December 22, 2006

Mr. G. R. Peterson
Vice President
McGuire Nuclear Station
Duke Power Company LLC
12700 Hagers Ferry Road
Huntersville, NC 28078

SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING IMPLEMENTATION OF ALTERNATIVE SOURCE TERM METHODOLOGY (TAC NOS. MC9746 AND MC9747)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 236 to Renewed Facility Operating License NPF-9 and Amendment No. 218 to Renewed Facility Operating License NPF-17 for the McGuire Nuclear Station (McGuire), Units 1 and 2 (McGuire 1 and 2). The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 20, 2005, supplemented by letters dated May 4 and August 31, 2006.

The amendments revise the McGuire 1 and 2 licensing basis to adopt a selective implementation of the alternative source term radiological analysis methodology in accordance with Part 50, Section 50.67 of Title 10 of the *Code of Federal Regulations*. The amendments also revise TS 3.9.4, Containment Penetrations.

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/

John Stang, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 236 to NPF-9
2. Amendment No. 218 to NPF-17
3. Safety Evaluation

cc w/encls: See next page

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DUKE POWER COMPANY LLC

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 236
Renewed License No. NPF-9

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. NPF-9, filed by the Duke Power Company LLC (licensee), dated December 20, 2005, as supplemented May 4 and August 31, 2006, 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 2.C.(2) of Renewed Facility Operating License No. NPF-9 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 236, are hereby incorporated into this 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-9
and the Technical Specifications

Date of Issuance: December 22, 2006

DUKE POWER COMPAPNY LLC

DOCKET NO. 50-370

MCGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 218
Renewed License No. NPF-17

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. NPF-17, filed by the Duke Power Company LLC (the licensee), dated December 20, 2005, as supplemented May 4 and August 31, 2006, 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 2.C.(2) of Renewed Facility Operating License No. NPF-17 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 218, are hereby incorporated into this 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-17
and the Technical Specifications

Date of Issuance: December 22, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 236

RENEWED FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

LICENSE AMENDMENT NO. 218

RENEWED FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License Pages	License Pages
NPF-9 page 3	NPF-9 page 3
NPF-17 page 3	NPF-17 page 3
TS Pages	TS Pages
3.9.4-1	3.9.4-1
B 3.9.4-1	B 3.9.4-1
B 3.9.4-2	B 3.9.4-2
B 3.9.4-3	B 3.9.4-3
B 3.9.4-4	B 3.9.4-4

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 236 TO RENEWED FACILITY OPERATING LICENSE NPF-9

AND

AMENDMENT NO. 218 TO RENEWED FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY LLC

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By application dated December 20, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML060250098), as supplemented by letters dated May 4, 2006 (ADAMS Accession No. ML061300600), and August 31, 2006 (ADAMS Accession No. ML062560094), Duke Power Company LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the McGuire Nuclear Station, Units 1 and 2 (McGuire 1 and 2). The supplements dated May 4 and August 31, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 24, 2006 (71 FR 50105).

The proposed changes would modify the current containment closure requirements for fuel movements involving irradiated fuel that has not been part of a critical reactor core within the previous 72 hours. This proposed amendment will provide the licensee more flexibility in scheduling outage tasks when moving fuel that has been afforded 72 hours of fission product decay time. The licensee also proposes to revise the accident source term for the design-basis fuel-handling accident based on a selective implementation of the alternative source term radiological analysis methodology. This amendment also proposes to revise TS 3.9.4 so that the surveillance and containment closure requirements will only apply to recently irradiated fuel, where recently irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours.

