September 30, 2005

Mr. D. M. Jamil
Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB7014 AND MB7015)

Dear Mr. Jamil:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 227 to Renewed Facility Operating License NPF-35 and Amendment No. 222 to Renewed Facility Operating License NPF-52 for Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 25, 2002, as supplemented by letters dated November 13 and December 16, 2003, September 22, 2004, April 6, June 14, July 8, August 17, and September 8 and September 19, 2005.

The amendments include a full-scope implementation of an alternative source term for evaluating the consequences of design basis accidents at Catawba Nuclear Station, Units 1 and 2. The amendments also revise the TSs for the Ventilation Filter Testing Program, Annulus Ventilation System, Auxiliary Building Filtered Ventilation Exhaust System, Fuel Handling Ventilation Exhaust System, Control Room Area Ventilation System, and 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/

Sean E. Peters, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 227 to NPF-35
2. Amendment No. 222 to NPF-52
3. Safety Evaluation

cc w/encls: See next page

September 30, 2005

Mr. D. M. Jamil
Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB7014 AND MB7015)

Dear Mr. Jamil:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 227 to Renewed Facility Operating License NPF-35 and Amendment No. 222 to Renewed Facility Operating License NPF-52 for Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 25, 2002, as supplemented by letters dated November 13 and December 16, 2003, September 22, 2004, April 6, June 14, July 8, August 17, and September 8 and September 19, 2005.

The amendments include a full-scope implementation of an alternative source term for evaluating the consequences of design basis accidents at Catawba Nuclear Station, Units 1 and 2. The amendments also revise the TSs for the Ventilation Filter Testing Program, Annulus Ventilation System, Auxiliary Building Filtered Ventilation Exhaust System, Fuel Handling Ventilation Exhaust System, Control Room Area Ventilation System, and 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/
Sean E. Peters, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 227 to NPF-35
2. Amendment No. 222 to NPF-52
3. Safety Evaluation

cc w/encls: See next page
DISTRIBUTION: See next page

Package Number: ML052730317
Amendment Number: ML052730312
Tech Spec Number: ML052770150 *No Significant Changes to SE NRR-058

OFFICE	PDII-1/PM	PDII-1/LA	SPSB/SC (A)	EMCB/SC*	IROB/SC	OGC	PDII-1/SC
NAME	SPeters	EMarinos for CHawes	MKotzales	LLund	TBoyce	AHodgdon	EMarinos
DATE	09/30/05	9/30/05	09/27/05	07/01/05	09/30/05	09/30/05	9/30/05

OFFICIAL RECORD COPY

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB7014 AND MB7015)

Date: September 30, 2005

DISTRIBUTION:
PUBLIC
PDII-1 R/F
RidsNrrDlpmLpdii (EHackett)
RidsNrrDlpmLpdii1 (EMarinos)
RidsNrrPMSPeters (Hard Copy)
RidsNrrLACHawes (Hard Copy)
RidsOgcRp
RidsAcrsAcnwMailCenter
GHill (4 Hard Copies)
TBoyce
RidsRgn2MailCenter
RidsNrrDlpmDpr
MHart
LBrown
MKotzalas
KParczewski
EForrest

DUKE ENERGY CORPORATION

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 227
Renewed License No. NPF-35

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

A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-35 filed by the Duke Energy Corporation, acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc. (licensees), dated November 25, 2002, as supplemented by letters dated November 13 and December 16, 2003, September 22, 2004, April 6, June 14, July 8, August 17, and September 8 and September 19, 2005, 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-35 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 227, which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation 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, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 30, 2005

DUKE ENERGY CORPORATION

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 222
Renewed License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Corporation, acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency (licensees), dated November 25, 2002, as supplemented by letters dated November 13 and December 16, 2003, September 22, 2004, April 6, June 14, July 8, August 17, and September 8 and September 19, 2005, 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-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 222, which are attached hereto, are hereby incorporated into this license. Duke Energy Corporation 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, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 30, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 227

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND LICENSE AMENDMENT NO. 222

RENEWED FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

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
3.6.10-2	3.6.10-2
3.6.16-2	3.6.16-2
3.9.3-1	3.9.3-1
5.5-10	5.5-10
5.5-11	5.5-11
B3.6.10-1	B3.6.10-1
B3.6.10-2	B3.6.10-2
B3.6.10-3	B3.6.10-3
B3.6.10-4	B3.6.10-4
B3.6.10-5	B3.6.10-5
B3.6.10-6	B3.6.10-6
B3.6.16-2	B3.6-16-2
B3.6.16-3	B3.6.16-3
B3.6.16-4	B3.6.16-4
B3.7.10-1	B3.7.10-1
B3.7.10-2	B3.7.10-2
B3.7.10-3	B3.7.10-3
B3.7.10-4	B3.7.10-4
B3.7.10-5	B3.7.10-5
B3.7.10-6	B3.7.10-6
B3.7.10-7	B3.7.10-7
B3.7.12-1	B3.7.12-1
B3.7.12-2	B3.7.12-2
B3.7.12-3	B3.7.12-3
B3.7.12-4	B3.7.12-4
B3.7.12-5	B3.7.12-5
B3.7.12-6	B3.7.12-6
B3.7.12-7	B3.7.12-7
B3.7.13-1	B3.7.13-1
B3.7.13-2	B3.7.13-2
B3.7.13-3	B3.7.13-3
B3.7.13-4	B3.7.13-4
B3.7.13-5	B3.7.13-5
B3.9.3-3	B3.9.3-3
B3.9.3-4	B3.9.3-4
B3.9.3-5	B3.9.3-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 227 TO RENEWED FACILITY OPERATING

LICENSE NPF-35 AND

AMENDMENT NO. 222 TO RENEWED FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated November 25, 2002 (Ref. 1), as supplemented by letters dated November 13 and December 16, 2003, September 22, 2004, April 6, June 14, July 8, August 17, and September 8 and September 19, 2005 (Refs. 2 through 10), Duke Energy Corporation, et al. (Duke, the licensee), submitted a request for changes to the Catawba Nuclear Station (Catawba), Units 1 and 2, Technical Specifications (TS). The proposed changes included a full-scope implementation of an alternative source term (AST) for evaluating the consequences of design basis accidents at Catawba and would revise the TSs for several ventilation systems. Duke stated that the proposed changes would:

C Revise the Annulus Ventilation System (AVS) annulus pressure surveillance requirement in (SR) 3.6.16.2 and relocate it to SR 3.6.10.6 to enhance the ability of the licensee to determine that the reactor building annulus outside air inleakage is within the maximum assumed design value used in the dose analyses.

C Revise TS 5.5.11, "Ventilation Filter Testing Program (VFTP)," to make changes to the Catawba, Unit 2 AVS in-place penetration and bypass leakage criteria.

C Revise TS 5.5.11 for the Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) to request appropriate TS 5.5.11 limits in order to ensure that the ABFVES will continue to meet its design basis functions. Also it requested changes to the Catawba, Unit 2 ABFVES in-place penetration and bypass criteria.

C Revise TS 5.5.11 to make changes to the Catawba, Unit 2 Fuel Handling Ventilation Exhaust System (FHVES) in-place penetration and bypass leakage criteria.

C Revise TS 3.9.3, “Containment Penetrations,” and its Bases to make editorial changes.

C Make conforming changes to TS Bases 3.6.10, “Annulus Ventilation System (AVS).”

C Make conforming changes to TS Bases 3.6.16, “Reactor Building.”

- Make changes to TS Bases 3.7.10, "Control Room Area Ventilation System (CRAVS)," to describe the CRAVS operation and testing and also make editorial changes.

- Make changes to TS Bases 3.7.12, "Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)," to describe the ABFVES operation and testing.

- Make changes to Bases 3.7.13, "Fuel Handling Ventilation Exhaust System (FHVES)" to describe the FHVES operation and testing.

2.0 REGULATORY EVALUATION

The accident source term is the radioactive inventory that could be released to the environment following an accidental release of fission products from the damaged core into the containment. Under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.67, "Accident Source Term," most licensees can replace the traditional accident source term used in the design basis accident with an alternate radiological source term. Implementation of this AST involves re-analyzing design basis accidents following the procedure described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Reactors."

Duke submitted this amendment request to implement the AST methodology for evaluating radiological consequences of design basis accidents. The regulatory requirements for which the Nuclear Regulatory Commission (NRC) staff based its acceptance of the analyses are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183, and 10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC-19), "Control Room," as supplemented by Section 6.4 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP)." Except where the licensee proposed a suitable alternative, the NRC staff utilized the regulatory guidance provided in SRP Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms," in performing this review. The NRC staff also considered relevant information in the Catawba updated final safety analysis report and TSs.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of design basis accidents, performed by Duke in support of its proposed license amendments. Information regarding these analyses was provided in Appendix A to Attachment 3 of the submittal (Ref. 1) and in the supplementary letters (Refs. 2 through 10). The NRC staff reviewed the assumptions, inputs, and methods used by Duke to assess the impact of the proposed changes to the Catawba, Units 1 and 2 TS and licensing basis. The NRC staff also performed independent calculations to confirm the conservatism of the licensee's analyses. However, the findings of this Safety Evaluation (SE) input are based on the descriptions of the licensee's analyses and other supporting information provided by Duke. The NRC staff only relied upon docketed information to make its safety finding.

3.1 Loss-of-Coolant Accident Radiological Consequences Analysis

The licensee performed an analysis of the radiological consequences of the design basis loss-of-coolant accident (LOCA) using an AST. To show compliance with 10 CFR 50.67, Duke calculated the maximum total effective dose equivalent (TEDE) at the exclusion area boundary (EAB) for any 2 hour period, the TEDE at the boundary of the low population zone (LPZ) over the duration of the accident, and the TEDE in the control room over the duration of the accident. The NRC staff reviewed the description of the analysis as submitted in Appendix A to Attachment 3 of the November 25, 2002 (Ref 1.), letter, with further clarifications by subsequent letters, and finds that the licensee generally followed the guidance in RG 1.183 regarding AST calculation methodologies. A discussion of the NRC staff's review of the licensee's analysis, with emphasis on exceptions to the guidance, follows.

