October 7, 2004

Mr. H. B. Barron
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church Street
Charlotte, NC 28202

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION CONCERNING MIXED OXIDE FUEL LEAD TEST ASSEMBLIES (TAC NOS. MB7863 AND MB7864)

Dear Mr. Barron:

By letter to the U.S. Regulatory Commission (NRC) dated February 27, 2003, Duke Energy Corporation (Duke) submitted an application for amendments to the renewed facility operating licenses for Catawba Nuclear Station, Units 1 and 2 (Catawba). The proposed amendments would revise the Technical Specifications to allow the use of four mixed oxide lead test assemblies at Catawba.

By letter to the NRC dated August 31, 2004, Duke indicated that certain information previously provided to describe the radiological consequences of accidents was based on out-of-date values. Duke's letter dated September 20, 2004, provided additional information regarding this concern. The NRC staff has reviewed this information and has identified a need for additional information as indicted in the enclosure..

Please contact me at (301) 415-1493 if you have any other questions on these issues.

Sincerely,

/RA by MJRoss-Lee for/

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure: As stated

cc w/encl: See next page

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DATE	10/05/04	10/05/04	09/16/04	09/16/04	10/07/04

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Catawba Nuclear Station, Units 1 & 2

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REQUEST FOR ADDITIONAL INFORMATION

ON APPLICATION FOR MIXED OXIDE (MOX) LEAD TEST ASSEMBLIES (LTAs)

DUKE POWER COMPANY

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

Dose Consequences Calculation

1. Duke's letter dated September 20, 2004, states that:

> The Responses to Radiological Questions in Reference 2 addressed doses from design basis accidents postulated to occur in cores containing four MOX fuel lead assemblies. Evaluations were performed for accidents in which a substantial number of fuel rods are assumed to fail, i.e., loss of coolant accident (LOCA), rod ejection accident (REA), and locked rotor accident (LRA). These accidents were characterized by the total number of rods in the four MOX fuel lead assemblies being a relatively small fraction of the number of rods that are assumed to fail and release radionuclides during the accident. The method used for the evaluations . . . involves increasing the baseline LEU [low enriched uranium] core dose for the accident by a factor to account for (i) higher iodine-131 initial inventory in the MOX fuel rods, (ii) a higher assumed fission gas gap release fraction, and (iii) the fraction of failed fuel rods that could come from a MOX fuel assembly. . . .

   Duke's letter continues by noting that the baseline LEU doses were based on outdated information from the Updated Final Safety Analysis Report (UFSAR) and that the problem affects only the offsite and control room (CR) doses for the REA and LRA and the CR doses for the LOCA.

   The specific action or lack of action resulting in the failure to update the UFSAR is not provided in Duke's September 20, 2004, letter. Please provide a detailed discussion of the specific action or lack of action which resulted in the use of outdated information. The information provided should address the basis for Duke's claim regarding the scope of the analyses that were affected.

2. The first and second sentences in the second paragraph on page 2 appear to conflict with each other with respect to whether the CR LOCA dose requires revision. Please provide a clarification.

   Is the root cause of the need to revise the CR dose analysis the same as the root cause discussed in Item 1 above?

3. Pages 3 and 4 of Duke's letter describe a process for the REA and LRA wherein bounding values for the number of failed fuel rods, and thus doses, are chosen that will encompass a

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period of multiple fuel cycles while a cycle-specific assessment made for the LRA analysis for Cycle 16 shows that no fuel failures will result. Please expand this discussion to include the expected cycle-specific results for the REA and the LRA for the specific fuel cycles that the MOX LTAs are planned to be used in. The response should be based on assessments using approved licensing-basis methodology and inputs.

4. Please provide a third column in the tables in Attachments 2 and 3 that indicates what value was used in the dose consequence analyses results previously submitted to the NRC in support of the February 27, 2003, MOX application or clarify whether the first column represents that information.

5. Please provide the basis for each of the indicated changes in the tables for Attachments 2 and 3. This could be done by an expansion of the notes.

   For example, Table 1A indicates that initial core power was changed from 3565 to 3479 megawatts thermal. What was the basis for this change - a previously submitted change to a licensing-basis analysis that was approved by the NRC staff?

6. Please provide an assessment of the relative effects of the changes indicated in the tables in Attachments 2 and 3 on the resulting calculated doses.

7. For the LRA, on page 3 of the letter, Duke states that Units 1 and 2 fuel failures are 9.5 percent and 5 percent, respectively. However, Duke provides a discussion that states that fuel failures are not projected at all for this cycle, and provides a commitment on page 7 to do a cycle-by-cycle analysis to confirm no failure. Duke is requested to clarify whether it proposes the bounding analysis or the cycle-specific analysis as the licensing-basis analysis. Considering that the updated LEU-MOX scaling analyses assumed 9.5 percent and 5 percent failures, the commitment to monitor for zero failures appears to be unnecessary. Please clarify.

