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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
Docket Numbers 50-413 and 50-414
Proposed Technical Specifications and Bases Amendment
Technical Specification and Bases 3.6.10
Annulus Ventilation System (AVS)
Technical Specification and Bases 3.6.16
Reactor Building
Technical Specification Bases 3.7.10
Control Room Area Ventilation System (CRAVS)
Technical Specification Bases 3.7.12
Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)
Technical Specification Bases 3.7.13
Fuel Handling Ventilation Exhaust System (FHVES)
Technical Specification and Bases 3.9.3
Containment Penetrations
Technical Specification 5.5.11
Ventilation Filter Testing Program (VFTP)
TAC Numbers MB7014 and MB7015

References:

1. Letters from Duke Energy Corporation to NRC, dated November 25, 2002, November 13, 2003, December 16, 2003, and September 22, 2004

2. Letter from NRC to Duke Energy Corporation, dated May 25, 2004

In Reference 2, the NRC provided a Request for Additional Information (RAI) concerning the subject Catawba license amendment request submittal. The Reference 1 September 22, 2004 letter provided a partial response to this RAI. The purpose of this letter is to provide the remaining response to this RAI.

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In addition, this letter contains a response to a chemistry related question provided by one of your technical reviewers in a conference call. The RAI responses are contained in Attachment 1 to this letter. Attachment 2 to this letter contains a revised markup of two of the previously transmitted Technical Specification markup pages. These pages were changed by recent unrelated license amendments that affected these pages. Duke Energy Corporation has determined that the original No Significant Hazards Consideration Analysis contained in the license amendment request submittal of November 25, 2002 requires revision as a result of this RAI response. Attachment 3 to this letter contains the revised analysis. The Environmental Analysis contained in the license amendment request submittal of November 25, 2002 is unchanged as a result of this RAI response.

Pursuant to 10 CFR 50.91, a copy of this letter is being sent to the appropriate State of South Carolina official.

Inquiries on this matter should be directed to L.J. Rudy at (803) 831-3084.

Very truly yours,

D.M. Jamil

Attachments

D.M. Jamil affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

D.M. Jamil, Vice President

Subscribed and sworn to me: 4/7/05
Date

Notary Public

My commission expires: 7/2/2014
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ATTACHMENT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

The following question pertaining to the use of MOX lead fuel assemblies was received on May 25, 2004.

In the Duke Power amendment request for use of mixed oxide (MOX) fuel lead test assemblies (LTAs), by letter dated September 23, 2003, you indicated that you plan to insert the 4 MOX LTAs into the Catawba Unit 1 core during Cycle 16, with Unit 2 as a backup if necessary. Considering that plan for the current license amendment which requests full implementation of an alternative source term for both units at Catawba, please evaluate all affected design basis accident dose analyses assuming that the 4 MOX LTAs are in a Catawba core. Accidents which assume release from the fuel, such as the loss of coolant accident, control rod ejection, and locked reactor coolant pump rotor are of concern for the use of MOX LTAs.

Response: Release of fission products from the fuel pin gaps and the fuel pins themselves are postulated for the design basis loss of coolant accident (LOCA). Releases from the gaps of some of the fuel pins following projected departure from nucleate boiling (DNB) are postulated for the following design basis accidents:

1) Locked rotor accident (10% of the fuel pins are assumed to enter DNB and experience clad failure),

2) Rod ejection accident (50% of the fuel pins are assumed to enter DNB and experience clad failure),

3) Fuel Handling Accident (all fuel pins in 1 fuel assembly are assumed to fail), and

4) Weir Gate Drop (all fuel pins in 7 fuel assemblies are assumed to fail).

An evaluation of the effect of the MOX lead fuel assemblies on radiological consequences of the design basis fuel handling accident (FHA) and weir gate drop (WGD) using the method of alternative source terms (AST) has been provided (Ref. 1). The AST analyses for the design basis LOCA, locked rotor accident (LRA), and rod ejection accident (REA) are presented here.

A baseline analysis of radiological consequences of the design basis LOCA with the method of AST was completed and presented to the Staff for review (Ref. 2). The submittal of this analysis was supplemented by responses to an earlier request for additional information (Ref. 3-5). This response presents an additional analysis of the design basis

LOCA with the source term revised to account for 4 MOX lead fuel assemblies (LFAs).

When the NRC posed this question, no baseline analysis of the design basis LRA or REA had been completed with the method of AST. This response presents a baseline analysis of radiological consequences of the design basis LRA and REA using the method of AST. It presents a supplemental analysis of radiological consequences of these design basis accidents accounting for the insertion of 4 MOX LFAs.

Design Basis LOCA

The analysis of radiological consequences of the design basis LOCA (Ref. 2, 4, 5) was repeated to account for insertion of 4 MOX LFAs. The preparation for this analysis included the following:

1) The source term and release rates for the design basis LOCA were modified as described below to account for the insertion of 4 MOX LFAs.

2) The baseline analysis of post LOCA iodine chemistry was reviewed in light of a predicted increase of the amount of iodine released from the core. In particular, the iodine partition fractions for leakage of Engineered Safety Features (ESF) systems in the Auxiliary Building were recalculated. This accounts for changes in the core inventory of iodine isotopes with the insertion of 4 MOX LFAs.

3) Lower bound values only were taken for the Control Room Area Ventilation System (CRAVS) airflow rates through the control room. The CRAVS recirculation airflow rate through the control room was set to 1,500 cfm only while the CRAVS total airflow rate to the control room was set to 3,500 cfm. It has been shown that the computed control room doses for the design basis LOCA are higher with lower values for the CRAVS total and recirculation airflow rates to the control room. Thus, this assumption is conservative (Ref. 2).

4) This analysis made use of dose coefficients from Federal Guidance Report (FGR) 11 and FGR 12 (Ref. 6, 7). These dose coefficients were used in the revised analysis of the radiological consequences of the design basis LOCA provided in response to a question in the first request for additional information (Ref. 5).

All other inputs and assumptions are the same as those reported in the original submittal of November 25, 2002 (Ref. 2).

Limiting radioactivity levels have been calculated for a single MOX LFA. The method of calculation was the same as that reported to the NRC concerning the development of the low enriched uranium (LEU) fuel assembly source term used for the AST analysis of the design basis FHA and WGD (Ref. 8, 9). In particular, enveloping values of burnup, enrichment, and power level (including the ECCS evaluation uncertainty) were identified and used in the calculation. Limiting values of radial peaking factors as a function of burnup were used to determine the source term. The calculations were completed with the computer code SCALE (Ref. 15). The resulting values of radioactivity levels in a MOX LFA are listed in Table 1. The radioactivity levels in an all LEU fuel core were presented previously (Ref. 2 Page A43-A45).

For all radioisotopes in the MOX LFAs but those of noble gases, the release fractions cited by the NRC for the post LOCA gap release and early in-vessel phases (Ref. 10 Table 2) were multiplied by 1.5. For noble gases in the MOX LFAs, the release fraction for the gap release phase was multiplied by 1.5 to increase it from 0.05 to 0.075. It is assumed that all noble gases in the core are released to containment over the post LOCA gap release and early in-vessel phases combined (Ref. 10 Table 2). Therefore, the early in-vessel release fraction for noble gases in the MOX LFAs was decreased from 0.95 to 0.925.

The Staff regulatory positions for AST set time spans of 0.5 and 1.3 hr respectively (beginning at 30 sec) for the gap release and early in-vessel phases for a design basis LOCA at a pressurized water reactor (Ref. 10 Table 4). These (the gap release and early in-vessel phase release fractions and the activities for the radioisotopes in an all LEU core and a core with 4 MOX LFAs) can be used to calculate radioisotope release rates for the gap release phase and the early in-vessel phase for a core containing 4 MOX LFAs.

The post LOCA release of iodine (both stable and radioactive isotopes) from a core with 4 MOX LFAs was calculated to increase by 2.52 moles (or less than 3%). The baseline analysis of post LOCA iodine chemistry (Ref. 2) was reviewed in light of this. The iodine partition factors for leakage from Engineered Safety Features (ESF) systems in the Auxiliary Building were recalculated to account for this. The methodology reported earlier to the Staff (Ref. 2) was used to repeat the calculations. The only change in the data used was to set the amount of iodine in the sump to 87.64 moles.

The iodine partition fractions for ESF leakage in the Auxiliary Building following a design basis LOCA at a Catawba nuclear unit with all LEU fuel have been reported to the NRC (Ref. 2 Enclosure 9). The results are shown in Table 2. There was no change from any of the baseline values except for ESF leakage downstream of the Residual Heat Removal System (RHRS) and Containment Spray System (CSS) Heat Exchangers for the design basis LOCA with either an Annulus Ventilation System (AVS) pressure transmitter failure or initially closed CRAVS outside air intake (Table 2 Notes 4 and 6). The initial value for these scenarios increased from 0.013 (revised from the value of 0.010 reported earlier - Ref. 2) to 0.014. The baseline analysis showed that ESF backleakage to the Refueling Water Storage Tank (RWST) contributed insignificantly to the post LOCA radiation doses. For this reason, the calculation of iodine partitioning from the RWST was not repeated. In addition, a review of the time constants and decontamination factors for CSS washout of fission products (Ref. 2 Pages A-7, A-8, and Enclosure 2) showed margin in these characteristics. Therefore, they were not recalculated.

Radiation doses associated with post LOCA releases from a nuclear unit with 4 MOX LFAs were calculated for the same design basis LOCA scenarios as were considered in the baseline analysis (Ref. 2).

1) Design basis LOCA with a Minimum Safeguards failure.

2) Design basis LOCA with failure of a pressure transmitter of the AVS, causing the affiliated AVS train to operate continuously in full Exhaust instead of modulation between Exhaust and Recirculation. The operators secure the affected AVS train in 2.5 hours after the initiating event.

3) Design basis LOCA with failure of cooling water flow through a Heat Exchanger of the RHRS or CSS.

4) Design basis LOCA with an initially closed CRAVS outside air intake. The operators open the closed CRAVS outside air intake within 10 hours of the initiating event.