3.1.1 LOCA Source Term

The radiological consequences analysis of the design basis LOCA assumes full core melting with release of the radioactive material to the reactor coolant system and then to the containment over a total period of 1.8 hours. Release of radioactive material to the environment is assumed to occur through leakage from the containment, leakage of containment sump water from the emergency core cooling system (ECCS) components outside containment after recirculation begins, and backleakage of containment sump water into the refueling water storage tank (RWST).

The licensee's analysis source term assumes full power operation, including calorimetric uncertainty, to give 102 percent of rated power. Duke also assumed limiting values of fuel enrichment and the limiting burnup for each radioisotope in the core to give the core isotopic inventory. The licensee used the Oak Ridge National Laboratory developed SCALE computer code (NUREG/CR-0200) to calculate the core isotopic inventory. The SCALE code includes the ORIGEN-ARP isotope generation and depletion code, which is listed in RG 1.183 as an acceptable computer code to use for generating core isotopic inventories.

By letter dated March 3, 2005 (Ref. 11), Amendment No. 220 to Renewed Facility Operating License NPF-35 and Amendment No. 215 to Renewed Facility Operating License NPF-52, the Catawba, Units 1 and 2 TS were revised to allow the use of four mixed oxide fuel (MOX) lead test assemblies (LTAs) in one of the two Catawba units. The LOCA radiological consequences analysis for this amendment request proposing TS changes and implementation of an AST also takes into account the impact of loading four MOX LTAs into the Catawba Unit 1 or Unit 2 core, as described in the revised analyses submitted in Refs. 5 through 10.

The licensee's assumptions with regard to the source term release fractions, release timing, and radionuclide composition follow the guidance in Regulatory Position 3 of RG 1.183. The RG 1.183 iodine chemical forms of 95 percent particulate as cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodine compounds can be assumed if the containment sump pH can be shown to be at values of 7 or higher to prevent iodine re-evolution. Duke provided a calculation showing the sump pH is expected to remain above 7 for the duration of the accident

According to NUREG-1465," Accident Source Terms for Light-Water Nuclear Power Plants," iodine released from the damaged core to the containment after a LOCA is composed of

95 percent CsI which is a highly ionized salt, soluble in water. Iodine in this form does not present any radiological problems since it stays dissolved in the sump water and does not enter the containment atmosphere. However, in a radiation field existing in the containment, some of this iodine could be transformed from the ionic (I^-) to the elemental form (I_2) which is scarcely soluble in water and can be, therefore, released to the containment atmosphere. Conversion of iodine to the elemental form depends on several parameters, of which pH is one of the more important. Maintaining pH basic in the sump water (pH≥7) will ensure that this conversion will be minimized. The licensee used the method described in NUREG/CR-5950, “Iodine Evolution and pH Control” (Ref. 12) for calculating generation of this elemental iodine. Its calculations have indicated that at the higher sump water pH, fewer iodides are converted into elemental form and at pH 7 or higher elemental iodine generated from this source becomes insignificant relative to the elemental iodine released directly to the containment from the damaged core. At Catawba, the sump water pH is controlled by the presence of sodium tetraborate released from the melting ice in the ice condenser. Sodium tetraborate is a salt derived from a strong base and a weak acid; it acts, therefore, as a buffer and helps to stabilize the sump water pH. After a LOCA several acids are either generated or are added to the containment. Relative amounts of these acids and that of sodium tetraborate determine the value of pH reached by the containment sump water.

3.1.2 Acids in Containment

The following acids will accumulate in the containment sump after a LOCA:

3.1.2.1 Boric acid

After a LOCA, boric acid from the reactor coolant system, cold leg accumulators and borated water storage tank is discharged into the sump. The licensee assumed that in all these systems, the concentration of boron is 3075 ppm. It is a conservative assumption, because the concentration of boron in the reactor coolant varies during a fuel cycle and is much lower toward the end of cycle.

3.1.2.2 Hydrochloric acid (HCl)

Hydrochloric acid is generated by decomposition of a cable insulation made from chlorinated polymers. The licensee used a generation rate of 4.6×10^{-4} moles of HCl per pound of insulation per Mrad which is consistent with the value in NUREG-5950. The resulting concentration of hydrochloric acid in the containment water was determined to increase with time reaching the concentration of a little below 1 ppm.

3.1.2.3 Nitric acid (HNO_3)

Following a LOCA, nitric acid is formed in the containment by irradiation of water and air. The amount of nitric acid produced is proportional to the time-integrated dose rate for gamma and beta radiation. Based on the information provided in NUREG-5950, the licensee calculated a generation rate of 7.3×10^{-6} moles of HNO_3 liter per Mrad. The concentration of nitric acid in the containment was determined to increase with time, reaching the value of a little over 3 ppm.

3.1.3 Calculation of Containment Pool pH

Duke has calculated sump water pH using an EXCEL spreadsheet with the Visual Basic Program PHSC. The EXCEL spreadsheet was used to calculate the time dependent concentration of the chemicals dissolved in the sump water and the PHSC program was used to determine the resulting time dependent pH. The inventories of chemicals and water in the containment sump were calculated by solving separate time dependent mass balance equations. These chemicals included boron in the form of boric acid, sodium in the form of sodium tetraborate, lithium in the form of lithium hydroxide added to the sump water for chemistry control, chlorides in the form of hydrochloric acid, and nitrates in the form of nitric acid. The PHSC program used the mathematical model based on correlations and data for solutions of boric acid, sodium hydroxide, and other acids and bases from the Electric Power Research Institute reports NP-5561-CCML and TR-105714. The licensee-calculated values of pH varied with time. Its lowest value was 7.3.

As recommended by NUREG-5950, Appendix C, Duke determined all of the calculated pH values for the temperature of 77 °F. However, in calculating conversion of ionic to molecular iodine, the licensee used a pH at the containment sump water temperature that was about one pH unit lower. This introduced significant conservatism in the licensee's calculation. Using this pH, Duke calculated iodine conversion from ionic to elemental form. The calculated value was so low that its concentration in the containment atmosphere was negligible compared to the concentration of molecular iodine directly released to the containment atmosphere from the damaged core. Because of its conservatism and its results, the NRC staff finds the licensee's methodology for calculating sump water pH to be acceptable.

3.1.4 Calculation of pH in RWST

In addition to the leakage of radioactive iodine in molecular form directly from the containment atmosphere to the outside, Duke identified a possible leakage path through the Engineering Safety Features (ESF). The containment sump water can leak either to the RWST or to the Auxiliary Building. Water can leak to the RWST, mostly through the valves that isolate the ESF recirculating system from the RWST. Since, after an accident, this water will contain radioactive iodine and since the RWST is vented to the atmosphere, this back leakage will constitute a path for the radioactive iodine to leak to the outside and contribute to the external dose rates. In its submittal Duke analyzed this source of iodine release for a LOCA. Although in the post-LOCA conditions the sump water pH is maintained at the value of equal to or higher than 7, when it mixes with the water remaining in the RWST after the injection phase, its pH could drop to the value significantly below 7. Low pH favors conversion of the dissolved iodine from a soluble ionic form to the scarcely soluble elemental form. Some of this iodine will be, therefore, released to the RWST air space from which it could leak to the atmosphere.

The licensee assumed that the sump water leaks into the RWST at a rate of 20 gpm. As more of this water leaks into the RWST, both its pH and the concentration of iodine increase. Initially sump water pH is 4.30 and the iodine concentration is 2.36×10^{-9} mol/liter. At the end of the 30-day post-LOCA period, pH rises to 7.01 and the iodine concentration rises to 3.20^{-5}mol/liter. In this condition, the fraction of iodine converted to elemental form is very low - somewhere between 4.4×10^{-5} and 5.0×10^{-6} and the amount of elemental iodine released from the RWST is insignificant.

3.1.5 Effect of MOX fuel on iodine release

The licensee performed its AST re-analysis for the post-LOCA conditions with the core containing four MOX fuel assemblies. The results of this re-analysis, performed with the methods of calculation for a conventional fuel, have indicated that the presence of MOX fuel produced only an increase of 3 percent or less in the amount of iodine in the sump. Such a small increase in iodine concentration will not have a meaningful effect on the conversion of iodine to the elemental form.

3.1.6 Summary of Iodine Release

The NRC staff reviewed the licensee's assumptions, methodology, and conclusions regarding the pH of sump water and the corresponding fraction of the iodine dissolved in sump water converted into the elemental form. The calculations were made for the 30-day period following a LOCA. From the results of these calculations the NRC staff concluded that although the value of pH varied with time, it never dropped below 7.3. Maintaining pH above 7 resulted in negligible fraction of the dissolved iodine converted into elemental form and low release of radioactive iodine to the environment. The NRC staff also concluded that the amount of iodine that would be released from the RWST as a result of backleakage would be negligible because of the low concentration of iodine when the tank pH is below 7.

3.1.7 LOCA Containment Release

Each Catawba unit has an ice condenser containment. This containment is divided into a lower compartment, an upper compartment and the ice condenser. The reactor and reactor coolant system (RCS) are located in the lower compartment. The only connections between the lower and upper compartments are the containment air return system (ARS) fans, which blow air from the upper to the lower containment, and the ice condenser, through which the LOCA blowdown passes from the lower to the upper containment. However, since the gap release from the fuel is not assumed to occur until 30 seconds after the initiation of the event, the blowdown phase is not assumed to include the LOCA fission product source term. Therefore, the licensee did not take any credit for airflow from the lower compartment to the upper compartment until the ARS fans activate at 600 seconds. Also, Duke assumed that the fission product source term is released to and mixes homogeneously in the lower compartment volume only. The licensee's modeling of the LOCA release into containment and subsequent containment mixing is in accordance with the guidance in RG 1.183.

