Duke Power
A Duke Energy Company

Duke Power
Energy Center
P.O. Box 1006
Charlotte, NC 28201-1006

March 1, 2004

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

Subject: Duke Energy Corporation
Catawba Nuclear Station Units 1 & 2, Docket Nos. 50-413, 50-414
Response to Request for Additional Information (TAC Nos. MB7863, MB7864)
Mixed Oxide Fuel Lead Assemblies (Radiological)

By letter dated February 27, 2003 Duke Energy submitted an application to amend the licenses of McGuire and Catawba to allow the use of four mixed oxide fuel lead assemblies. As part of the review of this application the Nuclear Regulatory Commission staff requested that Duke provide additional information in a letter dated February 4, 2004. The responses to these questions are included in Attachment 1. Inquiries on this correspondence should be directed to M.T. Cash at (704) 382-5826.

H.B Barron
Executive Vice President – Nuclear Generation
Duke Energy Corporation

A001

attachments

cc: w/attachments

L. A. Reyes
U. S. Nuclear Regulatory Commission Regional
Administrator, Region II Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

R. E. Martin (addressee only)
NRC Project Manager
U. S. Nuclear Regulatory Commission
Mail Stop O-8G9
Washington, DC 20555-0001

E. F. Guthrie
Senior Resident Inspector
U. S. Nuclear Regulatory Commission
Catawba Nuclear Station

J. B. Brady
Senior Resident Inspector
U. S. Nuclear Regulatory Commission
McGuire Nuclear Station

Diane Curran
Harmon, Curran, Spielberg & Eisenberg, LLP
1726 M Street, N.W.
Suite 600
Washington, DC 20036

Mary Olson
Director, Southeast Office
Nuclear Information and Resource Service
P.O. Box 7586
Asheville, NC 28802

H. J. Porter, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental Control
Columbia, SC 29201

bcc: w/attachments

Richard Clark-DCS
Patrick Rhoads-DOE
David Alberstein-DOE
Don Spellman-ORNL
NCMPA-1
NCEMC
PMPA
SRE

bcc: w/attachment (via email)

S. P. Nesbit
M. T. Cash
F. J. Verbos
J. L. Eller
S. P. Schultz
L. F Vaughn
M. W. Scott
L. J. Rudy
J. Hoerner – Framatome ANP
G. A. Meyer – Framatome ANP

bcc: w/attachments (paper copy)

NRIA File/ELL - EC05O
MOX File 1607.2304
Catawba Document Control File 801.01– CN04DM
Catawba RGC Date File (J. M. Ferguson – CN01SA)

Oath and Affirmation

I affirm that I, HB Barron, am the person who subscribed my name to the foregoing, and that all the matters and facts set forth herein are true and correct to the best of my knowledge.

HB Barron

HB Barron

Subscribed and sworn to before me on this 1st day of March 2004.

Michael T. Cash

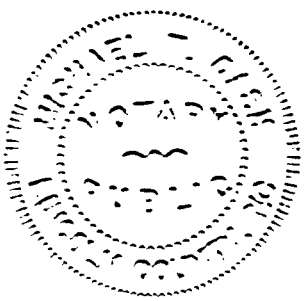
Notary Public

My Commission expires:

January 22, 2008

Date

MICHAEL T. CASH
Notary Public
Lincoln County, North Carolina
Commission Expires January 22, 2008



Attachment 1

REQUEST FOR ADDITIONAL INFORMATION

ON APPLICATION FOR MOX LEAD TEST ASSEMBLIES

DUKE POWER COMPANY

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

Radiological Consequences

1. In its submittal, the licensee established an increase of 9 percent in the iodine-131 (I-131) inventory of a mixed oxide (MOX) lead test assembly (LTA) over that in an "equivalent" low enrichment uranium (LEU) assembly. This factor is then applied to previously calculated results for the various design basis accidents to estimate what the dose could be with the MOX LTAs in place. This approach is accurate only if the I-131 concentration used in establishing the percentage increase is the same as that which was used to determine the previously calculated doses. The Nuclear Regulatory Commission (NRC) staff is concerned that the licensee's use of data for an equivalent LEU assembly in lieu of the current licensing basis data may have underestimated the impact on previously analyzed doses. The following examples are illustrative of the staff's concerns. They are based upon information that the staff has available. The licensee may be aware of other information, not provided to the NRC staff in the submittal or its supplements, that may be relevant to the NRC staff's concerns.