The total effective dose equivalents (TEDEs) for these design basis LOCA scenarios associated with a reactor core with all LEU fuel and a reactor core with 4 MOX LFAs are presented in Table 3. For all scenarios, the TEDEs for post LOCA ESF leakage increased with the insertion of the MOX LFAs. The TEDEs for this post LOCA release pathway are dominated by iodine radioisotopes which increase with the insertion of the MOX LFAs. For the design basis LOCA scenarios with an AVS pressure transmitter failure and an initially closed CRAVS outside air intake, the increases in the TEDEs associated with ESF leakage are higher than the increases in TEDEs for the remaining design basis LOCA scenarios. This reflects the impact of insertion of the MOX LFAs on the iodine partition fractions for these scenarios. The TEDEs for post LOCA containment leakage increased slightly for some scenarios while decreasing slightly for others. Many fission products other than noble gas and iodine isotopes are entrained in containment leakage. The activities for these in a MOX LFA may be lower than the activities in a LEU fuel assembly for some of these isotopes.

The changes in total TEDEs for the limiting design basis LOCA scenarios with the insertion of the 4 MOX LFAs are presented in Table 3. The design basis LOCA with failure of a RHRS or CSS Heat Exchanger is limiting for the TEDEs at the Exclusion Area Boundary (EAB) and boundary of the Low Population Zone (denoted as the LPZ). The TEDEs at the EAB and LPZ are as follows:

EAB TEDE (All LEU fuel): 5.41 Rem
EAB TEDE (4 MOX LFAs): 5.46 Rem
Change with insertion of MOX LFAs: +0.9%

LPZ TEDE (All LEU fuel): 3.11 Rem
LPZ TEDE (4 MOX LFAs): 3.14 Rem
Change with insertion of MOX LFAs: +1.1%

The design basis LOCA with an initially closed CRAVS outside air intake is limiting for the control room TEDE. The control room TEDEs for this scenario are as follows:

Control room TEDE (All LEU fuel): 2.13 Rem
Control room TEDE (4 MOX LFAs): 2.13 Rem
Change with insertion of MOX LFAs: +0.1%

The regulatory limits for post LOCA TEDEs are 25 Rem at the EAB and LPZ and 5 Rem in the control room. The radiation doses following the design basis LOCA at Catawba Nuclear Station are within these limits and remain within limits with the insertion of 4 MOX LFAs.

Design Basis Locked Rotor and Rod Ejection Accidents

Overview

Analyses of radiological consequences of the design basis LRA and REA using the method of AST have been completed. The analyses include a baseline analysis of these design basis accidents assuming that the source term is associated with LEU fuel pins only. A supplemental analysis of these accidents also was completed in which it was assumed that the source term included clad failure of all the pins of the 4 MOX LFAs. A synopsis of these analyses is reported below: baseline analysis of the design basis LRA, baseline analysis of the design basis REA, and supplemental analysis of the effects of insertion of the 4 MOX LFAs.

Baseline Analysis of the Design Basis LRA

The design basis LRA is postulated to be followed by entry of some of the fuel pins into departure from nucleate boiling (DNB) and subsequent clad failure of these fuel pins, release of the radioactivity into the reactor coolant, leakage into the steam generators (SGs), and release to the environment with SG boiloff.

The Design Basis Scenario

The design basis LRA is defined to be a LRA with loss of offsite power and a Minimum Safeguards failure (Ref. 11-13). The upper limit for the fractions of pins in DNB for a LRA at each nuclear unit with offsite power available is set to ensure that the radiation doses for the LRA with loss of offsite power are bounding.

Separate design basis LRA scenarios were analyzed for Unit 1 and Unit 2. The separate calculations account for the differences in reactor coolant inventories and time spans of SG tube bundle uncovery.

Design Basis LRA Source Terms

The radioactive source terms for the design basis LRA include the following constituents:

1) Initial activity in the reactor coolant amplified by the accident initiated iodine spike (Ref. 21). This is a conservative change with respect to the current license basis analysis (Ref. 11-13). Initial radioactivity levels in both the reactor coolant and unit secondary systems are included in this analysis

to confirm that they do not contribute significantly to the radiation doses for this design basis accident. The initial activities for noble gas and iodine radioisotopes are set pursuant to the plant technical specifications (Ref. 14 Technical Specification TS 3.4.16). The multiplier for accident initiated iodine spike is set to 335. This is the multiplier for the accident initiated iodine spike endorsed by the Staff for use in the analysis of radiological consequences of the design basis steam generator tube rupture (SGTR - Ref. 20). The reactor coolant pressure does not decrease significantly following a design basis LRA; no safety injection is predicted. With respect to the conditions associated with an accident initiated iodine spike, the design basis LRA would be bounded by the design basis SGTR. This justifies setting the multiplier for the accident initiated iodine spike to 335 for the design basis LRA. The data used to calculate the initial reactor coolant activity and the iodine appearance rates for the accident initiated iodine spike are presented in Table 4A.

2) Initial iodine activity in the secondary plant. This is a conservative change with respect to the current license basis analysis (Ref. 11-13). The initial radioactivity levels were set as follows: The initial iodine activities in the SGs were set in accordance with the plant technical specification (Ref. 14 TS 3.7.17). Equilibrium activities in the rest of the unit secondary systems were calculated based on a SG boiloff iodine partition factor of 100 (Ref. 19) and perfect (100%) scrubbing in the main condenser. The resulting radioactivity levels were assumed for the unit secondary system tanks (Upper Surge Tanks, Auxiliary Feedwater Condensate Storage Tank, and main condenser hotwell). Iodine isotopes were assumed to be transported from these tanks to the SG secondary side by the Auxiliary Feedwater System until the tanks were calculated to be emptied. The data used to calculate the initial activities in the unit secondary systems are presented in Table 4B.

3) Entry of some fuel pins into DNB and subsequent clad failure. The number of fuel pins assumed to enter DNB and experience clad failure following a design basis LRA at either Catawba nuclear unit was set to 10% of the core. The gap fractions for the fission products taken to be in the fuel pin gaps were set pursuant to the germane NRC regulatory position (Ref. 10 Table 3).

The DNB source term also is based on the limiting fission product radioactivity levels in a fuel assembly. The fuel assembly isotopics were developed in the same manner as for the MOX LFAs (cf. above) and also for the design basis FHA and WGD (Ref. 8, 9). The data for calculation of the baseline source term for the design basis LRA and REA are presented in Table 4C. The baseline fuel assembly isotopics for the design basis LRA and REA are presented in Table 5. These calculations model limiting values of isotopic conditions over the range of the limiting allowable burnup dependent radial peaking history curve. This assumption is used in the current license basis analysis of radiological consequences of the design basis LRA (Ref. 13).

Radioactivity Transport and Release

A brief synopsis of the simulation of radioactivity transport in the Reactor Coolant System (RCS) and SGs and releases to the environment is presented below (Comp. Ref. 21).

All fission products in the fuel pins assumed to enter DNB are assumed to immediately mix homogeneously in the reactor coolant. Tube leakage based on the limiting values in the plant technical specifications (Ref. 14 TS 3.4.13.d) is assumed to occur in each SG. During the event, the leakage may vary with pressures in the RCS and in the secondary sides of the SGs. The transient values presented earlier for primary to secondary leak rate (PSLR) to the NRC (Ref. 13) were modified to correspond to standard conditions. This modification was achieved in a conservative manner by multiplying the PSLR values in the current license basis by the factor 1.43. Trip of the affected unit and loss of offsite power are assumed at event initiation. Post trip steam releases from the SGs to the environment are assumed to occur immediately. Iodine activity in the SG steam releases is assumed to be partitioned from the water with a partition factor of 100.

All noble gases entrained in SG tube leakage are assumed to escape directly to the environment. At different times during the event, the tubes of the different SGs are calculated to become uncovered. For the time span over which the tubes of any SG are predicted to become uncovered, all fission products entrained in the tube leakage for that SG are assumed to escape directly to the environment. For the time spans for which the tubes of a SG are predicted to be submerged, all fission products entrained in the tube leakage for that SG except noble gases are assumed to mix instantly and homogeneously with the water in its secondary side.

Beginning at 30 minutes, the control room operators are assumed to cool the affected nuclear unit at 50 °F/hr. This assumption is consistent with the current license basis analysis (Ref. 13). When the affected nuclear unit is cooled to an average temperature of 350 °F at 5 hr after the initiating event, it is assumed that the operators place one train of the RHRS on-line. The cooldown is assumed to continue at 50 °F/hr or the limiting rate consistent with operation of one RHRS train until entry into Mode 5 (200 °F). Primary to secondary leakage and steam releases are assumed to continue until 211 °F at 31.9 hr after the initiating event. The following limiting values for boiloff rates for each SG for average RCS temperature below 350 °F were calculated:

891 lbm/min for average RCS temperature down to 240 °F at 8.0 hr after the initiating event and
120 lbm/min for average RCS temperature down to 211 °F at 31.9 hr after the initiating event.

The simulation of post accident SG boiloff to 211 °F is consistent with the regulatory positions for AST analysis of design basis accidents featuring releases of fission products from the SGs (Ref. 19). It also is consistent with the plant procedures by which the operators would bring the affected unit to cold shutdown following a design basis accident.

The data pertaining to activity transport and release to the environment for the design basis LRA is the same as presented to the Staff September 20 and October 29, 2004 (Ref. 11, 13) with the following two exceptions. First, the PSLR values for the design basis LRA were adjusted to correspond to standard conditions instead of reactor coolant conditions. This was done in a conservative manner by multiplication of each of the PSLR values by 1.43. Second, simulation of SG releases was continued to average reactor coolant temperature of 211 °F.

Post Accident Atmospheric Dispersion Factors (χ/Q Values)

The values used for the χ/Qs for transport and dispersion of fission products to the EAB and LPZ are unchanged from those values presented to the NRC on November 25, 2002 (Ref. 2) and October 29, 2004 (Ref. 13).