Duke took no credit for the scrubbing of radioactive materials in the ice condenser. The licensee also took no credit for the removal of iodine in the containment through natural processes. It did, however, take credit in the upper containment compartment for removal of fission products other than organic iodine and noble gases by the containment spray system (CSS). The methodologies used by Duke to calculate the time constants for elemental iodine removal and particulate removal by sprays follows the guidance in RG 1.183 and more specific guidance in SRP 6.5.2, Rev. 2, "Containment Spray as a Fission Product Cleanup System." Duke did not assume credit for fission product removal by sprays until the start of the ARS fans, which is consistent with the assumption that the fission products are initially released in the lower compartment which has no spray. The licensee took credit for elemental iodine removal until a decontamination factor (DF) of 200 was reached. The licensee also assumed that the particulate spray removal was reduced to 10 percent of the calculated value for the duration of

the spray, considering that a DF of 50 would be reached very soon after the initiation of the ARS fans. The licensee's modeling of fission product removal in containment follows the guidance in RG 1.183, and is acceptable.

The secondary containment, or annulus is provided with the AVS to filter primary containment leakage before release to the environment. The AVS is activated by a safety injection (SI) signal, and draws the annulus to a negative pressure. Duke analyzed the post-LOCA conditions in the annulus and the AVS response for the three following DBA LOCA scenarios:

1) DBA LOCA with minimum safeguards - only one of the two trains responds as designed.

2) No AVS failures, with failure of a CSS or residual heat removal system heat exchanger.

3) Failure of one AVS pressure transmitter - one train operates in exhaust mode until operators secure it 2.5 hours after the initiating event.

The licensee calculated and reported doses for each of the above failure scenarios for both the containment leakage and ECCS leakage pathways. Based on its analysis, Duke assumed that the AVS brings the annulus to the required negative pressure and begins filtration within 41.1 seconds for the DBA LOCA with minimum safeguards, and within 30.5 seconds for the other LOCA scenarios with two AVS fans in operation. Until the time the AVS is assumed operational, all of the primary containment leakage of 0.3 volume percent per day is assumed to be released directly to the environment. After that time, 93 percent of the primary containment leakage is assumed to be mixed in 50 percent of the annulus volume, then filtered for release by the AVS. These assumptions on the annulus mixing and percentage filtered by the AVS are consistent with the current licensing basis.

3.1.8 LOCA ECCS Leakage

Duke also calculated the dose due to leakage from ECCS components outside containment. The analysis not only considered release from ECCS components inside the auxiliary building, but also considered the potential release through backleakage to the RWST. The duration of the ECCS leakage for both cases is assumed to start when ECCS recirculation initiates and continues at a constant rate until the end of the accident period at 30 days. The ECCS is assumed to leak 0.5 gallons per minute (gpm) into rooms where the ABFVES is initially aligned, and 0.5 gpm into rooms where the ABFVES is manually aligned after 3 days. Therefore, only half of the ECCS leakage in the auxiliary building is filtered before release for the first 3 days, but all is filtered thereafter. The leakage into the RWST is assumed to be 20 gpm. The source term for the ECCS leakage is the radioactivity in the containment sump water and is assumed to consist of only the iodine isotopes and their precursors. The iodine species in the ECCS leakage release are assumed to be 97 percent elemental iodine and 3 percent organic iodine compounds. These assumptions are in accordance with guidance in RG 1.183.

Duke used a methodology to calculate Catawba site-specific iodine release fractions (partitioning) from the ECCS leakage. It made separate calculations for the releases from ECCS components in the auxiliary building and the releases from the RWST. RG 1.183,

Appendix A, Section 5.5 states that for leakage with a temperature less than 212 °F, "...the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates." Duke based the calculation of the iodine partitioning on the methodology in NUREG/CR-5950, "Iodine Evolution and pH Control." Using this methodology, the licensee calculated the formation of volatile elemental iodine based on the concentrations of iodine and iodide ions in the leakage. The total amount of iodine in the containment sump water, including stable I-127 and long-lived I-129, was used in the pH calculations.

The NRC staff evaluated the licensee's pH calculation, as discussed above. The licensee calculated and submitted time dependent water temperatures, iodine concentrations in the liquid and vapor, and the water pH for both the ECCS fluid and RWST. Using these calculated values, Duke determined iodine release rates from the RWST atmosphere to the environment and from the ECCS leakage areas to the environment. These releases were assumed to be filtered as discussed above. The displacement of air from the RWST with the backleakage and diurnal expansion of the airspace in the RWST were both accounted for in the licensee's calculations. For the ECCS leakage into the auxiliary building, the licensee also accounted for the ventilation system rates in the auxiliary building ECCS rooms.

The NRC staff has some concerns with assuming iodine partitioning of less than 10 percent. The NRC staff structured regulatory positions in RG 1.183, Appendix A, Section 5 to be deterministic and conservative in order to compensate for the lack of research into iodine release, partitioning and speciation from systems outside the containment, and the uncertainties of applying laboratory data to the post-accident environment of the plant. By letter dated
July 8, 2005 (Ref. 7), Duke addressed the NRC staff's specific questions on uncertainty and conservatism of the licensee's iodine partitioning calculation. Duke provided information to show that conservative assumptions were made in its calculation of the iodine partitioning wherever possible. The evaluation of the licensee's responses to the concerns follows.

Duke bases the percent release of iodine from the ECCS and RWST on NUREG/CR-5950, "Iodine Evolution and pH Control." NUREG/CR-5950 provides several fits to experimental data from controlled experiments. The release rates in NUREG/CR-5950 are for very specific laboratory conditions that do not appear to match those for the LOCA accident condition. For example, NUREG/CR-5950 does not appear to address the impact of impurities present due to core damage and other chemicals present. The data fit also contains very large errors when compared with the experimental data. Duke provided information to show that the values used from NUREG/CR-5950 are applicable and conservative for postulated ECCS leakage at Catawba. The licensee addressed the impact of impurities in the RCS and RWST fluids on the pH and iodine partitioning, differences between the very specific laboratory conditions and actual plant conditions, and the uncertainty within the curve fits of the data in NUREG/CR-5950.

In particular, the NUREG/CR-5950 methodology is based on reactions of I_2, I^-, water and hydrogen peroxide. The mathematical model for formation of I_2 makes use of equilibrium and reaction rate constants that are referenced at 25 °C (77 °F). NUREG/CR-5950 states that the model based on these constants overpredicts the formation of I_2 at temperatures above 30 °C (86 °F). The ECCS and RWST liquids would be at a greater temperature for the duration of the

event and the licensee did not modify the constants for a greater temperature. Therefore, the model calculating the elemental iodine release is conservative, based on this temperature effect. Duke pointed out information in NUREG/CR-5950 that can help in estimating the conservatism due to the use of the model for temperatures over 30 °C. However, the NRC staff notes that the information in the document does not appear to account for the uncertainty in the experimental data cited. The NRC staff thinks that there is some conservatism in the model with regard to the equilibrium and reaction rate constants, but considers the degree of conservatism to be unclear.

Duke stated that the analysis maximized the calculated iodine partitioning by assuming that no iodine is in the airspace above the ECCS leakage in the auxiliary building, and assuming the maximum iodine concentration in the sump water for each time interval. The assumption for the iodine in the airspace above the RWST was not as conservative, but the NRC staff thinks it is reasonable.

Additional margin was added in the licensee’s use of the calculated iodine partitioning in the dose analysis. Duke set a lower bound of 1 percent for the iodine partitioning for the ECCS leakage in the auxiliary building. The licensee stated that by assuming the partitioning does not go lower than 1 percent, the dose analysis gives a release margin ranging from 1.3 to 6.7, compared to the release fractions calculated by the NUREG/CR-5950 method. The dose calculations did not take credit for plate-out or dilution of iodine in the auxiliary building. Duke made conservatively higher leakage rate assumptions, as well. The NRC staff agrees that these assumptions are conservative and would lead to calculating a higher dose. Therefore, the NRC staff finds that Duke has appropriately addressed its concerns with regard to the uncertainty in the calculation of an ECCS leakage iodine partitioning factor less than 10 percent.

The NRC staff also asked the licensee whether it had considered contaminants in the area of the leakage and the effect the contaminants would have on the amount of organic iodine released or conversion of elemental iodine to organic forms. The licensee stated that procedural controls prohibit the storage of flammable and combustible materials (including organic compounds) in the vicinity of safety related systems, structures and components, including all ECCS components in the auxiliary building. Therefore, the amount of organic compounds in the area around the ECCS leakage is insignificant and would not impact the percentage of organic iodine released.

With regard to the calculation of iodine partitioning in the RWST, Duke provided information to show that flashing of the fluid does not happen along the pipes or in the RWST for the duration of the event. Duke further provided information on its modeling of mixing and stratification in the RWST. The ECCS fluid is assumed to mix homogeneously with the RWST fluid for each calculated time interval. Any ECCS backleakage would enter the RWST through the RWST outlet to the ECCS and the CSS. These systems connect to the RWST near its base, and any stratification would cause a decrease with height of the iodine concentration. The analysis assumption of homogeneous mixing in the RWST increases the iodine release rate over what would be calculated for a stratified RWST. However, the licensee stated that ECCS backleakage might also enter the RWST from lines that are near the top of the RWST, although the post-accident configuration of the plant makes leakage through these lines very unlikely. Duke identified this backleakage as a potential non-conservatism, and the NRC staff finds that neglecting this potential non-conservatism is reasonable based on the plant

conditions. Additionally, Duke stated that there are no significant amounts of organic compounds in the RWST or the area around the RWST that may impact the percentage of organic iodine released. The RWST is fabricated with all exposed internal surfaces constructed of uncoated stainless steel components. The NRC staff finds that Duke has appropriately addressed its concerns with regard to the fluid in the RWST.

