implementation of the AST, as described in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” and pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.67, “Accident Source Term.” The use of an AST is addressed in 10 CFR 50.67, which provides a mechanism for licensed power reactors to replace the traditional source term used in their DBA radiological consequence analyses.

The NRC staff evaluated the radiological consequences of affected DBAs against the dose criteria specified in 10 CFR 50.34(b)(2); these criteria are 0.25 Sieverts (Sv), or 25 rem, total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2-hour period following the onset of the postulated fission product release, 0.25 Sv (25 rem) TEDE at the outer boundary of the low population zone (LPZ) for the duration of exposure to the release cloud, and 0.05 Sv (5 rem) TEDE in the control room.

The NRC staff used applicable guidance in Standard Review Plan (SRP) Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Term,” and RG 1.183. Other relevant regulatory documents used by the NRC staff are (1) General Design Criterion (GDC) 19, “Control Room,” of Appendix A to 10 CFR Part 50 and NUREG-0737, Item III.D.3.4, as they relate to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases, and (2) 10 CFR 50.36(c)(2), “Limiting Conditions for Operation” in TSs, as it relates to criteria for including an item as a limiting condition for operation in TSs.

Oconee Updated Final Safety Analysis Report (UFSAR) Section 3.1.17, Criterion 17, “Monitoring Radioactivity Releases (Category B),” provides the licensee’s interpretation and intent of agreement with the original Atomic Energy Commission GDC proposed in 1967 for monitoring radioactivity releases that the licensee adopted as part of the Oconee initial plant design-basis.

Oconee UFSAR Section 3.1.70, Criterion 70, “Control of Releases of Radioactivity to the Environment (Category B),” states that the facility design shall include those means necessary to maintain control over the plant’s radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified: a) on the basis of 10 CFR Part 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur, and b) on the basis of 10 CFR Part 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence, except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides a definition of defense-in-depth and guidance on using a qualitative approach to risk assessment.

Defense-in-depth is a long standing and widely accepted design philosophy used in the NRC regulatory process. Defense-in-depth is used to assure that there are multiple barriers or systems available to assure that the public and operating staffs are protected from the release of radioactive elements during both accident and normal operating events. Defense-in-depth is invoked in RG 1.183, which provides the guidance on the use of the AST.

TSTF-51 is an industry-initiated and NRC-accepted method for allowing some engineered safety feature (ESF) systems and components to be non-operable during a part of the refueling outage, subject to a defined decay period and acceptable shutdown administrative controls.

3.0 TECHNICAL EVALUATION

3.1 Loss-of-Coolant Accident

The current radiological consequence analysis for the postulated LOCA is based on the accident source term described in TID-14844 and it is provided in UFSAR Section 15.15, "Maximum Hypothetical Accident." To demonstrate that the ESFs designed to mitigate the radiological consequences at Oconee will remain adequate after implementing the changes requested in this license amendment, the licensee re-analyzed the offsite and control room radiological consequences of the postulated LOCA. The licensee has implemented an AST in this re-analysis.

The licensee submitted the results of its offsite and control room dose calculations and provided the major assumptions and parameters used in its dose calculations. As documented in its submittals, the licensee has determined that after implementation of the changes requested in this license amendment with use of an AST, the existing ESF systems at Oconee will still provide assurance that the radiological consequences of the postulated LOCA at the EAB, in the LPZ, and in the control room will meet the acceptable radiation dose criteria specified in 10 CFR 50.67(b)(2). As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole-body and thyroid dose guidelines of 10 CFR 100.11 and GDC 19.

The NRC staff has reviewed the licensee's analyses and has performed an independent confirmatory radiological consequence dose calculation for the following three potential fission product release pathways:

(1) reactor building (RB) leakage,
(2) leakage from ESF systems outside containment, and

(3) emergency core cooling system (ECCS) back-leakage to the borated water storage tank (BWST) through check valves in the normal suction line from the BWST.

3.1.1 Reactor Building Leakage

The Oconee RB completely encloses the reactor coolant system. It is designed for an internal pressure of 59 psig and the current RB design-basis leak rate is 0.25 percent by weight per day (% per day). For the radiological consequence analysis, this rate is followed by 0.125% per day after 24 hours following a LOCA for the duration of the accident (30 days). In addition, 50% of the 0.25% per day leakage is processed by the PRVS. The remaining 50% of the 0.25% per day leakage is assumed to be released directly to the environment.

The licensee proposed to lower the RB leak rate specified in the current TS Section 5.5.2 to 0.20% per day from 0.25% per day. For the radiological consequence analysis, the licensee also proposed to lower the RB leak rate to 0.1% per day from 0.125% per day after 24 hours following a LOCA for the duration of the accident.

In its original submittal dated October 16, 2001, the licensee proposed to delete the PRVS from the TSs. The PRVS is an ESF system designed to collect and process RB leakage to minimize the release of radioactivity to the environment following a LOCA. This system uses high efficiency particulate air (HEPA) filters and charcoal adsorbers.

In a subsequent supplement dated September 22, 2003, the licensee revised its original request to retain the PRVS in the TS without crediting the removal of fission products by this system in the LOCA radiological consequence analyses for the site boundaries and for the control room. In the supplement dated November 20, 2003, the licensee proposed again to delete the PRVS from the TS without crediting the removal of fission products.

As a result of not crediting the removal of fission products by the PRVS, the licensee proposed to delete the requirement from the TSs to measure the RB leakage rate to the penetration room. The licensee assumed all leakage (100 percent) from the RB is released directly to the environment without filtration by the PRVS.

The fission products in the RB atmosphere following the postulated LOCA are mitigated by natural deposition of fission products in aerosol form and by the RB spray system (RBSS). The licensee stated in its letter dated September 12, 2002, that it has conservatively neglected aerosol deposition in the RB and excluded any credit for the removal of fission products in aerosol form by natural deposition in the radiological consequence re-analyses. The radiological consequence analyses performed by the licensee showed that Oconee would still meet the relevant dose criteria specified in 10 CFR 50.67 without any credit for removing fission products by natural deposition processes in the RB.

The RBSS is an ESF system and is designed to provide RB cooling and fission product removal in the RB following the postulated LOCA. The RBSS consists of two spray pumps that are automatically started by a high RB pressure signal (15 psig). The licensee assumed that spray flow is initiated within 96 seconds from the initiation of the postulated LOCA. During the RBSS operation, the licensee assumed a mixing rate of two unsprayed volumes per hour between the sprayed and unsprayed portions of the RB atmosphere that is consistent with the guidance provided in RG 1.183. This mixing rate is 30,500 cubic feet per minute.