2.0 REGULATORY EVALUATION

This safety evaluation input addresses the impact of the proposed changes on previously analyzed design-basis accident (DBA) radiological consequences and the acceptability of the revised analysis results. The regulatory requirements for which the Nuclear Regulatory

Commission (NRC) staff based its acceptance are:

- Title 10 of the *Code of Federal Regulations* (10 CFR), Part 100, Section 100.11 (10 CFR 100.11), as supplemented by accident-specific criteria in Section 15 of the *Standard Review Plan* (SRP).
- 10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC-19).
- 10 CFR 50.36, "Technical specifications."
- 10 CFR 50.67, "Accident source term."
- SRP Section 6.4, "Control Room Habitability Systems."
- SRP Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms."
- Regulatory Guide (RG) 1.25 (Safety Guide 25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized-Water Reactors."
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants."
- Technical Specification Task Force (TSTF) Change Traveler TSTF-51A, Revision 2, approved by the NRC on October 13, 1999, which provides for the relaxation of some TS requirements during refueling after a sufficient decay period has occurred.

The NRC staff also considered relevant information in the McGuire 1 and 2, "Updated Final Safety Analysis Report [UFSAR]," and TSs.

3.0 TECHNICAL EVALUATION

This license amendment request (LAR) proposes to revise the McGuire 1 and 2 licensing bases by adopting the alternative source term (AST) radiological analysis methodology, as allowed by 10 CFR 50.67 for the fuel-handling accident (FHA). This adoption represents a selective implementation of the AST as described in RG 1.183. The amendment request proposes to revise TS 3.9.4, "Refueling Operations, Containment Penetrations," and its associated bases based on Technical Specifications Task Force (TSTF) Change Traveler TSTF-51A, Revision 2, "Revise Containment Requirements During Handling Irradiated Fuel and Core Alterations."

The licensee evaluated the FHA in containment, in the spent fuel pool (SFP), and a SFP weir gate drop accident using the AST methodology. The scenarios were modeled with a fuel decay time of 72 hours. The licensee also evaluated the SFP weir gate drop onto irradiated fuel with a minimum decay time of 17.5 days (420 hours), based on restrictions for movement of the weir gate. For each scenario, the licensee calculated the total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), low-population zone (LPZ), and in the control room without crediting containment closure. The results of the licensee's evaluations, as shown in Tables 1 and 2 (Attachment 1), are within the dose acceptance criteria of 10 CFR 50.67, as supplemented by the accident-specific criteria in SRP Section 15.0.1 and Regulatory Position 4.4 of RG 1.183.

3.1 FHA Analysis

The licensee submitted calculations for the evaluation of TEDE, using AST methodology, for the design-basis FHAs in containment and in the SFP and the weir gate drop accident. No credit was taken for containment integrity or for filtration by the containment purge exhaust system (CPES) for the FHA in containment. The licensee did not credit the fuel-handling ventilation exhaust system (FHVES) for scenarios in the SFP.

As stated in its application, the licensee requests approval of selective implementation of the AST for the FHA and requests approval for the amendments of TS 3.9.4 and Surveillance Requirement (SR) 3.9.4.1. The amendment of TS 3.9.4 would limit the requirement for containment isolation during movement of irradiated fuel assemblies to "recently" irradiated fuel assemblies. "Recently" irradiated fuel is defined as fuel that has occupied part of a critical reactor core within the previous 72 hours. The addition of the term "recently," as associated with handling irradiated fuel in all of the containment function TS requirements, is only applicable to licensees who have demonstrated by analysis that, after sufficient radioactive decay has occurred, the off-site doses resulting from FHA remain below the SRP 15.0.1 limits (well within 10 CFR 50.67).

The licensee also proposes to modify TS 3.9.4 by deleting "CORE ALTERATIONS" from the applicability and action statements. The FHA is the only event during CORE ALTERATIONS that is postulated to result in fuel damage and radiological release. The limiting condition for operation and required actions will remain applicable during activities which could result in a FHA with fuel damage and radiological releases.

The licensee has calculated radiation doses for the design-basis fuel-handling and weir gate drop accidents postulated to occur at McGuire 1 and 2. The AST methodology was used in this DBA. The radioactive source terms were developed and the analyses were conducted pursuant to RG 1.183.