Duke also identified and addressed two areas for potential non-conservatism in the modeling of iodine partitioning from the ECCS and RWST. The licensee did not model spraying nor streaming to a floor drain of the ECCS leakage in the auxiliary building. In the modeling of the RWST backleakage, the assumption that the iodine and other solutes in the ECCS mix instantly and homogeneously may not be conservative, depending on where the backleakage enters the RWST.

The licensee did not consider the effects of the leaked fluid evaporating to dryness. Duke judged that evaporation to dryness would not be likely to occur at Catawba, based on the continuous leakage, pH of the fluid and test data. The NRC staff finds the licensee's conclusion to be fairly reasonable but, because of the uncertainty as to whether the condition may actually exist, it considers that not addressing evaporation to dryness is potentially an additional non-conservatism in Duke's calculation.

Although the NRC staff does not know the true amount of iodine partitioning from the leaked ECCS or RWST, the analysis performed by Duke appears reasonable overall. The NRC staff's determination considers that the licensee does not propose to remove any safety grade filtration systems used to mitigate the accident releases from the ECCS areas in the auxiliary building. The determination also considers the NRC staff's sensitivity analysis that indicates that if the RG 1.183 assumption of 10 percent partitioning is used, the total design basis accident (DBA) doses are expected to remain acceptable. The NRC staff finds that, considering the conservative nature of LOCA analysis as a whole and considering the licensee's conservative assumptions in the use of the NUREG/CR-5950 methodology, the licensee's analysis including iodine partitioning less than 10 percent for ECCS leakage and RWST backleakage is reasonable.

3.1.9 LOCA Control Room Modeling

The CRAVS automatically starts following an SI signal or a blackout signal on the CRAVS 4160 volt switchgear. To model the control room response to the LOCA with loss of offsite power, Duke assumed that the CRAVS initiates at the time of the core damage, and begins pressurization and filtration of the control room. Duke also assumes that one train of the CRAVS is immediately secured by the operators; therefore, only one train of CRAVS is credited in the dose analysis. The limiting value for asymmetry in the airflow into the two CRAVS outside air intakes was determined to be 60/40, and the control room atmospheric dispersion factors (χ/Q) were adjusted to account for the imbalance. The NRC staff's review of the licensee's control room and offsite χ/Qs is discussed below in Section 3.3 of this SE. The licensee assumed the lower limits for the CRAVS filter train recirculation flow rates. It also calculated doses for both the lower limit and upper limit for the CRAVS intake flow rate. The licensee assumed that the unfiltered inleakage into the control room was 100 cubic feet per minute (cfm) for the duration of the event, which bounds the Catawba tracer gas test results from years 2001 and 2002.

3.1.10 LOCA Results

The NRC staff reviewed the information provided in the licensee's AST submittal and supplements and the Catawba current licensing basis. The NRC staff also performed an independent calculation that confirmed the licensee's dose results. Duke's LOCA dose analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 2. The licensee's limiting calculated LOCA dose results are given in Table 1. The licensee's calculated doses from any system failure scenario are within the SRP 15.0.1 radiological dose acceptance criteria for a LOCA. These TEDE criteria are 25 rem at the EAB for the worst two hours, 25 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident.

3.2 Other DBAs

The licensee proposed changes to the assumed accident operation of the AVS and the ABFVES. The LOCA is the bounding DBA for the mitigative function of these systems. The design basis LOCA assumes a release from the reactor core into the containment, and from there into the annulus where it may be filtered by the AVS before release to the environment. The design basis LOCA also assumes release into the containment sump water, which then leaks into the auxiliary building from emergency core cooling system components following recirculation. The ABFVES filters some of the activity from the ECCS leakage before release to the environment.

The other design basis accidents that assume these two release pathways are the control rod ejection accident (REA) and the locked rotor accident (LRA). Because the REA and LRA both assume a percentage of the fuel in the core has failed fuel rod cladding due to departure from nucleate boiling (DNB), Duke performed revised dose analyses using an AST for these accidents for a baseline case assuming that the core only includes low-enriched uranium (LEU) fuel, and a supplemental case assuming that the four MOX LTAs are included in the damaged fuel population. Separate analyses were performed for Catawba, Units 1 and 2.

3.2.1 Rod Ejection Accident

Catawba's design basis REA is currently defined to be an REA with offsite power available and a minimum safeguards failure. Duke considered two separate scenarios for fission product release to the environment: (1) primary-to-secondary leakage and secondary steaming from the steam generators (SGs) until cold shutdown, or (2) a break in the containment boundary and containment leakage for 30 days. Duke additionally considered fission product release from the ECCS components outside of containment and through backleakage to the RWST, which is more conservative than the RG 1.183 guidance. For each of the three evaluated release pathways, Duke assumed that the entire REA activity source term was released only through that pathway. The licensee added the higher of the dose from the REA containment leakage pathway or from the ECCS leakage including backleakage to the RWST to the dose for the SG secondary steaming pathway. While the actual dose from an REA would be a composite of the pathways, an acceptable dose from each pathway, modeled as if it were the only pathway, would show that the composite dose would also be acceptable.

For the REA source term, the licensee's dose analysis assumed 50 percent of the core fuel rods had experienced DNB and immediately released the fission products within the gap to the

RCS coolant. The limiting fuel assembly isotopic inventory was used in calculating the fuel rod gap source term and included a fuel radial peaking factor of 1.65. Duke assumed that the fraction of
gap fission product activity released for the LEU fuel was 0.1 for iodine and noble gases, and 0.12 for the alkali metals. For the MOX LTAs, Duke multiplied the above gap fractions by a scaling factor of 1.5, which is consistent with what was previously approved in the MOX LTA amendments for the halogen and noble gas isotopes. No fuel melting was assumed.

For the SG secondary coolant steaming scenario, the radioactive materials in the reactor coolant are transported through primary-to-secondary leakage into each of the SGs. The licensee's methods for simulating the transport of activity in the SGs and the release of activity from the SGs are the same as those used in the current licensing basis. These assumptions were previously found acceptable by the NRC in the review of the MOX LTA license amendments. Duke assumes that the primary-to-secondary leakage is initially the current TS allowable leakage of 150 gallons per day (gpd) per SG with subsequent leakage varying over time, based on the accident transient analysis and adjusted so that the leakage values would correspond to standard conditions. The licensee performed its transient analysis using analysis methodologies previously approved by the NRC staff. The activity was assumed to be released to the environment through secondary coolant steaming through the atmospheric dump valves and main steam safety valves (MSSVs) until the primary coolant system was cooled down to 211 °F.

For the containment leakage release scenario, Duke assumed that all the available activity from the damaged fuel and the initial primary coolant activity was immediately released into the containment through the ejected rod housing. The containment leakage release pathway model assumed that the containment was leaking at the design value of 0.5 percent volume per day for the first 24 hours, then leaked at half that value until the end of the accident, which was assumed to be 30 days. No credit was taken for any mechanism by which iodine would be removed from the containment atmosphere and moved to the containment sump. Duke modeled natural deposition of aerosols other than those including iodine. The 10^{th} percentile data and correlations from NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," (also called the Powers natural deposition model) were used by Duke to calculate the time constants and decontamination factors for natural deposition of aerosols including rubidium, cesium and bromine isotopes. Using the Powers aerosol natural deposition 10^{th} percentile model is in accordance with guidance in RG 1.183.

For the REA, cold leg recirculation is assumed to begin 2 hours after the initiating event. The licensee modeled the release of iodine isotopes through ECCS leakage both into the auxiliary building and to the RWST beginning at 2 hours. This scenario assumes that the entire fuel gap and initial primary coolant activity source term is immediately released to the containment sump. The rate of ECCS leakage to rooms to which the ABFVES is initially aligned was set to 0.5 gpm for the duration of the accident. An additional rate of ECCS leakage to rooms not initially aligned to the ABFVES was also set to 0.5 gpm. For this ECCS leakage pathway, the operators would manually align the rooms to the ABFVES at 3 days, which is the same assumption made in the LOCA dose analysis. Duke assumed that the 10 percent of the iodine included in the ECCS leakage would be released to the environment. Duke calculated backleakage to the RWST and iodine partitioning using the same methods, assumptions and data as for the LOCA, except using the REA source term and assuming 10 gpm of

backleakage. The NRC staff finds that Duke's modeling of the ECCS leakage, including RWST backleakage, follows RG 1.183 guidance and is acceptable.

The licensee assumed the control room was isolated and the CRAVS was placed into operation automatically by plant response to the accident. Filtered pressurization of the control room occurs following an SI signal or a blackout signal. As a conservative measure, Duke assumed that control room operators immediately secured one of the redundant CRAVS filter trains. Duke also assumed that 60 percent of the control room air intake flow rate was through one of the redundant CRAVS emergency outside air intakes, while the remaining 40 percent entered through the other emergency intake. This asymmetric air flow gives an increase in the calculated radiation doses to the control room operators. The licensee further assumed the lower design values for the CRAVS intake flow rate and recirculation flow rate in the REA dose analysis. The licensee set the unfiltered inleakage into the control room assumption to 100 cfm for the duration of the event, which bounds the tracer gas test results from years 2001 and 2002.

