Duke Power®
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GARY R. PETERSON
Vice President
McGuire Nuclear Station

Duke Power
MG01VP / 12700 Hagers Ferry Road
Huntersville, NC 28078-9340

704 875 5333
704 875 4809 fax
grpeters@duke-energy.com

April 22, 2004

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Corporation (Duke)
McGuire Nuclear Station Units 1 and 2
Docket Numbers 50-369 and 50-370
Technical Specifications Amendment
Request for Additional Information (RAI); TS 3.7.15 - Spent Fuel Assembly Storage, and TS 4.3 - Fuel Storage

Reference: (1) Duke letter to NRC, dated September 29, 2003, and
(2) NRC letter to Duke, dated March 8, 2004 (TAC NOS. MC0945 AND MC0946)

This letter provides additional information that was requested by the NRC staff in the above referenced NRC letter and further clarified during several teleconference calls. The NRC staff's questions and Duke's responses are provided in the following attachment. Also, as discussed during the teleconference call on April 15, 2004, Duke will be submitting responses to Questions No. 11 and 19 at a later date. The responses to these two questions will be submitted to the NRC prior to May 21, 2004.

Please contact Norman T. Simms of Regulatory Compliance at 704-875-4685 with any questions with respect to this matter.

Very truly yours,

G. R. Peterson

Attachments

A001

xc: (w/attachment)

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

J.B. Brady
NRC Senior Resident Inspector
McGuire Nuclear Station

J.J. Shea, Project Manager (addressee only)
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop O-7D1A
11555 Rockville Pike
Rockville, MD 20852-2738

Beverly O. Hall, Section Chief
Radiation Protection Section
1645 Mail Service Center
Raleigh, N.C. 27699-1645

Gary R. Peterson, being duly sworn, states that he is Vice President of McGuire Nuclear Station; that he is authorized on the part of Duke Energy Corporation to sign and file with the U.S. Nuclear Regulatory Commission these revisions to the McGuire Nuclear Station Facility Operating Licenses Nos. NPF-9 and NPF-17; and, that all statements and matters set forth therein are true and correct to the best of his knowledge.

Gary R. Peterson, Vice President
McGuire Nuclear Station
Duke Energy Corporation

Subscribed and sworn to before me on April 22, 2004.

Notary Public

My Commission Expires: August 17, 2006

ATTACHMENT 1

Responses to Nuclear Regulatory Commission Request for Additional Information Related to McGuire Nuclear Stations's License Amendment Request for No-Boraflex Credit

Question 1

In the note at the bottom of the proposed Technical Specifications (TS) Tables 3.7.15-1 through 3.7.15-4, Duke Energy Corporation (Duke, the licensee) states the following: "Fuel which differs from those designs used to determine the requirements of Table 3.7.15-[*] may be qualified for use as a Region 2 [**] Assembly by means of an analysis using NRC approved methodology to assure that the k_{eff} [effective multiplication factor] is less than 1.0 with no boron and less than or equal to 0.95 with credit for soluble boron." Where * indicates the applicable table number and ** indicates the appropriate fuel assembly classification; Unrestricted, Restricted, Filler, or Empty Checkerboard. Please provide additional information to identify and describe the Nuclear Regulatory Commission (NRC) approved methodology that will be employed to qualify assemblies for storage as one of the aforementioned fuel assembly classifications.

Response

The NRC approved methodology for qualifying assemblies for storage referred to in the note at the bottom of the proposed Technical Specifications Tables 3.7.15-1 through 3.7.15-4 refers specifically to the methodology supporting this License Amendment Request which NRC is reviewing. This language and approach is consistent with that used in previous license amendments and TS fuel qualification tables for the McGuire spent fuel pools.

Question 2

After reviewing the licensee's proposed TS changes, the NRC staff determined that the use of Boral poison panel inserts should be described in Section 4.0, "Design Features," of the MNS TSs. The staff determined that a reference to the Boral inserts satisfies 10CFR50.36, "Technical Specifications." Specifically, 10CFR50.36(c)(4), *Design Features*, states that, "Design features to be included [in the TSs] are those features of the facility such as materials of construction and geometric

arrangements, which, if altered or modified, would have a significant effect on safety..."

Since Boral is a material of construction, which if altered or modified would have a significant effect on safety, please provide a revised technical specification requirement that accounts for the use of Boral in this application.

Response

A new technical specification is added as 4.3.1.1.f, which states the following: *"Neutron absorber (Boral) installed between fuel assemblies in the Region 1 racks."* (See Attachment 2)

Question 3

In Attachment 6, Section 3, "Fuel Assembly Designs Considered," the licensee stated that the Burnable Poison Rod Assembly (BPRA) designs used in the MkBI and MkBW fuel can have variable Boron-10 content. Additionally, the licensee stated that it assumed boron carbide (B_4C) loadings of 1.4 weight percent for MkBI assemblies and 4.0 weight percent for MkBW assemblies and that these loadings are at, or "very near" to, the highest boron concentrations used in the BPRAs for these fuel types. Please describe the analysis or evaluation that was performed to determine the maximum B_4C loading used in these assembly types. Additionally, if the licensee could not ascertain the maximum bounding B_4C loading used, the NRC staff requests that the licensee justify why it did not include an appropriate bias or uncertainty in the spent fuel pool (SFP) criticality analyses.

Response

The MkBI fuel type in the McGuire SFP criticality analysis represents, as Section 3 in Attachment 6 of the LAR states, old Oconee MkB2, MkB3, and MkB4 irradiated fuel. Of the 300 MkBI fuel assemblies currently being stored in the McGuire SFP Region 2 racks, only twenty (20) once contained BPRAs with a B_4C concentration (1.43 wt %) that exceeded the 1.40 wt % value used for the SFP criticality analysis.

A re-examination of the MkBI criticality cases shows that increasing the maximum B_4C concentration from 1.40 wt % to 1.43 wt % yields a k_{eff} increase of less than 0.00007 Δk for discharge MkBI fuel stored in the McGuire SFPs. While this is practically

a negligible increase, it is also noteworthy that these 20 MkBI assemblies were discharged in May 1977, and so currently have a cooling time of over 26 years. For the discharge MkBI fuel assemblies in the McGuire SFPs, an additional seven years of cooling time beyond the maximum of 20 years allowed for credit (see page 35 in Attachment 6 of the LAR) is worth nearly another 0.01 Δk reduction. This unused margin is more than 100 times the amount of the reactivity penalty for increasing the MkBI BPRA concentration from 1.40 to 1.43 wt % B_4C.