The transport and release of radioactivity to the environment for the design-basis FHA and the weir gate drop accident were modeled incorporating the following assumptions:

- 72 hours of decay was credited in the single assembly FHA analyses, based upon the definition of "recently irradiated" fuel.
- 17.5 days of decay was credited in the multiple assembly weir gate drop accident analysis, based upon the minimum permitted time to move a weir gate.
- No filtration credit was taken for either the CPES or the FHVES.
- The retention of iodine in either the reactor cavity water or the SFP water was modeled. For both accident locations the effective decontamination factor (DF) was set to 200.

Additional assumptions are provided in Table 3 Attachment 2. New atmospheric dispersion factors (χ/Q values) for transport of radioactivity to the control room area ventilation system (CRAVS) outside air intakes were calculated. The new 0–2 hour χ/Q values were accepted by the staff for use in this license amendment application, but these 0–2 hour χ/Q values, along with the longer averaging time χ/Q values presented in the licensee's submittal, are not acceptable for use in any other DBA control room dose analyses (see Section 3.3).

Radiological consequences were calculated for the following scenarios:

1. FHA in containment with the equipment hatch open (releases through the open equipment hatch).

2. FHA in the containment with the personnel air locks open (releases from the unit vent stack).

3. FHA in the SFP (releases from the unit vent stack).

4. Weir gate drop in the SFP (releases from the unit vent stack).

The licensee calculated the TEDE for off-site locations and the control room. For the calculation of the control room TEDE, following a design-basis FHA or weir gate drop accident, manual operator action to start one CRAVS filter train at 30 minutes after the initiating event was credited.

For all scenarios, the licensee has shown that the off-site (EAB and LPZ) and control room operator dose consequences are within the dose acceptance criteria. The NRC staff confirmed the that the offsite and control room doses meet acceptance criteria through independent calculations.

3.2 <u>TS Changes</u>

The licensee proposed an amendment of the TS to limit fuel-handling restrictions to "recently" irradiated fuel. With the proposed change, containment closure will no longer be required when moving fuel assembles that have not occupied a critical reactor core within the previous 72 hours ("non-recently" irradiated fuel). It will also change the surveillance and testing requirements in TS 3.9.4, "Containment Penetrations." Specifically, the term "CORE ALTERATION" will be removed and "recently" will be added to designate irradiated fuel that would require containment isolation. The NRC staff reviewed the proposed TS change and found that it is consistent with the TSTF-51A Traveler program which permits some TS requirements to be inactive once fuel has decayed a sufficient period of time. This period of time is defined by the term "recently" as documented in the plant TS bases document. The staff finds that the required action to suspend core alterations is not needed since the dose results are bounded by the FHA. In accordance with TSTF-51A, reference to 10 CFR Part 100 will be replaced by reference to 10 CFR 50.67.

In a letter dated August 31, 2006, the licensee described the administrative controls they will establish to promptly close containment penetrations, if necessary, during refueling operations. The licensee has committed to the following:

1. Controls will be established to maintain the containment equipment hatch closed and held in place by a minimum of four bolts during movement of irradiated fuel assemblies within containment. This closure control will remain in place until a prompt closure method is developed.

2. Other penetrations open during movement of irradiated fuel assemblies within containment shall have the ability to be closed promptly within approximately 30 minutes.

3. The penetration closure method will be robust enough to limit dose and allow the containment purge system to function adequately.

4. The procedure changes necessary to implement the administrative controls will be in place prior to implementing an approved license amendment.

5. Personnel training for prompt closure will be conducted prior to opening a penetration during movement of irradiated fuel assemblies within containment.

These commitments are consistent with the guidance of NUMARC 91-06 (Nuclear Mangement and Resources Council (now NEI, Nuclear Energy Institute)), and therefore are acceptable to the staff.