The NRC staff reviewed the information provided in the licensee's AST submittal and supplements and the Catawba current licensing basis. It also performed an independent calculation that confirmed the licensee's dose results. Duke's REA dose analysis used assumptions and inputs that follow the guidance in RG 1.183. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 2. The licensee's calculated REA dose results are given in Table 1. Duke's calculated doses from either release scenario are within the SRP 15.0.1 radiological dose acceptance criteria for a rod ejection accident. These TEDE criteria are 6.3 rem at the EAB for the worst 2 hours, 6.3 rem at the LPZ for the duration of the accident and 5 rem in the control room for the duration of the accident.

3.2.2 Locked Rotor Accident

Duke calculated the dose consequences of a postulated design basis LRA. The LRA begins with either the instantaneous seizure of the rotor or a break in the shaft of a reactor coolant pump, resulting in a sudden decrease in reactor coolant flow through the core. The licensee has also assumed a coincident loss of offsite power and a minimum safeguards failure resulting in radionuclide release through steaming through the steam generator power operated relief valves (PORVs) and safety valves. The release is assumed to continue until the reactor coolant has reached 211 °F.

The licensee assumed 10 percent of the core experiences fuel failure as a result of the fuel experiencing DNB. The limiting fuel assembly isotopic inventory was used in calculating the fuel rod gap source term, and included a bounding fuel radial peaking factor of 1.65. The licensee used the fuel rod gap inventory assumptions given in RG 1.183, Table 3. For the MOX LTAs, Duke multiplied the fuel gap fractions by a scaling factor of 1.5, which is consistent with what was previously approved in the MOX LTA amendments for the halogen and noble gas isotopes. No fuel melting was assumed.

The licensee assumed that the initial activity in the primary reactor coolant is at the equilibrium TS limit of 1 μCi/gm Dose Equivalent I-131 and is increased by an accident-initiated iodine spike at 335 times the equilibrium iodine appearance rate. Duke also assumed that the initial activity in the secondary coolant is at the equilibrium TS limit of 0.1 μCi/gm Dose Equivalent I-

131. The coolant activity assumptions are a conservative change with respect to the current licensing basis analysis, and are conservative compared to the guidance in RG 1.183, Appendix G, for the LRA. The licensee chose the accident-initiated iodine spike multiplier of 335 based on guidance in RG 1.183, Appendix E, for the SG tube rupture considering the similarity in the primary coolant pressure reduction for each accident type. The NRC staff finds this assumption to be reasonable. Duke's modeling of the transport of the radionuclides from the primary coolant to the secondary coolant system, and into the environment is consistent with guidance in Appendix E of RG 1.183.

Duke calculated the dose consequences of an LRA using the offsite and bounding control room atmospheric dispersion factors for the release from the PORVs to the normal control room air intake. The atmospheric dispersion factors are discussed below in Section 3.3. The licensee assumed the control room was isolated and the CRAVS was automatically placed into operation by plant response to the accident. Filtered pressurization of the control room occurs following an SI signal or a blackout signal. As a conservative measure, Duke assumed that control room operators immediately secured one of the redundant CRAVS filter trains. The licensee assumed that 60 percent of the control room air intake flow rate is through one of the redundant CRAVS emergency outside air intakes, while the remaining 40 percent enters through the other emergency intake. This asymmetric air flow gives an increase in the calculated radiation doses to the control room operators. The licensee also assumed the lower design values for the CRAVS intake flow rate and recirculation flow rate in the LRA dose analysis. The licensee assumed that the unfiltered inleakage into the control room was 100 cfm for the duration of the event, which bounds the tracer gas test results from years 2001 and 2002.

The NRC staff reviewed the information provided in the licensee's submittal and supplements and the Catawba current licensing basis. It also performed independent calculations that confirmed the licensee's dose results. Duke's LRA analysis used assumptions and inputs that follow the guidance in RG 1.183 and are acceptable. Assumptions used by the licensee and evaluated by the NRC staff are listed in Table 2. The licensee's calculated LRA dose results are given in Table 1. Duke's calculated dose results are within the dose limits in 10 CFR 50.67 and GDC 19. The offsite dose results are also within the dose acceptance criteria in SRP 15.0.1 for the LRA, which is a small fraction of 10 CFR 50.67 limits (i.e., 2.5 rem TEDE).

3.2.3 MOX Lead Test Assemblies Non-LOCA Gap Activity

The SE for Amendment No. 220 to NPF-35 and Amendment No. 215 to NPF-52 (Ref. 11) included by reference a portion of the SE, which had been issued separately by letter dated April 5, 2004 (Ref. 12). Section 3.2.4 of the April 5, 2004, SE evaluated the fuel rod gap fission product source term assumption for the MOX LTAs and also provided some limitations. The NRC staff stated that Duke's assumption of a 50 percent increase in the gap fractions in Table 3 of RG 1.183 was acceptable for the purposes of the MOX LTA amendment request only, and that it should not be construed as a precedent for another licensing action at Catawba or any other reactor site. The NRC staff further stated that the gap fraction analyses are strongly dependent on the projected power history, and that the gap fraction evaluation should be re-visited if the actual power history is to deviate significantly from the projected power history.

The assumption of a 50 percent increase in the gap fractions in Table 3 of RG 1.183 is not affected by the subject AST license amendment request for the halogen and noble gas isotopes. The April 5, 2004, MOX LTA SE analysis showed that the adjusted gap fractions

were bounding and acceptable for I-131 and other halogens and noble gases other than Kr-85. The SE also stated that although Duke's adjustment of the gap fraction was not bounding for Kr-85, considering the relatively low significance of Kr-85 as a dose contributor and the relatively small Kr-85 inventory in the core, its impact on the fuel handling accident and waste gas decay tank rupture postulated doses would be negligible. The NRC staff similarly considers that the underestimation, by approximately 10 percent, of the Kr-85 dose contribution for the four MOX LTAs would have a small effect on the calculated doses for the REA and LRA.

In the April 5, 2004, SE, the NRC staff stated that if Duke should implement an AST at Catawba in the future, the gap fraction associated with Cs-137 would need to be explicitly addressed in the DBA analyses. Duke accounted for the increase in the Cs-137 gap activity by multiplying the RG 1.183 Table 3 non-LOCA gap activity fraction by a scaling factor of 1.5, such as was used for iodine and nobles gases in the MOX LTA amendments. In the April 5, 2004, SE, the NRC staff estimated a peak value of 21.6 percent for the alkali metals non-LOCA gap fraction. By multiplying the RG 1.183 gap fractions by a scaling factor of 1.5, the alkali metals gap fraction was increased to a value of 18 percent. This value is not bounded by the NRC staff's analysis. However, the NRC staff notes conservatism in the licensee's REA and LRA dose analyses that counteracts the potential nonconservatism in the assumption for the Cs-137 gap fraction.

The MOX LTAs are only a small portion of the overall fuel in the core, approximately 2 percent. For the REA, the licensee assumed that 50 percent of the core is damaged. The four MOX LTAs are assumed to be within the damaged fraction. There are a total of 193 assemblies in the core. Four MOX assemblies out of 97 contribute approximately 4 percent of the total REA release. For the LRA, the licensee assumed that 10 percent of the core is damaged. Four MOX assemblies contribute to approximately 21 percent of the total LRA release.

For the REA and LRA, the licensee took the limiting burnup dependent radial peaking factor for all failed fuel assemblies. For the REA, because the licensee assumed that 50 percent of the core is melted, not all the failed fuel rods would experience the maximum radial peaking and using the maximum radial peaking factor to increase the gap activity is conservative. The alkali metal gap activity fraction would be equivalent to 29.7 percent, when multiplied by the radial peaking factor, which would bound the value from the NRC staff's evaluation for the MOX LTA amendment.

For both the REA and LRA, the licensee's set of dose analysis assumptions are conservative, and the dose results would not be expected to increase significantly if the Cs-137 gap fraction were to be increased to 21.6 percent for the four MOX LTAs. Duke stated that the MOX LTA gap fraction was made specifically for the insertion of the MOX LTAs and that the licensee will submit an analysis of MOX fuel rod gap fractions for non-LOCA accidents in support of a future license amendment request pertaining to batch loading of MOX fuel assemblies. Based on the above, the NRC staff extends its acceptance of Duke's assumption of a 50 percent increase in the gap fractions in Table 3 of RG 1.183 for the present amendment request only. This acceptance should not be construed as a precedent for another licensing action at Catawba or any other reactor site. The conclusions regarding the fuel gap fractions in the MOX LTA amendment SE dated April 5, 2004, remain applicable.

3.3 Atmospheric Dispersion Estimates

Duke generated new atmospheric dispersion factors (χ/Q values) for use in evaluating the radiological consequences of the LOCA, REA, and LRA in the Catawba control room. The licensee made calculations for a number of postulated release scenarios from Catawba, Units 1 and 2 to the Unit 1 and Unit 2 control room outside air intakes, then selected the limiting χ/Q values for use in the dose assessments. Duke used existing χ/Q values to perform offsite dose assessments for the Catawba EAB and LPZ.

3.3.1 Meteorological Data

The licensee generated new control room χ/Q values for this license amendment request using meteorological data collected at the Catawba site from 1994-1999. Duke previously provided the 1994-1999 meteorological database to support Catawba, Units 1 and 2 License Amendment Nos. 198 and 191 issued April 23, 2002 (Ref. 13).

During the period from January 1994 through mid-1996, wind speed and wind direction data were measured at the 10-meter and 40-meter levels and stability class was calculated using the temperature difference between the 40-meter and 10-meter levels. Following the installation of a new meteorological measurement tower in mid-1996, the wind speed and wind direction data were measured at the 10-meter and 60-meter levels and the stability class was calculated using the temperature difference between the 60-meter and 10-meter levels. The NRC staff reviewed and discussed the 1994-1999 data in the SE associated with Amendment Nos. 198 and 191.