Two pumps start taking suction initially from the BWST and initiate building spray through two spray headers until the water in the BWST reaches a pre-set low level 25 minutes after the accident. The spray pump suction is then transferred manually to the RB sump and the spray water is re-circulated until the spray operation is terminated 112.8 hours after the accident. Two pump operation is more conservative than one pump operation in the radiological analysis since two pump operation will draw down the water in the BWST earlier, starting recirculation, which provides a lower iodine removal efficiency. The licensee used the models and guidance provided in RG 1.183 to determine the removal rates by the RBSS for iodine in elemental form and fission products in particulate form. Therefore, the NRC staff finds these removable rates acceptable. The major parameters and assumptions used by the licensee, including the spray removal rates, are listed in Table 2.

3.1.2 Post-LOCA Leakage From Engineered Safety Features Outside Containment

Any leakage water from ESF components located outside the RB releases fission products during the recirculating phase of long-term core cooling following a postulated LOCA. The leakage from ESF components occurs in the auxiliary building. The licensee assumed the leakage into the auxiliary building to start at 25 minutes following a postulated LOCA when the RBSS switches its water suction from the BWST to the RB sump. The leakage rate is assumed to be 50 gallons per hour, which is 2 times the allowable leakage limit. The licensee assumed that 10 percent of the total iodine activity in the leaked fluid becomes airborne and is released directly to the environment throughout the entire period of the accident. These assumptions (ESF leak rate and iodine release fraction) are consistent with the guidelines provided in RG 1.183, and therefore, the NRC staff finds they are acceptable. The major parameters and assumptions used by the licensee are listed in Table 2.

3.1.3 Emergency Core Cooling System Back-Leakage to Borated Water Storage Tank

During a postulated LOCA, the suction water source for the ECCS is switched from the BWST to the RB sump. The licensee assumed the leakage into the auxiliary building to start at 25 minutes following a postulated LOCA when the ECCS switches its water suction from the BWST to the RB sump. In this configuration, check valves in the normal suction line from the BWST provide isolation between the RB sump and the BWST. The licensee assumed, in its radiological consequence analysis, 5 gallons per minute (gpm) back-flow leakage through check valves into the BWST from the RB sump. The licensee stated in its letter dated May 20, 2002, that the back-leakage to the BWST is tested every reactor outage and that the 5 gpm leakage value assumed in the radiological consequence analyses bounds actual test results.

In determining the radiological dose calculation through this pathway, the licensee developed its own methodology which is partly based on a model described in NUREG/CR-5950, "Iodine Evolution and pH Control." The licensee calculated and submitted time dependent BWST water temperature, iodine concentrations in the BWST liquid and vapor, and the BWST water pH. Using these calculated values, the licensee determined iodine release rates from the BWST atmosphere to the environment. The NRC staff reviewed the model used and calculation performed by the licensee for fission product release through this pathway. The NRC staff finds the licensee's release model and assumptions to be conservative, and therefore, acceptable. The major parameters and assumptions used by the licensee are listed in Table 2.

3.1.4 Radiological Consequence of Loss-of-Coolant Accident

The licensee re-evaluated the radiological consequences resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB and LPZ are within the dose criteria specified in 10 CFR 50.67. To verify the licensee's radiological consequence analyses, the NRC staff performed its confirmatory radiological consequence dose calculation and found its results are also within the dose criteria specified in 10 CFR 50.67. Although the NRC staff performed its independent radiological consequence dose calculation as a means of confirming the licensee's results, the NRC staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculation are provided in Table 1 and the major parameters and assumptions used by the licensee and found acceptable by the NRC staff are listed in Tables 2 and 4. The radiological consequences of the LOCA at the EAB and at the LPZ calculated by the licensee and by the NRC staff are within the dose criteria specified in 10 CFR 50.67 and the control room dose limit as established by GDC 19.

3.2 Fuel-Handling Accident

The current radiological consequence analysis for the postulated design-basis FHA is based on the accident source term described in TID-14844, and it is provided in UFSAR Section 15.11. In the TS change request, the licensee proposed to revise TS Section 3.9.3 to allow the containment air lock doors, the equipment hatch and each penetration providing direct access from the containment atmosphere to the outside atmosphere to remain open during fuel movement and refueling operations. The licensee re-evaluated the radiological consequences resulting from a postulated FHA in the spent fuel pool (SFP) area and in the containment with the personnel air locks open, the equipment access hatch open, and the containment penetrations open providing direct access from the containment atmosphere to the environment. The licensee implemented the AST in its re-evaluation of a FHA. Each of the Oconee unit's containment is equipped with two containment air lock doors and an equipment access hatch.

The licensee considered two fuel-handling events for the postulated FHA: (1) the drop of a single fuel assembly in the containment or SFP, and (2) the drop of a fuel transport cask or an independent spent fuel storage installation (ISFSI) transfer cask (with multiple fuel assemblies) in the SFP. The licensee concluded that the radiological consequences resulting from the postulated FHA in the SFP area and in the containment with open personnel air locks, the equipment access hatch open, and the containment penetrations providing direct access from the containment atmosphere to the environment are within the dose acceptance criteria specified in SRP 15.0.1 for the EAB and in 10 CFR 50.67 for the control room.

The licensee reached this conclusion as a result of:

(1) implementing the AST,

(2) taking no credit for removal of fission products by the PRVS, SFPVS, or the RB purge exhaust filtration system as proposed in the TS change request,

(3) assuming no containment isolation (open air lock doors, equipment hatch, and containment penetrations) as proposed in this TS change request,

(4) using an overall decontamination factor of 183 for iodine in elemental and organic forms in the SFP water with minimum water depth of 21.34 feet consistent with the guidelines provided in RG 1.183,

(5) using the ARCON96 model documented in RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” in determining the control room intake air dispersion factors (X/Q values) (see Section 3.4 below),

(6) taking credit for dual control room air intake locations, reducing its X/Q values by assuming that wind flow is in the direction of the intake with the poorer atmospheric dispersion conditions and that this intake is drawing in 60 percent of the air (see Section 3.4 below),

(7) using new relocated control room air intake locations on the northeast and southeast corners of the turbine building roof as proposed in this TS change request,

(8) assuming all fuel rods in one fuel assembly with an axial power peaking factor of 1.65 and a peak rod average fuel burnup of 62,000 megawatt days per metric ton uranium (MWD/MTU) are damaged to the extent that its entire gap activity inventory of the damaged fuel rods is released to the surrounding water for single fuel assembly drop event consistent with the guidelines provided in RG 1.183 (the corresponding maximum linear heat generation rate is 6.0 kW per foot based on 418.6 effective full power days, 3 cycles, an active fuel height of 11.86 feet, and 208 pins per fuel assembly), and

(9) using a fission product decay period of 72 hours (time period from the reactor shutdown to the first fuel movement) for single fuel assembly drop event.

