RAS 8581

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September 20, 2004

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DOCKETED
USNRC

October 5, 2004 (4:35pM)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Re: In the Matter of Duke Energy Corporation
Catawba Nuclear Station, Units 1 and 2
<u>Docket Nos. 50-413-OLA, 50-414-OLA</u>

Dear Administrative Judges:

Attached for the information of the administrative judges and the parties in this proceeding is a copy of a letter, submitted today, by Duke Energy Corporation ("Duke") to the NRC Staff regarding the issue addressed in my correspondence of August 31, 2004. Specifically, in the attached letter Duke is supplying the NRC Staff with updated information on the dose consequences of certain design basis accidents addressed in the license amendment request ("LAR") for the mixed oxide ("MOX") fuel lead assemblies. Duke's letter also addresses the impact of these corrections on the NRC's safety evaluation and proposed no significant hazards consideration determination. Duke has concluded that the regulatory conclusions regarding the MOX fuel lead assembly LAR previously reached by the NRC Staff remain valid.

As is clear in the attached correspondence, Duke has also confirmed the conclusion reflected in my prior communication on this matter: the out-of-date information in the MOX fuel license amendment request, and the updated design basis accident doses being provided to the NRC Staff today, are <u>not</u> material to the MOX fuel Loss of Coolant Accident ("LOCA") analyses that are the subject of Contention I in this proceeding. Contention I relates to the MOX fuel design basis LOCA thermal-hydraulic analyses, and relates to compliance with the emergency core cooling system acceptance criteria of 10 C.F.R. § 50.46. Those LOCA analyses and criteria are not implicated by the accident consequence (*i.e.*, dose) information discussed in the attached correspondence. (See attached letter, at page 5.)

Template = SECY-043

SECY-02

The attached correspondence specifically provides revised information for the dose consequences of the design basis rod ejection and the locked rotor accidents, as I previously described on the record in this proceeding when this issue was first discussed at a conference with the judges and parties.

As also discussed in the attached letter to the NRC Staff, Duke has since determined that the calculated doses due to a design basis LOCA for the MOX fuel case were not affected in the re-analysis. However, Duke is providing in the attached correspondence updated values for control room doses for the baseline LOCA analysis for the all low-enriched uranium ("LEU") fuel case. Offsite dose values (*i.e.,* at the exclusion area boundary and for the low population zone) are not affected. (See attached letter, at pages 2, 4.) Note that LOCA dose issues have also been specifically determined to be beyond the scope of Contention I in this proceeding.[1] Doses are obviously also not material to the security contention remaining in this proceeding.

Duke conservatively forwarded the prior submittal on this topic for the information of the Licensing Board and parties because it addressed an issue that could relate to the schedule for issuance of the requested license amendment. *Compare Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-143, 6 AEC 623, 625 ((1973) (parties are to apprise licensing boards of new information that is "relevant and material to the matters being adjudicated"). Duke is forwarding the attached submittal at the request of the Licensing Board. In Duke's view, however, any further proceedings on this issue are subject to the NRC's rules of practice, including 10 C.F.R. § 2.734. That regulation establishes the high standard for reopening a closed record such as the record on Contention I.[2]

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1 Order (Confirming Matters Addressed at April 6 Telephone Conference), April 8, 2004 (at p. 2) ("With respect to Contention I, this contention encompasses those calculations involved in the determination of events up to and including LOCAs and DBAs, but does not include analyses related to any releases either in containment or offsite."); *see also* Tr. 1726-36.

2 *Cf. Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-82-39, 16 NRC 1712, 1714-15 (1982) (where a motion to reopen relates to a previously uncontested issue, the moving party must satisfy both the standards for admitting late-filed contentions and the criteria for reopening the record).

Very truly yours,

David A. Repka

David A. Repka
Counsel for Duke Energy Corporation

Enclosure
cc: Service List (w/Enclosure) via U.S. mail and e-mail

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September 20, 2004

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Subject: Duke Energy Corporation Catawba Nuclear Station Units 1 & 2, Docket Nos. 50-413, 50-414
Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (Revised Dose Evaluations)

This letter and its attachments provide revised information in support of the Reference 1 license amendment request (LAR) to receive and use four MOX fuel lead assemblies at the Catawba Nuclear Station. In response to Requests for Additional Information (RAIs) from the NRC, Duke provided evaluations of the impact of four MOX fuel lead assemblies on design basis accident doses (Reference 2). Duke subsequently made minor corrections to those MOX fuel lead assembly dose evaluations (Reference 3). As noted in Reference 4, while in the process of performing cycle-specific reload analyses for Catawba 1 Cycle 16 Duke recognized that certain dose information in References 1-3 was based on out-of-date input values for design basis accident doses with low enriched uranium (LEU) cores.

Duke has performed revised dose evaluations for Catawba cores containing four MOX fuel lead assemblies. Like the previous dose values, these revised design basis accident dose values for Catawba cores containing four MOX fuel lead assemblies are lower than the applicable acceptance criteria. These updated dose results support the conclusion that MOX fuel lead assemblies can be used at Catawba without presenting an undue hazard to the health and safety of the public. The revised dose results are consistent with the overall conclusions reached by the Nuclear Regulatory Commission (NRC) in its safety evaluation (Reference 5), environmental assessment (Reference 6), and Proposed Finding of No Significant Hazards (Reference 7).

This letter summarizes the results of the revised evaluations of the dose impacts of using four MOX fuel lead assemblies at Catawba. In addition, updates to the application materials that reflect the revised evaluations are provided as attachments. Finally, corrective actions associated with this problem are addressed.

<u>Description and Scope of Problem</u>

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 is described in Reference 3 and involves increasing the baseline LEU 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. This method was reviewed and accepted by the NRC (Reference 5), except that the NRC applied a slightly higher adjustment to reflect a larger MOX-LEU difference in initial iodine-131 inventory.

While the methodology continues to be valid, some of the baseline LEU doses that Duke used in its evaluations were outdated. The LEU values for dose to the public were taken from Table 15-14 of the Catawba Updated Final Safety Analysis Report (UFSAR). However, the REA and LRA dose values in the UFSAR are based on calculations that have been superseded by more recent work. The current LEU values are higher than those shown in the UFSAR for the REA and LRA, as shown in Attachment 1. Attachment 2 provides a summary of the changes in the baseline LEU REA and LRA dose analyses that resulted in the higher dose values.

The calculated LOCA doses at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room were not affected by the re-analyses discussed above. However, in reviewing the MOX fuel dose information previously provided to NRC, it was noted that the baseline LEU LOCA Control Room dose values (taken from UFSAR Table 15-41) also needed to be updated. The current (updated) LEU LOCA Control Room doses are higher than those shown in the UFSAR, as shown in Attachment 1. Attachment 3 provides a summary of the changes in the baseline LEU LOCA Control Room dose analyses that resulted in the higher dose values.

With respect to the MOX fuel LAR and related correspondence, these problems affect only the REA and LRA information (offsite and Control Room doses) and LOCA information (Control Room dose only). As discussed in Attachment 3, the baseline LEU LOCA offsite dose calculations are unaffected by the update to the Control Room dose calculations. The MOX fuel handling accident and weir gate drop were explicitly analyzed for both MOX and LEU fuel, and those results are also not affected. As noted in the NRC Safety Evaluation (Reference 5, Section 3.2.2), the MOX fuel lead assemblies will have no impact on the doses resulting from steam generator tube rupture, main steamline break, instrument line break, waste gas decay tank rupture, and liquid storage tank rupture. That conclusion and the associated rationale remain valid.

Revised Dose Evaluations

The appropriate baseline LEU thyroid dose values were used to perform revised evaluations of doses resulting from REA, LRA, and LOCA (Control Room only) for cores containing four MOX fuel lead assemblies. The results are summarized in Attachment 4 and discussed below. With respect to dose to the public, the discussion focuses on EAB doses, which are more limiting than LPZ results. The applicable dose acceptance criteria are taken from Reference 13, Attachment 2, Table Q12-3.

Rod Ejection

For the REA, the baseline LEU thyroid dose increased from 1 rem at the EAB (both units) to 21.8 rem for Unit 1 and 30.7 rem for Unit 2. These numbers are based on 50% failed fuel, which is the same as the prior evaluation. The EAB thyroid dose acceptance criterion for this accident is 75 rem.