   - In Table Q3(f)-1, provided in the licensee's letter of November 3, 2003, the I-131 concentration of a 5% MOX assembly with fuel handling accident (FHA) peaking factors was identified as 8.81E+05 curies. In Attachment 6 of the licensee's letter dated December 20, 2001, (Reference 4), an I-131 concentration of 7.46E+05 curies for an LEU assembly, including the radial peaking factor, was identified. The NRC staff questions whether the values in Attachment 6 are part of the current licensing basis for a FHA since that amendment request was approved based in part on those results. The increase in the I-131 concentration associated with the MOX LTA is 18.1 percent, double the value apparently assumed in the licensee's present comparative analysis.

   - Table 15.12 of the Catawba Updated Final Safety Analysis Report provides a core inventory I-131 value of 8.9E+07 curies for a power level of 3636 megawatts thermal (MWt). This power level is 6.5 percent greater than the power level of 3411 MWt identified in Table 1 of the licensee's letter dated December 10, 2003. Since the core inventory is directly proportional to power level, the adjusted I-131 inventory would be 8.9E+07 / 1.065 = 8.36E+07 curies for the core or 8.36E+07 / 193 = 4.33E+05 curies for an average LEU assembly. Removing the peaking

factor from the Table Q3(f)-1 I-131 value yields 8.81E+05 / 1.65 = 5.34E+05 curies. The resulting increase from a current LEU assembly to the MOX LTA (4.33E+5 to 5.34E+05) is 23 percent rather than 9 percent.

For the accidents identified in Tables Q3(b)-1 through Q3(b)-4, please provide the following information:

(1) The quantity of I-131, in curies, that was utilized in the current analysis of record that provided the tabulated LEU dose

(2) The quantity of I-131, in curies, in the MOX LTA used for comparison.

(3) The rated power plus uncertainty that the LEU and MOX LTA radionuclide inventories were based upon.

(4) The percent increase in the I-131 concentrations identified in (1) and (2), adjusted for differences identified in (3)

If the resulting percent increases differ from those used in the licensee's analyses reported in Tables Q3(b)-1 through Q3(b)-4, please revise the submittal, or provide a justification of why the licensee's approach should be found acceptable.

Response Overview

The submittal (References Q1-1 and Q1-2) was designed to isolate the effect on dose due to only the change to mixed oxide (MOX) fuel. This was accomplished by performing detailed comparisons between fission product inventory for a MOX fuel assembly and an equivalent LEU fuel assembly. This approach was taken in order to provide the clearest representation of the impact on the accident consequences analyses due to the change in fuel type. A source term inventory was prepared for the MOX fuel using an analytical approach which developed a set of assumptions and input values for use in a computer code as was described in previous RAI responses (Reference Q1-2). To provide the most appropriate one-to-one comparison between MOX and low enrichment uranium (LEU) fuel, an LEU calculation was performed using consistent input values and the same computer code.

Direct use of the existing licensing basis source terms in Reference Q1-3 (Design Basis Accidents except for the Fuel Handling Accidents) and Attachment 6 of Reference Q1-4 (Fuel Handling Accidents) was avoided intentionally because their comparison to the MOX source term would fail to isolate the differences between the two different types of fuel. A comparison involving these LEU source terms carries with it the differences in computer codes (or versions) and the differences in the analytical approach which are manifested in different sets of assumptions. The information in the submittal (References Q1-1 and Q1-2) provides the most direct comparison of the effects from a switch to MOX fuel relative to the consequences from using (equivalent) LEU fuel. If the licensing basis

source term inventory models were applied in a consistent manner to a MOX fuel assembly, it would be expected that the change in I-131 inventory would be similar in magnitude to what has been calculated and published in Reference Q1-2.

The application of the MOX - LEU comparison results to determine the impact of four MOX fuel lead assemblies on licensing basis dose calculations for events other than those involving fuel handling accidents presumes that the approved licensing basis source term inventory for LEU fuel and the resultant computed dose methodology are an acceptable baseline. The intent is to determine a reasonable and conservative estimate of the dose impact of four MOX fuel lead assemblies on the current licensing basis dose calculation results. This is accomplished by applying a factor for the change in fuel assembly fission product inventory, combined with a factor for the difference in fission product inventory release. The results demonstrate minor impact from the inclusion of four MOX fuel lead assemblies and adequate margin to all applicable limits.