For multiple fuel assembly events involving a drop of a fuel transport cask or an ISFSI transfer cask in the SFP, the number of fuel assemblies involved in these events varies depending on the location of the cask drop (i.e., the Unit 1 and 2, or Unit 3, SFP) and the type of cask dropped (i.e., transport cask or ISFSI cask). Oconee is designed with two SFPs: the Unit 1 and 2 SFP and the Unit 3 SFP. In its letter dated May 30, 2000, the licensee provided a table summarizing the number of fuel assemblies that would be involved in each cask drop event in each SFP ranging from 518 to 1024 fuel assemblies. Since the number of fuel assemblies involved in these events is greater than the amount of fuel recently discharged from a core, (i.e., 177 fuel assemblies per core), the licensee assumed two different decay times for each event: one for the fuel recently discharged from the core (55 to 70 days) and one for the other fuel involved in the event (one year). The NRC staff finds the licensee’s assumption conservative and, therefore, acceptable.

The licensee performed 12 different radiological consequence cases for these FHA events, using a combination of different type of cask dropped (i.e., transport cask or ISFSI cask), different location of the cask drop (i.e., the Unit 1 and 2, or Unit 3, SFP), and the different control room air intakes (i.e., Unit 1 and 2, or Unit 3, control room). Oconee has two control rooms, one for Unit 1 and 2 and one for Unit 3. The licensee determined that (1) the bounding single fuel assembly FHA event is the FHA in either the Unit 1 and 2, or Unit 3, SFP to the Unit

1 and 2 control room air intake, and (2) the bounding multiple-fuel assembly FHA event is the transport cask drop event in either the Unit 1 and 2, or Unit 3, SFP to the Unit 1 and 2 control room air intake.

The NRC staff reviewed the methods, parameters, and assumptions that the licensee used in its radiological dose consequence analyses. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the FHA events that will produce the greatest radiological consequence. The NRC staff finds, and agrees with the licensee, that the bounding single fuel assembly FHA event is the FHA in the Unit 1 and 2 SFP to the Unit 1 and 2 control room air intake, and (2) the bounding multiple fuel assembly FHA event is the transport cask drop event in the Unit 1 and 2 SFP to the Unit 1 and 2 control room air intake. The radiological consequences calculated by the NRC staff are well within the dose criterion specified in 10 CFR 50.67 (5 rem TEDE in the control room) and meet the dose acceptance criterion specified in the SRP 15.0.1 (a 6.3 rem TEDE at the EAB).

Even though the NRC staff performed its confirmatory dose calculations, the NRC staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee, which are acceptable to the NRC staff, are listed in Table 3. The radiological consequences at the EAB, at the LPZ, and in the control room calculated by the licensee are all well within the dose criterion specified in 10 CFR 50.67 and meet the dose acceptance criterion specified in GDC 19.

3.3 Control Room Habitability

The Oconee control rooms are located in the auxiliary building. Units 1 and 2 have a shared control room while Unit 3 has a separate control room. The control room habitability system includes two control room ventilation system booster fan trains (CRVSBFTs). The CRVSBFT is provided to maintain a positive control room pressure to ensure outward leakage, preventing unfiltered air inleakage into the control room. The CRVSBFT consists of, among other things, two separate outside air booster fans, HEPA filters and charcoal adsorbers. Each booster fan is designed to provide 1350 CFM. The licensee assumed that the control room operator will start the control room ventilation system (CRVS) booster fans within 30 minutes of the DBA.

The licensee requested a revision to the acceptance criteria for the CRVSBFT charcoal adsorbers in TS Section 5.5.12 to greater than 97.5 percent radioactive methyl iodide removal from the current requirement of greater than 90 percent.

The licensee committed to relocate the existing control room outside air intakes for the shared Unit 1 and 2 control room and for the separate Unit 3 control room. The control room outside air intakes will be moved from the roof of the auxiliary building to the northeast and southeast corners of the turbine building roof. Dual intakes will be installed for each control room. The Oconee station presently has one outside intake for each control room. Moving the intakes further from the source-term release points would reduce the control room operator dose based on the ARCON96 dispersion model (see Section 3.4 below). Installation of dual intakes, as described in the licensee's planned modification, would divide the dose between the two control rooms, assuming that one intake for each control room drew in contaminated air and the second intake for each control room drew in clean air. The licensee's radiological consequence analyses to support this TS change request used the relocated control room outside air intakes and dual intake design to meet the relevant dose acceptance criterion.

On June 12, 2003, the NRC staff issued Generic Letter (GL) 2003-01, "Control Room Habitability." This GL identifies NRC staff concerns regarding the reliability of current surveillance testing to identify and quantify control room inleakage, and requests licensees to confirm the most limiting unfiltered inleakage into their control room envelope. On December 9, 2003, the licensee submitted a "60-day" response to this GL for Oconee. In this submittal, the licensee stated that tracer gas tests were performed at Oconee in 1998 and 2001 for both the Unit 1 and 2 control room, and the Unit 3 control room, to confirm that the inleakage into the control room envelope is less than or equal to the value assumed in the Oconee radiological consequence analysis. The purpose of the 2001 re-test was to demonstrate the effectiveness of the CRVS sealing program.

The test methodology and analyses in determining the unfiltered air inleakage rate to the control room used in the 1998 and 2001 tests were based on the American Society for Testing and Materials Standard E741-95, "Standard Test Method for Determining Air Change Rate in a Single Zone by Means of a Tracer Gas Dilution." The licensee provided the test results obtained in 1998 in its letter dated October 16, 2001, and provided the test results obtained in 2001 in its letter dated May 20, 2002. The test results from the 2001 test were not available when the October 16, 2001, letter was submitted.

Both test results are provided in Table 5. The 1998 test resulted in 80 cfm for the Unit 1 and Unit 2 control room and 73 cfm for the Unit 3 control room. The uncertainty associated with the testing and analyses was ± 55 cfm for the Unit 1 and Unit 2 control room and ± 25 cfm for the Unit 3 control room. The 2001 test resulted in zero measured inleakage (0 cfm) to both control rooms; the uncertainty associated with the testing and analyses is ± 18 cfm for the Unit 1 and Unit 2 control room and ± 13 cfm for the Unit 3 control room.

In its radiological consequence analyses, the licensee has chosen a value of 40 cfm as the current design-basis control room unfiltered air inleakage rate based on the most recent (performed in 2001) tracer gas test result (in spite of zero measured inleakage to both control rooms) in conjunction with the total allowable ECCS leakage rate of 25 gph. The licensee stated that the selection of bounding values for the control room unfiltered inleakage assumed in the radiological analyses provides the licensee with margin to accommodate changes in input assumptions that could be required to account for possible future plant operational changes (e.g., increase in ECCS leakage flow, imbalances ventilation system air flow, reductions in filtration efficiencies).