3.3 <u>Meteorological Data</u>

The licensee calculated new χ/Q values for use in evaluating the radiological consequences on control room operators of an FHA in containment, an FHA in the SFP area, and a weir gate drop (WGD) in the SFP area. These control room χ/Q values represent a change from those currently presented in the McGuire 1 and 2 UFSAR. The licensee used existing UFSAR χ/Q values to perform off-site dose assessments for the EAB and LPZ.

3.3.1 <u>Control Room χ/Q Values</u>

3.3.1.1 <u>Meteorological Data</u>

The licensee generated the new control room χ/Q values for this amendment application using meteorological data collected on-site at McGuire 1 and 2 during the 1995–1999 period. The licensee provided these data in its application dated December 20, 2005. The data were provided in the form of hourly data files for input into the ARCON96 computer code and contained lower and upper wind data and atmospheric stability class data.

Section 2.3.3.5 of the McGuire 1 and 2 UFSAR states that a new meteorological tower with new instrumentation became operational in September 1998. As part of a quality review of the 1995–1999 hourly meteorological database, the NRC staff generated wind roses comparing wind direction frequency distributions between the 1995–1997 data and the 1999 data for both the lower and upper tower levels. These wind roses showed considerable discrepancies for the NE and S clockwise to WSW wind direction sectors for both tower levels (e.g., for both tower levels, S and SSW winds occurred more frequently during the earlier monitoring program as compared to the new monitoring program, whereas NE, SW, and WSW winds occurred more frequently during the new monitoring program as compared to the earlier monitoring program).

Similarly, stability class frequency distributions comparing the 1995–1997 data set and the 1999 data set showed more frequent extreme stability class A (unstable) and stability class G (stable) during the earlier monitoring program, as compared to the new monitoring program. Consequently, the NRC staff asked the licensee to explain what might have caused the discrepancies in wind direction and stability class frequency distributions between the 1995–1997 and the 1999 data sets and propose which of these two data sets is considered to be more representative of current site conditions.

In its response, dated May 4, 2006, the licensee explained that the pre-September 1998 data were collected from two separate towers, one instrumented at the 10-meter level and the second instrumented at the 40-meter level. The licensee stated that these earlier towers, both located on the west side of the plant, were subjected to mechanical turbulence from the wakes of nearby buildings and the greater frequency of stability classes A and G is attributable to local terrain and building effects on temperature. According to the licensee, the post-September 1998 data were collected on a different tower located on the north side of the plant. This new tower is instrumented at the 10-meter and 60-meter levels. The licensee stated that the newer tower has an open exposure in all directions and the increased vertical separation distance between the upper and lower level temperature sensors provides a better categorization of the temperature differences by which stability class can be determined. Consequently, the licensee considers the post-September 1998 data from the newer tower to be more representative of ambient conditions at the plant site. Nonetheless, the licensee used the complete 5-year (1995–1999) data set as input to the ARCON96 model. The licensee stated that this introduced conservatism into the calculation because the earlier towers had generally lower wind speeds and more hours of extreme stable (stability class G) conditions.

3.3.1.2 <u>Control Room Atmospheric Dispersion Analysis</u>

McGuire has a common control room facility for both units. The CRAVS consists of two pressurization trains, each of which has two outside air intakes. Each train's air intakes are co-located with the other train's intakes (e.g., separated by only a few feet). The two air inlet locations are on either side of the plant complex (i.e., one intake is adjacent to the WSW side of the Unit 1 containment and the second intake is adjacent to the ESE side of the Unit 2 containment) and are sufficiently separated to minimize the possibility of both inlet locations being exposed to a radioactive plume at the same time. Either train is capable of providing a positive pressure in the control room. Although McGuire 1 and 2 can be classified as a dual intake plant without manual or automatic intake selection controls, the licensee conservatively assumed only one train and one intake were operational throughout the duration of each of the accident scenarios (i.e., the dilution credit typically available for a dual intake plant was not used).