The highest B_4C concentration ever used for BPRAs in MkBW fuel assemblies irradiated at McGuire is 4.0 wt %. Therefore, the McGuire SFP criticality analysis used a bounding BPRA model for the MkBW fuel design.

Question 4

In Attachment 6, Section 3, the licensee stated that the "WABA" and "Pyrex" BPRAs contained a standard Boron-10 content. However, in Table 4, "Design Data for Burnable Poison Rod Assemblies (BPRAs) Considered in the McGuire SFP Region 2 Criticality Analysis," the licensee did not provide the uncertainty in any of the key design parameters, such as poison pellet density, poison pellet inside and outside radii, or Boron-10 concentration. Please describe how the tolerances in these and the other parameters in Table 4 were accounted for in the Region 2 SFP criticality analysis.

Response

In accordance with past practice in quantifying the reactivity penalty associated with irradiation of fuel assemblies that once contained BPRAs, only the nominal data for the BPRA design parameters (poison density, pellet radial dimensions, and B-10 content) were considered for the analysis supporting this LAR. Uncertainties in these parameters are second-order effects which, when combined with the other independent reactivity contributors in the mechanical, method, and burnup uncertainties, would be expected to result in a negligible increase in the total 95/95 k_{eff}.

To confirm this, conservative uncertainties (±10% for poison density, ±2% for pellet outside radius, and ±5% for B-10 concentration) have been evaluated for "WABA" BPRAs in W-RFA fuel. Individually, these variations result in maximum SFP k_{eff} increases of 0.00045 Δk, 0.00077 Δk, and 0.00024 Δk, respectively. When these BPRA-related uncertainties are statistically combined with the independent uncertainties listed

in Table 22 of Attachment 6, the overall 95/95 k_{eff} increase is less than 0.00003 Δk. This is a negligible reactivity penalty, especially when compared to the k_{eff} margin to 1.0 in Table 22 (0.00112 Δk), as well as the conservatisms listed in the response to Question 10.

Question 5

In Attachment 6, Section 6, "Computation of the Maximum 95/95 k_{eff}," the licensee described the Fixed Poison Self-Shielding Bias it included in the criticality analysis for Region 1. Please provide additional information to describe the analysis that was performed to determine the value of the bias and how it determined that the bias was appropriately conservative.

Response

The Fixed Poison Self-Shielding Bias is defined as the bias that accounts for the slight self shielding effects associated with the clustering of boron carbide particles. This bias is only applicable to the Region 1 fuel storage racks, since Region 2 no longer takes credit for the installed Boraflex.

The methodology used to determine the extent of non-conservatism presented by the code's homogenization of the Boral can be represented in three steps:

1) Determination of self shielding factor (f_o)
2) Determining f_o's effect on transmission probabilities
3) Equating the changes in transmission probabilities to a value of Δk for use as a bias.

Calculations were performed using SCALE 4.4 with the 238 group cross section library. Input parameters of most importance are those related to the Boral panel (e.g., B-10 atomic weight percent), for which conservative values are assumed. The resulting bias is +0.0010 Δk and is the maximum value calculated from all evaluated variations and combinations of temperature and soluble boron concentration. This bias is conservatively applied to all conditions in the 95/95 k_{eff} calculations. Furthermore, it was assumed that the particle size distribution is that distribution that yields the most conservative bias, while the actual distribution would yield a smaller bias. For these reasons, the Fixed Poison Self-Shielding Bias is judged to be conservative.

Question 6

In Table 5, "Pertinent 95/95 Biases and Uncertainties to be Considered in the McGuire New Fuel Vault (NFV) and SFP Criticality Analysis," the licensee shows that the Monte Carlo Computational Uncertainty is not included in the SFP Region 2 analyses. However, the licensee stated in Section 4 that KENO V.a was used in the verification of the Checkerboard/Empty configurations used in the Region 2 analyses. Also the licensee stated that the SFP Region 2 calculations used 600 neutron generations for KENO V.a. Please identify whether the Monte Carlo Computational Uncertainty was included in any Region 2 analysis. For example, the NRC staff requests that the licensee state whether the Monte Carlo Computational Uncertainty was included in the verification analyses for the Checkerboard/Empty configurations.

Response

The biases and uncertainties that are included for the Region 2 criticality calculations, as shown in Table 5, are applicable to those calculations performed using CASMO-3/SIMULATE-3. It is important to note that all 95/95 k_{eff} calculations for Region 2 were performed using CASMO-3/SIMULATE-3. KENO V.a was used to verify the validity of the CASMO-3/SIMULATE-3 Checkerboard/Empty model (Table 17 of the LAR).

In Table 17 of the LAR, the KENO V.a results presented are the KENO V.a calculated k_{eff} without correction for the monte carlo uncertainty. Although these are not presented with correction for the monte carlo uncertainty, this uncertainty was considered when comparing the CASMO-3/SIMULATE-3 results with the KENO V.a results. Table 6.1 includes the monte carlo uncertainty for the KENO V.a calculations, as well as all code related biases and uncertainties for all cases. It is noted that the KENO sigma is approximately 0.00090 for most cases and the 95/95 one-sided tolerance factor is 1.752 for 600 generations.

Table 6.1 Comparison Between SIMULATE and KENO V.a Cases for Various Checkerboard/"Empty" Storage Configurations
{all cases at 150 °F, 0 ppm boron, 2.00 w/o STD Fuel}

Region 2 Storage Configuration	SIMULATE k_{eff} using model with fissile material in "water hole"	Avg KENO V.a k_{eff} using model with fissile material in "water hole"	KENO V.a k_{eff} using model with NO fissile material in "water hole"
All Checkerboard Assemblies (4/4)	1.2050	1.2032	1.2037
3 Checkerboard / 1 "Empty" (3/4)	1.0690	1.0696	1.0616
2 Checkerboard / 2 "Empty" (2/4)	0.8677	0.8594	0.8319
1 Checkerboard / 3 "Empty" (1/4)	0.7628	0.7658	0.7463
All "Empty" Cells (0/4)	0.2083	0.2099	-

Question 7

In Attachment 6, Section 4, "Criticality Computer Code Validation," the licensee describes the mechanical tolerances considered in calculating the Mechanical Uncertainty term. To better aid the NRC staff in evaluating the acceptability of the Mechanical Uncertainty values used in the licensee's criticality analyses, please provide a table, similar to Table 5, listing the following: 1) a detailed list of all tolerances included in the criticality analysis, 2) a summary of which tolerances were considered in each criticality analysis, 3) the value of the tolerance, and 4) the reactivity effect (delta-k) for each tolerance.