The licensee provided a sensitivity study with a table showing various combinations of the control room unfiltered air inleakage rates and the total allowable ECCS leakage rates meeting the dose acceptance criteria. The licensee indicated that, based on the result of the next tracer gas test planned in the Control Room Habitability Program, the licensee may change to a different combination of the control room unfiltered air inleakage rates and the total allowable ECCS leakage rates, still meeting the dose acceptance criteria for the EAB, LPZ, and control room. The NRC staff finds the licensee’s approach to be acceptable. The NRC staff also used the 40 cfm control room unfiltered air inleakage rate and the 25 gph total allowable ECCS leakage rate in its confirmatory radiological consequence analyses.

For the first 30 minutes into the postulated LOCA prior to the operation of the CRVS in the emergency pressurization mode in its radiological consequence analyses, the licensee has chosen to use the bounding unfiltered air inleakage rate obtained from the 1998 test (1150 cfm), with the CRVS in the normal operational mode for the Unit 1 and 2 control room, since it is more conservative than that obtained in the 2001 test. The NRC staff also used this leakage value in its confirmatory radiological consequence analyses.

The licensee re-evaluated the control room habitability with the application of the AST using the bounding 40 cfm unfiltered air inleakage into the control room and concluded that the radiological consequences to the control room operator resulting from the postulated LOCA and FHA were within the 5 rem TEDE criterion specified in 10 CFR 50.67. To verify the licensee's radiological consequence assessments, the NRC staff performed its confirmatory radiological consequence dose calculations for the control room operator and finds its results were also within the 5 rem TEDE criterion specified in 10 CFR 50.67. Although the NRC staff performed its independent radiological consequence dose calculation as a means of confirming the licensee's results, the NRC staff's acceptance is based on the licensee's analyses.

The results of the licensee's radiological consequence calculation are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Tables 2 and 4. The radiological consequences for the control room operator calculated by the licensee and by the NRC staff are all within the dose criteria specified in 10 CFR 50.67.

Although the NRC staff has reviewed the licensee's response to GL 2003-01 for Oconee, as well as those received from other licensees, follow-on regulatory action has not been decided at this time. Nonetheless, the NRC staff has determined that there is reasonable assurance that the Oconee control rooms will be habitable during a LOCA and FHA and that these amendments may be approved prior to the NRC staff's formal review of the licensee's response to the GL for Oconee. The NRC staff bases this determination on (1) the Oconee tracer gas test results, which confirmed that the inleakage into the control room envelope is less than the value assumed in the Oconee radiological consequence analyses; (2) the NRC staff's confirming analysis; and (3) the resulting control room operator doses, which meet the acceptable dose limit. The NRC staff's approval of these amendments does not relieve the licensee of addressing the information requests in GL 2003-01 for Oconee and does not imply that the NRC staff would necessarily find the analysis in these amendments acceptable as a response to information request 1(a) in GL 2003-01.

3.4 Atmospheric Dispersion Factors

3.4.1 Meteorological Data

The licensee used five years of on-site meteorological data collected during calendar years 1991 through 1995 to estimate the atmospheric relative concentration (X/Q) values used in the radiological consequence analyses. These data were measured at 10 and 60 meters above grade at the Oconee station. The licensee stated that the program is maintained to comply with RG 1.23, "Onsite Meteorological Programs," and has further described the Oconee meteorological measurement program as follows. The tower in use since 1988 is located in a clearing, west of the plant. The area around the tower has been kept free of obstructions to ensure good exposure of the instruments. Weekly system checks and semi-annual calibration checks are performed to ensure that the instruments are maintained within specifications. During the semi-annual calibrations, all instruments on the tower are replaced with newly certified instruments. Data were reviewed by data reviewers, and validated and edited by the licensee's in-house Certified Consulting Meteorologist prior to archival. Annual data recoveries were approximately 96 percent during the first three years and about 93 percent during the second two years of the 1991 through 1995 period.

The NRC staff performed a review of the data, using the methodology described in NUREG 0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," on meteorological data quality assurance. Further review was performed using a computer spreadsheet. Examination of the data revealed some occurrence of wind data remaining unchanged for two or more consecutive hours. While the occurrence seemed somewhat higher to the NRC staff than expected by the meteorological factors identified by the licensee, even with the uncertainty, the NRC staff estimates that the recovery is still, at a minimum, near 90 percent and the data recovery uncertainties should not have a significant impact on the licensee's X/Q estimates. Although there were several occurrences of data outages of more than a week's duration, most outages were of relatively short duration.

With respect to atmospheric stability, the A stability class was infrequently reported to occur for a relatively long period of time, in excess of a day in several cases. Class A stability conditions were also reported to occur infrequently at night. The licensee attributed this to factors such as cold air advection aloft or post-frontal air mass changes. Regardless of the cause, the NRC staff judges the reported occurrences to have an insignificant effect on the X/Q estimates for the dose assessment described above. Wind speed and direction frequency occurrence at each of the two levels were fairly similar from year to year, with 1991 tending toward slightly lower wind speeds at both levels. Lower level winds were predominately from the northeast and southwest with secondary flow from the northwest. Upper level winds were most often from the north through east northeasterly sectors and from the southwest.

3.4.2 EAB and LPZ Relative Concentration Estimates

With respect to the EAB and LPZ X/Q values, the licensee has used values previously approved by the Directorate of Licensing, U.S. Atomic Energy Commission, in the "Safety Evaluation of the Oconee Nuclear Power Station, Units 2 and 3," issued in 1973. Because Unit 1 is in close proximity to Unit 2 when compared with the distances to the EAB and LPZ, these values should be acceptable for application to Unit 1. The NRC staff did not directly review the X/Q values as a part of this evaluation, but notes that the calculation methodology used by the licensee at that time, about 30 years ago, was different than current guidance and NRC staff practice. However, the NRC staff performed a comparison calculation using the 1991 through 1995 meteorological data and the PAVAN methodology that is based upon RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants." The results of the NRC staff calculations support the numeric acceptability of the licensee's EAB and LPZ X/Q estimates for this amendment. The licensee's values are listed in Table 4.

3.4.3 Control Room Relative Concentration Estimates

The licensee used the ARCON96 methodology discussed in RG 1.194 to make calculations for all possible source-receptor pairs and selected the most limiting cases for use in the dose assessment. All postulated releases were considered as ground level point releases except those from the equipment hatch and fuel handling building roll-up door. The licensee considered the postulated hatch and door releases as ground level diffuse releases. The NRC staff did not make a judgement as to the acceptability of the diffuse release assumptions, but did make comparison calculations assuming the releases were ground level point releases.

Resultant values were slightly higher than assuming a diffuse source, but were still bounded by the X/Q values from releases from the limiting plant vent used in the radiological consequence analyses.