The licensee calculated control room χ/Q values for two release locations at each unit: the equipment hatch and vent stack. Control room χ/Q values from each of the two release pathways for each unit were calculated to each of the two CRAVS air inlet locations and the most conservative resulting χ/Q values from each release pathway were used in the control room dose analysis. Unfiltered inleakage was also assumed to occur throughout the duration of each event and was modeled using the air intake χ/Q values.

The licensee used guidance provided in RG 1.194 to generate the new control room atmospheric dispersion factors. The licensee calculated these new control room χ/Q values using the ARCON96 computer code (NUREG/CR-6331, Revision 1, “Atmospheric Relative Concentrations in Building Wakes”). RG 1.194 states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in DBA radiological analyses.

Because the heights of both release locations are less than 2½ times the height of adjacent buildings, both release locations were modeled using the ARCON96 ground-level release option, in accordance with RG 1.194. Additional details on the licensee’s assessments of control room post-accident atmospheric dispersion conditions for each release location are as follows:

a. <u>Equipment Hatch Releases</u>: The equipment hatch pathway was used by the licensee to model releases associated with an FHA in containment with the equipment hatch open. The licensee modeled the Unit 1 equipment hatch release pathway to the Unit 1 air inlet location and the Unit 2 equipment hatch release pathway to the Unit 2 air inlet location as vertical area sources and as point sources. The licensee also modeled the Unit 1 equipment hatch release pathway to the Unit 2 air inlet location and the Unit 2 equipment hatch release pathway to the Unit 1 air inlet location as point sources. The resulting ARCON96 analyses showed that the Unit 1 equipment hatch point source release to the Unit 1 air inlet location was the bounding source-receptor combination that resulted in the highest (i.e., most conservative) χ/Q values for all time periods.

b. <u>Vent Stack Releases</u>: Each McGuire unit has a vent stack. The vent stack pathway was used by the licensee to model releases associated with (1) FHA in containment with the personnel air locks open, (2) FHA in the SFP, and (3) WGD in the SFP. The licensee modeled the vent stack releases as point sources. The vent stack release heights are 40.2 meters above ground level (AGL), as compared to the normal air intake height of 1.5 meters AGL. The resulting ARCON96 analyses showed that the bounding vent stack release to air intake source-receptor combination was the Unit 2 release to the Unit 2 air inlet location for the 0–2 hour interval and the Unit 1 release to the Unit 1 air inlet location for the rest of the time intervals.

The licensee executed ARCON96 using the entire 1995–1999 on-site hourly database as input. The 10-meter wind data from both the earlier (pre-September 1998) and newer (post-September 1998) monitoring program were provided as input to ARCON96 as the lower level wind data and were identified as being recorded at the 10-meter level. The 40-meter wind data from the earlier monitoring program and the 60-meter wind data from the newer monitoring program were provided as input to ARCON96 as the upper-level wind data and both sets of data were identified as being recorded at the 60-meter level. Stability class data were derived based on the earlier monitoring program temperature difference measurements between the 40-meter and 10-meter levels for the pre-September 1998 period and the newer monitoring program temperature difference measurements between from the 60-meter and 10-meter levels for the post-September 1998 period.

Because of the discrepancies in wind direction and stability class frequency distributions between the pre-1998 data set and the post-1998 data set, the NRC staff asked the licensee to justify why different control room χ/Q values should not be generated using a pre-1998 data set and a post-1998 data set and the more conservative resulting χ/Q values used in the control room dose assessment. In its response, dated May 4, 2006, the licensee explained that it believed the use of the combined years of meteorological data was conservative.

The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual siting, building arrangements, release characterization, source-receptor configuration, meteorological regimes, or terrain conditions that preclude the use of the ARCON96 model for the McGuire site. The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and site practice. The NRC staff made an independent evaluation of the resulting bounding atmospheric dispersion estimates by running the ARCON96 computer code using the 1999 meteorological database and compared its bounding results to the licensee's control room χ/Q values, presented in Table 5.1 of the licensee's December 20, 2005, submittal. The NRC staff found that the staff's 0–2 hour χ/Q values for both release pathways were similar to the licensee's values, but the staff's longer averaging period χ/Q values for both release pathways were higher than the licensee's values. These results imply that the licensee's use of a combined set of meteorological data (i.e., pre-September 1998 data from the earlier monitoring system and post-September 1998 data from the newer monitoring system) was not conservative for the longer averaging periods.