For the design basis LRA all fission products are assumed to be released through the SG Power Operated Relief Valves (PORVs). The SG PORVs are located in the SG doghouses. The control room χ/Q values for releases from the SG doghouse vents (cf. Ref. 4) are used in this analysis. These values correspond to transport of fission products to one CRAVS outside air intake. As the design basis LRA scenario includes a single failure, both CRAVS intakes are assumed to be open. The design basis value of 60/40 for the airflow split in the CRAVS outside air intakes was used to adjust the control room χ/Q values.

Control Room Model

The control room model used in the AST analysis of the design basis LRA is the same as that used for the AST analysis of the design basis LOCA (Ref. 2), with one clarification. Lower bound values were used for both CRAVS total airflow rate to the control room (3,500 cfm) and CRAVS recirculation airflow rate through the control room (1,500 cfm). The use of lower bound values for the CRAVS total and recirculation airflow rates is conservative for the calculation of control room radiation doses at Catawba Nuclear Station.

Conversion to Dose

Dose coefficients were taken from FGRs 11 and 12 (Ref. 6, 7). All other aspects of conversion from radioactivity to dose conform to the germane staff expectations for AST analysis (Ref. 10).

TEDEs Following the Design Basis LRA (Baseline Analysis)

The TEDEs following the design basis LRA are presented in Table 6. These are found to be within all of the germane regulatory limits (2.5 Rem at the EAB and LPZ and 5 Rem in the control room).

Baseline Analysis of the Design Basis REA

The AST analysis of the design basis REA shares many of the features as the AST analysis of the design basis LRA. These include a DNB source term, initial activity in the RCS and secondary unit systems, and simulation of SG tube leakage, tube bundle uncovery, and boiloff. Therefore, the discussion of the baseline AST analysis of the design basis REA frequently references the baseline AST analysis of the design basis LRA. Differences between simulation of these features for the LRA and REA are noted.

The AST analysis of the design basis REA also includes simulation of releases to containment or containment sump. In conformance to the current license basis analysis, the AST analysis of the design basis REA assumes that it is followed by a LOCA in that the radiation doses from releases to containment are added to the radiation doses from post accident SG releases. There is a departure from the current license basis analysis in that the higher of the radiation dose from post REA containment or ESF leakage including backleakage to the Refueling Water Storage Tank (RWST) is added to the radiation doses for post REA SG releases. The justification for this is given below.

The Design Basis Scenario

The design basis REA is defined to be a REA with offsite power available and a Minimum Safeguards failure as it is in the current license basis analysis (Ref. 11-13). The fraction of pins in DNB following a design basis REA is set to the same limit regardless of availability of offsite power. With offsite power available, the RCS pumps remain on-line, adding to the heat removal load for the SGs and thus the SG tube bundle uncovery time spans. Therefore, offsite power is assumed to be available in the AST analysis of this design basis accident.

Separate design basis REA scenarios were analyzed for Unit 1 and Unit 2 as they were in the AST analysis of the design basis LRA.

Design Basis REA Source Terms

The radioactive source terms for the design basis REA are computed in the same manner as for the design basis LRA with the following two exceptions:

1) No accident initiated iodine spike is included.

2) The fraction of fuel pins assumed to be in DNB is set to 50%. No fuel melt is assumed to occur (Ref. 13, 16).

The limiting fuel assembly isotopic inventory is used in calculating the DNB source term for the design basis REA as it is for the design basis LRA. This is a conservative departure from the current license basis analysis in which average full power was assumed for the pins in DNB. Isotopics based on limiting values for burnup dependent radial peaking currently in place were used to develop the DNB source term for the design basis REA. In the future, Duke Energy Corporation (Duke) may develop a DNB source term that is closer to the lowest upper bound (supremum) for the design basis REA.

Post REA Activity Transport and Releases From the SGs

The methods for simulating the transport of activity in the SGs and releases of activity from the SGs are the same as those used in the AST analysis of the design basis LRA. The data pertaining to the activity transport to and releases from the SGs are the same as that associated with the current license basis analysis (Ref. 13) with the exception of simulating cooldown to 211 ^{o}F. (Comp. Ref. 22)

Time dependent values were taken for SG tube leakage (PSLRs). This is consistent with the current license basis analysis for the design basis REA (Ref. 11-13). Simulation of the REA as a LOCA yields the calculation of time-dependent PSLRs in the transient thermal-hydraulic analysis (RETRAN) and is the reason for addition of the constituents to the radiation doses from releases to the SGs and containment. The values presented earlier (Ref. 13) were multiplied by 1.43 (as were the PSLR values for the design basis LRA) so that the PSLR values would correspond to standard conditions.

Overview of Post REA Releases to the Containment

Separate scenarios are postulated in which all of the activity in the RCS (the entire DNB source term and initial activity in the reactor coolant) is released either to the containment or the containment sump. This is in contrast to the current license basis in which 25% of the RCS source term was released to the containment and 50% to the sump. In addition, no credit is taken for either the CSS or ice condenser for removal of any fission product. Credit is taken for natural deposition of fission products to internal structures but not for noble gas or iodine isotopes. In summary, no credit is taken for any mechanism by which iodine would be transported from the containment to the containment sump. Furthermore, only iodine isotopes are assumed to be released from the containment sump. This provides the basis for calculating radiation doses for post REA containment and ESF leakage separately, and adding the higher of the two constituents to the radiation doses calculated for post REA SG releases.

Post REA Containment Transport and Leakage

As noted above, all fission products in the gaps of the fuel pins in DNB (and initial RCS activity) are assumed to be released immediately to containment. In particular, the fission products in the REA source term are assumed to be released instantaneously to the lower compartment as they are for the design basis LOCA. As noted in the submittal (Ref. 2), the lower and upper compartments are separated by a divider deck with very limited ventilation exchange. Furthermore, the design basis REA has a break flow closer to the small break LOCA than to the design basis LOCA. This justifies placing the source term in the lower compartment.

The aspects of activity transport in primary and secondary containment and releases with containment bypass leakage and containment leakage filtered by the AVS are generally the same as simulated in the AST analysis of the design basis LOCA with a Minimum Safeguards failure. The exceptions are noted as follows:

1) No analyses of post REA containment response (ice melt) and containment sump pH have been completed. In lieu of these analyses, the conservative assumption is made that all iodine in the source term consists of diatomic iodine (97%) and organic iodine compounds (3%). No particulate iodine compounds (e.g., CsI) is assumed. In the future, Duke may complete a calculation of post REA ice melt and a calculation of post REA pH in the containment sump.

2) No credit is taken for the CSS. It is consistent with the current license basis for the design basis REA (Ref. 11, 13).

3) Credit is taken for deposition of fission products. The method of NUREG/CR-6189 (Ref. 17) is used to calculate the time constants and decontamination factors (DFs) for natural deposition. This methodology simulates natural deposition of aerosols or particulates. Since the iodine isotopes were not assumed to be in particulate form, natural deposition of iodine was not simulated. The 10^{th} percentile data and correlations were used to make the calculations. The time constants and DFs for natural deposition are presented in Table 7.

Post REA ESF Leakage Overview

Cold leg recirculation is assumed to begin at 2 hours after the initiating event as it is in the current license basis analysis of the design basis REA (Ref. 11, 13). Releases of iodine isotopes with ESF leakage both in the Auxiliary Building and to the RWST is simulated beginning at this time.

Post REA ESF Leakage in the Auxiliary Building

Iodine partition factors for post LOCA ESF leakage in the Auxiliary Building have been calculated (Ref. 2). These iodine partition factors are based on a calculation of post LOCA containment sump pH. The iodine partition factor for ESF leakage following a design basis REA is set to 10% or 0.1 (Ref. 18). In the future, Duke may reevaluate this assumption on the basis of a calculation of post REA ice melt and containment sump pH.

As was the case for the design basis LOCA, the ESF leakage was partitioned as follows:

1) The rate of ESF leakage in rooms to which the Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) is initially aligned is set to 0.5 gpm.

2) The rate of ESF leakage in rooms to which the ABFVES is not initially aligned also is set to 0.5 gpm. It was assumed that the operators would align the ABFVES to these rooms within 3 days after the initiating event. This assumption also was made in the AST analysis of the design basis LOCA (Ref. 2).

ESF Backleakage to the RWST Following a Design Basis REA

Analyses of iodine transport, partitioning, and releases from the RWST with ESF backleakage following a design basis REA were completed. The analyses employed the same methods (the computer program IODEX), assumptions, and data as that for the design basis LOCA (Ref. 2) with the following exceptions. Separate analyses were completed for the cases of the source term associated with all LEU fuel and including 4 MOX LFAs.

1) The leakage included no melted ice. This assumption was put into effect by setting the sodium and boron concentrations in the leakage to 0 ppm and 3,075 ppm, respectively. In the future, Duke may complete analyses of ice melt and containment sump pH following a design basis REA and use these as the bases for a reanalysis of post REA ESF backleakage to the RWST.

2) The amount of iodine in the sump for the design basis REA was computed to be 10.64 moles for all LEU fuel and 10.86 moles for four MOX LFAs in the source term.

3) In order to establish sufficient margins to the acceptance criteria for TEDEs for the design basis REA, the rate of ESF backleakage was lowered from 20 gpm (Ref. 2) to 10 gpm.

The iodine release fractions and equivalent rate of unfiltered ESF leakage from the containment sump directly to the environment are shown in Table 8.

Post Accident Atmospheric Dispersion Factors (χ/Q Values)

The values used for the χ/Qs for transport and dispersion of fission products to the EAB and LPZ, are the same as those used in the AST analysis of the design basis LRA.

The values used for the χ/Qs for transport and dispersion of fission products from SG releases to the control room are the same as used in the AST analysis of the design basis LRA. The values used for the χ/Qs for transport and dispersion of fission products from the containment and ESF leakage to the control room are the same as used in the AST analysis of the design basis LOCA (Ref. 2).