Applying the same methodology used by Duke in Reference 3, the corresponding EAB REA thyroid doses for a core containing four MOX fuel lead assemblies are 22.3 rem for Unit 1 and 31.5 rem for Unit 2. As noted in Reference 2, this approach is conservative in that it assumes that all of the rods in the MOX fuel lead assemblies preferentially fail. In reality, with MOX fuel in an unrodded, non-limiting location, it is extremely unlikely that all of the MOX rods would fail. Nevertheless, while the baseline LEU REA thyroid dose has increased, the incremental impact of MOX fuel lead assemblies is small and the resulting doses are within the acceptance criterion.

Locked Rotor

For the LRA, the baseline LEU thyroid dose increased from 3.7 rem at the EAB (both units) to 23.6 rem for Unit 1 and 22 rem for Unit 2. These revised dose values are based on failed fuel fractions of 9.5% for Unit 1 and 5% for Unit 2, which are smaller failed fuel fractions than the 11% failed fuel fraction assumed in the original evaluation. The EAB thyroid dose acceptance criterion for this accident is 30 rem.

Again applying the same methodology as was used by Duke in Reference 3, the corresponding EAB LRA thyroid doses for a core containing four MOX fuel lead assemblies are 26.9 rem for Unit 1 and 27.8 rem for Unit 2. The LRA exhibits a higher percentage dose increase due to MOX fuel lead assemblies than does the REA. This is attributable to the fact that fewer total rods are assumed to fail in the LRA, so the relative impact of the MOX fuel assemblies (in which all rods are conservatively assumed to preferentially fail) is greater. Nevertheless, while the baseline LEU dose has increased, the resulting doses from MOX fuel lead assembly cores are within the acceptance criterion. Recognizing the large conservatism inherent in applying this methodology to the LRA, Duke further evaluated the LRA, as discussed below.

The actual amount of fuel that is predicted to fail during a LRA varies from cycle to cycle. Core thermal-hydraulic characteristics and core peaking influence the number of fuel rods that are calculated to experience DNB. Cycle-specific checks on calculated core peaking are performed to ensure that the amount of failed fuel from a LRA is less than the amount assumed in the current dose evaluation. Cycle-specific assessments for recently operated or operating Catawba cores resulted in no calculated fuel failures during a LRA. Using the methodology of Reference 8, Section 4.3, Duke has recently performed a cycle-specific LRA analysis for Catawba 1 Cycle 16 (C1C16). The analyzed core design includes 72 feed LEU assemblies [Westinghouse Robust Fuel Assembly (RFA) design] and four feed MOX fuel lead assemblies. The design also incorporates eight once-burned Westinghouse Next Generation Fuel (NGF) lead test assemblies.

Thermal-hydraulic models specific to each fuel type (RFA, NGF, and MOX) were used to generate maximum allowable peaking limits for various axial power shapes. Core peaking values were calculated for C1C16 and compared to the maximum allowable peaking limits. The analysis, performed over a range of Cycle 16 conditions, showed that the calculated peaking does not exceed the DNB peaking limits for any fuel type (either MOX or LEU), so no fuel failures are calculated to occur during a C1C16 LRA.

With no MOX fuel failure, there would be no incremental dose associated with a LRA that is attributable to the MOX fuel lead assemblies. Duke commits to informing the NRC in the event an analysis of any cycle operating with the MOX fuel lead assemblies indicates that a MOX fuel rod would fail during a design basis LRA.

Locked Rotor and Rod Ejection Control Room Doses

Duke had not previously explicitly addressed the impact of MOX fuel lead assemblies on Control Room dose following a LRA or REA, due to the fact that LOCA Control Room thyroid doses were clearly more limiting. Given the higher baseline LEU doses to the public from LRA and REA, Duke evaluated the impact of MOX fuel lead assemblies on the Control Room thyroid dose following a REA and LRA. These evaluations used the same failed fuel fractions as discussed above – 50% for both units during a REA, and 9.5% and 5.0% for Units 1 and 2, respectively, during a LRA. The Control Room thyroid doses from a baseline LEU core, as well as the impact of using four MOX fuel lead assemblies, are shown in Attachment 4. Like the EAB and LPZ cases, these Control Room dose evaluations of MOX fuel lead assembly cores used the Reference 3 methodology which conservatively assumes that all MOX fuel pins preferentially fail. The resulting Control Room thyroid doses are within the acceptance criterion of 30 rem for both accidents, and the impact of MOX fuel is small.

LOCA Control Room Dose

Finally, Duke has revised the evaluation of the impact of MOX fuel lead assemblies on the Control Room thyroid dose following a LOCA. The baseline LEU LOCA Control Room thyroid dose is 21 rem (both units). Applying the methodology of Reference 3, the corresponding Control Room dose from a core containing four MOX fuel lead assemblies following a LOCA is 21.3 rem (both units). While the baseline LEU Control Room thyroid dose has increased, the impact due to MOX fuel is small, and the resulting doses from MOX fuel lead assembly cores are within the acceptance criterion of 30 rem.

Summary

The following conclusions can be drawn from these results.

- The revised REA and LRA doses to the public from a MOX fuel lead assembly core are within the acceptance criteria.
- Four MOX fuel lead assemblies do not significantly increase the accident dose to the public following a REA or LRA.

- A C1C16 cycle-specific analysis demonstrates the substantial conservatism that is inherent in the LRA dose evaluation.
- Control Room doses following a REA, LRA, or LOCA from a MOX fuel lead assembly core are within the acceptance criteria.
- Four MOX fuel lead assemblies do not significantly increase the Control Room dose following a REA, LRA, or LOCA.
- Other dose evaluations for the MOX fuel lead assemblies (e.g., LOCA doses to the public, fuel handling accident) are unaffected and remain valid.

It is noted that the NRC calculated slightly different MOX fuel lead assembly core doses than Duke in the NRC Safety Evaluation (Reference 5, Section 3.2.5). Duke has also calculated the revised MOX fuel lead assembly core dose values using the NRC approach, and those values are summarized in Attachment 5 to this letter. The conclusions reached above remain valid using the NRC approach to calculating doses.

Additional information is provided in the revised LAR material (see below).

Affected Sections of the Application

Duke has reviewed the MOX fuel lead assembly LAR and associated correspondence and identified the following affected documents.

1. Radiological Consequences RAI response dated November 3, 2003 (Reference 2), as modified by the March 16, 2004 letter (Reference 3).
2. Radiological Consequences RAI response dated March 1, 2004 (Reference 9), as modified by the March 16, 2004 letter (Reference 3).
3. LAR No Significant Hazards Consideration analysis (Reference 1, Attachment 4), as modified by the March 26, 2004 letter (Reference 10).
4. LAR Environmental Report (Reference 1, Attachment 5), as modified by the March 26, 2004 letter (Reference 10).

Attachment 5 to this letter revises the radiological consequence information that was previously submitted (items 1 and 2, above). Attachment 6 to this letter revises the No Significant Hazards Consideration analysis (item 3, above). Attachment 7 to this letter revises the Environmental Report (item 4, above).

As is reflected in Attachments 5-7, while these changes to the dose analyses modify some of the details of the application, there is no change to the conclusions that are supported by the LAR.

The dose analysis corrections provided in this letter have no impact on the analyses demonstrating that the MOX fuel lead assemblies comply with the emergency core cooling system acceptance criteria in 10 CFR 50.46. More specifically, LAR Section 3.7.1 (Reference 1, Attachment 3) and the responses to RAIs 12-17, 21-22, and 24-28 (Reference 2) are not affected.

Corrective Actions Related to the MOX Fuel LAR

The problem with the MOX fuel lead assembly dose analyses has been entered into the Duke Corrective Action Program, and assessment and corrective action work are underway.

As the NRC is aware, Duke previously identified a discrepancy related to the presence of Westinghouse Next Generation Fuel (NGF) lead test assemblies in Catawba Unit 1, as described in the April 16, 2004 Duke letter to the NRC (Reference 11). Pursuant to that issue, Duke performed a review that was summarized in a May 13, 2004 letter (Reference 12). That review was performed by personnel involved in the generation of the LAR and associated RAI responses. The NGF review effectively identified a number of clarifications to the application materials. However, the review did not identify the error in the source document (Catawba UFSAR) that was used for the MOX fuel dose evaluations discussed herein.