In its initial assessment, the licensee divided the limiting X/Q values by a factor of two. Use of this reduction factor is not part of the current Oconee licensing basis. The licensee based the reduction on plans to modify the CRVS to use dual intakes, rather than a single air intake for each control room as is the current design. This assumed that both of the dual intake channels would perform continuously and simultaneously during the course of an accident to individually provide one-half of the needed air and that air from at least one of the two intakes could be assumed to be uncontaminated. However, the licensee acknowledged that the final design of the control room intakes and air intake flow rates may vary slightly from the assumptions used in the preliminary design assessment. Therefore, the licensee performed additional calculations using projected design variations that would potentially impact the X/Q values. This included an assumption that the intake locations would be 10 feet higher than intake locations used in the preliminary design and an intake flow rate imbalance of 55 and 45 percent, rather than each intake providing 50 percent of the flow.

The intake flow imbalance assumption was based on operational experience with dual intakes at one of the licensee's other plants. The licensee also assumed that wind flow is in the direction of the intake with the poorer atmospheric dispersion conditions and that this intake is drawing in 55 percent of the flow. In its letter dated September 22, 2003, the licensee provided revised control room X/Q values. The resultant values for postulated releases from the plant vent and BWST used in the limiting dose calculations for these proposed amendments are listed in Table 4. Present NRC guidance permits reduction of the X/Q values for dual intakes when the intakes for each individual control room are adequately separated to provide a low contamination intake and when these intakes are adequately designed. The NRC staff finds the physical separation of the intakes for each individual control room adequate. The NRC staff also finds the X/Q values listed in Table 4 acceptable predicated on the assumption that both of the dual intake channels for each individual control room are designed and operated to be capable of performing continuously and simultaneously to individually provide the air flow assumed in the licensee's analysis and that air from at least one of the two intakes could be assumed to be uncontaminated for the duration of an accident.

4.0 <u>TECHNICAL SPECIFICATION CHANGES</u>

The NRC staff has reviewed these proposed changes to the TSs and has made the following assessments based on the descriptions of the changes provided by the licensee, the licensee's UFSAR, and the application of the regulations identified in Section 2.0.

4.1 TS 3.3.6 Engineered Safeguards Protective System (ESPS) Manual Initiation

Limiting Conditions for Operations (LCO) 3.3.6 c is being changed to delete the reference to "Penetration Room Ventilation."

The licensee states that the PRVS will not be credited for control room and off-site doses based on the revised radiological analyses of the Maximum Hypothetical Accident (MHA). The NRC staff has confirmed that the PRVS does not need to be credited in the MHA analysis and that the analysis results are acceptable. As such, the PRVS is not on the primary success path in the mitigation of a DBA. Thus, the ESPS manual initiation channels for the PRVS specified in LCO 3.3.6 are not required to be OPERABLE to support the mitigation of a DBA. The NRC staff has found the proposed change to remove reference to the PRVS to be acceptable.

4.2 TS 3.7.10 Penetration Room Ventilation System (PRVS)

The licensee proposed to remove this section from the TS. The PRVS will not be credited for evaluating potential control room and off-site doses. This change results in an operational efficiency that is achievable from implementing the AST. The revised radiological analyses of the MHA are performed without taking credit for the PRVS filters and the results of this analysis show that the off-site and control room doses remain below the guidance provided in RG 1.183. Removal of this system from the TSs eliminates the requirement to demonstrate the effectiveness of this system in operation. This simplifies testing design and performance tasks.

The NRC staff notes that the licensee is requesting that the PRVS filter system be removed from the TS. As currently licensed, the assumed unfiltered leakage from the containment is 0.1375% per day (based on 90% filter efficiency, and 0.25% per day leakage in which 50% of the leakage passes through the PRVS filters). With the proposed TS changes, the assumed unfiltered leakage per day increases to 0.2% per day. This is a 45 percent increase in the assumed unfiltered containment leakage that is released to the environment. However, the resulting dose calculations show that the release will remain within the licensing regulatory limits with substantial margin.

The NRC staff determined that, if the licensee takes credit for higher control room filter efficiency and for its newly installed dual control room intakes, the impact of a 45 percent larger unfiltered release from the containment can be mitigated for the control room dose and the licensee's AST analysis would produce acceptable dose results without taking credit for the PRVS filters. The licensee in the May 20, 2002, submittal stated that 50 percent of the unfiltered leakage was assumed to be released from the plant vent. This statement implies that the fans must be operational to drive the unfiltered containment leakage to the plant vent.

The plant vent that was used as the release point in the MHA. A release from this location is considered a ground release and that its dispersion factor is conservative with respect to the equipment hatch release dispersion factor. Thus, a vent release bounds other potential release paths (i.e., the equipment hatch release path) and provides additional assurance of acceptability.

The NRC staff evaluated the removal of the PRVS filters to determine the impact on defense-in-depth. The penetration rooms serve as a secondary containment for containment leakage. The filters provide a barrier to the release of radiation to the environment from containment penetration leakage during an accident. The NRC staff concludes that the impact on defense-in-depth is acceptable based on the low frequency of the initiating event and the impact on doses resulting from the release, which are well below the regulatory limits.

UFSAR Section 3.1.17, Criterion 17, requires that releases of radioactive elements from the facility be monitored. Use of the PRVS provides for monitoring a portion of the containment releases during an accident by routing the flow to the plant vent. The licensee conservatively takes credit for the release from the plant vent in its MHA analysis. This provides reasonable assurance that this part of the release will be monitored.

UFSAR Section 3.1.70 commits to those means necessary to maintain control over the plant radioactive effluents. It states specifically that the design for radioactivity control will be based on the basis of 10 CFR Part 100 dosage level guidelines for potential reactor accidents of exceeding low probability. The MHA in which the containment could be pressurized is an accident of low probability. The RG 1.183 limits referenced by the licensee are taken from 10 CFR 50.67 and are the equivalent of 10 CFR Part 100 dosage limits for application of the AST. As such, the licensee is in compliance with the design requirements stated in its UFSAR.

In producing an integrated assessment of the proposed change, the NRC staff also considered, on a qualitative basis, the overall increase in risk. The principal DBA to be considered is the MHA that could pressurize the containment and be the driving force for leakage from the containment. The MHA has a very low frequency of occurrence. The MHA assumption that the entire core will be damaged is conservative. Assumptions about plate out of iodine and retention of iodine in containment fluids are also conservative. Core damage frequency and large early release fractions are unaffected by the PRVS. The resulting dose calculations show that the release would be within the licensing regulatory limits with substantial margins. These considerations provide some assurance that the uncertainties in parameters and the progression of the accident will not induce a situation where the regulatory limits would be exceeded.

The NRC staff has concluded that the proposed change has no appreciable impact on plant safety. The proposed change will enhance efficiency of operation and reduce unnecessary regulatory burden by eliminating TS requirements for this system. Public confidence is maintained by carefully assessing the risk associated with this proposed change and ensuring that the impact on public health and safety is minimal and that any potential increase in exposure is conservatively within established regulatory guidelines.