Response

Represented in Table 7.1 are the values of individual uncertainties due to mechanical tolerances, and the total mechanical uncertainty that is applied in the 95/95 calculation of k_{eff}. The values of the mechanical tolerances are not provided since this data is proprietary, however, the mechanical uncertainties are the maximum calculated using the maximum and minimum bounds of the tolerances. The total mechanical uncertainty shown for the New Fuel Vault (NFV), is the largest value calculated over the range of fuel types and moderator density, and is applied to all conditions. The values for the total mechanical uncertainties shown for Region 1 and Region 2

of the spent fuel pool are calculated for both the no-soluble-boron and soluble boron conditions and have been shown to be bounding over the burnup and cooling ranges used in the 95/95 calculation of k_{eff}.

Table 7.1 Table of Mechanical Uncertainties

	NFV	Region 1 (no boron)	Region 1 (310 ppm boron)	Region 2 (no boron)	Region 2 (800 ppm boron)
Enrichment	0.00366	0.00242	0.00345	0.00197	0.00257
Theoretical Density	0.00316	0.00256	0.00390	0.00009	0.00201
Fuel Pellet Dishing	0.00202	0.00183	0.00335	0.00006	0.00121
Fuel Pellet OD	0.00158	0.00200	0.00195	0.00003	0.00033
Fuel Clad OD	0.00481	0.00258	0.00366	0.00079	0.00020
Assembly CTC Spacing	0.00178	0.00299	0.00597	N/A	N/A
Assembly Eccentricity	0.00176	0.00246	0.00157	0.00169	0.00165
Storage Cell ID	N/A	0.00589	0.00647	0.00625	0.01100
Storage Cell Thickness	N/A	0.00227	0.00558	0.00876	0.00442
Boral Width	N/A	0.00247	0.00191	N/A	N/A
Plenum Thickness	N/A	0.00203	0.00164	N/A	N/A
Total Mech. Unc.	**0.00785**	**0.00973**	**0.01324**	**0.01110**	**0.01247**

Notes: 1) The NFV does not have cell walls, therefore, there are no uncertainties related to the cell ID or cell wall thickness.
2) In Region 1, only the Boral width uncertainty is evaluated, all other Boral parameters are modeled at the bounding values.
3) In Region 2, the CTC spacing uncertainty is inherent in the cell ID uncertainty because of the egg-crate design. Furthermore, there are no uncertainties related to Boral or the plenum due to the rack design.

This following equation is used to calculate the final mechanical uncertainty:

$$\Delta k_{MechUnc} = \sqrt{\sum_i [(k_i - k_{nom})^2 + (f_{95} \cdot \sigma_i)^2] + (f_{95} \cdot \sigma_{nom})^2}$$

where,
k_{nom} = CASMO or KENO Calculated k_{eff} for nominal mechanical parameters,
σ_{nom} = KENO one-sigma uncertainty for nominal mechanical parameters,
k_i = CASMO or KENO Calculated k_{eff} for mechanical

tolerance (i),

σ_i = KENO one-sigma uncertainty for mechanical tolerance (i),

f_{95} = 95th percentile one-sided tolerance factor (1.727 for 1000 generations, Region 1 assumed a more conservative value of 1.778)

It is noted that for Region 2, CASMO was used to calculate the individual uncertainties except for the assembly eccentricity. Therefore, for all but the assembly eccentricity, there is no associated value for σ. Also note that the individual uncertainties in Table 7.1 are represented by the following term:

$$\Delta k_{unc} = \sqrt{(k_{unc} - k_{nom})^2 + (f_{95} \cdot \sigma_{unc})^2}$$

Table 7.2 provides the data for a sample calculation of the mechanical uncertainty of the New Fuel Vault.

Table 7.2 Sample Calculation of the New Fuel Vault Mechanical Uncertainty

	k-eff	sigma
Nominal	0.93229	0.00087
Enrichment	0.93545	0.00107
Theoretical Density	0.93502	0.00092
Fuel Pellet Dishing	0.93337	0.00099
Fuel Pellet OD	0.93278	0.00087
Fuel Clad OD	0.93682	0.00094
Assembly CTC Spacing	0.93330	0.00085
Assembly Eccentricity	0.93302	0.00093
Total Mechanical Uncertainty = 0.00785		

Question 8

In Attachment 6, Section 4, the licensee described the Burnup Computational Uncertainty used in the MNS spent fuel pool criticality analyses. The licensee stated that it had determined the bounding uncertainty as a function of burnup and provided the equation it will use to calculate this uncertainty. However, the licensee did not provide detailed information to demonstrate that it had determined the appropriate burnup dependent uncertainty or that its equation was indeed bounding. Therefore, please provide additional information describing the methodology employed to determine the burnup dependent uncertainty as well as the means used to demonstrate that this equation was truly bounding.

Response

The Burnup Computational Uncertainty is defined as the uncertainty in the ability of the CASMO-3/SIMULATE-3 codes to accurately determine the isotopic content and the associated k_{eff} of a system with this isotopic concentration. This uncertainty is represented as a linear function of assembly average burnup.

The overall approach for the calculation of the Burnup Computational Uncertainty is to find the difference between the measured and predicted values for the reactivity of burned fuel assemblies with the same burnup (i.e., uncertainty in the calculated isotopics). Specifically, the reactivity difference is determined based upon the difference in the measured to predicted boron concentration, and the related boron worth, from the McGuire operating cycles. In this sense, the reactivity difference (Δk) was computed over a range of fuel burnups. From the results of this analysis it was seen that there was no observable trend in the reactivity difference with respect to burnup. This is attributed to the fact that the calculated reactivity difference included other uncertainties (e.g., uncertainties related to core operating conditions, and the B-10 concentration, etc.) in addition to the uncertainty in calculated isotopics. Due to limitations in the methodology, these uncertainties could not be separated to determine the reactivity difference solely attributable to the calculated isotopics. Therefore, it was assumed that the maximum burnup computational uncertainty calculated over the range of evaluated burnups, 0.00454 Δk, would occur at 50 GWd/MTU, and would be zero at 0 GWd/MTU. This approach is conservative since the uncertainty being applied is a composite of several uncertainties, of which the uncertainty in calculated isotopics represents only a minor component.

Question 9

Table 6 of Attachment 6 provides the bounding criticality analysis for storage of fuel in the NFV. The licensee stated that fuel assemblies are stored in the NFV without any location restrictions. Please identify whether the NFV criticality analysis assumed a uniform loading of the highest reactivity assembly type or a co-location of various assembly types in adjacent cells within the NFV. Please describe how the bounding storage configuration for the NFV was determined.