When reviewing potential source-receptor pairs associated with control room χ/Q values for the current license amendment request, the NRC staff noticed that winds during the 1994–1999 time period were reported to occur infrequently from the north northeast, east, and east southeast directions. The NRC staff requested that Duke confirm that it was not likely that measurement of winds from the easterly directions was affected by factors such as vegetation, plant structures, tower structure, instrument location on the tower, or local topography near the meteorological measurement tower. In its September 8, 2005, response, Duke compared measurements for the 1994-1999 time period with data from other time periods and concluded that the measurements were generally similar during the periods of comparison and around the site region. In addition, Duke also provided a quantitative discussion of the meteorological measurement tower and the types and locations of vegetation, structures, and local terrain with regard to height and possible wake cavity effects.

Based on the meteorological measurements program and meteorological database review described above and in the SE associated with Catawba, Units 1 and 2 License Amendment Nos. 198 and 191, the NRC staff has concluded that the 1994–1999 site meteorological database provides an acceptable basis for making atmospheric dispersion estimates for use in the design basis accident assessments performed in support of this license amendment request.

3.3.2 Control Room Atmospheric Dispersion Factors

Duke calculated new χ/Q values to evaluate the impact of Catawba, Unit 1 and Unit 2 releases on the Unit 1 and Unit 2 control room outside air intakes for the LOCA, REA and LRA. The licensee used guidance provided in RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” to generate

these new control room atmospheric dispersion factors. Duke calculated the new control room χ/Q values using the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes") and the 1994–1999 hourly data from the site meteorological tower. RG 1.194 states that ARCON96 is an acceptable methodology for assessing control room χ/Q values for use in design basis accident radiological analyses.

The licensee generated χ/Q values for numerous postulated release scenarios. Source and receptor pairs were determined by reviewing plant drawings and documentation and performing plant walkdowns. All releases were considered to be ground level. Distances from the release location to the receptor were either the shortest straight line distance or arc length around circular buildings. If flow was assumed to go over a building, the straight line distance between the source and receptor was used as if the building was not there. The licensee considered the potential impact of loss of offsite power (LOOP) or other single failures in its assessment. Specific areas of note follow:

- Duke provided input information and χ/Q values for a number of postulated release and intake scenarios, not all of which were used in the Catawba LOCA, REA and LRA dose assessments. Some of the information and χ/Q values were provided to show which scenarios were most limiting.

- Catawba is a two unit site with a common control room and dual control room outside air intakes. Thus, each release scenario has four possible sets of χ/Q values, one set for a release from each of the two units to each of the two control room intakes. Duke calculated χ/Q values for each release-receptor combination to identify the more favorable air intake (i.e., the air intake with the lower χ/Q value) and the less favorable air intake (i.e., the air intake with the higher χ/Q value).

- In the early 1990's, Duke became aware that under some conditions a LOOP could result in closure of the isolation valves. Subsequently, in response to Licensee Event Report (LER) 413/91-08 dated March 20, 1991, Duke removed all control room isolation valve controls except those for manual operation. Thus, no valid single failure can cause an intake to close inadvertently. However, as a condition of resolution of the LER, Duke made a commitment to assume closure of one of the intakes for the first 10 hours of those DBA scenarios that could experience a single failure. The 10-hour time interval was selected assuming that personnel would recognize an intake was closed within the first 8 hours and that the intake could be opened within the following 2 hours. This administrative control was to account for possible closure for maintenance activities such as actuator and chlorine detector preventive maintenance, actuator refurbishment every few years, and to conduct differential pressure testing.

  In its September 8, 2005, response to Question 2, which requested an estimate of the amount of time per year an intake would be closed for maintenance activity, Duke stated that each control room air intake is closed a maximum of approximately 2 days per year. Thus, both intakes are normally open. In dose assessments assuming that one intake is initially closed, Duke used χ/Q values associated with the less favorable air intake for the first 10 hours and weighted χ/Q values for the remaining period of the control room dose assessment. Since the ARCON96 computer code does not directly calculate χ/Q values for 8 to 10-hour and 10 to 24-hour time periods, Duke generated χ/Q values for

these intervals by arithmetically averaging χ/Q values from other time periods. When both intakes were postulated to be open, a 60/40 split was assumed with the less favorable intake drawing in 60 percent of the total inflow. Weighted χ/Q values were generated following the methodology applicable to dual control room intakes described in Section C.3.3.2 of RG 1.194. The NRC staff requested that Duke identify the χ/Q values that were directly input into the Catawba LOCA, REA and LRA dose assessments. In response to this request, Duke provided a summary table of χ/Q values in its September 8, 2005, response to Question 1 and supplementary response dated September 19, 2005. These χ/Q values are listed in Table 3.

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 Catawba site. The NRC staff qualitatively reviewed the inputs to the ARCON96 calculations and found them generally consistent with site configuration drawings and NRC staff practice. The NRC staff performed a random check of the resulting atmospheric dispersion estimates by running the ARCON96 computer code and obtained similar results. On the basis of this review, the NRC staff has concluded that the χ/Q values for Catawba LOCA, LRA and REA releases to the control room as presented in Table 1 are acceptable for use in the DBA control room dose assessments performed in support of this license amendment request.

3.3.3 EAB and LPZ Atmospheric Dispersion Factors

Duke used existing χ/Q values that were accepted by the NRC staff in a previous licensing proceeding to evaluate the impact of the Catawba, Units 1 and 2 postulated releases to the EAB and LPZ. These values are presented in Table 4. Based on the review described in the SE associated with Catawba License Amendment Nos. 220 and 215, dated March 3, 2005, and a review of the licensee’s use of these χ/Q values, the NRC staff has concluded that the EAB and LPZ χ/Q values, as presented in Table 4, are acceptable for use in the DBA assessments performed in support of this license amendment request.

3.4 <u>Technical Evaluation of Proposed TS Changes for Catawba</u>

The licensee proposes the following changes to the TS:

<u>Revise SR 3.6.16.2 and relocate to SR 3.6.10.6</u>

Currently, SR 3.6.16.2 states:

Verify each AVS train produces a pressure equal to or more negative than -0.5 inch water gauge in the annulus within 1 minute after a start signal.	18 months on a STAGGERED TEST BASIS

Proposed SR 3.6.10.6 states:

Verify each AVS train produces a pressure equal to or more negative than -0.88 inch water gauge when corrected to elevation 564 feet.	18 months

Proposed SR 3.6.16.2 states:

Verify that during the annulus vacuum decay test, the vacuum decay time is $\geq$ 87 seconds	18 months

Technical Evaluation:

In reviewing the change, the NRC staff considered that the AVS drawdown time is an input to the dose analysis in the sense that the AVS filtration is not credited for the first 23 seconds. After 23 seconds, the containment leakage release is considered to be filtered by the AVS. In the November 25, 2003 (Ref. 1), submittal, the licensee stated that 23 seconds is the upper limit for the AVS system to begin operation at full speed. The analysis indicates that the AVS draws down the annulus to the required negative pressure in 41.4 seconds after initiation of a design basis LOCA with only one fan in operation. Through the analysis, the licensee showed that the key variable for confirmation of drawdown of the annulus is the leakage rate from the reactor building. The vacuum decay test proposed as the new SR 3.6.16.2 assures that the leakage from the reactor building to the annulus is sufficiently low to validate the assumptions in the drawdown analysis. The proposed SR 3.6.10.6 provides assurance that each AVS fan is capable of drawing down the annulus to the required negative pressure. The NRC staff finds, based on the information presented by the licensee, that the proposed changes are acceptable and provide assurance of the proper operation of the AVS system in its accident mitigation function.

<u>Revise TS 5.5.11, "Ventilation Test Program," for the AVS, ABFVES, and FHVES</u>

The licensee proposes to change TS 5.5.11a and TS 5.5.11b for Unit 2 to revise the criterion for the high-efficiency particulate air (HEPA) and carbon filter penetration and system bypass leakage from "<0.05%" to "<1%." It is anticipated that the requested change would allow the licensee to reduce the frequency of replacing the carbon adsorber and to operate the system with an unfiltered bypass leakage of 1.0 percent.

The carbon filter penetration and system bypass leakage test is an in-place leakage test as described in RG 1.52, Revision 2, C.5 done in accordance with ANSI N510-1975. It is a system leak tightness test that indicates the amount of effluent that bypasses the HEPA or carbon adsorber beds. All of the effluent that bypasses the filter beds is released to the environment as an unfiltered release.

The licensee noted an increase in unfiltered bypass after it began continuous operation of the ABFVES in 1998 to address a single failure vulnerability of a non-safety ventilation damper. To prevent having to replace the filter bed media on a more frequent basis, Duke requests that this SR acceptance criterion for unfiltered bypass leakage be changed to 1.0 percent.

The licensee provided in-place test data in a September 22, 2004, letter to support its proposed change to 1 percent. The data encompasses the period from 1998 through 2004 and covers all trains of the AVS, FHVES and ABFVES. The licensee stated that the ABFVES failed to meet

the surveillance test requirement after periods of operation. Duke contends that moisture in the bed interferes with the adsorption and de-adsorption of the R-11 refrigerant used in the test.

Duke provided a dose assessment, which demonstrates that with a 1 percent bypass criterion both 10 CFR Part 100 and GDC 19 doses are met. The NRC staff recognizes that with the implementation of the alternative source term, dose limits could be met with larger levels of unfiltered bypass, provided these levels were explicitly accounted for in the dose analyses and appropriate surveillance requirements and criteria were established to demonstrate that the filtration system would function as assumed in the dose analysis.

Since the licensee has explicitly accounted for this increase in penetration and bypass leakage in its dose analysis to demonstrate that its licensing basis dose limits are met, the NRC staff finds the proposed change acceptable.