The source terms described in Reference Q1-2 were purposely derived to be more conservative and bounding than previous source terms. The source term in Reference Q1-3 provides the basis for the majority of the design basis accident analyses. Since the time the source term in Reference Q1-3 was developed and the present, the industry has become more knowledgeable and has developed advanced tools for source term analysis. The source term inventory calculation performed for this submittal used this improved approach. It is a detailed and complex evaluation that considers fuel type and design, enrichment, assembly power histories, and burnup histories of the fuel. Multiple permutations of conditions have been calculated and the results were combined to determine the appropriate, conservative source term inventory for use in accident evaluations.

For each fuel cycle the accident source term in Reference Q1-3 is reviewed against the proposed design to ensure that the accident analyses will continue to be valid. Among the issues investigated are the source term input values related to the core design to assure their continued applicability. This effort is undertaken as part of the Reload Design Review process. A more bounding source term (such as that described in References Q1-1 and Q1-2) would require less cycle specific assessment.

<u>Response to Subquestion (1)</u>
The "all-LEU" results in Reference Q1-2 Tables Q3(b)-1 and Q3(b)-2 are licensing basis results. The LEU source term for these licensing basis cases comes from UFSAR Table 15-12. This table provides an I-131 core activity of 8.9E7 Curies of I-131. This reduces to 4.61E5 Curies I-131 per fuel assembly. This value is a core average, core wide source term inventory and does not include a peaking factor such as that applied for Fuel Handling Accident source terms. UFSAR Table 15-12 was derived for a power level of

3636 MW thermal (MWth). Accident analyses are typically performed with uncertainties and the UFSAR Table 15-12 power level includes added conservatism.

On the other hand, the LEU results in Reference Q1-2 Tables Q3(b)-3 and Q3(b)-4 are based upon the development of the LEU equivalent fuel assembly Fuel Handling Accident source term inventory as reported in Reference Q1-2 Table Q3(f)-2. This computed value is 8.06E5 Curies I-131, which is the basis for the increase reported and used in the MOX evaluations for Tables Q3(b)-1 and Q3(b)-2. Thermal power uncertainties and peaking are included in the derivation of this value. The Fuel Handling Accident source term modeled used bounding peaking conditions (peaking factor = 1.65, high burnup, as described below). It showed a relatively large inventory percentage difference for I-131 between the two fuel types (about 9%). Percentage differences for other inventory modeling approaches that are used for DNB event source terms or the LOCA accident were lower.

Response to Subquestion (2)

The source term analyses in the submittal were developed to support the use of MOX lead assemblies in either McGuire Nuclear Station (MNS) or Catawba Nuclear Station (CNS). As reported in Reference Q1-2, Table Q3(f)-1, the MOX assembly activity is 8.81E5 Curies I-131. This inventory applies to modeling fuel handling accidents where an assembly peaking factor of 1.65 has been imposed. This peaking factor is used in licensing analyses for MNS and is consistent with the approach in Regulatory Guide 1.25 (Reference Q1-5). For the CNS analyses (Reference Q1-4) a peak power history is employed consistent with the guidance in Regulatory Guide 1.183 (Reference Q1-6), based upon the maximum full power operation of the core as determined by the cycle safety analysis. This explains the difference between the LEU source term results provided in this submittal as compared to the December 2001 CNS submittal (Reference Q1-4).

Response to Subquestion (3)

The nominal rated power for Catawba Nuclear Station (CNS) is 3411 MWth. A standard 2% thermal uncertainty factor is typically included in current source term inventory calculations as a conservatism and was applied in the generation of the MOX and equivalent LEU source terms. This factor accounts for the worst case allowable error in those parameters which affect the thermal power calculation, including feedwater flow. The applicability of this uncertainty is discussed in the description of fission product inventory calculations in Regulatory Guide 1.183.

In contrast, the data in UFSAR Table 15-12 were produced nearly fifteen years ago. A core power uncertainty of 6.6% was assumed. The current licensing basis loss of coolant, locked rotor, and rod ejection accidents were based on this source term. A 4.5% conservatism was assumed in those analyses (UFSAR Tables 15-22, 26, 40, and 41).