Duke self-identified this MOX fuel lead assembly LAR dose analysis problem during work related to C1C16 cycle-specific reload analyses. Duke has performed a thorough review of the MOX fuel lead assembly dose analyses and corrected all analyses affected by the errors. Those corrections are reflected in this letter. In addition, Duke performed an independent review of the entire MOX fuel lead assembly LAR and its supporting documentation. The review was carried out by qualified personnel who were not involved with the development of the LAR and associated correspondence. The review team has preliminarily determined that the supporting documentation adequately supports the statements made in the LAR. Duke will provide the NRC with a summary of the results of the independent review once it has been documented. The projected completion date is October 1, 2004.

Other Corrective Actions

Other aspects of the problem beyond the MOX fuel lead assembly LAR are being addressed through the Duke Corrective Action Program. Upon identification of the problem, immediate corrective actions were performed. Duke confirmed that no current "operable but degraded" evaluations or active license amendment requests used the outdated UFSAR dose information. Subsequently, Duke initiated a review of Licensee Event Reports, Notices of Enforcement Discretion, Generic Letter responses, bulletin responses, and license amendment requests between 1996 and the present. Duke intends to review operability and 10 CFR 50.59 evaluations that were performed over the same time period. Duke is also in the process of updating all Chapter 15 dose information in the Catawba UFSAR. Finally, Duke is verifying the accuracy of the Chapter 15 dose information in the McGuire and Oconee UFSARs.

In addition, the broader issue related to updates of the Catawba UFSAR is being assessed. Corrective actions will be identified and implemented, as appropriate, based on the outcome of that assessment.

Commitments

The NRC commitments made in this letter are summarized below.

1. Duke will inform the NRC if any reload analysis for cycles with MOX fuel lead assemblies indicates that any MOX fuel rod would fail during a design basis LRA. In these circumstances, Duke would determine the need for NRC approval in accordance with the requirements of 10 CFR 50.59.
2. Duke will provide the NRC with a summary of the results of the MOX fuel lead assembly LAR independent review that has been performed under the Duke Corrective Action Program.

Summary and Conclusion

Duke relied on outdated dose information in developing responses to NRC RAIs related to the MOX fuel lead assembly LAR. The problem occurred as a result of a failure to update the Catawba UFSAR dose analyses and to ensure that accurate inputs were used for the LAR dose evaluations.

Duke has revised the information in the MOX fuel LAR. The revised dose information, provided to NRC in this letter, continues to demonstrate that MOX fuel lead assemblies would have at most a minor impact on doses resulting from accidents that assume a large number of fuel failures in the reactor core. The doses are within the applicable acceptance criteria.

The regulatory conclusions associated with the MOX fuel lead assembly LAR remain valid. There is reasonable assurance that four MOX fuel lead assemblies can be used at Catawba with no undue risk to the health and safety of the public. Using four MOX fuel lead assemblies at Catawba will have no significant impact on the environment. The three standards of 10 CFR 50.92(c) are satisfied such that the MOX fuel lead assembly LAR involves no significant hazards consideration.

This error was self-identified by Duke. In addition to addressing the immediate problem as it relates to the MOX fuel LAR, Duke is investigating the root causes and implementing corrective actions to prevent recurrence. Duke will work proactively with the NRC to resolve the issue in a timely and comprehensive manner. If you have any questions, please contact Steve Nesbit at (704) 382-2197.

Sincerely,

Henry R Barron

Henry B. Barron

References

(1) Letter, M. S. Tuckman (Duke) to U. S. Nuclear Regulatory Commission, Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies, February 27, 2003.

(2) Letter, M. S. Tuckman (Duke) to U. S. Nuclear Regulatory Commission, Response to Request for Additional Information, November 3, 2003.

(3) Letter, W. R. Mc Collum (Duke) to U. S. Nuclear Regulatory Commission, Corrections to Dose Information, March 16, 2004.

(4) Letter, W. R. Mc Collum (Duke) to U. S. Nuclear Regulatory Commission, Dose Inputs, August 31, 2004.

(5) U. S. Nuclear Regulatory Commission, Safety Evaluation for Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies, April 5, 2004.

(6) U. S. Nuclear Regulatory Commission, Duke Energy Corporation; Concerning the Application for Irradiation of Mixed Oxide Lead Test Assemblies at Catawba Nuclear Station, Units 1 and 2, Environmental Assessment and Finding of No Significant Impact, 69 Federal Register 51112, August 17, 2004.

(7) U. S. Nuclear Regulatory Commission, Duke Energy Corporation, et al., Catawba Nuclear Station, Units 1 and 2; Notice of Opportunity To Comment and Proposed No Significant Hazards Consideration Determination, 69 Federal Register 41852, July 12, 2004.

(8) DPC-NE-3002-A, Revision 3, McGuire Nuclear Station and Catawba Nuclear Station UFSAR Chapter 15 System Transient Analysis Methodology, May 1999.

(9) Letter, Barron, H. B. (Duke) to U. S. Nuclear Regulatory Commission, Response to Request for Additional Information (Radiological), March 1, 2004.

(10) Letter, Barron, H. B. (Duke) to U. S. Nuclear Regulatory Commission, Additional Information, March 26, 2004.

(11) Letter, Barron, H. B. (Duke) to U. S. Nuclear Regulatory Commission, Use of MOX Fuel With Next Generation Fuel Lead Test Assemblies, April 16, 2004.

(12) Letter, Barron, H. B. (Duke) to U. S. Nuclear Regulatory Commission, Correspondence Review, May 13, 2004.

(13) Letter, Mc Collum, W. R. (Duke) to U. S. Nuclear Regulatory Commission, Response to Request for Additional Information (Environmental, Radiological, and Materials), February 2, 2004.

Attachments (7)

Attachment 1 – Current LEU Accident Thyroid Doses
Attachment 2 - Changes to Baseline REA and LRA Dose Analyses for LEU Cores
Attachment 3 - Changes to Baseline LOCA Control Room Dose Analysis for LEU Cores
Attachment 4 - Accident Thyroid Doses – MOX Fuel Lead Assembly Cores
Attachment 5 – Updated LAR Radiological Consequences Information
Attachment 6 – Updated No Significant Hazards Consideration Evaluation
Attachment 7 – Updated Environmental Report

Oath and Affirmation

I affirm that I, H.B. 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.

H B Barron

H.B. Barron

Subscribed and sworn to before me on this 20 day of September 2004.

Betty Sharpe Gregory

Notary Public

BETTY SHARPE GREGORY NOTARY My Comm. Exp. March 09, 2009 PUBLIC MECKLENBURG COUNTY, N.C.

My Commission expires:

3/9/09

Date

cc: w/attachments

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

Attachment 1

Current LEU Accident Thyroid Doses

Accident	Unit	Exclusion Area Boundary		Low Population Zone		Control Room	
		Limit (rem)	Dose (rem)	Limit (rem)	Dose (rem)	Limit (rem)	Dose (rem)
Locked Rotor	1	30	23.6 (3.7)	30	4.1 (1.2)	30	0.9
	2	30	22.0 (3.7)	30	3.6 (1.2)	30	1.1
Rod Ejection	1	75	21.8 (1.0)	75	17.4 (0.1)	30	6.4
	2	75	30.7 (1.0)	75	19.3 (0.1)	30	8.7
LOCA	Both	300	89 (89)	300	25 (25)	30	21 (5.3)

Note 1: The values in parentheses are the LEU thyroid dose values from the Catawba UFSAR that were used in the MOX fuel lead assembly dose evaluations.

Note 2: Control Room doses from locked rotor and rod ejection were not provided in the MOX fuel lead assembly LAR or subsequent correspondence.

Attachment 2

Changes to Baseline REA and LRA Dose Analyses for LEU Cores

The table below presents the key assumptions and parameters used in the analyses of radiological consequences of the licensing basis locked rotor accident (LRA) and rod ejection accident (REA). It provides a comparison of these assumptions and inputs between the analyses reported in the Catawba Nuclear Station (CNS) Updated Final Safety Analysis Report (UFSAR) and the current licensing basis calculations. For cases in which there are differences, the values from the calculations are marked in **bold letters.**

The CNS UFSAR reports the analysis of radiological consequences for the LRA in Section 15.3.3.3 and Table 15-22. The analysis of radiological consequences of the REA is reported in CNS UFSAR Section 15.4.8.3 and Table 15-26.