8. Regarding Table 1A, Page 3 of Attachment 2:

   (a) Please explain how "SG [steam generator] leak rate time dependence included" is modeled in the calculation. Please provide a table of the leakage values assumed versus time. If the assumed leakage into each SG is different, please tabulate for each SG as well.

   (b) The values for control room area ventilation system (CRAVS) CR outside airflow rate (2667) and CRAVS CR recirc airflow rate (1333) differ from the corresponding values used in the LOCA CR dose re-analysis discussed in Duke's Attachment 3. Please explain why the 2000 cfm flow rate used in the LOCA analysis is conservative given the corresponding value of 1333 cfm used in previous analyses (i.e., filter mitigation proportional to flow).

9. Regarding Table 1B, Page 4 and Table 1C, Page 6 of Attachment 3, please explain the purpose of the "Total for Unit 1" and "Total for Unit 2" entries as used in the modeling. Were the individual SG uncovery times used in a model that treated the four SGs separately, or were the four SGs modeled as a single "node" with an uncovery time equal to the total?

10. Please explain Footnotes 18 and 19 on Page 8 of Attachment 2. It is not clear why the footnote references the CRAVS when the table was discussing the annulus ventilation system (AVS). Is this a typographical error?

11. Please provide the values for the reactor coolant system mass and SG masses assumed in the analyses.

Meteorology

1. Regarding Table 1 of Attachment 2 to the September 20, 2004, letter:

   (a) Is the exclusion area boundary (EAB) χ/Q value for the limiting 2-hour time period and are the low population zone (LPZ) and CR χ/Q values for the 0 - 8 hour period?

   (b) Does the REA analysis sequence cease at 8 hours so no other χ/Q values are needed or were these values used for the duration of the accident because they are the most limiting? If other values were used, provide a list of the values and a reference citing when the values were previously approved by the NRC or provide a description of the methodology, inputs and assumptions used to calculate the values.

2. Regarding the LOCA analysis for the CR:

   (a) Is the CR χ/Q value in Table 1 fifty percent of the 0 - 8 hour value in the table on Page 3 of Attachment 3 for the LOCA?

   (b) Are all releases for the LOCA, REA and LRA from the same release location?

   (c) Is/are the χ/Q value(s) for this location the most limiting value(s)? If not, what other χ/Q values were used? Provide a reference citing when the values were previously approved by the NRC or a description of the methodology, inputs and assumptions used to calculate the values.

3. (a) Item 10 of the Table 1 Notes states, “The control room χ/Q value assumes that both intakes are open because no common mode failure could cause an outside air intake valve to close and fail an Auxiliary Feedwater System (AFWS) pump (the limiting failure).” Please explain the relationship between failure of a CR air intake and an AFWS pump.

   (b) Does Note 10 address the discussion on Page 15.6-21 of the Catawba UFSAR, (revision date, March 27, 2003), by saying that a single failure of a CR air intake cannot occur for either the REA or LRA? If so, provide further information on the relationship between the CR air intakes and AFWS pumps explaining why. If not, provide justification for reduction of the CR χ/Q value and for use of 50 percent of the limiting χ/Q value rather than 60 percent, as previously discussed with the staff.

4. Regarding the table on Page 3 of Attachment 3 to the September 20, 2004, letter:

   (a) Provide a reference citing when the χ/Q values were previously approved by the NRC staff or a description of the inputs and assumptions used with the Murphy-Campe methodology to calculate the values.

   (b) The table does not list the EAB and LPZ χ/Q values used in the LOCA dose assessment. Were the values used in the Catawba UFSAR or those listed in Table 1A, referenced above? If the UFSAR values were used, why are they appropriate for the LOCA, when they may not be necessary for the REA and LRA?

Reactor Systems

1. Duke's letter indicated that the baseline LEU doses were based on outdated information from the UFSAR and that the problem affects only the offsite and CR doses for the REA and LRA and the CR doses for the LOCA. In order to confirm that this outdated information in UFSAR does not affect the results of the licensing analyses (thermal-hydraulic) of the REA and LRA for supporting the assumptions used in dose calculation for these events, please respond to the following questions.

   (a) Identify any out of date information in Section 15.3.3 (LRA) and Section 15.4.8 (REA) of the current UFSAR and discuss their effects on the results of the analyses for these events.

   (b) Provide detailed results of the cycle-specific accident analyses for the LRA and REA considering the following: a) the calculated number of fuel pins experiencing departure from nucleate boiling; and b) SG water levels during the transients and the total time period while the SG tubes are not covered by water such that all fission products entrained with primary to secondary leakage are released to the environment.

   (c) Confirm that the analyses in Item (b) above were performed using NRC-approved methodology. Discuss the review and approval status of these cycle-specific analyses.

   (d) The current dose assessment assumes 50 percent fuel failure for the REA, and assumes 9.5 percent (Unit 1) and 5 percent (Unit 2) fuel failure for the LRA. Discuss the commitment made to perform new dose assessments when the future cycle-specific analyses result in the calculated amount of fuel failure exceeding the assumptions used in the dose calculations.