However, the NRC staff has determined that Duke is not performing the in-place bypass leakage surveillance in the mode that is credited in the design basis accident analysis. The design basis accident analysis credits one single train of filter operation for 30 days after a LOCA. The licensee has stated that the ESF operation of the system is at a flow of 6500 cfm, which continues for 3 days. For the remaining 27 days, the system operates as a single train at 37,000 cfm. The system surveillance test is performed with two trains operating at a flow of 30,000 cfm per train. The bypass leakage determined at a filter flow rate of 30,000 cfm easily bounds the bypass leakage that would be expected at a flow of 6500 cfm. However, the bypass leakage test results at 30,000 cfm may not be conservative when the flow is increased to 37,000 cfm. The test conducted by Catawba does not assure that the test flow rate bounds the assumed post-LOCA operating flow of 37,000 cfm. Therefore, the licensee needs to evaluate this condition within the VFTP. After its evaluation, if the licensee finds that the TS surveillance value is non-conservative, it must follow the process in NRC Administrative Letter 98-10, which calls for licensees to institute administrative controls for the TS surveillance and to submit a license amendment request to correct the requirement.

<u>Paragraphs 5.5.11a, 5.5.11b, and 5.5.11c.</u>

The licensee proposes to change the word “charcoal" to "carbon" in TS 5.5.11b, TS 5.5.11c, and TS 5.5.11d. The term "carbon" is the correct terminology for discussing nuclear grade activated carbon filter media (reference the previous identical change). This is an editorial change that the NRC staff finds acceptable.

<u>Paragraphs 5.5.11a, 5.5.11b, and 5.5.11d</u>

The licensee proposed to add "(2 fans)" after "Aux. Bldg. Filtered Exhaust" and change the flowrate entry for the system from "30,000 cfm to "60,000 cfm." These changes clarify that two filtered exhaust fans are operating in parallel with a total flowrate of 60,000 cfm for each unit.

The licensee has withdrawn this TS change request as documented in its letter dated September 8, 2005.

<u>Paragraph 5.5.11c</u>

The licensee proposed to add "(Note 1)" after "Aux. Bldg. Filtered Exhaust" and add the following material following the list of ventilation systems:

> "Note 1: The Auxiliary Building Filtered Exhaust System carbon adsorber samples shall be tested at a face velocity of 48 ft/min instead of the 40 ft/min specified in ASTM D3803-1989. 48 ft/min is the nominal limiting velocity the carbon adsorber may be exposed to under post accident conditions as a result of certain postulated failures. The results from this test shall then be corrected to a 2.27 inch bed in accordance with the guidance provided in ASTM D3803-1989 prior to comparing them to the Technical Specification criteria. 2.27 inches is the actual bed depth for the filter unit."

Technical Evaluation:

The licensee presented information in the September 22, 2004, letter that when the ABFVES is operating in a single train accident mitigation mode the flow rate increases to 37,000 cfm, which is significantly above the rated flow (30,000 cfm) of the system. Changing the velocity to 48 ft/min, which corresponds to 37,000 cfm, assures that the laboratory test bounds the actual conditions for which the filter system is credited in the design basis analysis. Therefore, the NRC staff finds that this proposed change is acceptable.

Revise TS 5.5.11 table column heading:

The licensee proposes to change the column heading in TS 5.5.11a and TS 5.5.11b from "Penetrations" to "Penetrations and System Bypass." This is an editorial change that the NRC staff finds acceptable.

Revise TS 3.9.3, LCO c.2:

The licensee proposes to change "charcoal" to "carbon." This is an editorial change that the NRC staff finds acceptable.

3.5 Summary

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by Duke to assess the radiological impacts of revised TS 3.6.10, TS 3.6.16, TS 3.9.3, and TS 5.5.11, for Catawba, Units 1 and 2. Changes have also been proposed for the TS Bases for the preceding systems and also for the CRAVS, the ABFVES, and the FHVES. The NRC staff finds that Duke used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2.0 above. The NRC staff compared the doses estimated by the licensee to the applicable criteria identified in Section 2.0. The NRC staff finds, with reasonable assurance, that the licensee's estimates of the EAB, LPZ, and control room doses will continue to comply with these criteria. Therefore, the proposed changes to TS 3.6.10, TS 3.6.16, TS 3.9.3, and TS 5.5.11 are acceptable with regard to the radiological consequences of postulated design basis accidents.

This licensing action is considered to be a full implementation of the AST. With this approval, the previous accident source term in the Catawba, Units 1 and 2, design bases is superceded by the AST proposed by Duke. The previous offsite and control room accident dose criteria

expressed in terms of whole body, thyroid and skin doses are superceded by the TEDE criteria of 10 CFR 50.67 or fractions thereof, as defined in RG 1.183. All future radiological analyses performed to demonstrate compliance with regulatory requirements shall address all characteristics of the AST and the TEDE criteria as described in the Catawba, Units 1 and 2 design bases.

The NRC staff finds that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the manner proposed by the licensee in its changes to SR 3.6.16.2, SR 3.6.10.6, TS 3.9.3, and TS 5.5.11.

4.0 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.

5.0 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 a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (68 FR 18272 and 70 FR 29789). 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.

6.0 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.

7.0 REFERENCES

1. Letter from Gary R. Peterson, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," November 25, 2002 (ADAMS Accession No. ML023380432)

2. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," November 13, 2003 (ADAMS Accession No. ML033280022)

3. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical

Specifications and Bases Amendment...," December 16, 2003 (ADAMS Accession No. ML033580622)

4. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," September 22, 2004 (ADAMS Accession No. ML042720440)

5. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," April 6, 2005 (ADAMS Accession No. ML051050438)

6. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," June 14, 2005 (ADAMS Accession No. ML051730410)

7. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," July 8, 2005 (ADAMS Accession No. ML052030337)

8. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," August 17, 2005 (ADAMS Accession No. ML052410103)

9. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," September 8, 2005 (ADAMS Accession No. ML052630397)

10. Letter from D. M. Jamil, Duke Energy Corporation to USNRC, " Proposed Technical Specifications and Bases Amendment...," September 19, 2005 (ADAMS Accession No. ML05)

11. Letter from Robert E. Martin, USNRC to H. B. Barron, Duke Energy Corporation, "Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB7863 and MB7864)," March 3, 2005 (ADAMS Accession No. ML042320059)

12. Letter from Robert E. Martin, USNRC to H. B. Barron, Duke Energy Corporation, "Safety Evaluation for Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies (TAC Nos. MB7863, MB7864, MC0824, AND MC0825), April 5, 2004 (ADAMS Accession No. ML040970046)

13. Letter from Chandu P. Patel, USNRC to G. R. Peterson, Duke Energy Corporation, "Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB3758 and MB3759)," April 23, 2002 (ADAMS Accession No. ML021140431)

Principal Contributors: Michelle Hart, Edwin Forrest, Leta Brown, and Krzystof Parczewski

Date: September 30, 2005

Table 1
Licensee Calculated DBA Radiological Consequences

Accident	TEDE (rem) at location		
	EAB	LPZ	Control Room
Either unit:			
LOCA, limiting case, all LEU	5.50	3.15	2.20
LOCA, limiting case, w/ MOX LTAs	5.55	3.19	2.21
Unit 1:			
LRA, 10% fuel failure, all LEU	0.95	0.24	0.73
LRA, 10% fuel failure, w/ MOX LTAs	1.02	0.26	0.80
REA, 50% fuel failure, all LEU	4.14	3.26	2.47
REA, 50% fuel failure, w/ MOX LTAs	4.23	3.31	2.50
Unit 2 :			
LRA, 10% fuel failure, all LEU	1.63	0.35	1.36
LRA, 10% fuel failure, w/ MOX LTAs	1.77	0.38	1.48
REA, 50% fuel failure, all LEU	4.89	3.37	2.85
REA, 50% fuel failure, w/ MOX LTAs	4.97	3.43	2.89

Table 2
Analysis Assumptions

Loss of Coolant Accident

Limiting dose scenario assumptions	
Core power (includes 2% uncertainty), MWt	3479
Source term model	RG 1.183
Iodine species fraction	
Elemental	0.0485
Particulate	0.95
Organic	0.0015
Containment volumes, ft^3	
Total	1,016,454
Upper	669,559
Lower	346,895
Sump	79,000

Containment Leakage Pathway

Containment leak rate, %/day	Collected	Bypass	Total
Prior to annulus drawdown	0	0.3	0.3
Drawdown to 24 hours	0.279	0.021	0.3
After 24 hours	0.1395	0.0105	0.15
Annulus drawdown, sec			23
Annulus filtration credited, sec			30.5
Annulus volume, ft^3 (assuming 50% credit)			242,045
Annulus ventilation flow rate per fan, cfm			8100
Annulus iodine filtration efficiency, %			
Particulate and elemental			95
Organic			80
Containment air return flow rate per fan (after 600 seconds), cfm			40,000
Containment spray removal credit for one train, hr^{-1}			
Particulate/Aerosol			
0 - 600 sec			0
600 - 3000 sec			0.951
3000 - 3180 sec			0.786
3180 - 86400 sec			1.74
Elemental			
0 - 600 sec			0
600 - 3000 sec			20
3000 - 3180 sec			0.20
3180 - 30,000 sec			0.44
30,000 - 40,000 sec			0.42
40,000 - 50,000 sec			0.40
50,000 - 60,000 sec (DF = 200)			0.37
Ice condenser iodine removal credit			None
Iodine natural deposition credit			None