The radiation doses reported for these events in the current licensing basis calculations are significantly higher than those in the UFSAR. This is attributed to the incorporation of modeling to predict the duration of tube bundle uncovery in the steam generators (SGs) and analyze its effects on the radiological consequences of these events. Duke began accounting for tube bundle uncovery in its calculations of radiation doses for the LRA and REA in 1992. Tube bundle uncovery was simulated in the calculations for the LRA and REA radiation doses Duke reported to the Nuclear Regulatory Commission (NRC) on March 15, 1996 and August 27, 1996 in support of SG replacement. The dose values were subsequently updated again to reflect modifications to the auxiliary feedwater systems of both units which were implemented in 1997 (Unit 2) and 1998 (Unit 1).

The increase in dose is primarily attributable to consideration of SG tube uncovery effects in the dose analyses, as explained below. REA and LRA may lead to fuel rod departure from nucleate boiling (DNB) during the accident. The rods that experience DNB are assumed to release radionuclides to the reactor coolant. An important pathway for radionuclide release outside containment is from reactor coolant that flows into the SGs via assumed primary-to-secondary leakage. Eventually, radionuclides are released to the atmosphere by discharge through the SG relief valves. During time spans of tube bundle uncovery (i.e., whenever the tubes are not completely submerged), Duke conservatively assumes that all fission products entrained with primary-to-secondary leakage (iodine isotopes as well as noble gases) are released to the environment. This approach increases offsite and Control Room doses because it minimizes partitioning and retention of iodine in the SGs. When the tubes are submerged, Duke assumes that the iodine isotopes entrained with primary-to-secondary leakage mix instantaneously with the water in the SG secondary side and are released only with the steam releases and with a partition fraction of 0.01.

Modeling of tube bundle uncovery yields a very substantial increase in thyroid radiation doses. The following serves as an example of the importance of tube bundle uncovery. Subsequent to a simulated LRA on Unit 2, tube bundle uncovery for the SG in the affected loop was computed to last for the first 137 sec after the initiating event. That time span contributed 60% to the Exclusion Area Boundary (EAB) thyroid radiation dose and 25% to the Low Population Zone (LPZ) thyroid radiation dose from that SG. The other three SGs were calculated to experience approximately 0.5 hour of tube bundle uncovery. For these three SGs, that time span contributed 98% to the EAB thyroid radiation dose and 90% to the LPZ thyroid radiation dose from each SG. This demonstrates the magnitude of the effect of SG tube bundle uncovery on radiation doses for design basis accidents including releases of radioactivity from the SGs.

Beginning with the analyses to support SG replacement, different dose values are reported for Unit 1 and Unit 2. A key difference between units is the tube bundle uncovery time, which is significantly longer for Unit 2 with the original SGs, as compared to Unit 1 with the replacement SGs.

In the following table, for each case where a parameter differs between the UFSAR and current analysis, a note is written to explain the difference.

Table 1
Comparison of Analysis Model Parameters
For the Locked Rotor Accident and Rod Ejection Accident

Table 1A
Parameters Common to the LRA and REA

Parameter	Values in the Analysis which Supports the UFSAR	Current Licensing Basis Analysis Value	Notes
Data Pertaining to the Radioactive Source Term			
Initial core power (MWt)	3565	**3479**	1
Isotopic Inventory	Table 2	**Table 2**	2
Initial reactor coolant activity included?	Yes	**No**	3
Iodine composition fraction			
Elemental iodine	0.91	0.91	
Particulate iodine	0.05	0.05	
Organic iodine compounds	0.04	0.04	
Fission product gap fraction (%)			
Kr-85	30	**10**	4
Other fission products	10	10	
Data Pertaining to Transport and Release of Radioactivity			
Fraction of the gap inventory released to the primary coolant (%)	100	100	
SG leak rate (gpd per SG)	360	**150**	5
SG leak rate time dependence included?	No	**Yes**	6
SG tube bundle uncovery considered?	No	**Yes**	7
SG iodine partition fraction	0.01	0.01	
Dispersion Data			
EAB χ/Q (sec/m^3)	5.5×10^{-4}	$\mathbf{4.78\times10^{-4}}$	8
LPZ χ/Q (sec/m^3)	1.8×10^{-5}	$\mathbf{6.85\times10^{-5}}$	8
Control Room χ/Q (sec/m^3)		$\mathbf{9.9\times10^{-4}}$	9, 10
Data Pertaining to the Control Room and the Control Room Air Ventilation System (CRAVS)			
Control room (CR) volume (cu.ft.)		**89,200**	9
CRAVS CR outside airflow rate (cfm)		**2667**	9
CRAVS CR recirc airflow rate (cfm)		**1333**	9
CRAVS filter efficiency (%)			
Elemental iodine		**99**	9
Particulate iodine		**99**	9
Organic iodine compounds		**95**	9
Rate of CR unfiltered inleakage (cfm)		**30**	9

Table 1B
Parameters Pertaining to the LRA

Parameter	Values in the Analysis which Supports the UFSAR	Current Licensing Basis Analysis Value	Notes
Data Pertaining to the Radioactive Source Term			
Fraction of the core in DNB (%)	10	**9.5 (Unit 1)** **5.0 (Unit 2)**	11
Data Pertaining to Transport and Release of Radioactivity			
Duration of plant cooldown after the initiating event (hr)	8	**8**	
Time spans of SG tube bundle uncovery (seconds)			
Unit 1			
SG 1	N/A	**1205**	7
SG 2	N/A	**0**	
SGs 3 & 4 apiece	N/A	**1121**	
Total for Unit 1	N/A	**3447**	
Unit 2			
SG 1	N/A	**1870**	7
SG 2	N/A	**137**	
SGs 3 & 4 apiece	N/A	**1770**	
Total for Unit 2	N/A	**5547**	
Total steam released (lbm)			12
Unit 1	515,247 (0-2 hr) 1,040,910 (2-8 hr)	**748,330 (0-2 hr)** **989,069 (2-8 hr)**	
Unit 2	515,247 (0-2 hr) 1,040,910 (2-8 hr)	**736,434 (0-2 hr)** **966,906 (2-8 hr)**	

Table 1C
Parameters Pertaining to the REA

Parameter	Values in the Analysis which Supports the UFSAR	Current Licensing Basis Analysis Value	Notes
Data Pertaining to the Radioactive Source Term			
Fraction of the core in DNB (%)	10	**50**	13
Fraction of the core that is melted (%)	0.25	**0**	14
Data Pertaining to Transport and Release of Radioactivity			
Fraction of the gap inventory released to containment (%)			
Noble gases	100	100	
Iodine isotopes	25	25	
Fraction of the gap inventory released to the containment sump	0	**50**	15
Fraction of the gap inventory released to the primary coolant	100	100	16
Containment volume (cu.ft.)	1,015,000	**1,196,000**	17
Containment leak rate (volume % /day)			
0-24 hr	0.3	0.3	
Time > 24 hr	0.15	0.15	
Containment bypass leak rate (% containment leak rate)	7	7	
Annulus Ventilation System (AVS) flow rate (cfm)	9000	**8100**	18
AVS flow reduction penalty included (flow reduction of 15% at 15 minutes)	Yes	**No**	19
Annulus volume credited (%)	50	50	
Containment air return fan flow rate (cfm)	40,000	40,000	
Ice condenser efficiency (%)			
Elemental iodine	30	30	
Particulate iodine	0	0	
Organic iodine compounds	0	0	
Credit included for the Containment Spray System?	Yes	**No**	20
ECCS leakage iodine partition fraction (%)	N/A	**10**	15
Duration of plant cooldown after the initiating event	2 min	**8 hr**	

Parameter	Values in the Analysis which Supports the UFSAR	Current Licensing Basis Analysis Value	Notes
Time spans of SG tube bundle uncovery (seconds)			
Unit 1			
SG 1	N/A	751	7
SG 2	N/A	662	
SGs 3 & 4 apiece	N/A	2458	
Total for Unit 1	N/A	6329	
Unit 2			
SG 1	N/A	2425	7
SG 2	N/A	2605	
SGs 3 & 4 apiece	N/A	810	
Total for Unit 2	N/A	6650	
Total steam released (lbm)			13, 21
Unit 1	44,500 (0-2 min)	748,330 (0-2 hr) 989,069 (2-8 hr)	
Unit 2	44,500 (0-2 min)	736,434 (0-2 hr) 966,906 (2-8 hr)	
Dose Analysis Results			
See Attachment 1 to this letter			

Table 1 Notes

1) The value for power level used in the current calculations is 102% rated power.