The NRC staff accepts the use of the licensee's 0–2 hour control room χ/Q values for both the unit vent and equipment hatch release pathways for the DBAs analyzed in this amendment application because (1) the modeled accident scenarios (i.e., an FHA in containment and SFP area and a WGD in the SFP area) assumed a release duration of 2 hours, (2) the staff calculated 0–2 hour χ/Q values similar to those of the licensee, and (3) the highest calculated dose to the control room (e.g., 3.4 roentgen equivalent man (rem) TEDE associated with an FHA in containment with a release through the equipment hatch) was less than 70 percent of the dose limit of 5 rem TEDE. These 0–2 hour χ/Q values are presented in Table 4 (Attachment 3). However, 0–2 hour χ/Q values, along with the 0–8 hour, 8–24 hour, 1–4 day, and 4–30 day control room atmospheric dispersion factors presented in Table 5.1 of the licensee's December 20, 2005 submittal, are not acceptable for use in any other DBA control room dose analyses. These χ/Q values should be regenerated using a minimum of 3 years of data collected by the new meteorological monitoring program before being used in any other design-basis control room dose analysis.

3.3.2 <u>Offsite χ/Q Values</u>

The licensee evaluated off-site doses using the EAB and LPZ χ/Q values provided in Section 2.3.4 of the McGuire 1 and 2 UFSAR. These values are presented in Table 5 (Attachment 4). The NRC staff reviewed the licensee's use of existing McGuire 1 and 2 UFSAR EAB and LPZ χ/Q values and has found the existing values remain bounding for the FHA. On the basis of this review, the NRC staff concludes that the EAB and LPZ χ/Q values presented in Table 5 are acceptable for use in the design-basis offsite dose assessments performed in support of this amendment application.

4.0 <u>SUMMARY</u>

The NRC staff has reviewed the licensee's LAR. In this review, the NRC staff relied upon information placed on the docket by the licensee, and where deemed necessary, on staff confirmatory calculations.

The NRC staff accepts the use of the licensee's 0–2 hour control room χ/Q values for both the unit vent and equipment hatch release pathways for the design-basis accidents analyzed in this amendment application. However, the 0–2 hour χ/Q values, along with the 0–8 hour, 8–24 hour, 1–4 day, and 4–30 day control room atmospheric dispersion factors presented in Table 5.1 of the licensee's December 20, 2005, submittal, are not acceptable for use in any other DBAs control room dose analyses. These χ/Q values should be regenerated using a minimum of 3 years of data collected by the new meteorological monitoring program before being used in any other design-basis control room dose analysis.

The licensee's analysis demonstrated that the radiological consequence of an FHA would remain within applicable regulatory dose limits. The NRC staff compared the doses estimated by the licensee to the results estimated by the staff in its selective confirmatory calculations. The NRC staff finds, with reasonable assurance, that the licensee's dose estimates based on the amended TSs comply with the regulatory requirements in 10 CFR 50.67 and GDC-19.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on December 22, 2006 (71 FR 77071). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

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.