Control Room Model

The control room model used in the AST analysis of the design basis REA is the same as that used for the AST analysis of the design basis LOCA (Ref. 2) with the clarification that lower bound values were taken for the CRAVS total and recirculation airflow rates to the control room.

Conversion to Dose

Dose coefficients were taken from FGRs 11 and 12 (Ref. 6, 7). All other aspects of conversion from radioactivity to dose conform to the germane staff expectations for AST analysis (Ref. 10).

TEDEs Following the Design Basis REA (Baseline Analysis)

The TEDEs following the design basis REA are presented in Table 9. The constituents for post REA SG releases, containment leakage, and ESF leakage (in the Auxiliary Building and to the RWST combined) are presented. (The TEDEs for post REA ESF leakage are taken over the time spans

2-4 hr.) In all cases (Units 1 and 2, EAB, LPZ, control room), the TEDEs for post REA containment leakage exceeded the TEDEs for post REA ESF leakage and therefore were added to the post REA SG leakage to obtain the TEDEs for the design basis REA scenarios.

These are found to be within all of the germane regulatory limits (6.3 Rem at the EAB and LPZ and 5 Rem in the control room).

Effect of Insertion of the 4 MOX LFAs (LRA and REA)

The AST analyses of radiological consequences of the design basis LRA and REA were repeated to account for insertion of the 4 MOX LFAs. The only changes made to the input were to the source term for both accidents and iodine release rates for ESF backleakage to the RWST following a design basis REA (already discussed above). These changes implement the assumption that for the design basis LRA and REA all the fuel pins in the 4 MOX LFAs enter DNB and experience clad failure.

Input Preparation

The DNB source terms for the design basis LRA and REA were developed as follows:

1) All fuel pins in the 4 MOX LFAs (1,056 fuel pins) were assumed to enter DNB and experience clad failure. This was the basis for development of composite DNB source terms for the design basis LRA and REA.

2) The limiting fuel assembly isotopics presented in Tables 1 and 5 were used to develop the DNB source term. Note that of the radioisotopes represented in Table 1, only those of the noble gases (krypton and xenon), the halogens (iodine and bromine), and alkali metals (rubidium and cesium) were assumed to be in the fuel pin gaps (Ref. 10 Table 3).

3) The MOX LFA gap fraction for each isotope was set by multiplying the corresponding gap fractions by 1.5 in conformance to earlier assumptions made by Duke (Ref. 1). This assumption was made specifically in support of the license amendment request concerning insertion of the 4 MOX LFAs (Ref. 1, 23). Duke will submit an analysis of MOX fuel pin gap fractions for non LOCA accidents in support of a future license amendment request pertaining to batch loading of MOX fuel assemblies.

As discussed above, iodine release fractions for post REA ESF backleakage to the RWST were calculated with the assumption that the iodine inventories in all fuel pins in the MOX LFAs were released following DNB induced clad failure.

The radiation doses for these design basis accidents and their comparison to radiation doses for the design basis LRA and REA are presented in Tables 10-12.

Analysis Results

Radiation doses for the design basis LRA with the source term including 4 MOX LFAs are presented in Table 10. These are shown in comparison with the radiation doses for the design basis LRA with the source term associated with all LEU fuel only. Insertion of the 4 MOX LFAs produces an increase in each of the post LRA TEDEs. The largest relative increases with insertion of the MOX LFAs are to the EAB TEDEs. These increases are caused predominately by the projected increase in iodine release from the fuel pin gaps following the design basis LRA with MOX LFAs versus all LEU fuel.

Table 11 lists the TEDEs following the design basis REA scenarios with the source term including 4 MOX LFAs along with their constituents. The total post REA TEDEs for a source term including the gap activities of the MOX LFAs versus a source term associated with all LEU fuel are compared in Table 12. Insertion of the four MOX LFAs produces increases in all post REA TEDEs as it does for the design basis LRA.

Conclusions

Analyses of the effects of insertion of the MOX LFAs on the radiological consequences of the design basis LOCA, LRA, and REA have been completed with the method of AST. To perform this comparison, baseline analyses of the design basis LRA and REA were completed. In all cases, the TEDEs were found to fall below the germane regulatory limits with significant margins. The increase in post accident radiation doses with insertion of the 4 MOX LFAs are small, even with assumptions made to increase to a maximum the potential effects.

The following question was asked pertaining to the analysis of backleakage through Engineered Safety Features (ESF) systems to the Refueling Water Storage Tank (RWST):

One of the sources of release of elemental iodine to the environment is Refueling Water Storage Tank (RWST) - Appendix A, Attachment 3.

The mechanism, as I understand it, consists of the following phases:

- *Sump water back leaks into the RWST (20 gpm).*
- *Because of pH control the sump water contains most of its iodine in ionic form (I-) and only insignificant amount in elemental form (I2).*
- *The backleaking sump water comes in contact with the RWST water which is acidic containing boric acid and not any basic chemicals.*
- *The pH of the combined RWST and sump water will be slightly acidic and this produces conversion of some ionic iodine into elemental iodine.*
- *Some of this elemental iodine is released from liquid phase to the gas environment.*

If this is the mechanism for radioactive iodine release from the RWST, the following parameters will play controlling role.

Parameter	***Its Value and Source***
.Backleakage of sump water	*20 gpm - Enclosure 5*
.Iodine inventory in the sump	*85.06 moles*
.Sump water pH	*Figure 1 in Enclosure 7*
.Amount of water in the RWST	*RWST at low level setpoint? Enclosure 5*
.Boric acid concentration in the RWST	*3075 ppm.*
.Resultant pH in the RWST	*???*
.Fraction of I- converted to I2 in the RWST	*???*
.Partition coefficient for I2 in the RWST	*???*
.Iodine release fraction from the RWST	*Enclosure 6*

Response: ESF leakage to the RWST was postulated and evaluated not only for the design basis LOCA but also for the design basis rod ejection accident (REA). Separate responses are given regarding these design basis accidents.

Design Basis LOCA

The processes analyzed for the LOCA were completed as outlined in this question; the understanding is correct. The values of the parameters in the table are correct except that the amount of iodine in the sump actually used in the analysis was 85.12 moles. The water assumed to be in the RWST at the start of cold leg recirculation does correspond to the low level setpoint. Tables 13A and 13B show the pH of the solution in the RWST, amount of iodine in the RWST, amount of diatomic iodine or I_2 in the RWST, and I_2 partition coefficient at selected times after the initiating event. The fraction of iodine converted to I_2 is found by dividing the amount of I_2 in the RWST by the amount of total iodine in the RWST.

Design Basis REA

No ice melt was assumed as discussed in the baseline evaluation of the design basis REA provided in the response to the first question. Specifically, the sodium concentration in the entrained leakage was set to 0 ppm while the boron concentration was set to 3075 ppm (the upper limit assumed for boron concentration in the reactor coolant and the water in the cold leg accumulators and the RWST). The iodine inventory in the sump for this design basis accident was calculated to be 10.64 moles. To provide additional margin to the regulatory acceptance criteria for TEDEs for the design basis REA, the rate of ESF backleakage to the RWST was set to 10 gpm. All of the other input values are as noted above. The pH of the solution, inventory of iodine and I_2 in the RWST, and the I_2 partition coefficient for ESF leakage to the RWST following a design basis REA are shown in Tables 13C and 13D.

REFERENCES

1) H.B. Barron (Duke Energy Corporation) to U.S. Nuclear Regulatory Commission, "Catawba Nuclear Station Units 1 and 2; Dockets Nos. 50-413, 50-414 McGuire Nuclear Station; Docket Nos. 50-369, 50-370 Response to Request for Additional Information Regarding the Use of Mixed Oxide Lead Fuel Assemblies," November 3, 2003.

2) G.R. Peterson to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 (Docket Nos. 50-413 and 50-414, Proposed Technical Specifications and Bases Amendment, Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS), Technical Specification and Bases 3.6.16 Reactor Building, Technical Specification and Bases 3.7.10 Control Room Area Ventilation System (CRAVS), Technical Specification and Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), Technical Specification and Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES), Technical Specification and Bases 3.9.3 Containment Penetrations, Technical Specification 5.5.11 Ventilation System Testing Program," November 25, 2002.

3) Robert E. Martin (USNRC) to D.M. Jamil (Duke Energy Corporation), "Catawba Nuclear Station, Units 1 and 2 Re: Request for Additional Information (TAC Nos. MB7014 and MB7015)," September 11, 2003.

4) D.H. Jamil to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 (Docket Nos. 50-413 and 50-414, Proposed Technical Specifications and Bases Amendment, Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS), Technical Specification and Bases 3.6.16 Reactor Building, Technical Specification and Bases 3.7.10 Control Room Area Ventilation System (CRAVS), Technical Specification and Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), Technical Specification and Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES), Technical Specification and Bases 3.9.3 Containment Penetrations, Technical Specification 5.5.11 Ventilation System Testing Program, TAC Numbers MB7014 and MB7015," November 13, 2003.

5) D.H. Jamil to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station, Units 1 and 2 (Docket Nos. 50-413 and 50-414, Proposed Technical

Specifications and Bases Amendment, Technical Specification and Bases 3.6.10 Annulus Ventilation System (AVS), Technical Specification and Bases 3.6.16 Reactor Building, Technical Specification and Bases 3.7.10 Control Room Area Ventilation System (CRAVS), Technical Specification and Bases 3.7.12 Auxiliary Building Filtered Ventilation Exhaust System (ABFVES), Technical Specification and Bases 3.7.13 Fuel Handling Ventilation Exhaust System (FHVES), Technical Specification and Bases 3.9.3 Containment Penetrations, Technical Specification 5.5.11 Ventilation System Testing Program, TAC Numbers MB7014 and MB7015," December 16, 2003.

6) K.F. Eckerman, A.B. Wolbarst, A.C.B. Richardson, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal Guidance Report No. 11, 1988.

7) K.F. Eckerman and J.C. Ryman, External Exposure to Radionuclides in Air, Water, and Soil, Federal Guidance Report No. 12, 1993.