2) The values in Table 2 are different for the CNS UFSAR and the current analysis. The current values are associated with use of ORIGEN 2.1 and representative of current burnup, fuel enrichment, fuel cycle lengths, and batch sizes.

3) The CNS UFSAR assumes pre-existing reactor coolant activity, citing Table 11-4. The current analysis recognizes that the radiation doses for these accidents are almost completely dominated by fuel failure and corresponding fission product releases assumptions.

4) The value of 10% used for the gap fraction for each fission product (both LRA and REA) is based on R.G. 1.77.

5) The UFSAR currently lists the total SG leak rate as 1 gpm and is based on outdated limits. The current technical specifications (TS 3.4.13) set the individual SG leak rate at 150 gpd per SG.

6) Time-dependent values are taken from system analysis (RETRAN) calculations.

7) Inclusion of SG tube bundle uncovery is a conservative approach. This dramatically increases thyroid radiation doses.

8) The value used in the current analysis is based on more recent site meteorological data than the UFSAR analysis.

9) Control room radiation doses were not reported in the CNS UFSAR for the LRA or the REA.

10) 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).

11) CNS UFSAR Table 15-22 should have reported that the amount of "defective fuel" following the LRA was 10%. A value of 11% was used in the MOX fuel lead assembly evaluations (letters to NRC dated 11/3/03 and 3/16/04). In the current calculations, the assumed amount of fuel clad failure is as shown in Table 1B.

12) At the time of SG replacement, the post accident steaming rates for the LRA and REA were recalculated with a model for post accident long-term cooldown. Subsequently the steaming rates were revised with the analysis of the effects of the modifications to the auxiliary feedwater system. More conservative and bounding input assumptions were used for each nuclear unit.

13) Allowable failed fuel fraction for DNB was increased to 50% to provide additional margin for cycle-specific design. CNS UFSAR Table 15-26 lists the data and assumptions for the analysis of radiological consequences of the REA. This table lists two columns of data, one for a design basis analysis and one for a realistic analysis. The column associated with the realistic analysis is ignored for this comparison.

14) The thermal-hydraulic safety analysis for the REA that supports the calculations of radiation doses predicts no fuel melt.

15) The calculation supporting UFSAR 15.4.8.3 did not consider post accident ECCS leakage while the current analysis conservatively accounts for such leakage.

16) CNS UFSAR 15.4.8.3 Assumption 3 states that 50% of the iodine inventory in the gap is assumed to be released to the reactor coolant. This UFSAR number is incorrect; the associated UFSAR calculation used 100%.

17) Actual physical containment free volume is modeled in the current calculations, although the radiological consequences are essentially unaffected by this input value.

18) The CRAVS airflow rate in the current analysis reduces the required value of 9000 cfm by 10%. This adds conservatism to the computed radiation doses.

19) The penalty for CRAVS flow reduction, which was linked to assumed use of the Containment Hydrogen Purge System at CNS, required a minimum of two random failures in the equipment provided for post accident reduction of hydrogen in containment. This was eliminated in the current analysis as a minor over-conservatism.

20) The current analysis calculation conservatively does not take credit for operation of the Containment Spray System following an REA.

21) The explicit assumption of steam dump releases for only 2 minutes was dropped as a pathway for release of radioactivity. This pathway is not significant compared to primary-to-secondary leakage with tube bundle uncovery.

Table 2
Comparison of Core Radioisotope Inventory
Locked Rotor Accident and Rod Ejection Accident

Radioisotope	Core Radioisotope Inventory (Curies)	
	Values in the Analysis which Supports the UFSAR	Current Licensing Basis Analysis Value
I-130	0.0E0	3.3E6
I-131	8.9E7	9.1E7
I-132	1.3E8	1.3E8
I-133	1.9E8	1.9E8
I-134	2.1E8	2.0E8
I-135	1.8E8	1.7E8
Xe-131m	1.3E6	2.1E5
Xe-133m	5.6E6	5.4E6
Xe-133	1.9E8	1.9E8
Xe-135m	3.5E7	3.2E7
Xe-135	1.9E8	6.2E7
Xe-137	1.7E8	1.7E8
Xe-138	1.6E8	1.8E8
Kr-83m	1.1E7	1.5E7
Kr-85m	2.5E7	3.6E7
Kr-85	4.3E5	1.1E6
Kr-87	4.7E7	7.2E7
Kr-88	6.5E7	1.0E8
Kr-89	8.2E7	1.3E8

Attachment 3

Changes to Baseline LOCA Control Room Dose Analysis for LEU Cores

The current UFSAR LOCA Control Room dose results reflect no Emergency Core Cooling System (ECCS) leakage, and the assumed Control Room unfiltered inleakage value is 10 cubic feet per minute (cfm). Accordingly, the analysis was updated to reflect ECCS leakage. In addition, a more conservative unfiltered inleakage assumption of 30 cfm was used, based upon Control Room tracer gas testing performed at Catawba in support of the Alternative Source Term analysis. The result of these changes is an increase in the LEU LOCA Control Room doses as shown in Attachment 1.

Unlike the current UFSAR LOCA Control Room dose analysis, the current UFSAR LOCA offsite dose calculations and results already account for ECCS leakage. Therefore, no changes are required to the UFSAR LOCA offsite dose calculations, and the results used previously in the MOX fuel lead assembly dose evaluations remain valid. The incorporation of ECCS leakage into the UFSAR LOCA Control Room dose analysis makes that analysis consistent with the LOCA offsite dose calculation.

The table below provides a listing of the key input parameters to the LOCA Control Room dose analysis. It provides a comparison of these inputs between (i) the results currently reported in the UFSAR and (ii) the results of the current Duke calculations (which incorporate the changes that are discussed above). The changes are shown in **bold**. In the current Duke calculations, the results of the analysis supporting the UFSAR have been conservatively adjusted to account for ECCS leakage and greater Control Room unfiltered inleakage.

The ECCS leakage model is based on Standard Review Plan (Revision 2) Section 15.6.5 Appendix B "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment." ECCS leakage starts at the time of sump recirculation. The ECCS leakage increased the Control Room thyroid doses by 8.5 rem from the current UFSAR value.

The change in Control Room unfiltered inleakage from 10 cfm to 30 cfm increases the Control Room thyroid dose by 7.2 rem.

The inclusion of both effects (ECCS leakage and higher Control Room unfiltered inleakage) raises the Control Room thyroid dose from the previously cited 5.3 rem to the currently calculated value of 21 rem.

Comparison of Control Room Dose Analysis Model Parameters for Loss of Coolant Accidents

Parameter	Values in the Analysis which Supports UFSAR[1]	Current Licensing Basis Analysis Value
Failed Fuel Percentage	100%	100%
Noble Gases Fraction Released to Containment	100%	100%
Iodine Fraction Released to Containment	25%	25%
Iodine Fraction Released to the Sump	50%	50%
Organic Iodine Fraction	4%	4%
Particulate Iodine Fraction	5%	5%
Elemental Iodine Fraction	91%	91%
Containment Volume	1.0154E6 ft^3	1.0154E6 ft^3
Upper Containment Volume	6.7E5 ft^3	6.7E5 ft^3
Lower Containment Volume[2]	3.45E5 ft^3	3.45E5 ft^3
Containment Leakage Rate • 0 – 24 hrs • > 24 hrs	 0.3% 0.15%	 0.3% 0.15%
Bypass Leakage Fraction	7%	7%
Control Room Pressurization Rate	2800 cfm	2800 cfm
Control Room Filtered Recirculation Rate	2000 cfm	2000 cfm
Control Room Unfiltered Leakage Rate	10 cfm	**30 cfm**
Control Room Volume	8.92E4 ft^3	8.92E4 ft^3
Control Room Iodine Filter Efficiency	99%	99%
Control Room Area Flow Rate	1200 cfm	1200 cfm
ECCS Leakage Rate[3] • 0.47 – 24 hrs • 24 – 24.5 hrs • >24.5 hrs	 0 0 0	 **7520 cc/hr** **50 gpm** **7520 cc/hr**
Time to Start Sump Recirculation[3]	0	**1700 sec**
ECCS Leakage Iodine Partition Fraction[3]	0	**10%**
Annulus Ventilation Iodine Filter Efficiency	95%	95%
Annulus Ventilation Flow Rate • 0 – 900 sec • > 900 sec	 9000 cfm 7650 cfm	 9000 cfm 7650 cfm
Time of Negative Annulus Pressure	95 sec	95 sec
Annulus Volume	4.84E5 ft^3	4.84E5 ft^3
Annulus Volume Credited	50%	50%