The NRC staff finds that the removal of the PRVS from the TS is acceptable. Although there is an increase in the unfiltered leakage from an accident and a reduction in defense-in-depth and the ability to mitigate increased leakage, the AST analysis shows acceptable dose results below the limits established in 10 CFR 50.67 and GDC 19. A qualitative assessment of the increased risk indicates that there are reasonable safety margins and that the increased risk is not significant. The licensee has committed to the completion of modifications that support this change including the change in control room filter efficiency testing, the change in containment integrated leak rate test requirements, the use of dual air intakes on the control room, and the modification to limit ECCS leakage during post-LOCA recirculation. Each of these modifications is being implemented under the 10 CFR 50.59 process consistent with the existing plant licensing basis. This finding of acceptability is contingent upon these modifications being completed prior to the implementation of the changes that have been found to be acceptable in this SE.

4.3 TS 3.7.17 Spent Fuel Pool Ventilation System

a. The licensee proposed to delete two notes in LCO 3.7.17:

   Note 1. LCO 3.0.3 is not applicable
   Note 2. Not applicable during reracking operations with no fuel in the spent fuel pool.

   In the February 18, 2004, letter, the licensee stated that during the 72 hours in which fuel is considered to be recently irradiated and the TS is applicable, there are no actions required by entry in LCO 3.0.3, and thus, Note 1 serves no function. The licensee also states that Note 2 is unnecessary since during the short 72-hour period in which the TS is applicable there would be no reracking operations and thus it doesn't serve a function. The NRC staff concurs that these notes may be deleted without impact on public health and safety.

b. The licensee proposed changing the APPLICABILITY from "during movement of fuel in the spent fuel pool" to "during movement of <u>recently irradiated</u> fuel assemblies in the spent fuel pool."

   The proposed change is consistent with TSTF-51, which recognizes that the OPERABILITY of some systems can be relaxed after a sufficient period of time for fuel decay has passed. The period of time defined by "recently" must be equal to or longer than the time used in the FHA design-basis analysis for decay of fuel. The licensee used a period of 72 hours in its FHA DBA for fuel decay. The TS Bases are being revised to provide a definition of "recently irradiated" fuel. The NRC staff finds this change is acceptable.

c. The licensee proposed deleting the statement "During crane operations with loads over the spent fuel pool" from the APPLICABILITY.

   In the February 18, 2004, letter, the licensee stated that this requirement is being deleted since the revised FHA does not take credit for the SFPVS in the event of a Spent Fuel Pool Building Accident. The licensee also stated that for the 72 hours after shutdown in which the SFPVS is required to be operational, fuel can not be physically moved. The NRC staff concurs that crane operations with loads over the SFP for times outside of the initial 72 hours after shutdown pose no threat to public health and safety because the consequences of an FHA, after a decay time of at least 72 hours, are well below regulatory limits.

d. The licensee proposed modifying the ACTIONS in CONDITION A and CONDITION B by changing the wording in the REQUIRED ACTION from "Suspend movement of fuel in the spent fuel pool" to "Suspend movement of recently irradiated fuel assemblies in the spent fuel pool."

The proposed change is consistent with TSTF-51, which recognizes that the OPERABILITY of some systems can be relaxed after a sufficient period of time for fuel decay has passed. The period of time defined by recently must be equal to or longer than the time used in the fuel-handling accident design-basis analysis for decay of fuel. The licensee used a period of 72 hours in its FHA DBA for fuel decay. The TS Bases are being revised to provide a definition of "recently irradiated" fuel. The NRC staff finds this change is acceptable.

e. The licensee proposed modifying the ACTIONS in CONDITION A and CONDITION B by deleting the REQUIRED ACTION to "Suspend crane operations with loads over the spent fuel pool."

   In the February 18, 2004, letter, the licensee stated that this requirement is being deleted since the revised FHA does not take credit for the SFPVS in the event of a Spent Fuel Pool Building Accident. The licensee also stated that for the 72 hours after shutdown in which the SFPVS is required to be operational, fuel can not be physically moved. The NRC staff concurs that crane operations with loads over the SFP for times outside of the initial 72 hours after shutdown pose no threat to public health and safety because the consequences of an FHA, after a decay time of at least 72 hours, are well below regulatory limits.

f. The licensee proposed changing SURVEILLANCE REQUIREMENT SR 3.7.17.1 FREQUENCY from "31 days" to "Within 31 days prior to the movement of recently irradiated fuel assemblies."

   The NRC staff concurs that the planned testing of the SFPVS train in the period of time of less than 31 days prior to the movement of irradiated fuel provides assurance that the system will be functioning and available, i.e., OPERABLE. Although testing at other times may be useful as part of an overall maintenance or surveillance program, the NRC staff concludes that the testing during the immediate period before moving recently irradiated fuel is sufficient for the TS surveillance requirement.

4.4 TS 5.5.2 Containment Leakage Rate Testing Program

a. The licensee proposed changing the maximum allowable containment leakage rate, $L_{a,}$ at P_a, from 0.25% to 0.20% of containment air weight per day.

   The NRC staff reviewed the licensee's request to change the leakage rate criteria from 0.25% per day to 0.20% per day. This is a conservative change in the sense that it allows a smaller leakage rate from the containment. It is a non-conservative change in that it justifies a smaller assumed containment leakage rate in the MHA DBA with the overall effect of reducing the calculated dose. However, a review of test results from the licensee's most recent containment leakage testing shows that the licensee typically has containment leakages less than the 0.20% value that has been proposed. Thus, the change proposed by the licensee is reasonable based on operating experience and the NRC staff finds that the proposed change to the more restrictive leakage rate test criteria is acceptable.

b. The licensee proposed to delete the criterion item b. that states Leakage >0.5 L_a shall be to the penetration room.

The licensee does not take credit for containment leakage to the penetration room and its filter system in its MHA. The licensee assumes that all of the containment leakage, L_a, is released directly to the environment. This is conservative and the NRC staff finds that this change is acceptable.

4.5 TS 5.5.12 Ventilation Filter Testing Program

a. TS 5.5.12 items a and c.

The licensee proposed deleting the PRVS HEPA filter test in item a and the PRVS carbon adsorber test in item c.

As discussed previously, the PRVS system is not credited in any DBA. As such, the NRC staff has determined that the system may be removed from the TS. Similarly, test requirements for the PRVS may also be removed. Requirements for testing this system will be controlled by other plant documents, if applicable. The NRC staff finds this change acceptable.

b. TS 5.5.12 item e.

The licensee proposed to delete reference to the PRVS and to change the laboratory test acceptance criteria for the CRVS booster fan filter from 90% to 97.5% and states that the criteria for the SFPVS carbon adsorber would remain at ≥90%.