8) G.R. Peterson to U.S. Nuclear Regulatory Commission, "Proposed Amendment for Partial Scope Implementation of the Alternate Source Term and Proposed Amendment to Technical Specifications (TS) 3.7.10, Control Room Area Ventilation System, TS 3.7.11, Control Room Area Chilled Water System, TS 3.7.13, Fuel Handling Ventilation Exhaust System, and TS 3.9.3, Containment Penetrations," December 20, 2001.

9) C.P. Patel (USNRC) to G.R. Peterson, "Catawba Nuclear Station, Units 1 and 2, RE: Issuance of Amendments (TAC Nos. MB3758 and MB3759)," April 23, 2002.

10) USNRC, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Regulatory Guide 1.183, July 2000.

11) W.R. McCollum to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station Units 1 & 2, Docket Nos. 50-413, 50-414 Proposed Amendment to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (Revised Dose Inputs)," September 20, 2004.

12) R.E. Martin to H.B. Barron, "Catawba Nuclear Station, Units 1 and 2 - Request for Additional Information Concerning Mixed Oxide Fuel Lead Test Assemblies (TAC Nos. MB7863 and MB7864)," October 7, 2004.

13) W.R. McCollum to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation Catawba Nuclear Station Units 1 & 2, Docket Nos. 50-413, 50-414 Proposed Amendment to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (Response to Request for Additional Information on Dose Inputs)," October 29, 2004.

14) Catawba Nuclear Station Technical Specifications, with Amendments Through 220/215.

15) Oak Ridge National Laboratory, SCALE: A Modular Code System for Performing Standardized Analyses for Licensing Evaluation, NURGE/CR-0200, USNRC, March 1997.

16) Duke Energy Corporation, Multidimensional Reactor Transients and Safety Analysis Physics Parameters, Duke Topical Report DPC-NE-3001-PA (Rev 0), November 15, 1991.

17) D.A. Powers and K.E. Washington (Sandia National Laboratories), S.B. Burson (USNRC), and J.L. Sprung (SNL), A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments, NUREG/CR-6189 (SAND 94-0407), July 1996.

18) Ref. 10 Op. Cit., Appendix A.

19) Ibid., Appendix E.

20) Ibid., Appendix F.

21) Ibid., Appendix G.

22) Ibid., Appendix H.

23) M.S. Tuckman (Duke Energy Corporation) to U.S. Nuclear Regulatory Commission, "Catawba Nuclear Station Units 1 and 2; Dockets Nos. 50-413, 50-414 McGuire Nuclear Station; Docket Nos. 50-369, 50-370 Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Lead Fuel Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50," February 27, 2003.

Table 1
Radioactivity Levels in a MOX LFA

Noble Gases		Tellurium Group		Noble Metals	
Kr83m	6.25E+04	Se81	3.36E+04	Mo99	1.24E+06
Kr85m	1.17E+05	Se83	3.32E+04	Mo101	1.19E+06
Kr85	4.36E+03	Se83m	2.79E+04	Mo102	1.20E+06
Kr87	2.18E+05	Se84	1.01E+05	Tc99m	1.11E+06
Kr88	2.93E+05	Se87	4.97E+04	Tc101	1.19E+06
Kr89	3.33E+05	Sb127	9.00E+04	Tc104	1.16E+06
Xe131m	8.93E+03	Sb128	1.63E+04	Ru103	1.24E+06
Xe133m	4.71E+04	Sb128m	1.32E+05	Ru105	1.04E+06
Xe133	1.35E+06	Sb129	2.82E+05	Ru106	6.94E+05
Xe135m	3.38E+05	Sb130	1.06E+05	Ru107	6.71E+05
Xe135	7.73E+05	Sb130m	2.97E+05	Rh103m	1.24E+06
Xe137	1.23E+06	Sb131	5.11E+05	Rh105	9.62E+05
Xe138	1.06E+06	Sb132m	2.74E+05	Rh106m	2.53E+04
		Te127m	1.30E+04	Rh107	6.73E+05
Halogens		Te127	8.21E+04	Pd109	4.53E+05
		Te129	2.51E+05	Pd111	6.34E+04
Br83	6.25E+04	Te129m	4.78E+04	Pd112	2.75E+04
Br85	1.17E+05	Te131	6.00E+05		
Br87	1.61E+05	Te132	1.04E+06	**Cerium Group**	
I130	2.82E+04	Te133	6.93E+05		
I131	7.21E+05	Te133m	5.82E+05	Ce141	9.79E+05
I132	1.08E+06	Te134	1.04E+06	Ce143	9.06E+05
I133	1.39E+06			Ce144	6.43E+05
I134	1.48E+06	**Alkali Earth Metals**		Ce145	6.22E+05
I135	1.31E+06			Ce146	5.13E+05
		Sr89	3.37E+05	Np237	6.95E-02
Alkali Metals		Sr90	3.25E+04	Np238	1.03E+05
		Sr91	5.51E+05	Np239	1.36E+07
Rb86	1.15E+03	Sr92	6.45E+05	Np240	4.60E+04
Rb88	3.02E+05	Sr93	7.90E+05	Pu236	2.40E-01
Rb89	3.82E+05	Ba139	1.14E+06	Pu238	2.08E+03
Rb90	3.21E+05	Ba140	1.09E+06	Pu239	1.29E+03
Cs134	1.93E+05	Ba141	1.05E+06	Pu240	9.77E+02
Cs136	6.89E+04	Ba142	9.44E+05	Pu241	2.69E+05
Cs137	9.35E+04			Pu242	4.96E+00
Cs138	1.19E+06			Pu243	9.63E+05
Cs139	1.09E+06				

Table 1, Continued
Radioactivity Levels in a MOX LFA

Lanthanides		Lanthanides, Cont'd		Lanthanides, Cont'd	
Y90	3.42E+04	La143	9.00E+05	Pr142	5.81E+04
Y91	4.86E+05	Nd147	4.07E+05	Pr143	8.55E+05
Y91m	3.20E+05	Nd149	2.64E+05	Pr144	6.49E+05
Y92	6.49E+05	Nd151	1.56E+05	Pr144m	9.03E+03
Y93	5.44E+05	Pm147	9.21E+04	Pr145	6.23E+05
Y94	9.11E+05	Pm148	1.15E+05	Pr146	5.20E+05
Y95	9.78E+05	Pm148m	1.85E+04	Pr147	4.24E+05
Zr95	8.90E+05	Pm149	3.90E+05	Am241	4.40E+02
Zr97	1.05E+06	Pm151	1.56E+05	Am242m	3.56E+01
Nb95	8.81E+05	Sm153	4.66E+00	Am242	2.10E+05
Nb95m	9.88E+03	Sm156	2.36E+04	Am243	8.28E+01
Nb97	1.06E+06	Eu154	1.26E+04	Am244	2.03E+05
La140	1.11E+06	Eu155	3.99E+03	Cm242	8.48E+04
La141	1.07E+06	Eu156	3.45E+05	Cm244	5.41E+03
La142	1.01E+06	Eu157	3.48E+04		

Table 2
Iodine Partition Fractions for
ESF Leakage in the Auxiliary Building
Following a Design Basis LOCA Postulated for a
Catawba Nuclear Unit with 4 MOX LFAs

End of Time	Iodine Partition Factors				
Interval (hr)	**Note 1**	**Note 2**	**Note 3**	**Note 4**	**Note 5**
Note 6	0.100	0.100	0.010	0.014	0.100
72	0.022	0.028	0.010	0.010	0.024
720	0.010	0.010	0.010	0.010	0.010

Notes:

1) This scenario involves ESF System leakage in a room to which Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) is aligned. In the limiting cases, filtered ESF leakage also is upstream of the Residual Heat Removal - RHR - and Containment Spray - CSS - Heat Exchangers. This scenario also includes a design basis LOCA with Minimum Safeguards. One ABFVES train is affected by the failure and is not available.

2) This scenario also involves ESF System leakage in a room to which the ABFVES is aligned and also upstream of the RHR and CSS Heat Exchangers. This scenario includes a design basis LOCA with either a failure of a pressure transmitter of the Annulus Ventilation System (AVS), failure of a RHR or CSS Heat Exchanger to remove heat, or closed Control Room Area Ventilation System (CRAVS) Outside Air Intake. For these scenarios, all ABFVES trains are in operation.

3) This scenario includes ESF System leakage in the Mechanical Penetration Room (MPR) at EL 577 (downstream of the RHR and CSS Heat Exchangers). The ABFVES is not aligned to this room following the LOCA and Safety Injection Signal. Credit is taken for the operators aligning the ABFVES to this room three (3) days after the LOCA. Therefore, ESF System leakage in this room is not filtered for the first 3 days following the initiating event. This scenario includes a design basis LOCA with Minimum Safeguards. One ABFVES train is not available.

4) This scenario includes ESF System leakage in the MPR at EL 577. The ABFVES is not aligned to this room for the first 3 days. The scenario also includes a design basis LOCA with AVS pressure transmitter failure or closed CRAVS Outside Air Intake. All ABFVES trains are in operation. The baseline value for the iodine partition fractions here were 0.013, 0.010, and 0.010. They were reported as 0.010, 0.010, and 0.010.

5) The ESF System leakage occurs in the MPR at EL 577 and is not filtered before release to the environment for the first 3 days. The scenario includes a design basis LOCA with failure of a RHR or CSS Heat Exchanger. All ABFVES trains are in operation.

6) This time period ends at 2.5 hr after the initiating event for Cases 1 and 2 and 2.9 hr after the initiating event for Cases 3, 4,and 5.