Parameter	Values in the Analysis which Supports UFSAR[1]	Current Licensing Basis Analysis Value
Ice Condenser Elemental Iodine Removal Efficiency • 0 – 600 sec • 600 - 4191 sec • > 4191 sec	 0% 30% 0%	 0% 30% 0%
Containment Spray Iodine Decontamination Factors • Elemental • Particulate • Organic	 5.9 100 0	 5.9 100 0
Spray Lambdas for Iodine Removal Rate • Elemental • Particulate • Organic	0.9 / hr 2.4 / hr 0	0.9 / hr 2.4 / hr 0
Control Room Atmospheric Dispersion Factors • 0 – 8 hrs • 8 – 10 hrs • 10 – 24 hrs • 1 – 4 days • > 4 days	 1.98E-3 sec/m^3 1.44E-3 sec/m^3 7.2E-4 sec/m^3 5.1E-4 sec/m^3 2.8E-4 sec/m^3	 1.98E-3 sec/m^3 1.44E-3 sec/m^3 7.2E-4 sec/m^3 5.1E-4 sec/m^3 2.8E-4 sec/m^3

Parameter	Values in the Analysis which Supports UFSAR[1]	Current Licensing Basis Analysis Value
Core Radioisotope Inventory		
I-131	8.9E7 Ci	8.9E7 Ci
I-132	1.3E8 Ci	1.3E8 Ci
I-133	1.9E8 Ci	1.9E8 Ci
I-134	2.1E8 Ci	2.1E8 Ci
I-135	1.8E8 Ci	1.8E8 Ci
Xe-131m	1.3E6 Ci	1.3E6 Ci
Xe-133m	5.6E6 Ci	5.6E6 Ci
Xe-133	1.9E8 Ci	1.9E8 Ci
Xe-135m	3.5E7 Ci	3.5E7 Ci
Xe-135	1.9E8 Ci	1.9E8 Ci
Xe-137	1.7E8 Ci	1.7E8 Ci
Xe-138	1.6E8 Ci	1.6E8 Ci
Kr-83m	1.1E7 Ci	1.1E7 Ci
Kr-85m	2.5E7 Ci	2.5E7 Ci
Kr-85	4.3E5 Ci	4.3E5 Ci
Kr-87	4.7E7 Ci	4.7E7 Ci
Kr-88	6.5E7 Ci	6.5E7 Ci
Kr-89	8.2E7 Ci	8.2E7 Ci
Dose Results		
See Attachment 1		

[1] List of parameters based on the "conservative case" of UFSAR Tables 15-40 and 15-41. Isotope list taken from Table 15-12.
[2] Typographical error in current UFSAR table. Value should be 3.45E5 ft^3.
[3] This model existed in the offsite dose analysis, it was a change to the Control Room analysis.

Attachment 4

Accident Thyroid Doses – MOX Fuel Lead Assembly Cores
Duke Methodology

Accident	Unit	Exclusion Area Boundary		Low Population Zone		Control Room	
		Limit (rem)	Dose (rem)	Limit (rem)	Dose (rem)	Limit (rem)	Dose (rem)
Locked Rotor	1	30	26.9 (23.6)	30	4.6 (4.1)	30	1.0 (0.9)
	2	30	27.8 (22.0)	30	4.5 (3.6)	30	1.4 (1.1)
Rod Ejection	1	75	22.3 (21.8)	75	17.8 (17.4)	30	6.6 (6.4)
	2	75	31.5 (30.7)	75	19.8 (19.3)	30	8.9 (8.7)
LOCA	Both	300	90.2 (89)	300	25.3 (25)	30	21.3 (21)

Note 1: The values in parentheses are the current LEU thyroid dose values for Catawba.

Note 2: This increase in dose due to MOX fuel is calculated as described by Duke in Reference 3.

Attachment 5

Updated LAR Radiological Consequences Information

November 3, 2003 Response to NRC Request for Additional Information Dated July 25, 2003

The Responses to Radiological Questions 3b and 3e are modified to reflect changes from (i) the March 16, 2004 Duke letter and (ii) changes arising from this letter. The modified responses are provided in their entirety below. Modified sections of the responses are **bold** and in brackets[ ].

Question 3:

b. Provide the numeric results of the analyses discussed in Sections 3.7.3, 4.2.1.3, and 5.6.3.1 in terms of the whole body and thyroid dose quantities, or total effective dose equivalent (TEDE), as appropriate to the licensing basis of Catawba and McGuire. Include offsite and control room doses.

Response

The MOX fuel lead assembly license amendment request has been amended to apply to Catawba only. Therefore, this response provides the requested information for Catawba only.

Evaluations

As was discussed in the response to Radiological Consequences Question 3(a), the thyroid dose is typically the limiting dose in that there is less margin between calculated thyroid dose and thyroid dose limit than between calculated whole body dose and whole body dose limit. **[Since the thyroid doses are more restrictive, the evaluation of the TID based accidents is performed based upon the thyroid doses for these accidents.]**

As also discussed in the response to Radiological Consequences Question 3(a), all of the mixed oxide (MOX) fuel lead assemblies are assumed to be in the affected fuel population for all of the accidents examined. Thus, for those scenarios where a significant portion of the core is affected, the increase in dose is calculated by taking the relative change in I-131 multiplied by the number of affected MOX lead assemblies (4), divided by the total population of the 193 fuel assemblies in the core affected. A 50% increase in the TID or AST (Reference Q3(b)-2) release fractions (discussed in the response to Radiological Consequences Question 3(g)) for the MOX fuel lead assemblies is modeled by multiplying by 1.5. For the LOCA, all fuel assemblies are assumed to be affected; **[for the locked rotor accident, 9.5% of the core is assumed to be affected for Unit 1 and 5.0% of the core for Unit 2;]** and for the rod ejection accident 50% of the core is assumed to be affected.

Because the existing analysis is based on a full core of low enrichment uranium (LEU) fuel assemblies, the evaluation is concerned with quantifying the increase in the dose result from replacing four LEU fuel assemblies with four MOX fuel lead assemblies. The dose result already includes the effects of the four LEU fuel assemblies, so the evaluation quantifies the increase from the replacement by accounting for the difference in the effects of these two types of fuel assemblies. It is this difference that results in the change in dose.

The LOCA evaluation is illustrated as follows:

- **[The number of MOX fuel lead assemblies assumed to be failed (4) is multiplied by the increase in I-131 inventory in MOX relative to LEU (1.09) and a factor to account for the increase in the release fraction being assumed for I-131 for MOX (1.5) to give 6.54. Note: the increase in I-131 release fraction is per the response to Radiological Consequences Question 3(g).**
- **The 6.54 value is added to the number of LEU assemblies assumed to be failed (193 - 4 = 189) to give 195.54.**
- **The 195.54 sum is divided by the total number of failed assemblies (193) to give 1.0132, the dose multiplier for a LOCA with four MOX fuel lead assemblies.**
- **Subtracting one from the dose multiplier gives 0.0132 or 1.32%, the fractional or percentage increase in dose due to four MOX fuel lead assemblies.**
- **The existing dose results are multiplied by the dose multiplier of 1.0132 to calculate the projected thyroid dose with four MOX fuel lead assemblies. For the EAB thyroid dose for the TID LOCA scenario, the current dose is 89 Rem. Multiplied by 1.0132, this gives a total projected thyroid dose of 90.2 Rem with four MOX fuel lead assemblies.**

The same calculational process yields an increase in the locked rotor thyroid dose of 13.9% for Unit 1 and 26.2% for Unit 2, and an increase in the rod ejection thyroid dose of 2.6% (both units). The resulting locked rotor EAB thyroid doses are 26.9 rem (Unit 1) and 27.8 rem (Unit 2). The resulting rod ejection EAB thyroid doses are 22.3 rem (Unit 1) and 31.5 rem (Unit 2). These results are summarized in Tables Q3(b)-2a and Q3(b)-2b.

It should be noted that this methodology provides a conservatively high estimation of the dose impact of using MOX fuel. For the locked rotor and rod ejection accidents that involve less than 100% fuel failure, the methodology assumes that all of the MOX fuel rods fail while only some of the LEU fuel rods fail, thereby magnifying the relative impact of MOX fuel.