Deletion of the reference to the PRVS is acceptable based on the deletion of the PRVS from the TS as discussed above. Increasing the acceptance criteria for the CRVS booster fan carbon adsorber is conservative and allows the licensee to take credit for a higher adsorption efficiency in its DBA analysis. Leaving the SFPVS carbon adsorber efficiency at its current value of ≥90% does not adversely impact the operation of the system. The NRC staff finds these changes acceptable.

d. TS 5.5.12 item f.

The licensee proposed deleting this item since it pertains to the PRVS system which is being deleted from the TS.

The NRC staff finds that the deletion of this item is acceptable since the PRVS system is being removed from the TS. Operational and testing requirements, if applicable, more appropriately belong in other plant documents.

4.6 TSTF-51 Related Technical Specification Changes

Changes related to TSTF-51 establish a point in time following reactor shutdown when OPERABILITY of ESF systems that are typically used to mitigate the consequences of a FHA are no longer required to be OPERABLE in order to meet the SRP guidance for offsite dose limits (less than 25 percent of 10 CFR Part 100 limits or the limits specified in 10 CFR 50.67).

Fuel that has been part of a critical reactor core within the preceding 72 hours is referred to as "recently" irradicated fuel. Only recently irradiated fuel contains sufficient fission products to require OPERABILITY of an accident mitigation feature to meet the accident analysis assumptions. Therefore, the APPLICABILITY requirements for OPERABILITY FHA consequence mitigation features are revised. The requested changes would eliminate TS requirements for these features during core alterations involving fuel movement beyond the recently irradiated time period. The affected TS Limiting Conditions for Operation (LCO) are defined in the following TS sections:

TS Section Number	Title	Extent of change
TS 3.3.16 Applicability, Required Action, and SR 3.3.16.2	Reactor Building (RB) Purge Isolation - High Radiation	Deletes APPLICABILITY during CORE ALTERATIONS Adds recently in front of irradiated fuel Deletes Required Action A.2.1 Adds recently in front of irradiated fuel in Required Action A.2.2 and renumbers it A.2 Deletes CORE ALTERATIONS and adds recently in front of irradiated fuel in the Surveillance Requirement
TS 3.7.9	Control Room Ventilation System (CRVS) Booster Fans	Adds APPLICABILITY "during movement of recently irradiated fuel assemblies." Changes Condition D to limit applicability for Modes 1,2,3, or 4. Adds Condition E for Applicability "during movement of recently irradiated fuel" along with Required Action "suspend movement" and Completion Time "immediately"
Table 3.7.16	Control Room Area Cooling Systems (CRACS)	Adds APPLICABILITY "during movement of recently irradiated fuel assemblies." Changes Condition D to limit applicability for Modes 1,2,3, or 4. Renumbers Condition E to Condition F and limits applicability for Modes 1,2,3, or 4. Adds new Condition E for "required action and completion time not met with Required Actions and Completion Times. Adds new Condition G for "two CRACS trains inoperable during movement of recently irradiated fuel with Required action and Completion Time.
TS 3.8.2 Applicability and Actions A and B	AC Sources - Shutdown	Adds recently in front of irradiated fuel

TS Section Number	Title	Extent of change
TS 3.8.4 Applicability and Required Action A.2.2	DC Sources - Shutdown	Adds recently in front of irradiated fuel
TS 3.8.7 Applicability and Required Action A.2.2	Vital Inverters - Shutdown	Adds recently in front of irradiated fuel
TS 3.8.9 Applicability and Required Action A.2.2	Distribution Systems - Shutdown	Adds recently in front of irradiated fuel
TS 3.9.3 Applicability, Required Action, and SR 3.9.3.2	Containment Penetrations	Deletes APPLICABILITY during CORE ALTERATIONS Adds recently in front of irradiated fuel Deletes Required Action A.1 Adds recently in front of irradiated fuel in Required Action A.2 and renumbers it A.1 Deletes CORE ALTERATIONS and adds recently in front of irradiated fuel in the Surveillance Requirement
TS 3.9.6 Applicability and Required Action	Fuel Transfer Canal Water Level	Deletes APPLICABILITY “during CORE ALTERATIONS, except during latching and unlatching of CONTROL ROD drive shafts” Deletes Required Action A.1

In order to implement the above changes to APPLICABILITY and ACTION statements, the LCO for OPERABILITY need only apply when handling fuel that has recently been in the critical reactor core (i.e., “recently irradiated” fuel). The submittal proposes a revised FHA dose analysis for Oconee which takes credit for a radioactive decay period of 72 hours based on an AST pursuant to 10 CFR 50.67 and the guidance of RG 1.183. Given this decay period, the licensee is now proposing changes to redefine the TS requirements by requiring ESF systems, which are previously relied upon to mitigate an FHA, applicable only during movement of fuel that has been “recently irradiated.” The term “recently irradiated” is a cycle-specific number and represents the decay period for the reduction in radionuclide inventory available for release in the event of an FHA. For the upcoming refueling outage, the licensee has determined that the appropriate decay period will be 72 hours. In summary, once the reactor has been shutdown for a minimum of 72 hours, the licensee has demonstrated that the FHA reanalysis (that does not rely on either building integrity or the FHA mitigating systems) will not exceed dose limitations. The TS Bases are being revised to provide a definition of “recently irradiated” fuel.

In addition, consistent with the instructions in TSTF-51, Revision 2, regarding decreasing doses even further below that provided by natural decay, the licensee has committed to follow the guidelines of NUMARC 93-01, Revision 3, Section 11.3.6, "Assessment Methods for Shutdown Conditions," Subsection 5, "Containment - Primary (PWR)/Secondary (BWR)."

The deletion of the CORE ALTERATIONS term is justified since an FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release, and such FHAs will be fully enveloped by the proposed APPLICABILITY.

In addition to the above changes to the APPLICABILITY statements, the licensee proposed numerous corresponding changes to the ACTION statements, such as elimination of references to CORE ALTERATIONS and the insertion of “recently irradiated fuel” when referring to the movement of irradiated fuel. The proposed changes do not impact TS requirements for systems needed to prevent or mitigate CORE ALTERATION events other than the FHA. They also do not change the requirements for systems needed for decay heat removal or requirements to maintain the specified water levels over irradiated fuel. Since the proposed revisions to the TS do not result in changes to the design-basis, the NRC staff concludes that these revisions are acceptable.

4.7 TS 3.7.9 Control Room Ventilation System (CRVS) Booster Fans

In addition to the TSTF-51 changes shown in the table above, the licensee proposes to add a Note to the Completion Time for Condition B and Condition C.

Condition B Completion Time additional note:

Note: An additional 96 hours is allowed when entering this condition for implementation of Control Room intake/booster fan modification.

Condition C Completion Time additional note:

Note: An additional 48 hours is allowed when entering this condition for implementation of Control Room intake/booster fan modification

The licensee states that the Notes will allow for a one-time additional completion time extension to implement the Control Room Intake/Booster Fan modification. This extension is acceptable based on the current knowledge and experience in control room habitability. The completion times specified in the proposed required actions recognize the low probability of an accident occurring during the time period when the boundary is degraded.