Response

As stated in Section 7, there are no restrictions on the storage configuration of new fuel assemblies in the New Fuel Vault. The calculations that support this determination modeled the New Fuel Vault with the same fuel assembly type at the maximum permissible enrichment in every storage cell. In this sense, the mixing of different fuel types in the New Fuel Vault was not analyzed. The assumption that all fuel assemblies will be of the same type is derived from the historical and anticipated fresh fuel batch composition, which has always been and is always expected to be the same fuel type.

Question 10

The licensee stated that "Extensive historic and projected 3D burnup, temperature, boron, and burnable poison data are employed to appropriately quantify the isotopic content of the fuel designs considered." Please describe the analysis that was performed and assumptions that were used to demonstrate that appropriately conservative values of the aforementioned parameters were used and that the reactivity of the spent fuel assemblies was maximized.

Response

The McGuire SFP criticality analysis for Region 2 storage attempted to determine a conservative, but realistic, isotopic content for irradiated fuel. Instead of trying to choose bounding values for the five pertinent irradiation history parameters (viz., axial burnup, moderator temperature, fuel temperature, soluble boron concentration, and burnable poison exposure), the analysis used all available core follow data for the 3-D values of these variables in actual discharged fuel assemblies.

As noted in Section 8.2 of Attachment 6, complete historic 3-D

core follow data were available in CASMO-3 / SIMULATE-3 format for the W-STD, W-OFA, MkBW, MkBWb1, and MkBWb2 assemblies that have been employed in the McGuire reactors. These data are considered realistic and appropriate for use in the SFP criticality analysis, because they are effectively used as a benchmark against irradiated fuel core reactivity measurements, in the determination of the burnup computational uncertainty. The evaluation of this uncertainty, as discussed on page 42 of Attachment 6, showed no definitive bias, indicating that the core follow calculations are accurately determining the isotopic content of irradiated fuel in the McGuire reactors. Conservatism in using these core follow data for the W-STD, W-OFA, MkBW, MkBWb1, and MkBWb2 designs - in SFP storage applications - is assured by the following three considerations:

- As noted on page 26 in Attachment 6 of the LAR, the individual burnup "group" axial profile for BPRA exposure that yields the highest "average" BPRA exposure is applied to all burnup groups.
- As noted on page 8 in Attachment 6, the maximum possible number of BPRA rodlets (24) are assumed present for any fuel irradiation with BPRAs inserted. In reality, most of the assemblies irradiated in the McGuire reactors contained fewer than 24 BPRA rodlets (typically between 12 and 20 rodlets). A reduction in the BPRA rodlets from 24 to 16 reduces the assembly reactivity of discharge fuel stored in the SFP by more than 0.002 Δk.
- The total mechanical uncertainties discussed on page 15 in Attachment 6 were determined using fresh fuel. Calculations indicate that with typical discharge burnups of reactor fuel, this total mechanical uncertainty could be significantly reduced, enough that total Region 2 95/95 k_{eff}s would drop by 0.001 to 0.002 Δk or more.

Discharge data from "best-estimate" future core designs were used to quantify the isotopic content of W-RFA fuel, because this design has only recently been introduced into the McGuire reactors, and so the historic W-RFA data are currently very limited. For the MkBI fuel design, estimates of the five irradiation history parameters were taken from the earliest available CASMO-3 / SIMULATE-3 Oconee core follow data that included a fuel design similar to the MkBI. For the reasons stated on page 25 in Attachment 6 of the LAR, the chosen 3-D profiles of the five history parameters MkBI and W-RFA fuel designs are considered conservative. In addition, the conservatisms associated with the BPRA profile grouping and in the assessment of the total mechanical uncertainty (as noted in the previous paragraph for the other fuel designs) apply to the MkBI and W-RFA designs as well.

(Question 11 to be addressed in a future submittal)

Question 12

In Section 8.2, the licensee described the interpolation procedure to be used when cooling times and burnup limits fall between the values provided in TS Tables 3.15.1-4. The licensee stated that it quantified the maximum error associated with its proposed interpolation methodology. Please provide additional information describing how the maximum error was identified and how it was verified that this error is bounding.

Response

In attempting to identify the maximum interpolation error, it was suspected that the largest errors would occur at the midpoints between the specifically-evaluated enrichment and cooling time data points shown in Tables 18 through 21 of Attachment 6 in the LAR.

First, extra sets of standard CASMO-3 fuel depletion cases were performed at 4.75 wt % U-235, directly between the 4.50 wt % and 5.00 wt % U-235 enrichments specifically evaluated for the LAR. This high enrichment was chosen because the reactivity changes (and hence burnup requirement changes) with cooling time are greatest at the maximum (5.00 wt % U-235) enrichment, and thus were deemed most likely to yield the biggest interpolation errors.

Region 2 Unrestricted storage cases were then carried out using W-OFA and W-STD fuel designs, with both 4.75 wt % U-235 and 5.00 wt % U-235 enrichments, and cooling time k_{eff}s computed at 2.5-year intervals (rather than the 5-year intervals used for the cases that produced the Table 18 through 21 burnup requirements).

The results from these cases showed that the maximum 95/95 k_{eff} difference between "midway" interpolation burnup / enrichment data points (2.5, 7.5, 12.5, 17.5-yr cooling, 4.75 wt % enrichment) and their neighboring specifically-evaluated data points (0, 5, 10, 15, 20-yr cooling, 5.00 wt % enrichment), was 0.00036 Δk.

To confirm that this maximum error was bounding for other fuel types, enrichments, and storage configurations, actual polynomial fits to the Attachment 6, Table 18 through 21 minimum burnup requirements were examined. This survey showed that for

the different fuel storage configurations the fourth-order polynomial curves, as functions of cooling time, generally fit the burnup requirement data quite smoothly. However, for some of the Filler and Checkerboard fuel storage burnup limits in Attachment 6, Tables 20 and 21, there was a small degree of non-conservative "wiggle" in the polynomial curve fit between the 15- and 20-year cooling time data points. This wiggle is attributable to the fact that as cooling time accrues for these assemblies, the burnup requirements fall below the threshold between two of the defined Attachment 6, Table 9 burnup profile groups. The different axial profile data sets between these burnup groups cause calculated k_{eff} discontinuities between, for example, a fuel assembly evaluated at 19.99 GWD/MTU and at 20.01 GWD/MTU.