Table 3
Total Effective Dose Equivalents (TEDEs) Following a Design Basis LOCA at Catawba Nuclear Station
All LEU Fuel Assemblies vs 4 MOX LFAS

Table 3A
EAB TEDEs

CNS DB LOCA Scenario	Radioactivity Release Path	EAB TEDEs (Rem)		Difference
		All LEU Fuel	4 MOX LFAs	%
CNS DB LOCA, Minimum Sfgds failure	Cont Leakage	3.04	3.04	-0.1
	ESF Leakage	0.50	0.51	+2.0
	Total	3.54	3.55	+0.2
CNS DB LOCA, AVS pressure xmtr failure	Cont Leakage	3.96	3.97	+0.3
	ESF Leakage *	0.58	0.62	+6.9
	Total	4.54	4.59	+1.2
CNS DB LOCA, RHRS or CSS HX failure	Cont Leakage	2.68	2.67	-0.1
	ESF Leakage	2.74	2.79	+2.0
	Total	5.41	5.46	+0.9
CNS DB LOCA, init closed CRAVS intake	Cont Leakage	2.68	2.67	-0.1
	ESF Leakage *	0.58	0.62	+6.9
	Total	3.26	3.29	+1.1

Table 3B
LPZ TEDEs

CNS DB LOCA Scenario	Radioactivity Release Path	LPZ TEDEs (Rem)		Difference
		All LEU Fuel	4 MOX LFAs	%
CNS DB LOCA, Minimum Sfgds failure	Cont Leakage	1.83	1.84	+0.4
	ESF Leakage	0.53	0.54	+1.9
	Total	2.36	2.38	+0.7
CNS DB LOCA, AVS pressure xmtr failure	Cont Leakage	1.92	1.93	+0.4
	ESF Leakage *	0.55	0.56	+2.9
	Total	2.47	2.49	+1.0
CNS DB LOCA, RHRS or CSS HX failure	Cont Leakage	1.71	1.71	+0.4
	ESF Leakage	1.40	1.43	+2.0
	Total	3.11	3.14	+1.1
CNS DB LOCA, init closed CRAVS intake	Cont Leakage	1.71	1.71	+0.4
	ESF Leakage *	0.55	0.56	+2.9
	Total	2.25	2.28	+1.0

Table 3C
Control Room TEDEs

CNS DB LOCA Scenario	Radioactivity Release Path	Control Room TEDEs (Rem)		Difference
		All LEU Fuel	4 MOX LFAs	%
CNS DB LOCA, Minimum Sfgds failure	Cont Leakage	1.24	1.24	-0.2
	ESF Leakage	0.25	0.25	+2.0
	Total	1.49	1.49	+0.1
CNS DB LOCA, AVS pressure xmtr failure	Cont Leakage	1.31	1.31	-0.2
	ESF Leakage *	0.25	0.26	+2.8
	Total	1.56	1.57	+0.3
CNS DB LOCA, RHRS or CSS HX failure	Cont Leakage	1.16	1.16	-0.3
	ESF Leakage	0.64	0.66	+2.0
	Total	1.81	1.82	+0.5
CNS DB LOCA, init closed CRAVS intake	Cont Leakage	1.80	1.79	-0.4
	ESF Leakage *	0.33	0.34	+3.1
	Total	2.13	2.13	+0.1

Note *: The TEDE constituents for ESF leakage for these design basis LOCA scenarios are revised from the original values originally submitted. (Cf. Ref. 2, Appendix A, Enclosure 13, Page A-78.)

Table 4
Data Pertaining to the Radioactive Source Terms
For the Design Basis LRA and REA At Catawba Nuclear Station

Table 4A
Data Pertaining to the Initial Reactor Coolant Source Term

Parameter	Value	Ref./Notes
Equilibrium reactor coolant gross gamma activity	100/E-Bar uCi/gm	Ref. 14 SR 3.4.16.1
Equilibrium DEI reactor coolant specific activity	1 uCi/gm	Ref. 14 SR 3.4.16.2
Reactor coolant mass		
Unit 1	537,793 lbm	Ref. 13
Unit 2	481,637 lbm	Ref. 13
Concurrent iodine spike for		
LRA	Yes	
REA	No	
Conc iodine spike parameters		
Letdown flow	125 gpm	Note 1
Letdown flow density	62.4 lbm/cu.ft.	Note 2
Reactor coolant leak rate	11 gpm	Ref. 14 LCO 3.4.13.b & LCO 3.4.13.c
RCS leakage density	62.4 lbm/cu.ft.	Note 2
Spike multiplier	335	Note 3

Table 4B
Data Pertaining to the Secondary Systems Source Terms

Parameter	Value	Ref./Notes
Equilibrium DEI SG secondary coolant specific activity	0.1 uCi/gm	Ref. 14 LCO 3.7.17
SG iodine partition factor	100	Ref. 10 App E
Main condenser scrubbing efficiency	100%	Note 4
Initial SG secondary side water mass		
Unit 1	112,000 lbm	
Unit 2	77,300 lbm	
Initial condensate grade water inventory	1×10^{12} lbm	Note 5
Auxiliary Feedwater flow rate		
Unit 1	1894 lbm/min	
Unit 2	1896 lbm/min	

Auxiliary Feedwater flow density	62.4 lbm/cu.ft.	Note 2
Time after the initiating event the condensate grade sources are emptied	2.7 hr.	

Table 4C
Data Pertaining to the DNB Reactor Coolant Source Term

Parameter	Value	Ref./Notes
Fraction of fuel pins in DNB		
Design basis LRA	10%	Note 6
Design basis REA	50%	Ref. 13, 16
Post REA melted pins?	No	Ref. 13, 16
Radial peaking included?	Yes	Note 7
Number of fuel pins in a fuel assembly	264	
Number of fuel assemblies in the core	193	
Design Basis LRA gap fractions		Ref. 10 Table 3
I^{131}	0.08	
Kr^{85}	0.10	
Other noble gases	0.05	
Other halogens	0.05	
Alkali metals	0.12	
Design Basis REA gap fractions		Ref. 10 Table 3 and
Alkali metals	0.12	Footnote 11
All other isotopes	0.10	

Notes

1) It is assumed that two letdown lines are in operation.

2) Standard density is assumed.

3) The NRC endorses this value for use in simulating the accident initiated iodine for the design basis SGTR. Its application in the analysis of radiological consequences of the design basis LRA is justified in the text of this response.

4) Perfect scrubbing in the main condenser is assumed to calculate the upper bound of the activity of iodine radioisotopes in the condensate grade sources.

5) The inventory in the condensate grade sources is set to an arbitrarily high value. This ensures that no significant dilution of the specific activity in the condensate grade source is computed given loss of activity with Auxiliary Feedwater flow.

6) The fraction of fuel pins in DNB for the design basis REA is set in this analysis.

7) Taking the limiting burnup dependent radial peaking for all failed fuel assemblies assumed is a conservative assumption in the current license basis for the design basis LRA (Ref. 13). The same conservative approach is used to develop the radioactive source term for the design basis REA where 50% of the fuel pins in DNB are assumed to fail. In the future Duke may develop a DNB source term for the design basis REA that is closer to the lowest upper bound (supremum) for the limiting source term given the large fraction of fuel pins assumed to enter in DNB for this design basis accident (50%).

Table 5
Limiting Radioactivity Levels in a LEU Fuel Assembly

Radio Isotope	Activity (Ci)	Radio Isotope	Activity (Ci)	Radio Isotope	Activity (Ci)
Noble Gases		Halogens		Alkali Metals	
Kr83m	1.27E+05	Br83	1.27E+05	Rb86	1.68E+03
Kr85m	2.85E+05	Br85	2.85E+05	Rb88	8.48E+05
Kr85	7.31E+03	Br87	4.72E+05	Rb89	1.13E+06
Kr87	5.86E+05	I130	2.52E+04	Rb90	1.07E+06
Kr88	8.29E+05	I131	7.52E+05	Cs134	1.91E+05
Kr89	1.07E+06	I132	1.11E+06	Cs136	4.16E+04
Xe131m	9.63E+03	I133	1.60E+06	Cs137	9.15E+04
Xe133m	4.88E+04	I134	1.86E+06	Cs138	1.59E+06
Xe133	1.57E+06	I135	1.52E+06	Cs139	1.51E+06
Xe135m	3.20E+05				
Xe135	4.14E+05				
Xe137	1.48E+06				
Xe138	1.52E+06				

Table 6
Radiation Doses Following a CNS Design Basis LRA
(All LEU Fuel)

Type of Radiation Dose	Radiation Dose (Rem)
CNS Unit 1 Design Basis LRA with LOOP	
EAB TEDE	0.95
LPZ TEDE	0.24
Control Room TEDE	0.24
CNS Unit 2 Design Basis LRA with LOOP	
EAB TEDE	1.63
LPZ TEDE	0.35
Control Room TEDE	0.43

Table 7
Time Constants and DFs for Natural Deposition
Following a CNS Design Basis REA

Time Step End Point (Hours)		**Time Constant**	**Decontamination**
Beginning	**End**	**hour^{-1}**	**Factor**
0.00	0.50	0.02801	1.0134
0.50	1.80	0.05713	1.0944
1.80	3.80	0.06502	1.3220
3.80	11.80	0.09151	1.3220
11.80	13.80	0.09146	3.9270
13.80	22.22	0.09146	3.9270
22.22	27.78	0.03770	8.2920
27.78	33.33	0.02770	8.2920
33.33	720.00	0.00000	1.0000

Table 8
Iodine Release Fractions And
Equivalent ESF Leak Rates to the Environment For
ESF Backleakage to the FWST Following a Design Basis REA

Time Span	FWST Iodine Release	Equivalent Unfiltered
(Hours)	Fractions	ESF Leak Rate (cfm)
CNS Design Basis REA and All LEU Fuel in the Source Term		
0 - 2	0.000E+00	0.000E+00
2 - 8	3.135E-06	4.191E-06
8 - 10	1.266E-05	1.692E-05
10 - 24	2.332E-05	3.117E-05
24 - 96	8.910E-04	1.191E-03
96 - 720	2.415E-02	3.228E-02
CNS Design Basis REA and 4 MOX LFAs in the Source Term		
0 - 2	0.000E+00	0.000E+00
2 - 8	3.200E-06	4.278E-06
8 - 10	1.292E-05	1.727E-05
10 - 24	2.379E-05	3.180E-05
24 - 96	9.069E-04	1.212E-03
96 - 720	2.433E-02	3.252E-02