Response to Subquestion (4)

From the detailed source term work performed for this submittal, there is no direct or simple way to provide the requested adjustments. As described above, isolating the differences between the inventory of a MOX assembly versus an LEU assembly requires detailed analysis using a consistent analytical approach. The bounding percent increase in MOX I-131 relative to the equivalent LEU fuel assembly is 9% as was provided in Reference Q1-2. This comparison provides a direct correlation since it isolates other factors, allowing the difference computed to be attributed to the change to MOX fuel. These source terms were used to generate the results from Reference Q1-2 Tables Q3(b)-1 through Q3(b)-4.

Response Conclusion

The approach used in the submittal is acceptable because it provides the most appropriate and direct comparison to show the effect attributable solely to the change of fuel type. A comparison to the licensing basis source term inventory would be less useful because it would involve differences in fuel type and two analytically diverse source term calculations. Besides the differences in fuel types, the increase in MOX I-131 concentration relative to the licensing basis source term is also due to the differences in the analytical approach to the generation of the source term, the assumptions, and the computer codes (including versions) among other factors. If the MOX fuel inventory was calculated using assumptions and methods similar to the licensing basis case, it would be expected that the magnitude of the increase would be similar. The particular analysis comparison used for these calculations was selected because it showed a relatively large inventory percentage difference for the MOX fuel.

The consequence results which were computed by analyses (Fuel Handling Accidents) for the cases involving MOX fuel are unaffected by this discussion. Only those accidents which were evaluated based upon the relative change in I-131 (those accidents with large fuel failure populations) would be affected. The change in consequences for those accidents was so small for the four MOX lead assemblies that the changes relative to the licensing basis case do not affect the conclusions. Examining up to a 23% increase in I-131 activity due to MOX as suggested in the question would increase the calculated final dose results by at most 0.3 Rem from those reported in Reference Q1-2.

References

Q1-1 Letter from Tuckman, M. S. (DPC), February 27, 2003, to USNRC, "Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies and Request for Exemption from Certain Regulations in 10 CFR Part 50."

Q1-2 Letter from Barron, H. B. (DPC) to USNRC, November 3, 2003, "Catawba Nuclear Station Units 1 & 2, Docket Nos. 50-413, 50-414, McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370, Response to Request for Additional Information Regarding the Use of Mixed Oxide Lead Fuel Assemblies."

Q1-3 Catawba Nuclear Station Updated Final Safety Analysis Report, Revision 10, 2003 Update.

Q1-4 Letter from G. R. Peterson (DPC) to USNRC, December 20, 2001, "Proposed Amendment for Partial Implementation of 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."

Q1-5 Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Revision 0, March 1972.

Q1-6 USNRC Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

2. In a recent teleconference, the licensee explained the basis for the Table Q3(a)-2 release fraction value of 1.96E-4. The NRC staff has determined that it will be relying on this information in preparing its Safety Evaluation. As such, this information is requested to be submitted by letter. Please explain the derivation of the release fraction value in the forthcoming submittal, as discussed in the teleconference.

The NRC staff reviewed the NUREG/CR-6410 method cited by the licensee in the teleconference (i.e., Section 3.3.4.8). This section discusses the crushing of small right cylinders of brittle materials by forces applied over the entire upper surface of the specimen by a component with an area greater in size than that of the impacted surface. It appears that the experimental data were obtained for individual pellets. Please provide a justification of the applicability of these data to fuel pellets contained within fuel pins that are part of a larger fuel assembly. For example, how is the momentum of the fuel assembly structural components reflected in the energy density calculated from the pellet density? Does friction between the pellet surface and the cladding reduce the compressive force represented by the energy density?

Response

In Reference Q2-1 the consequence analysis for dropping a fresh mixed oxide (MOX) fuel assembly was described. In performing this analysis, it was necessary to quantify the portion of the fuel pellets which would be damaged and released to the atmosphere in a form that could be respired. Reference Q2-2 was utilized in the calculation of this release fraction. This reference describes tests performed by Argonne National Laboratory where a force was applied directly to the pellet with an object whose diameter was greater than that of the bare pellet. The pellet was pulverized in the testing. The resultant data were then fit with a log-log curve, which was represented by the following relationship:

$$\text{Release Fraction} = 3.27\text{E-}11 * E_d^{1.131}$$

subject to the condition that the energy density (E_d, J/m^3) is less than 1E7 J/m^3.

Given the energy density, the release fraction is readily calculated. The energy density is computed by multiplying the height of the drop, the density of the material, and the gravitational acceleration constant. A drop height of 30 feet was used to bound the maximum lift height of a fresh assembly during receipt operations at either the McGuire or Catawba Nuclear Stations. The McGuire Nuclear Station has since been removed from consideration for the MOX lead assembly demonstration program.