This effect, mentioned briefly as a burnup group transition concern on page 26 in Attachment 6 of the LAR, can be observed in the proposed TS figures corresponding to the minimum burnup requirements. Note, in particular, the MkBWb1 2.50 wt % U-235 limits for Filler storage, and the MkBWb1, MkBWb2, and W-RFA 4.00 wt % U-235 limits for Checkerboard storage. However, the minimum burnup requirements in these cases were set sufficiently high at the specific cooling time data points near the areas of concern to account for any non-conservative polynomial curve wiggle. Therefore, the +0.00036 Δk interpolation bias remains valid and bounding for use in the maximum 95/95 k_{eff} calculations.

Question 13

In its criticality analyses, the licensee assumed the temperature extremes permitted in the design basis of the McGuire spent fuel pools. For example, the licensee performed the criticality analyses at both 150 and 32 degrees Fahrenheit (°F) to ensure that it had bounded the criticality analyses based on the spent fuel pool moderator temperature coefficient (MTC). However, the maximum density of water occurs at 39.2 °F. Therefore, the licensee's analyses performed at 32 °F may not calculate the maximum reactivity in the SFP if the MTC is negative. Please review the criticality analyses to determine for each analysis whether the MTC is negative and add an appropriately conservative temperature bias to account for the difference in the density of the water at 39.2 °F as opposed to 32 °F.

Response

The McGuire SFP criticality calculations showed that for the Region 1 storage racks the maximum k_{eff} occurred at the lower pool water temperature (32 ºF). The Region 2 storage racks, on the other hand, have maximum k_{eff}s at the higher pool temperature of 150 ºF (for stored fuel meeting the minimum burnup requirements shown in Tables 18 through 21 in Attachment 6 of the LAR), due to the fact that no credit is taken for any remaining Boraflex in the Region 2 racks.

The Region 1 criticality computations were performed with SCALE 4.4 / KENO V.a. For 32 ºF conditions, a conservative water density of 1.00 g/cc was specified in the Region 1 KENO V.a input. This bounds the maximum water density at 39.2 ºF.

Question 14

In its proposed TS Figure 3.7.15-2, the licensee determined that it can store fuel assemblies meeting the burnup and cooling time requirements of TS Table 3.7.15-4 in a 3-of-4 checkerboard configuration. The remaining cell must remain empty in this configuration. Please provide additional information on the physical or administrative controls which will be used to ensure these cells remain empty.

Response

Existing Technical Specifications currently include checkerboard configurations for the storage of spent fuel in Region 1 and Region 2. These configurations are as follows:

- 2 out of 4 loading pattern, and
- 1 out of 4 loading pattern

The empty cell locations associated with the proposed checkerboard storage configurations will be handled in a similar manner as they are now for the above checker board configurations. Devices to physically block a cell are currently not used. As currently performed, administrative controls will be used to ensure that the designated cell is empty.

All fuel and component movements are controlled by approved procedures. Procedures require independent verification whenever a fuel assembly or component is being moved within the spent fuel pool. The individual preparing procedures for moving fuel assemblies or components verifies the proper configuration

of each fuel assembly or component being moved, including, that all required empty cell locations affected by the move are empty. The proposed fuel moves are then reviewed by a separate, independent individual, verifying that each fuel assembly affected by the move is in compliance with the fuel storage requirements in Technical Specifications.

Following completion of the development of the procedure for the movement of fuel, the procedure is provided to the fuel handling work crew for execution of the requested moves. Prior to moving a fuel assembly, the assembly is independently verified to be the one identified within the approved procedure. Prior to placing this assembly into a particular location, the location is independently verified to be the location identified in the approved procedure.

Question 15

The licensee has placed considerable emphasis on credit for burnup of the spent fuel for storage in the Region 2 racks. Please provide detailed information describing the methods that will be in place, either administratively or experimentally, to independently confirm the fuel burnup before an assembly is placed in the storage racks.

Response

The existing Technical Specifications requirements currently credits burnup for the safe storage of spent fuel in Region 2 racks. Administrative controls is the method currently used to determine the burnup for a given fuel assembly. The current method of administrative controls will be employed for determining the burnup for a given fuel assembly.

Each month, the burnup and isotopic weights for each fuel assembly is determined based on flux maps taken during cycle operation. These flux maps model the core power distribution that a fuel assembly experiences. Following end of cycle operation, the final burnup of the fuel assembly at that point in time is determined. This information is collected and maintained within a database. Prior to discharge of a fuel assembly from the core to the spent fuel pool, the database is updated to reflect the burnup history of the fuel assembly based on unit operation. As such, the final burnup for a fuel assembly is maintained within the database. This database is referred to when determining the storage qualification in the spent fuel pool for a given fuel assembly.

Question 16

In Attachment 3, the licensee stated that, "[t]he placement of an assembly between the rack and the pool wall would result in a lower k_{eff} relative to the criticality analysis due to the increased neutron leakage at the spent fuel pool wall because the criticality analysis assumes an infinite array of fuel assemblies." Sometimes, this inherent leakage was assumed in the original design of the spent fuel storage racks resulting in no poison inserts and smaller flux traps on the periphery of the racks. Please evaluate the center-to-center spacing that would exist between assemblies during this accident and verify that the spacing is greater than or equal that assumed in the criticality analyses.

Response

Review of the McGuire Region 1 spent fuel pool storage rack drawings shows that Boral panels are not required on the module faces for which the boundary is the spent fuel pool wall. The west direction in Unit 1 and the east direction in Unit 2 have a rack-to-wall spacing of 4-1/8" and 6-15/16", respectively. This distance does not permit placement of a fuel assembly between the fuel storage rack and spent fuel pool wall in these directions. In the south direction of both Unit 1 and Unit 2, the rack-to-wall spacing is 10", and the rack-to-wall spacing in the north direction of both units is 35-3/16". Therefore, it is feasible to inadvertently place a fuel assembly between the Region 1 fuel storage racks and the spent fuel pool wall in both the north and south directions. In this scenario, the fixed neutron poison (Boral) panels would not separate the fuel assemblies, and a smaller center-to-center spacing would exist than that which exists under normal storage conditions of fuel assemblies in the Region 1 fuel storage racks. Although the leakage at the edge of the fuel storage racks has not been credited in the criticality evaluation supporting the LAR, the NRC has requested that Duke Power verify that such credit for leakage and/or soluble boron is sufficient to off-set the reactivity increase due to placement of a single fuel assembly at the maximum enrichment in a location between the fuel storage racks and the spent fuel pool wall.