Table 9
Radiation Doses Following the CNS Design Basis REA
(All LEU Fuel)

	TEDE (Rem)			
	SG	Containment	ESF	
Radiation Dose (TEDE) Type	**Releases**	**Leakage**	**Leakage**	**Total**
CNS Unit 1 Design Basis REA with Offsite Power Available				
EAB TEDE	1.30	2.62	0.64	3.91
LPZ TEDE	0.22	2.74	2.60	2.96
Control Room TEDE	0.21	1.46	1.31	1.67
CNS Unit 2 Design Basis REA with Offsite Power Available				
EAB TEDE	2.02	2.62	0.64	4.63
LPZ TEDE	0.33	2.74	2.60	3.07
Control Room TEDE	0.32	1.46	1.31	1.78

Table 10
Radiation Doses Following the CNS Design Basis LRA
Effect of Insertion of 4 MOX LFAs

	TEDE (Rem)	
Radiation Dose Type	**LEU**	**MOX**
Unit 1 Design Basis LRA		
EAB TEDE	0.95	1.02
LPZ TEDE	0.24	0.26
Control Room TEDE	0.24	0.26
Unit 2 Design Basis LRA		
EAB TEDE	1.63	1.77
LPZ TEDE	0.35	0.38
Control Room TEDE	0.43	0.46

Table 11
Radiation Doses Following the CNS Design Basis REA
(Four MOX LFAs in the Source Term)

	TEDEs (Rem)			
	SG	Cont	ESF	
Radiation Dose (TEDE) Type	Releases	Leakage	Leakage	Total
CNS Unit 1 Design Basis Rod Ejection Accident				
EAB TEDE	1.32	2.65	0.65	3.97
LPZ TEDE	0.22	2.79	2.66	3.01
Control Room TEDE	0.22	1.48	1.34	1.70
CNS Unit 2 Design Basis Rod Ejection Accident				
EAB TEDE	2.05	2.65	0.65	4.70
LPZ TEDE	0.34	2.79	2.66	3.13
Control Room TEDE	0.33	1.48	1.34	1.81

Table 12
Radiation Doses Following the CNS Design Basis REA
Comparison of All LEU Fuel vs Insertion of 4 MOX LFAs

	TEDEs (Rem)	
Type of TEDE	All LEU	4 MOX LFAs
CNS Unit 1 Design Basis REA		
EAB TEDE	3.91	3.97
LPZ TEDE	2.96	3.01
Control Room TEDE	1.67	1.70
CNS Unit 1 Design Basis REA		
EAB TEDE	4.63	4.70
LPZ TEDE	3.07	3.13
Control Room TEDE	1.78	1.81

Table 13
Data Pertaining to ESF Backleakage to the RWST
Following a Design Basis Accident

Table 13A
Design Basis LOCA with
Failure of Cooling Water Flow
Through a RHRS or CSS Heat Exchanger

	Solution pH	RWST Iodine	RWST I_2	
Time	In the RWST	Concentration	Concentration	I_2 Partition
Seconds	(Note 1)	Mole/Liter	Mole/Liter	Coefficient
0	4.30	0	0	0
790	4.30	0	0	45.4
810	4.30	2.32E-09	9.97E-14	45.4
900	4.33	1.25E-08	2.65E-12	45.4
1200	4.42	4.42E-08	2.44E-11	45.3
1400	4.47	6.36E-08	4.20E-11	45.3
1800	4.57	1.00E-07	7.29E-11	45.2
3600	4.91	2.56E-07	1.25E-10	44.9
4800	5.07	3.51E-07	1.20E-10	44.7
6000	5.20	4.42E-07	1.12E-10	44.5
7200	5.30	5.31E-07	1.05E-10	44.3
28800	5.94	2.05E-06	8.89E-11	41.2
36000	6.03	2.52E-06	8.91E-11	40.2
86400	6.34	5.47E-06	1.03E-10	34.8
345600	6.55	1.46E-05	2.81E-10	22.2
2592000	6.68	3.16E-05	7.17E-10	9.4

Table 13B
Design Basis LOCA with
No Failure of Cooling Water Flow
Through a RHRS or CSS Heat Exchanger

	Solution pH	RWST Iodine	RWST I_2	
Time	In the RWST	Concentration	Concentration	I_2 Partition
Seconds	(Note 1)	Mole/Liter	Mole/Liter	Coefficient
0	4.30	0	0	0
790	4.30	0	0	45.4
810	4.30	2.36E-09	1.03E-13	45.4
900	4.33	1.27E-08	2.72E-12	45.4
1200	4.42	4.48E-08	2.50E-11	45.4
1400	4.47	6.45E-08	4.28E-11	45.3
1800	4.57	1.01E-07	7.40E-11	45.3
3600	4.92	2.59E-07	1.25E-10	45.1
4800	5.08	3.56E-07	1.21E-10	45.1
6000	5.21	4.48E-07	1.12E-10	45.0
7200	5.30	5.39E-07	1.05E-10	44.9
28800	5.95	2.08E-06	8.83E-11	43.2
36000	6.04	2.56E-06	8.78E-11	42.7
86400	6.38	5.54E-06	8.94E-11	39.5
345600	6.74	1.47E-05	1.21E-10	31.1
2592000	7.01	3.20E-05	1.61E-10	19.7

Table 13C
Design Basis Rod Ejection Accident with
A Minimum Safeguards Failure (All LEU Fuel)

	Solution pH	RWST Iodine	RWST I_2	
Time	In the RWST	Concentration	Concentration	I_2 Partition
Seconds	(Note 2)	Mole/Liter	Mole/Liter	Coefficient
0	4.30	0	0	0
7200	4.30	0	0	45.4
28800	4.30	1.02E-07	1.93E-10	44.5
36000	4.30	1.34E-07	3.40E-10	44.2
86400	4.26	3.49E-07	2.54E-09	42.3
345600	3.97	1.17E-06	5.83E-08	35.7
2592000	3.18	3.45E-06	9.67E-07	22.1

Table 13D
Design Basis Rod Ejection Accident with
A Minimum Safeguards Failure (4 MOX LFAs)

	Solution pH	RWST Iodine	RWST I_2	
Time	In the RWST	Concentration	Concentration	I_2 Partition
Seconds	(Note 2)	Mole/Liter	Mole/Liter	Coefficient
0	4.30	0	0	0
7200	4.30	0	0	45.4
28800	4.30	1.04E-07	2.01E-10	44.5
36000	4.30	1.37E-07	3.54E-10	44.2
86400	4.26	3.57E-07	2.64E-09	42.3
345600	3.97	1.19E-06	6.05E-08	35.7
2592000	3.18	3.53E-06	9.93E-07	22.1

Notes on Table 13

1) All pH values are computed at the temperature of the solution in the RWST.

2) All pH values are computed at the temperature of the solution in the RWST, as stated in Note 1. Increasing temperature of the solution in the RWST produces the decrease in pH noted here.

3) The transfer to cold leg recirculation is assumed to begin at 790 sec after the initiating event for the design basis LOCA. For the design basis REA the transfer to cold leg recirculation is assumed to begin 7200 sec after the initiating event.

ATTACHMENT 2

REVISED MARKED UP TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

and System Bypass

ESF Ventilation System	Penetration	Flowrate
Annulus Ventilation (Unit 1)	< 1%	9000 cfm
~~Annulus Ventilation (Unit 2)~~	~~< 0.05%~~	~~9000 cfm~~
Control Room Area Ventilation	< 0.05%	6000 cfm
Aux. Bldg. Filtered Exhaust (Unit 1) (2 fans)	< 1%	30,000 cfm
~~Aux. Bldg. Filtered Exhaust (Unit 2)~~	~~< 0.05%~~	~~30,000 cfm~~
Containment Purge (non-ESF) (2 fans)	< 1%	25,000 cfm
Fuel Bldg. Ventilation (Unit 1)	< 1%	16,565 cfm
~~Fuel Bldg. Ventilation (Unit 2)~~	~~< 0.05%~~	~~16,565 cfm~~

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows the following penetration and system bypass when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the flowrate specified below ± 10%.

carbon

and System Bypass

ESF Ventilation System	Penetration	Flowrate
Annulus Ventilation (Unit 1)	< 1%	9000 cfm
~~Annulus Ventilation (Unit 2)~~	~~< 0.05%~~	~~9000 cfm~~
Control Room Area Ventilation	< 0.05%	6000 cfm
Aux. Bldg. Filtered Exhaust (Unit 1) (2 fans)	< 1%	30,000 cfm
~~Aux. Bldg. Filtered Exhaust (Unit 2)*~~	~~< 0.05%~~	~~30,000 cfm~~
Containment Purge (non-ESF) (2 fans)	< 1%	25,000 cfm
Fuel Bldg. Ventilation (Unit 1)	< 1%	16,565 cfm
~~Fuel Bldg. Ventilation (Unit 2)~~	~~< 0.05%~~	~~16,565 cfm~~

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Annulus Ventilation	< 4%	95%
Control Room Area Ventilation	< 0.95%	95%
Aux. Bldg. Filtered Exhaust (Note 1)	< 4%	95%
Containment Purge (non-ESF)	< 6%	95%
Fuel Bldg. Ventilation	< 4%	95%

INSERT 4

*The Penetration bypass acceptance criteria for the charcoal adsorber for the 2B ABFVES train is changed to 0.20%. This will remain in effect until the next Unit 2 refueling outage in the spring of 2006.

(continued)

INSERT 4 for TS 5.5.11c

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.

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

carbon

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the flowrate specified below ± 10%.

ESF Ventilation System	Delta P	Flowrate
Annulus Ventilation	8.0 in wg	9000 cfm
Control Room Area Ventilation	8.0 in wg	6000 cfm
Aux. Bldg. Filtered Exhaust (2 fans)	8.0 in wg	6 30,000 cfm
Containment Purge (non-ESF) (2 fans)	8.0 in wg	25,000 cfm
Fuel Bldg. Ventilation	8,0 in wg	16,565 cfm

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI N510-1980.