ECCS Leakage Pathway

ECCS recirculation start, sec	790
ECCS leakage rate, gpm	
Initially aligned to ABFVES	0.5
Not initially aligned to ABFVES	0.5
Time to align ABFVES to all rooms, days	3
ABFVES iodine filter efficiency, %	
Particulate and elemental	95
Organic	80
ECCS leakage iodine partitioning, variable	July 8, 2005 letter, Tables Q1-1 and Q1-2
ECCS backleakage rate to RWST, gpm	20
RWST iodine partitioning, variable	July 8, 2005 letter, Tables Q1-3 and Q1-4

Parameters Common to Rod Ejection and Locked Rotor Accidents

Core power (includes 2% uncertainty), MWt	3479
LRA gap fractions, LEU	
I-131	0.08
Kr-85	0.10
Other noble gases	0.05
Other halogens	0.05
Alkali metals	0.12
REA gap fractions, LEU	
Halogens and noble gases	0.10
Alkali metals	0.12
MOX LTA gap scaling factor, all isotopes	1.5
Fraction of gap inventory released to primary coolant	1.0
Radial peaking factor	1.65
Initial primary coolant concentration, μCi/gm DEI-131	1.0
Primary coolant mass, lbm	
Unit 1	537,793
Unit 2	481,637
Initial secondary coolant concentration, μCi/gm DEI-131	0.1
Initial secondary coolant mass, lbm	
Unit 1	112,000
Unit 2	77,300
Initial SG leak rate, gpd/SG	150
SG tube uncovery	Yes
Iodine partitioning (with SG tubes covered)	0.01
Limiting boil-off rates per SG, lbm/min	
0 - 8 hr (RCS at 240 °F)	891
8 - 31.9 hr (RCS at 211 °F)	120

Locked Rotor Accident

Percent of core in DNB		10
Iodine spiking factor in primary coolant		335
Duration of plant cooldown, hrs		31.9
SG tube bundle uncovery, seconds	Unit 1	Unit 2
Steam Generator 1	1205	1870
Steam Generator 2	0	137
Steam Generator 3 and 4	1121	1770
Time-dependent SG tube leakage		
Time-dependent SG mass		
Time-dependent SG steam release		

Rod Ejection Accident

Percent of core in DNB			50
Percent of core that is melted			0
Iodine species fractions for containment, ECCS and SG releases			
Elemental			0.97
Particulate			0
Organic			0.03
Three scenarios			
1 - Fraction of gap inventory released to containment			1.0
2 - Fraction of gap inventory released to containment sump			1.0
3 - Fraction of gap inventory released to primary coolant			1.0
Containment volume, ft^3			1,196,000
Containment leak rate, %/day	Collected	Bypass	Total
Prior to annulus drawdown	0	0.3	0.3
Drawdown to 24 hours	0.279	0.021	0.3
After 24 hours	0.1395	0.0105	0.15
Annulus drawdown, sec			23
Annulus filtration credited, sec			30.5
Annulus volume, ft^3 (assuming 50% credit)			242,045
Annulus ventilation flow rate per fan, cfm			8100
Annulus iodine filtration efficiency, %			
Particulate and elemental			95
Organic			80
Containment spray iodine removal credit			None
Ice condenser iodine removal credit			None
Containment natural deposition for non-iodine aerosols			Powers 10th percentile
Containment sump volume, ft^3			79,000
ECCS recirculation start, hr			2
ECCS leakage rate, gpm			
Initially aligned to ABFVES			0.5
Not initially aligned to ABFVES			0.5
Time to align ABFVES to all rooms, days			3
ABFVES iodine filter efficiency, %			
Particulate and elemental			95
Organic			80
ECCS leakage iodine partitioning			0.1
ECCS backleakage rate to RWST, gpm			10
Iodine release fractions for RWST leakage			July 8, 2005 letter, Table Q1-5
Duration of plant cooldown, hrs			31.9
SG tube bundle uncovery, seconds		Unit 1	Unit 2
Steam Generator 1		751	2425
Steam Generator 2		662	2605
Steam Generator 3 and 4		2458	810
Time-dependent SG tube leakage			
Time-dependent SG mass			
Time-dependent SG steam release			

Control Room Parameters

Control room volume, ft^3	117,920
CRAVS makeup flow rate, cfm	
Lower bound	3500
Upper bound	5500
CRAVS intake flow split	60/40
Time to open a closed CRAVS intake, hr	10
Control room recirculation flow rate, cfm	1500
Control room recirculation filter efficiency	
Elemental	0.99
Particulate	0.99
Organic	0.95
Control room unfiltered inleakage, cfm	100

Table 3
Catawba Control Room Atmospheric Dispersion Factors

Unit Vent Stack Release Initially to the Limiting Outside Control Room Air Intake			
Accidents	**Time Interval (hrs)**	**χ/Q Values (sec/m^3)**	**No. of open intakes**
LOCA with an initially closed control room outside air intake. Release pathways include containment leakage to the annulus, containment bypass leakage, and engineered safety feature (ESF) leakage in the auxiliary building.	0 – 2	1.74×10^{-3}	One
	2 – 8	1.47×10^{-3}	One
	8 – 10	6.90×10^{-4}	One
	10 - 24	$3.68\text{x}10^{-4}$	Both
	24 – 96	2.67×10^{-4}	Both
	96 – 720	1.87×10^{-4}	Both

Refueling Water Storage Tank Release Initially to Limiting Outside Control Room Air Intake			
Accidents	**Time Interval (hrs)**	**χ/Q Values (sec/m^3)**	**No. of open intakes**
LOCA with an initially closed control room outside air intake . The release pathway consists of ESF back leakage to the refueling water storage tank (RWST) and releases from the RWST vent.	0 – 2	1.92×10^{-3}	One
	2 – 8	1.48×10^{-3}	One
	8 – 10	7.40×10^{-4}	One
	10 - 24	4.10×10^{-4}	Both
	24 – 96	2.86×10^{-4}	Both
	96 – 720	1.87×10^{-4}	Both

Table 3 (con't)

Unit Vent Stack Release to Both Control Room Air Intakes			
Accidents	**Time Interval (hrs)**	**χ/Q Values (sec/m^3)**	**No. of open intakes**
LOCA with either a minimum safeguards failure, an annulus ventilation system (AVS) pressure transmitter failure, or a residual heat removal system (RHR) or containment spray system (CSS) heat exchanger failure. All REA scenarios. Release pathways include containment leakage to the annulus, containment bypass leakage, and ESF leakage to the auxiliary building.	0 – 2	1.04×10^{-3}	Both
	2 – 8	8.82×10^{-4}	Both
	8 – 10	4.14×10^{-4}	Both
	10 - 24	$3.68x10^{-4}$	Both
	24 – 96	2.67×10^{-4}	Both
	96 – 720	1.87×10^{-4}	Both

Refueling Water Storage Tank Vent Release to Both Control Room Air Intakes			
Accidents	**Time Interval (hrs)**	**χ/Q Values (sec/m^3)**	**No. of open intakes**
LOCA with either a minimum safeguards failure, an AVS pressure transmitter failure, or a RHR or CSS heat exchanger failure. All REA scenarios. Release pathways consist of ESF back leakage to the RWST and releases from the RWST vent.	0 – 2	1.26×10^{-3}	Both
	2 – 8	9.78×10^{-4}	Both
	8 – 10	4.86×10^{-4}	Both
	10 - 24	4.10×10^{-4}	Both
	24 – 96	2.68×10^{-4}	Both
	96 – 720	1.87×10^{-4}	Both

Table 3 (con't)

S/G PORVs, MSSVs and AFW TDP Exhaust Vent to Both Intakes			
Accidents	**Time Interval (hrs)**	**χ/Q Values (sec/m^3)**	**No. of Open Intakes**
All LRA and REA scenarios. Release pathways consist of steam generator boiloff to the outboard steam generator doghouse.	0 - 2	7.14×10^{-3}	Both
	2 – 8	4.05×10^{-3}	Both
	8 – 10	2.24×10^{-3}	Both
	10 - 24	1.81×10^{-3}	Both
	24 – 96	1.24×10^{-3}	Both
	96 – 720	7.26×10^{-4}	Both

Table 4
Catawba EAB and LPZ Atmospheric Dispersion Factors

Receptor	Time Interval (hrs)	χ/Q Value (sec/m^3)
Exclusion Area Boundary	0 – 2	4.78×10^{-4}
Low Population Zone	0 – 8	6.85×10^{-5}
	8 – 24	4.00×10^{-5}
	24 – 96	2.00×10^{-5}
	96 – 720	7.35×10^{-6}

cc:

Mr. Lee Keller, Manager
Regulatory Compliance
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Ms. Lisa F. Vaughn
Duke Energy Corporation
526 South Church Street
P. O. Box 1006
Mail Code = EC07H
Charlotte, North Carolina 28201-1006

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P.O. Box 29513
Raleigh, North Carolina 27626

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

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

NCEM REP Program Manager
4713 Mail Service Center
Raleigh, North Carolina 27699-4713

North Carolina Electric Membership Corp.
P.O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
4830 Concord Road
York, South Carolina 29745

Mr. Henry Porter, Assistant Director
Division of Waste Management
Bureau of Land and Waste Management
Dept. of Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. R.L. Gill, Jr., Manager
Nuclear Regulatory Issues
and Industry Affairs
Duke Energy Corporation
526 South Church Street
Mail Stop EC05P
Charlotte, North Carolina 28202

Saluda River Electric
P.O. Box 929
Laurens, South Carolina 29360

Mr. Peter R. Harden, IV, Vice President
Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road
12th Floor
Charlotte, North Carolina 28210

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

cc:

Division of Radiation Protection
NC Dept. of Environment, Health,
and Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. Henry Barron
Group Vice President, Nuclear Generation
and Chief Nuclear Officer
P.O. Box 1006-EC07H
Charlotte, NC 28201-1006

Diane Curran
Harmon, Curran, Spielbergy &
Eisenberg, LLP
1726 M Street, NW
Suite 600
Washington, DC 20036