A criticality evaluation is performed for the placement of a single fuel assembly at the edge of the Region 1 spent fuel pool storage rack, with a minimum center-to-center spacing of 9.09" (Region 1 nominal center-to-center spacing is 10.40"). Separate conditions representing the placement of an assembly to the north and south side of the Region 1 spent fuel pool storage rack are considered. It is also assumed that the array of fuel

storage is infinite in the other three lateral directions. The results show that the calculated 95/95 k_{eff} is 0.9464 with credit for 1000 ppm of soluble boron. This is bounded by the weir gate drop accident which credits 2475 ppm of soluble boron.

Region 2 does not take credit for fixed neutron poison materials, and the rack is an egg-crate design, which would not significantly change the center-to-center spacing for an assembly placed between the Region 2 fuel storage rack and the spent fuel pool wall. Therefore, the placement of a fuel assembly between the Region 2 spent fuel pool storage racks and the pool wall is bounded by the Region 1 results.

Question 17

In Attachment 6, Section 8.2, the licensee stated that when predicted and measured burnup data was compared, the maximum individual assembly error observed was about 4.0 percent. The licensee then stated that "[w]hen an array of fuel assemblies large enough to affect system reactivity is evaluated for the McGuire SFP Region 2, and the distribution of predicted-to-measured burnup differences is accounted for, the maximum system reactivity increase observed is [approximately] 0.00125 delta-k." Please provide the following:

a. An explanation of whether the determination of the maximum system reactivity increase is based on the maximum individual assembly error observed (about 4.0 percent) or on a distribution of observed predicted-to-measured burnup differences.

b. If the maximum individual assembly error observed was used, provide a detailed explanation justifying why this error will remain bounding for future spent assemblies.

c. If a distribution of observed predicted-to-measured burnup differences was used, provide a detailed description of how the distribution used to arrive at the 0.00125 delta-k value was selected and an explanation of why this distribution is bounding for all potential spent fuel loading configurations.

Response

The burnup measurement uncertainty of ±0.00125 Δk is based on an evaluation that considered the distribution of predicted-to-measured burnup differences, rather than the worst-case individual assembly error. This approach was taken because it has been observed that, for moderate reactivity differences

between fuel assemblies - such as the Δk associated with a burnup error of less than 4% - fairly large arrays of perturbed fuel (3x3 or larger) are needed to significantly affect overall SFP system reactivity.

For this analysis, as page 42 in Attachment 6 stated, the measured burnup error for an individual fuel assembly was defined as the difference between the in-core measured burnup and the core follow predicted burnup. Note that it was assumed that this burnup difference is representative - when large numbers of fuel assemblies are considered - of the difference between the measured burnup and the "true" burnup of a fuel assembly (which is not known).

To quantify a bounding burnup measurement uncertainty in this manner, measured burnups (from in-core detector measurements) and predicted burnups (from SIMULATE-3 core follow calculations) were compiled for the discharge fuel assemblies from McGuire's entire reactor operation history to date. This totaled over 1900 fuel assemblies. The percent differences between the measured and predicted burnups were computed, and a sorted distribution of these differences was made. This distribution was then compared with a randomly-generated normal distribution of the same sample size, with the same standard deviation. These distributions are plotted together in Figure 17.1. Note that the distribution of burnup measurement errors approximates a normal distribution rather well. The skewness, which measures the symmetry of the distribution, is very close to the desired zero. The kurtosis of the measurement error distribution is somewhat higher than the normal zero value, which indicates that the distribution is more peaked (narrower) than a true normal distribution.

This distribution of predicted versus measured burnup errors was used to quantify the largest conceivable average burnup error for various clustered arrays of fuel assemblies, which were put together by randomly selecting assemblies (with replacement) from the large McGuire core-follow inventory of discharge fuel. The random-selection process was carried out with storage arrays ranging from a 3x3 to a 12x12 array. Over 1900 random samples were taken for each of these arrays. For each of these samples, the "average" burnup measurement error was determined by taking the average of the individual burnup errors within the pertinent array. The resulting "average" burnup measurement errors were then sorted, and the highest "average" error for each array size was conservatively taken for specific analysis of the maximum conceivable burnup measurement uncertainty. Table 17.1 documents these maximum "average" burnup errors for each of the array sizes selected.

This "average" burnup measurement error was used to model that whole array with fuel at a burnup different from the surrounding fuel by that "average" burnup error. Note that the surrounding fuel - that is, all of the fuel assemblies outside the evaluated burnup error array - would be expected to have a zero "average" burnup error. The largest system k_{eff} increase observed, among all array sizes evaluated, was conservatively chosen as the bounding burnup measurement uncertainty.

Note that this method assumes that fuel is stored in the McGuire SFPs without special consideration of, or regard for, the measured burnup errors of individual fuel assemblies. In other words, it is assumed that the station personnel storing fuel are not intentionally placing clusters of fuel assemblies with all positive (or all negative) measured burnup errors together. Such an assumption is actually fundamental to all the fuel-related uncertainties considered in the SFP criticality evaluation.

Note, also, that the idea of treating this uncertainty in a global sense is the same as the approach used in calculating the burnup computational uncertainty. The burnup computational uncertainty was determined by examining the entire McGuire reactor core, and comparing its predicted k_{eff} to measured k_{eff} at various cycle exposures.

Sensitivity analyses demonstrated that 5.00 wt % W-RFA fuel with a 20-year cooling time produced the largest system reactivity increase per % burnup reduction. This fuel assembly type / enrichment / cooling time combination was thus used to evaluate the "average" burnup errors identified for the arrays in Table 17.1.