ESF Ventilation System	Wattage @ 600 vac
Annulus Ventilation	45 ± 6.7 kW
Control Room Area Ventilation	25 ± 2.5 kW
Aux. Bldg. Filtered Exhaust	40 ± 4.0 kW
Containment Purge (non-ESF)	120 ± 12.0 kW
Fuel Bldg. Ventilation	80 + 8/-17.3 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

(continued)

ATTACHMENT 3

REVISED NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

No Significant Hazards Consideration Analysis

The following discussion is a summary of the evaluation of the changes contained in this proposed amendment against the 10 CFR 50.92(c) requirements to demonstrate that all three standards are satisfied. A no significant hazards consideration is indicated if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated, or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated, or

3. Involve a significant reduction in a margin of safety.

First Standard

Does operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated? No.

This license amendment request proposes amendments to the system TS and/or Bases and/or VFTP TS requirements for the AVS, ABFVES, FHVES, and CRAVS. It also proposes amendments to the TS and Bases for Containment Penetrations. The AVS is in standby during normal plant operations and operates only following a Safety Injection signal or during a test. It is not an accident initiator. The ABFVES is in operation during normal plant operations. However, the ABFVES is not used in direct support of any phase of power generation or conversion or transmission, shutdown cooling, fuel handling operations, or processing of radioactive fluids. Therefore, it is not an accident initiator. The FHVES is utilized to support fuel handling operations when moving recently irradiated fuel. It is not an accident initiator. The CRAVS operates during normal plant operations. However, it is not an accident initiator (the CRAVS being defined so as to exclude equipment that maintains an appropriately low temperature in the control room). The status of containment penetrations is required to be controlled so as to minimize the consequences of a fuel handling accident or a weir gate drop accident. The containment penetrations by themselves are not accident initiators. No accident initiators are associated with the changes proposed in this license amendment request. For these reasons, operation of the facility in accordance with this proposed amendment does not involve a significant increase in the probability of any accident previously evaluated.

In support of the proposed amendment, an analysis has been performed to determine the radiological consequences of the design basis LOCA at Catawba Nuclear Station. The analysis made use of the Alternative Source Term (AST) methodology and in general conformed to the regulatory positions of Regulatory Guide 1.183 and the draft regulatory positions of DG-1111. Total Effective Dose Equivalent (TEDE) radiation doses at the Exclusion Area Boundary (EAB), boundary of the Low Population Zone (LPZ), and to the control room operators were calculated and found to be acceptable. TEDEs were calculated for a design basis LOCA postulated for a Catawba nuclear unit operating with all low enriched uranium (LEU) fuel and with 4 mixed oxide (MOX) lead fuel assemblies (LFAs). It was found that insertion of 4 MOX LFAs did not produce a significant increase in the TEDEs for a design basis LOCA.

TEDEs have been estimated from the radiation doses with the current analysis (reported in the letter dated December 10, 2004) using the guidelines of Regulatory Guide 1.183. These TEDEs are compared to the limiting TEDEs from the proposed analysis (letter of December 16, 2003 and this letter) as follows:

TEDEs Following the Design Basis LOCA

	TEDEs (Rem)		
	Equivalent	**Proposed**	
Location	**UFSAR**	**All LEU Fuel**	**4 MOX LFAs**
EAB	8.17	5.41	5.46
LPZ	1.03	3.11	3.14
Control Room	1.48	2.13	2.13

The new value for the control room TEDE radiation dose is higher than the TEDE radiation dose equivalent to the radiation doses currently reported in the UFSAR. However, the limiting control room TEDE radiation dose reported in this submittal is lower than the acceptance criterion by 57%. The new LPZ TEDE radiation dose is higher than the equivalent TEDE radiation dose currently represented. On the other hand, the margin to the acceptance criterion is 88%. The TEDE radiation doses newly computed at the EAB for the design basis LOCA are lower than the corresponding equivalent EAB TEDE radiation dose currently represented in the UFSAR. The margin in the EAB TEDE radiation dose to the guideline value is 78%. In all cases, there is significant margin between the newly calculated post-LOCA TEDE radiation doses and the corresponding regulatory guideline values. In the sense that the margins to the germane regulatory guideline values are still large, the new values of TEDE radiation doses are comparable to the equivalent TEDE associated with the post-LOCA radiation doses currently listed in the UFSAR. Furthermore, these margins for the design basis LOCA do not significantly decrease with insertion of the 4 MOX LFAs. Therefore, the proposed amendment is determined to not result in a significant increase in accident consequences.

AST analyses also were completed for the design basis locked rotor accident (LRA) and rod ejection accident (REA). Again, these design basis accidents were postulated to occur at a Catawba nuclear unit operating with either an all LEU core or with 4 MOX LFAs. The TEDEs following these design basis accidents were compared to the equivalent TEDEs associated with the current license basis analyses. The equivalent TEDEs were computed from the post-accident whole body and thyroid radiation doses using the method prescribed in Regulatory Guide 1.183, as noted above. TEDEs only at offsite locations were compared as post-accident control room radiation doses are not reported for these design basis accidents in the Catawba UFSAR.

TEDEs following the design basis LRA are presented as follows:

	TEDEs (Rem)		
	Equivalent	**Proposed**	
Location	**UFSAR**	**All LEU Fuel**	**4 MOX LFAs**
EAB	0.96	1.63	1.77
LPZ	0.19	0.35	0.38
Control Room	0.05	0.43	0.46

For the EAB, LPZ, and control room, the post-LRA TEDEs are seen to increase from the values equivalent to the radiation doses from the current license basis analyses. (This is attributed primarily to the increase in assumed fraction of the fuel pins with clad failure following a design basis LRA at Unit 2 from 5% to 10%.) However, the margins to the acceptance criteria of 2.5 Rem at the offsite locations and 5 Rem in the control room are still significant. The limiting margin, associated with the TEDE at the EAB, is 34% (29% with insertion of the 4 MOX LFAs).

TEDEs following the design basis REA are presented as follows:

	TEDEs (Rem)		
	Equivalent	Proposed	
Location	UFSAR	All LEU Fuel	4 MOX LFAs
EAB	1.33	4.63	4.70
LPZ	0.81	3.07	3.13
Control Room	0.38	1.78	1.81

For the EAB, LPZ, and control room, the post-REA TEDEs are seen to increase from the values equivalent to the radiation doses from the current license basis analyses, as they did for the design basis LRA. (This is attributed to a number of reasons. These include increase in the fraction of gap activity released to containment, inclusion of limiting radial peaking in the source term, and inclusion of alkali metals.) However, the margins to the acceptance criteria of 6.3 Rem at the offsite locations and 5 Rem in the control room are still significant. The limiting margin, associated with the TEDE at the EAB, is 27% (24% with insertion of the 4 MOX LFAs).

The changes proposed to the TS for Containment Penetrations are editorial in nature and will have no effect upon accident consequences.

The changes proposed to the VFTP TS for the AVS, ABFVES, and FHVES will not result in a significant increase in any accident consequences. The changes to make the penetration values for Unit 2 consistent with Unit 1 for the AVS, ABFVES, and FHVES are acceptable because the appropriate safety factors as delineated in the applicable regulatory guideline documents are still maintained. The change to the flowrate specified for the ABFVES is consistent with the design basis operation of this system. Also, the editorial changes proposed to the VFTP TS will have no impact on any accidents.

Operation of the facility in accordance with the proposed amendment does not involve a significant increase in the consequences of an accident previously evaluated.

<u>Second Standard</u>

Does operation of the facility in accordance with the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated? No.

This proposed amendment does not involve addition, removal, or modification of any plant system, structure, or component. These changes will not affect the operation of any plant system, structure, or components as directed in plant procedures.

The analysis performed in support of this license amendment request, together with the analyses of the design basis fuel handling accident and weir gate drop reported in previously submitted and NRC approved license amendment requests, includes full scope implementation of AST methodology. This analysis does not represent any change in the post-accident operation of any plant system, structure, or component.

Operation of the facility in accordance with this amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

<u>Third Standard</u>

Does operation of the facility in accordance with the proposed amendment involve a significant reduction in the margin of safety? No.

Margin of safety is related to confidence in the ability of fission product barriers to perform their design functions following any of their design basis accidents. These barriers include the fuel cladding, the Reactor Coolant System, and the containment. The performance of these barriers either during normal plant operations or following an accident will not be affected by the changes associated with the license amendment request.

The AVS is associated with the containment fission product barrier. Its post-accident operation will not be affected by implementation of the amendment to its TS. The operation of the ABFVES either during normal plant operations or following an accident will not be affected by implementation of the amendment to its TS. The operation of the FHVES either during normal plant operations or following an accident will not be affected by implementation of the amendment to its TS. The operation of the CRAVS either during normal plant operations or following an accident will not be adversely affected by the proposed changes to its TS Bases. The operation of Containment Penetrations following an accident will not be adversely affected by the proposed change to its TS.

As noted, an analysis of radiological consequences of the design basis LOCA at Catawba Nuclear Station has been performed in support of this license amendment request. The design basis LOCA scenarios were selected based on extensive evaluations of Catawba, its design basis, and its anticipated response to a design basis LOCA. Credit was taken only for safety related systems, structures, and components in simulating the mitigation of radiological consequences of the LOCA. Limiting values were taken for performance characteristics of the Class 1E systems modeled in the analysis. The radiological consequences (TEDE radiation doses at the EAB, LPZ, and in the control room) are within the regulatory guideline values with significant margin.

The changes proposed to the VFTP TS for the AVS, ABFVES, and FHVES will not result in a significant reduction in the margin of safety. These changes are supported by regulatory guidance documents, and are consistent with existing system operation. Also, the editorial changes proposed to the VFTP TS will not have any impact on safety.

Operation of the facility in accordance with the proposed amendment does not involve a significant reduction in the margin of safety.

Based upon the preceding discussion, Duke has concluded that the proposed amendment does not involve a significant hazards consideration.