The results do not appreciably reduce the margin for LOCA and rod ejection. For locked rotor, both the LEU and MOX lead assembly doses are within the acceptance criteria, even with the conservative assumption that all MOX fuel rods preferentially fail.

Recognizing the large conservatism inherent in applying this methodology to the LRA in particular, Duke further evaluated that accident. The actual amount of fuel that is predicted to fail during a LRA varies from cycle to cycle. Core thermal-hydraulic characteristics and core peaking influence the number of fuel rods that are calculated to experience DNB. Cycle-specific checks on calculated core peaking are performed to ensure that the amount of failed fuel from a LRA is less than the amount assumed in the current dose evaluation. Cycle-specific assessments for recently operated or operating Catawba cores resulted in no calculated fuel failures during a LRA. Duke performed a cycle-specific LRA analysis for Catawba 1 Cycle 16 (C1C16). The analyzed core design includes 72 feed LEU assemblies [Westinghouse Robust Fuel Assembly (RFA) design] and four feed MOX fuel lead assemblies. The design also incorporates eight once-burned Westinghouse Next

Generation Fuel (NGF) lead test assemblies. Thermal-hydraulic models specific to each fuel type (RFA, NGF, and MOX) were used to generate maximum allowable peaking limits for various axial power shapes. Core peaking values were calculated for C1C16 and compared to the maximum allowable peaking limits. The analysis, performed over a range of Cycle 16 conditions, showed that the calculated peaking does not exceed the DNB peaking limits for any fuel type (either MOX or LEU), so no fuel failures are calculated to occur during a C1C16 LRA. With no MOX fuel failure, there would be no incremental dose associated with a LRA that is attributable to the MOX fuel lead assemblies.]

Analyses

The impacts to doses in the evaluations described above are small due to the low number of MOX fuel lead assemblies as a portion of the population affected. The fuel handling accident scenarios provide for the most impact because MOX fuel comprises most or all of the fuel that is assumed to be damaged. The single assembly fuel handling accident (FHA) and the weir gate drop (WGD) accident were analyzed. For the purpose of comparison, the doses were computed using a MOX fuel source term and an equivalent LEU source term (see response to Radiological Consequences Question 3(f) for further discussion of equivalent MOX and LEU fuel assemblies). The unit vent is the limiting release point.

As discussed in the response to Radiological Consequences Question 3(g), increases in AST release fractions were examined to determine the sensitivity to these values. These and other results of the analyses for MOX and LEU for the FHA and WGD are presented in Tables Q3(b)-3 and Q3(b)-4. Included in these tables is an evaluation of the results for the Catawba AST LOCA including MOX lead assemblies. This analysis is currently under NRC Staff review (Reference Q3(b)-3).

For the FHA, consequences increased by about 9%. This matches the single assembly FHA increase in I-131 activity for MOX fuel relative to LEU (see response to Radiological Consequences Question 3(a) "Ratio of I-131 for MOX and LEU Fuel for Accident Evaluations"). Because the WGD involves both MOX and LEU fuel, the effect of the MOX fuel is reduced. The increase in I-131 for the WGD accident was 5% and the increase in dose results was approximately the same. Additionally, when the MOX release fractions were increased to account for differences in gap fractions, the impact on the results (relative to MOX fuel analyses with standard release fractions) was an increase in similar proportion to the I-131 release fraction. These results demonstrate that methodology involving the use of an I-131 activity ratio for evaluating accident consequences (as discussed and demonstrated above) provides valid results.

The dose consequences of the fresh fuel assembly drop analysis are reported in Section 3.7.3.5 of Attachment 3 to Reference Q3(b)-4.

See responses to Reactor Systems Question 12 and Radiological Consequences Questions 3(a), 3(f) and 3(g) for additional information.

References

Q3(b)-1 USAEC, Technical Information Document (TID) 14844, "Calculation of Distance

Factors for Power and Test Reactor Sites," March 1962.

Q3(b)-2 USNRC Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

Q3(b)-3 Letter from G. R. Peterson (DPC) to USNRC, "Proposed Technical Specifications and Bases Amendment: Technical Specification and Bases 3.6.10 Annulus Ventilation System, 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 and Bases 5.5.1 Ventilation Filter Testing Program (VFTP)," November 25, 2002.

Q3(b)-4 Tuckman, M.S., February 27, 2003, Letter to U.S. Nuclear Regulatory Commission, 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.

Table Q3(b)-1
Offsite and Control Room Doses with LEU Cores and Projected Doses with MOX Lead Assembly Cores for LOCAs with TID and AST Releases

Receptor	**TID Dose Limit** (Rem Thyroid)	**All LEU Core & TID Releases** (Rem Thyroid)	**MOX Lead Assemblies & Increased TID Releases** (Rem Thyroid)	**TEDE Dose Limit** (Rem TEDE)	**All LEU Core & AST Releases** [2] (Rem TEDE)	**MOX Lead Assemblies & Increased AST Releases** (Rem TEDE)
EAB	300	89	90.2	25	7.2	7.3
LPZ	300	25	25.3	25	4.0	4.1
Control Room	30	[21]	[21.3]	5	2.7	2.7

[ ] Shows values changed or provided for the first time in this submittal
[2] Reference Q3(b)-3

Table Q3(b)-2a
Offsite and Control Room Thyroid Doses with Full LEU Cores and Projected Thyroid Doses with MOX Lead Assemblies for Locked Rotor and Rod Ejection Accidents for CNS Unit 1

Receptor	Locked Rotor Dose Limit (Rem)	Locked Rotor with all LEU Core[1] (Rem)	Locked Rotor with MOX Lead Assemblies[2] (Rem)	Rod Ejection Dose Limit (Rem)	Rod Ejection with all LEU Core[1] (Rem)	Rod Ejection with MOX Lead Assemblies[2] (Rem)
EAB	30	[23.6]	[26.9]	75	[21.8]	[22.3]
LPZ	30	[4.1]	[4.6]	75	[17.4]	[17.8]
[Control Room]	[30]	[0.9]	[1.0]	[30]	[6.4]	[6.6]

[1] Standard TID releases.
[2] Increased TID releases.
[ ] Shows values changed or provided for the first time in this submittal

Table Q3(b)-2b
Offsite and Control Room Thyroid Doses with Full LEU Cores and Projected Thyroid Doses with MOX Lead Assemblies for Locked Rotor and Rod Ejection Accidents for CNS Unit 2

Receptor	Locked Rotor Dose Limit (Rem)	Locked Rotor with all LEU Core[1] (Rem)	Locked Rotor with MOX Lead Assemblies[2] (Rem)	Rod Ejection Dose Limit (Rem)	Rod Ejection with all LEU Core[1] (Rem)	Rod Ejection with MOX Lead Assemblies[2] (Rem)
EAB	30	[22.0]	[27.8]	75	[30.7]	[31.5]
LPZ	30	[3.6]	[4.5]	75	[19.3]	[19.8]
[Control Room]	[30]	[1.1]	[1.4]	[30]	[8.7]	[8.9]

[1] Standard TID releases.
[2] Increased TID releases.
[ ] Shows values changed or provided for the first time in this submittal

Note: Tables Q3(b)-2a and Q3(b)-2b present information for Units 1 and 2, respectively. Previously, the results did not vary between units, so all information was provided in a single Table Q3(b)-2.

Question 3:

e. If the analyses used methods, inputs, or assumptions different from that in the current licensing basis (CLB) for Catawba or McGuire, provide a justification for each change from the CLB.

Response

The MOX fuel lead assembly license amendment request has been amended to apply to Catawba only. Therefore, this response addresses Catawba only.