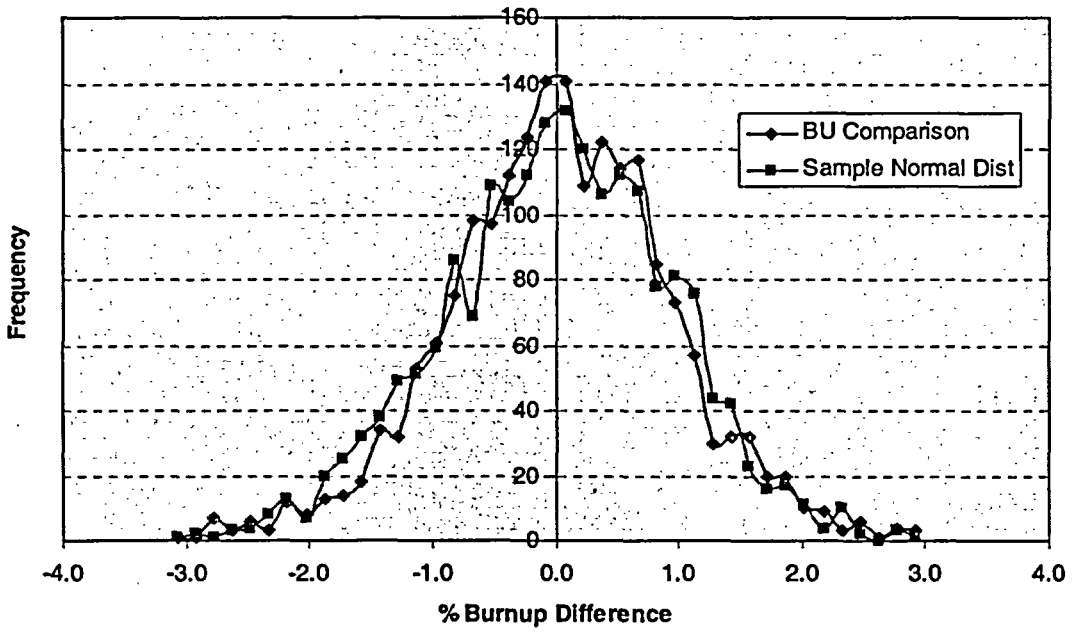
The Table 17.1 results show that the largest k_{eff} increase (0.00125 Δk) occurs with a 12x12 array of fuel, which has a worst-case average burnup error of 0.32%. This maximum burnup measurement uncertainty is judged to be appropriate and bounding for the following reasons:

- The uncertainty is based on the worst observed case of randomly selected real McGuire discharge fuel.
- The 12x12 array that yielded the highest k_{eff} increase is conservative because this is the total problem size; that is, there is no surrounding fuel in the model, so this is essentially an infinite array of fuel with a worst-case burnup error of 0.32%.

- As mentioned above, the fuel type with the most reactivity worth per % burnup reduction was used to quantify the magnitude of the worst-case measurement uncertainty.
- For the W-RFA fuel type, which does not currently have a significant inventory of discharge fuel in the McGuire SFPs, core design projections show higher W-RFA discharge burnups than the average of existing fuel in the McGuire SFPs. The distribution of predicted-to-measured burnup differences used in generating the measurement uncertainty shows a clear trend of smaller errors with higher discharge burnup.
- MkBI fuel is limited in presence in the McGuire SFPs (300 assemblies total), and research of early Oconee core operation reports indicates predicted-to-measured burnup errors in the same range as those observed for McGuire fuel.

Table 17.1. Burnup Measurement Uncertainty - System k_{eff} Increases with Varying Arrays of Worst-Case Higher-Reactivity Burnup Measurement "Errors" {Unrestricted Storage of 5.00 wt % U-235 W-RFA fuel at 150 ºF, 0 ppm boron, 20-year cooling time}

Nominal-Reactivity Burnup (GWD/MTU)	Higher-Reactivity Array Size	Worst-Case Average % Burnup Error in Higher-Reactivity Array	Higher-Reactivity Burnup (GWD/MTU)	SIMULATE system nominal k_{eff}	Δk increase from base case
45.57	0x0	--	--	0.97964	--
45.57	3x3	1.11	45.06	0.97991	0.00027
45.57	4x4	0.79	45.21	0.97998	0.00034
45.57	5x5	0.73	45.24	0.98014	0.00050
45.57	6x6	0.56	45.31	0.98020	0.00056
45.57	8x8	0.51	45.34	0.98051	0.00087
45.57	10x10	0.43	45.37	0.98080	0.00116
--	12x12	0.32	45.42	0.98089	**0.00125**



Statistics Measure	BU Comparison (Measured minus Predicted % burnup difference)	Sample Normal Distribution
Average	0.05	0.01
Maximum	3.99	5.04
Minimum	(3.11)	(3.04)
Standard Deviation	0.939	0.946
Skewness	0.080	(0.040)
Kurtosis	0.855	0.364

Figure 17.1. Comparison of McGuire Discharge Fuel "Measured minus Predicted" Burnup Distribution with Normal Distribution Sample

Question 18

In Attachment 6, Section 8.2, the licensee stated that axial profile k_{eff} errors compare rather well with a normal distribution. Additionally, the licensee identified that the largest individual assembly axial profile error calculated is +0.030 delta-k. However, the licensee determined the bounding axial profile uncertainty by considering a group of fuel assemblies large enough to affect system reactivity and taking into account the distributions of axial profile k_{eff} errors within that group. The licensee determined that the bounding axial profile uncertainty is ±0.00305 delta-k. Please provide detailed information on the methodology that was used to select the distribution used to calculate the axial profile uncertainty and why this distribution is bounding for all potential spent fuel loading configurations.

Response

To quantify the axial profile uncertainty, the same statistical approach employed for the burnup measurement uncertainty - evaluating worst-case "average" reactivity errors for different array sizes - was used. The first step in this process was to gather the actual McGuire core follow axial history profiles, which had been used to generate the average (estimated) axial profiles for each McGuire fuel type and burnup range. These actual profiles were imported into individual SIMULATE-3 cases that computed k_{eff}s for all of these end-of-cycle core follow data points (McGuire 1 Cycles 1 through 15, and McGuire 2 Cycles 1 through 14). These same cases were then re-run using the pertinent average axial profiles from Attachment 6 of the LAR, Tables 10 through 16. After filtering the final discharge fuel data from these cases, the axial profile Δk differences were compared with a sample normal distribution with the same standard deviation. These distributions are shown in Figure 18.1. Note that, as with the measured burnup errors in Figure 17.1, the axial profile Δk errors have a slightly more peaked distribution than the sample normal distribution.

Page 43 in Attachment 6 mentioned that groups of four or eight fuel assemblies are often symmetrically designed for reactor operation, and these fuel assembly groups will have the same axial profile characteristics when those assemblies are ultimately discharged together from the reactor. It is often convenient to keep such groups together or near each other if they meet the pertinent SFP storage criteria. For the statistical treatment of the axial profile errors the judgment was made that it was conservative to consider groups of four symmetric assemblies as a single unit. Clusters of these groups were evaluated together to determine the worst-case "average"

axial profile Δk errors in a particular storage array size.