The NRC staff has reviewed the inclusion of these two notes and finds that they are acceptable for a one-time modification of the control room intake/booster fans.

In summary, the NRC staff has reviewed the licensee’s analyses and performed confirmatory assessments of the radiological consequence of the postulated LOCA and FHA. The doses calculated by the licensee are listed in Table 1. The doses calculated by the licensee and by the NRC staff are all within relevant dose criteria specified in 10 CFR 50.67 and the SRP. Therefore, the NRC staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable predicated on a determination that both dual intake channels for each individual control room would perform continuously and simultaneously to individually provide the air flow assumed in the licensee’s analysis and that air from at least one of the two intakes could be assumed to be uncontaminated for the duration of an accident. The NRC staff further concludes that the proposed TS changes are acceptable and do not pose any significant risk to the health and safety of the public and control room operators.

5.0 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.

6.0 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 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 proposed findings that the amendments involve NSHC, and there has been no public comment on such findings (67 FR 2922, 68 FR 59215, and 68 FR 68660). 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.

7.0 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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Table 1
Radiological Consequences Expressed as TEDE[1]
(rem)

Design Basis Accidents	EAB[2]	LPZ[3]	Control Room
LOCA	11.8	3.3	4.4
Dose criteria[4]	25	25	5.0
Fuel handing accident for single fuel Assembly event	1.2	0.1	2.2
Dose criteria	6.3[5]	6.3[5]	5.0[4]
Fuel cask handing accident for multiple fuel Assembly event	1.8	0.2	3.4
Dose criteria	6.3[5]	6.3[5]	5.0[4]

(1) Submittal dated November 21, 2002 (Attachments 3 and 4)
(2) Exclusion area boundary
(3) Low population zone
(4) 10 CFR 50.67
(5) SRP 15.0.1

Table 2
Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident

Parameter	Value
Reactor power	2568 MWt
Containment volume	1.78E+6 ft^3
Sprayed area	8.66E+5 ft^3
Unsprayed area	9.17E+5 ft^3
Containment leak rates	
0 to 24 hours	0.2% per day
24 to 720 hours	0.1% per day
Containment mixing rates	
Sprayed to unsprayed	30,567 cfm
Unsprayed to sprayed	30,567 cfm
Aerosol removal rates by containment spray (per hour)	
Time	Rates
0 to 96 seconds	0
96 seconds to 25 minutes	9.7
25 minutes to 3.5 hours	6.7
3.5 to 112.8 hours	0.67
Elemental iodine removal rates by spray (per hour)	
Time	Rates
0 to 96 seconds	0
96 seconds to 25 minutes	20
25 minutes to 1.8 hours	0
1.8 to 8 hours	0.06
8 to 13.8 hours	0.09
13.8 to 24 hours	0.13
24 to 96 hours	0.071
96 to 112.8 hours	0.002

Table 2
Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident
(Continued)

Containment sump volume	4.81E+4 ft^3
ECCS leak rates	
<u>Time</u>	<u>Rates</u>
0 to 25 minutes	0
25 minutes to 720 hours	50 gph
Iodine partition factor	10%
ECCS leak rates to BWST	
<u>Time</u>	<u>Rates</u>
0 to 25 minutes	0
25 minutes to 720 hours	5 gpm
Control room	
Volume	8.64E+4 ft^3
Unfiltered air inleakage rates	
0 to 30 minutes	1150 cfm
30 minutes to 720 hours	40 cfm
Filter efficiencies	
Aerosol	99%
Elemental iodine	99%
Organic iodine	95%

Table 3
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor power	2568 MWt
Radial peaking factor	
Single fuel assembly event	1.65
Multiple fuel assembly cask drop event	1.20
Fission product decay period	
Single fuel assembly event	72 hours
Multiple fuel assembly cask drop event	55 days to 1 year
Number of fuel assembly damaged	
Single fuel assembly event	One
Multiple fuel assembly cask drop event	518 to 1,024
Fuel pool water depth	21.34 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5%
Kr-85	10%
I-131	8%
Alkali metals	12%
Fuel pool decontamination factors	
Iodine	183
Noble gases	1
Duration of accident	2 hours

Table 4
Meteorological Data
used for
Loss-of-Coolant (LOCA) and Fuel Handling (FHA) Accidents

Offsite Relative Concentration (X/Q) Values
for LOCA and FHA

Time (hr)	Receptor	X/Q (sec/m^3)
0-2	EAB	2.2 E-4
0 to 8	LPZ	2.35 E-5
8 to 24	LPZ	4.70 E-6
24 to 96	LPZ	1.50 E-6
96 to 720	LPZ	3.30 E-7

Control Room Relative Concentration (X/Q) Values

Time (hr)	Accident	X/Q (sec/m^3)
0 to 2	LOCA (unit vent release)	4.79 E-4*
2 to 8	LOCA (unit vent release)	3.40 E-4*
8 to 24	LOCA (unit vent release)	1.40 E-4*
24 to 96	LOCA (unit vent release)	1.09 E-4*
96 to 720	LOCA (unit vent release)	8.86 E-5*
0 to 2	LOCA (Borated water storage tank (BWST))	2.13 E-4*
2 to 8	LOCA (BWST release)	1.61 E-4*
8 to 24	LOCA (BWST release)	6.66 E-5*
24 to 96	LOCA (BWST release)	5.19 E-5*
96 to 720	LOCA (BWST release)	4.06 E-5*
0 to 2	FHA	5.38 E-4**
2 to 8	FHA	3.74 E-4**
8 to 24	FHA	1.57 E-4**
24 to 96	FHA	1.24 E-4**
96 to 720	FHA	1.01 E-4**

* LOCA (unit vent release) and LOCA (BWST release) control room X/Q estimates are based upon an assumption that wind flow is in the direction of the intake with the poorer atmospheric dispersion conditions and that this intake is drawing in 55 percent of the air.

** FHA control room X/Q estimates are based upon an assumption that wind flow is in the direction of the intake with the poorer atmospheric dispersion conditions and that this intake is drawing in 60 percent of the air. Further, the intakes are assumed to be 10 feet higher than for the LOCA and BWST control room estimates. These assumptions were made as part of a “sensitivity” study the licensee performed. Duke has stated that the final FHA doses calculated after the intakes are installed will reflect the as-tested control room air inflow and are expected to be lower than the dose values reported in Table 1.

Table 5
Measured Unfiltered Air Inleakage Rates

1998 Test

Control Room	Values
Unit 1 and 2	80 +/- 55 cfm
Unit 3	73 +/- 25 cfm

2001 Test

Control Room	Values
Unit 1 and 2	0 +/- 18 cfm
Unit 3	0 +/- 13 cfm

Oconee Nuclear Station

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