Attachments: 1. Tables 1 and 2,
Off-Site Dose Results for Fuel-Handling Accidents (FHA) and Weir Gate Drop Accidents Utilizing Alternate Source Term (AST), and Control Room Dose Results for FHA and Weir Gate Drop Accidents Utilizing AST
2. Table 3,
FHA Analysis Assumptions
3. Table 4,
McGuire 1 and 2 Control Room Atmospheric Dispersion Factors
4. Table 5,
McGuire 1 and 2 Off-site Atmospheric Dispersion Factors

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Date: December 22, 2006

TABLE 1
Off-Site Dose Results for Fuel-Handling Accidents (FHAs) and Weir Gate Drop Accidents Utilizing Alternate Source Term (AST)

Design-Basis Accident	EAB TEDE (REM)	LPZ TEDE (REM)	Dose Criteria (REM)

FHA in Containment	2.9	.26	6.3
FHA in the SFP	2.9	.26	6.3
Weir Gate Drop	5.4	.49	6.3

TABLE 2
Control Room Dose Results for Fuel-Handling Accidents and Weir Gate Drop Accidents Utilizing AST

Design-Basis Accident	Control Room TEDE (REM)	Dose Criteria (REM)
FHA in Containment	3.4	5
FHA in the SFP	1.4	5
Weir Gate Drop	2.8	5

TABLE 3
Fuel-handling Accident (FHA) Analysis Assumptions

Reactor power	3411 megawatts thermal
Radial peaking factor	1.65

TABLE 3 (Continued)

Fission product decay period, FHA in containment or in spent fuel pool (SFP)	72 hours
Fission product decay period, Weir gate drop	17.5 days
Number of fuel rods damaged, FHA in containment or in SFP	264
Number of fuel rods damaged, Weir gate drop	1848
Number of rods in core (equivalent full length)	50,952
Fuel gap fission product inventory I–131 Kr–85 Other iodines and noble gases Alkali metals	 8% 10% 5% 12%
Iodine species fractions Elemental Organic Particulates	 0.9985 0.0015 none
Water depth above damaged fuel	23 feet
Pool iodine effective decontamination factor	200
Chemical form of iodine above Elemental Organic	 57% 43%
Release modeling Immediate release from fuel through reactor cavity pool to building 100% release from building in 2 hours No credit for building hold-up or filtration prior to release	
Control room envelope volume	141,860 ft^3
Control room area ventilation system (CRAVS) Outside airflow rate - single train	2000 cubic feet per minute (cfm)
CRAVS System start delay time, minutes	30 minute
Unfiltered outside inleakage With the CRAVS OAPFT in standby With the CRAVS OAPFT in operation	 625 cfm 210 cfm
CRAVS filter efficiency Elemental Organic Particulate	 98.05% 98.05% 99%

Control room occupancy factors 0–24 hr 24–96 hr 96–720 hr	 1.0 0.6 0.4
Control room breathing rate	3.5×10^{-4} m^3/sec
Off-site breathing rate 0–8 hr 8–24 hr 24–720 hr	 3.5×10^{-4} m^3/sec 1.8×10^{-4} m^3/sec 2.3×10^{-4} m^3/sec

TABLE 4
McGuire 1 and 2
Control Room Atmospheric Dispersion Factors
(χ/Q values in sec/m^3)

Release Pathway	Accident	Time Interval
		0–2 hrs
Equipment Hatch	• Fuel-handling Accident (FHA) in containment with equipment hatch open	4.06×10^{-3}
Vent Stack	• FHA in containment with personnel air locks open • FHA in the spent fuel pool (SFP) • Weir gate drop (WGD) in the SFP	1.68×10^{-3}

These 0–2 hour control room χ/Q values, along with the 0–8 hour, 8–24 hour, 1–4 day, and 4–30 day control room atmospheric dispersion factors presented in Table 5.1 of the licensee's December 20, 2005, submittal, are not acceptable for use in any other design-basis accident control room dose analyses.

TABLE 5
McGuire 1 and 2
Off-site Atmospheric Dispersion Factors
(χ/Q values in sec/m^3)

Receptor	Time Interval	χ/Q Value
Exclusion Area Boundary	0–2 hrs	9.0×10^{-4}
Low Population Zone	0–8 hrs	8.0×10^{-5}
	8–24 hrs	5.2×10^{-6}
	24–96 hrs	1.7×10^{-6}
	96–720 hrs	3.7×10^{-7}

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