As discussed the response to Radiological Consequences Question 3(j), the current source term licensing basis for Catawba is transitioning from classical source term modeling (TID, Reference Q(e)-1) to Alternative Source Term (AST, Reference Q(e)-2), but the evaluations and analyses that were performed in support of the mixed oxide (MOX) fuel lead assemblies submittal were in keeping with the current licensing basis (CLB) of Catawba at the time it was submitted. The only difference between the current licensing basis analyses and the MOX fuel analysis for the spent fuel handling accidents is the use of MOX fuel, which is the subject of the license amendment request (LAR). Methods, inputs and assumptions not directly related to MOX fuel were not changed for the evaluations and analyses to accommodate MOX fuel lead assemblies (see response to Radiological Consequences Question 3(a)). The models used in the evaluations and analyses of accidents with MOX fuel lead assemblies were based upon those in place for the analyses supporting the Catawba CLB. **[The Catawba UFSAR is being updated to ensure consistency with the CLB.]**

Formal analyses were performed for the single spent fuel assembly accident and the weir gate drop accident (involving seven fuel assemblies). Evaluations were performed for those accidents where the four MOX lead assemblies would account for **[a smaller fraction]** of the damaged fuel population (see responses to Radiological Consequences Questions 3(a) and 3(b) for further information). These evaluations were based upon the comparison of isotopics between a MOX and an equivalent low enrichment uranium (LEU) fuel assembly (see further discussion in response to Radiological Consequences Question 3(f)).

Catawba is currently licensed to use Alternative Source Term (Reference Q3(e)-2) technology for fuel handling accidents. AST-based MOX fuel release calculations were performed to provide a baseline analysis using MOX fuel source terms and AST gap fractions. Cases were also run with a "raised" AST-based source term. This source term increased the release fractions but maintained the AST isotopic release distribution. The raised AST release analyses were performed to provide a bounding safety case of higher gap release fractions for the MOX fuel lead assemblies. The use of these gap fractions cases are discussed in response to Radiological Consequences Question 3(g).

There is no formal analysis of a dropped fresh LEU fuel assembly in air for Catawba. For MOX fuel it is appropriate to analyze this variation of the FHA because the potential for dose consequences are more severe, due to the presence of plutonium in the fuel pellets.

See responses to Radiological Consequences Questions 3(a), 3(b), 3(f), 3(g), and 3(j) for further information.

References

Q3(e)-1 USAEC, Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962.

Q3(e)-2 USNRC Regulatory Guide 1.183, "Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

March 1, 2004 Response to NRC Request for Additional Information Dated February 4, 2003

The "Response Conclusion" is modified as described below to reflect changes arising from this letter.

Delete the last sentence ("Examining up to a ...").

Add the following paragraph.

> The doses for loss of coolant accident (LOCA), rod ejection accident, and locked rotor were also calculated assuming the NRC value of I-131 in MOX fuel relative to low enriched uranium fuel (15.8% higher), rather than the Duke value of 9% (see discussion in Reference Q1-7, Section 3.2.5). These doses reflect operation with four MOX fuel lead assemblies and include the conservative assumption that all rods in all four MOX fuel lead assemblies preferentially fail. As seen in Tables Q1-1, Q1-2, and Q1-3 below, the dose values using the NRC I-131 ratio value, while slightly greater than those in Tables Q3(b)-1, Q3(b)-2a, and Q3(b)-2b in this letter, are nonetheless within the applicable acceptance criteria.

Add the following tables.

Table Q1-1
Projected NRC Computation of Locked Rotor, Rod Ejection, and LOCA EAB Thyroid Doses for CNS Units 1 and 2 with Four MOX Lead Fuel Assemblies

Accident	CNS Unit 1	CNS Unit 2	Limit
Locked Rotor	27.4 Rem	28.7 Rem	30 Rem
Rod Ejection	22.4 Rem	31.7 Rem	75 Rem
LOCA	90.4 Rem	90.4 Rem	300 Rem

Table Q1-2
Projected NRC Computation of Locked Rotor, Rod Ejection, and LOCA LPZ Thyroid Doses for CNS Units 1 and 2 with Four MOX Lead Fuel Assemblies

Accident	CNS Unit 1	CNS Unit 2	Limit
Locked Rotor	4.7 Rem	4.6 Rem	30 Rem
Rod Ejection	17.9 Rem	19.8 Rem	75 Rem
LOCA	25.4 Rem	25.4 Rem	300 Rem

Table Q1-3
Projected NRC Computation of Locked Rotor, Rod Ejection, and LOCA Control Room Thyroid Doses for CNS Units 1 and 2 with Four MOX Lead Fuel Assemblies

Accident	CNS Unit 1	CNS Unit 2	Limit
Locked Rotor	1.0 Rem	1.4 Rem	30 Rem
Rod Ejection	6.6 Rem	9.0 Rem	30 Rem
LOCA	21.3 Rem	21.3 Rem	30 Rem

Add the following reference.

Q1-7 U. S. Nuclear Regulatory Commission, Safety Evaluation for Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies, April 5, 2004.

Attachment 6

Updated No Significant Hazards Consideration (NSHC) Evaluation

Section 4.2.1.3 of the NSHC is updated to read as follows. Modified sections of the responses are **bold and in brackets[ ]**.

4.2.1.3 Consequences Evaluation

In order for a postulated accident to result in a significant increase in consequences, it must be shown that the accident results in a significant increase in dose to the public or to the Control Room operators. The UFSAR for both McGuire and Catawba contain the results of dose calculation for those accidents which have offsite or Control Room operator dose consequences. The dose consequences of these accidents were conservatively evaluated for a core consisting of four MOX fuel assemblies and 189 LEU fuel assemblies. The limiting design basis accidents for operations involving MOX fuel assemblies are the fuel handling accident and weir gate drop accident. **[The calculated dose consequences are within the acceptance criteria for both the fuel handling accident and the weir gate drop accident.]**

The insertion of MOX fuel lead assemblies would have a small effect on calculated radiation doses **[for other accidents. For the locked rotor accident, the maximum increase in thyroid dose to the public would be 5.8 rem at the Unit 2 Exclusion Area Boundary. This evaluation conservatively assumes that all MOX fuel rods preferentially fail and release radionuclides during the accident. The conservatism of this assumption was demonstrated by a cycle-specific evaluation of Catawba 1 Cycle 16, which calculated that no MOX fuel rods should experience departure from nucleate boiling and fail during a design basis locked rotor accident. For loss of coolant accidents (LOCAs) and rod ejection accidents, the increase in thyroid dose consequences due to MOX fuel lead assemblies would be even smaller, with a maximum increase of 1.2 rem at the Exclusion Area Boundary following a LOCA. The calculated doses of all accidents are within the acceptance criteria.]**

Based on this evaluation, it is concluded that amending the McGuire and Catawba licenses to allow the receipt, handling, storage, and use of MOX fuel lead assemblies does not result in a significant increase in the consequences of any accident previously evaluated in the UFSAR.

Attachment 7

Updated Environmental Report

Section 5.6.3.1 of the Environmental Report is updated to read as follows. Modified sections of the responses are **bold and in brackets[ ]**.

5.6.3.1 Design Basis Accidents

Based on a review of the various accident scenarios in the respective Safety Analysis Reports for McGuire and Catawba, it was determined that MOX fuel lead assemblies had the most impact on the results of the fuel handling and weir gate drop accidents. **[The calculated dose consequences for the MOX fuel lead assemblies are within the acceptance criteria for both accidents.**

The insertion of MOX fuel lead assemblies would have a small effect on calculated radiation doses for other accidents. For the locked rotor accident, the maximum increase in thyroid dose to the public would be 5.8 rem at the Unit 2 Exclusion Area Boundary. This evaluation conservatively assumes that all MOX fuel rods preferentially fail and release radionuclides during the accident. The conservatism of this assumption was demonstrated by a cycle-specific evaluation of Catawba 1 Cycle 16, which calculated that no MOX fuel rods should experience departure from nucleate boiling and fail during a design basis locked rotor accident. For loss of coolant accidents (LOCAs) and rod ejection accidents, the increase in thyroid dose consequences due to MOX fuel lead assemblies would be even smaller, a maximum of 1.2 rem higher at the Exclusion Area Boundary following a LOCA. The calculated doses of all accidents are within the acceptance criteria.]

The consequences of a drop of a fresh MOX fuel assembly in air were also calculated. The analysis assumed the drop of a complete MOX fuel assembly with resultant damage to the assembly. Specifically, cladding damage was postulated to occur and fuel pellet damage was assumed, which resulted in the airborne release of a respirable fraction of particulate nuclides. The activity was then transported to a receptor at the site boundary with resulting exposure from the particulate activity. Exposure was computed using Federal Guidance Report 11 (Reference 4) conversion factors. Even using extremely conservative assumptions with no credit for ventilation system filters, the resulting calculated dose was less than 0.4 rem, which is well below regulatory limits for design basis accidents.

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[4] EPA (Environmental Protection Agency) Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion*, EPA-520/1-88-020, September 1988.