The fuel material's density was calculated using a weighted average of the densities of UO_2 and PuO_2. The weighting factor is the percent of PuO_2 loading (5%), which was maximized to provide a conservative estimate for the impact from PuO_2 in the fuel matrix. Typically, fuel mixtures result in a density that is approximately 95% of theoretical density. In order to maximize the energy density and bound any potential mixture variations, full theoretical density was conservatively used for both materials.

The calculated energy density meets the criterion for application of the experimentally derived relationship. A value of 1.96E-4 was computed using this relationship and applied as the release fraction in the dropped fresh MOX fuel assembly model.

The effect of the structural cage materials on the energy density was neglected. The fuel pellets account for over 78% of the weight of the fuel assembly and are the most dense components of the fuel assembly. By using the theoretical fuel density (11 g/cm^3), the model assumes that the entire fuel assembly is composed of fuel pellets. If the other fuel assembly materials are included in this model, their effect would be to reduce the density of the fuel assembly modeled since each material is less dense than the fuel. Because the density of the assembly modeled would be reduced, the energy density would be less and the release fraction would be reduced (by about 10%). Therefore, the model that was applied is conservative and bounds lower energy densities which would be derived by including the effect of the less dense fuel assembly cage and cladding materials and lower PuO_2 concentrations.

Friction between the pellets and the cladding would tend to absorb energy and reduce compressive forces. This should be a minor effect because in new fuel there is generally little friction between the polished pellets and the clad. Pellets are typically loaded into cladding tubing with vibration or by hand. By neglecting this friction, the modeling approach increases the impact to the pellets and the potential release. Since the model employed does not take credit for the cladding, it neglects the friction between it and the pellets and is conservative and bounding since it maximizes the impact to the pellets and the release fraction.

It is recognized that several components of this release, dispersion, and respiration model are extremely conservative, but the approach was adopted as a direct and conservative use of the most applicable literature available on this topic. As applied to the fresh MOX fuel assembly drop analysis, this approach inherently assumes that the fuel assembly structural components and cladding do not exist. No credit is taken for any protection that the structural cage and cladding would provide. Since this is one of the functions of the structural cage, it would be expected that it would provide energy absorption in this scenario. The model essentially assumes that all of the bare fuel pellets in the MOX assembly are dropped to the floor (or more literally, that each pellet has the force equivalent to a 30 foot drop applied to it by an external implement of greater diameter). Material is dispersed ignoring the cladding barrier.

In the extremely unlikely event of a fresh fuel assembly drop prior to placing the assembly in or above the fuel pool, the event would progress much differently with more benign consequences. The fuel assembly would be lifted by the upper endfitting to an elevation above the spent fuel pool deck level and slowly transported horizontally to the spent fuel pool. It would drop from a near vertical orientation to the floor where it would impact the floor with the lower endfitting. The force of the impact would be borne by the

structural cage (endfittings and guide tubes), which would likely exhibit some deformation. Any compression of the cage could cause the fuel pins to slide upward through the grids as permitted by the as-built shoulder gap. Any force applied to the lower end cap by the lower endfitting would be transmitted to the cladding. The total force would be divided and spread over the fuel pins. After the initial impact, the fuel assembly would then fall on its side. Since the endfitting cross sectional area is greater than the fuel pitch, the upper endfitting would impact the floor first, the fuel pins would be supported by the grids, and the assembly structure and fuel grids would absorb much of this energy.

Therefore, many features of the MOX new fuel assembly drop accident model are very conservative, including the assumption that the cladding is breached for all fuel pins and the modeling and calculation of the pellet release fraction. Since the model takes no credit for the energy absorption and protective functions of the structural cage and the cladding, it essentially models direct energy impact to the fuel pellet. Thus, the release fraction could not be worse than that modeled and calculated and the evaluation concludes that these results far bound the realistic potential release scenarios.

References

Q2-1 Letter from Barron, H. B. (DPC) to USNRC, November 3, 2003, "Catawba Nuclear Station Units 1 & 2, Docket Nos. 50-413, 50-414, McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370, Response to Request for Additional Information Regarding the Use of Mixed Oxide Lead Fuel Assemblies."

Q2-2 NUREG/CR-6410, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," March 1998.