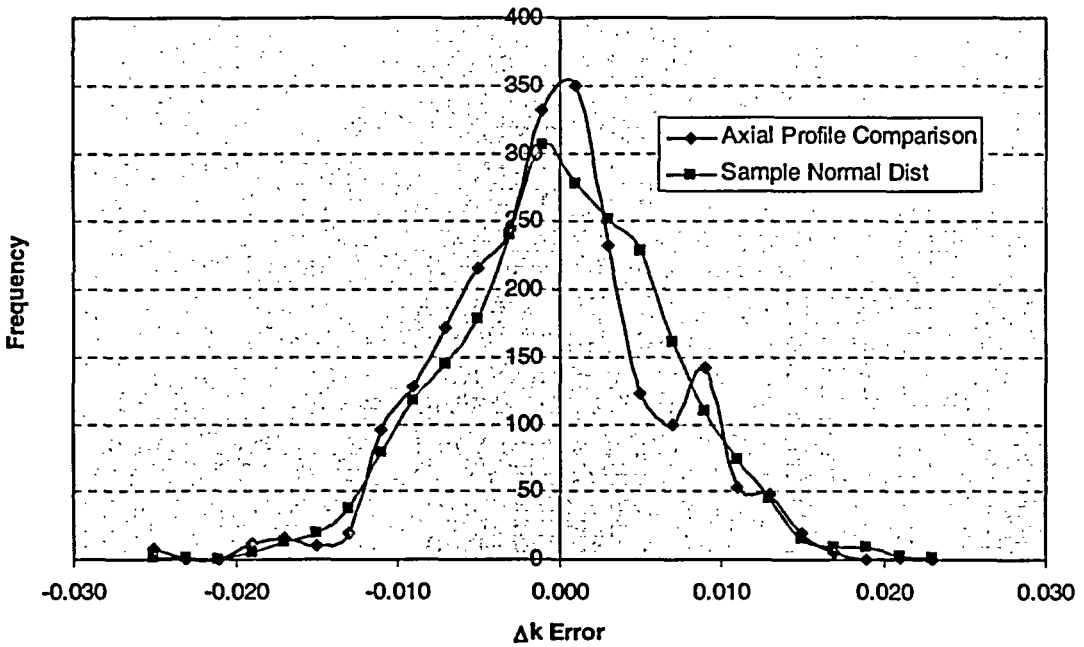
As with the burnup measurement uncertainty, the distribution of individual axial profile Δk errors was used to determine the largest feasible SFP system axial profile Δk uncertainty. Various arrays of symmetric assembly quartets were put together by randomly selecting assemblies (with replacement) from the large McGuire core-follow inventory. The random selection process was carried out with storage arrays ranging from a 2x2(x4) to a 6x6(x4) array. Over 2300 random samples were taken for each of these arrays. For each of these samples, the "average" axial profile Δk error was determined by taking the average of the individual Δk errors within the pertinent array. The resulting "average" Δk errors were then sorted, and the highest "average" error for each array size was conservatively taken for specific analysis of the maximum feasible system axial profile uncertainty. Table 18.1 documents these maximum "average" Δk errors for each of the array sizes selected. Note that for each of these maximum "average" Δk errors, a single assembly from the McGuire core follow database was selected with the matching Δk error for that particular array size.

For each case listed in Table 18.1 the center array within the total 12x12 system matrix used the actual axial history profiles from the core follow data of the chosen fuel assembly for that particular array size. Any surrounding assemblies making up the balance of the 12x12 system matrix used the pertinent "average" axial profiles (from Tables 10 through 16 in Attachment 6) for the chosen fuel assembly. The Table 18.1 results show that the largest system k_{eff} increase occurs with a 6x6(x4) array of fuel, which yields a worst-case average axial profile error of 0.00305 Δk. This maximum axial profile uncertainty is judged to be bounding for the following reasons:

- The uncertainty is based on the worst observed case of randomly selected real McGuire discharge fuel, with the conservative assumption that symmetric quartets of fuel assemblies remain stored together.
- The 12x12 array that yielded the highest k_{eff} increase is conservative because this is the total problem size; that is, there is no surrounding fuel in the model, so this is essentially an infinite array of fuel with a worst-case axial profile error of ±0.00305 Δk.
- For each of the fuel types documented in the LAR, including the W-RFA and MkBI assemblies, a conservative average burnable poison exposure profile was employed for all fuel burnup. The use of these conservative profiles for McGuire fuel is evident in Figure 18.1, which shows a slight negative bias (-0.0006 Δk).

Table 18.1. Axial Profile Uncertainty - System k_{eff} Increases with Varying Arrays of Worst-Case Higher-Reactivity Axial Profile "Errors" {SFP Region 2 at 150 ºF, 0 ppm boron, 0-year cooling time}

Higher-Reactivity Array Size	Worst-Case Average Δk Error in Higher-Reactivity Array	ID # of McGuire Fuel Assembly with same individual Axial Profile Δk Error	SIMULATE system k_{eff} for 12x12 array of Fuel Assemblies using "estimated" Axial Profiles	SIMULATE system k_{eff} for Higher-Reactivity array of Fuel Assemblies using actual core-follow Axial Profiles, within 12x12 array using "estimated" Axial Profiles	Δk increase from 12x12 "estimated" case
2x2 (x4)	+0.0124	NJ07D1	1.04279	1.04467	0.00188
3x3 (x4)	+0.0064	LM09VT	1.01283	1.01463	0.00180
4x4 (x4)	+0.0043	LM0L13	1.02421	1.02628	0.00207
5x5 (x4)	+0.0042	LM09XA	0.98772	0.99075	0.00303
6x6 (x4)	+0.0030	NJ09G5	0.99241	0.99545	**0.00305**



Statistics Measure	Axial Profile Comparison (Δk Error – Actual Profile k_{eff} minus "Average" Profile k_{eff})	Sample Normal Distribution
Average	(0.0006)	0.0001
Maximum	0.0302	0.0253
Minimum	(0.0257)	(0.0246)
Standard Deviation	0.0066	0.0066
Skewness	(0.074)	(0.036)
Kurtosis	0.760	0.082

Figure 18.1. Comparison of McGuire Discharge Fuel Axial Profile Δk (Actual Profile minus "Average" Profile) Error Distribution with Normal Distribution Sample

(Question 19 to be addressed in future submittal)

Question 20

In its analysis of accident conditions in the McGuire SFPs, the licensee discussed abnormal temperatures up to 212 °F. However, the licensee did not provide sufficient information to demonstrate accident conditions such as voiding (boiling) in the SFPs would not cause an increase in reactivity. Please provide additional information demonstrating that either sufficient soluble boron is present in the spent fuel pool to offset any reactivity increase caused by voiding or that voiding in the McGuire SFPs will insert negative reactivity. Also, provide additional information describing the design basis temperatures for the spent fuel pool including any differences in these design temperatures between the regions.

Response

Examination of the calculated k_{eff} for Region 1 conditions (32 °F and 212 °F) shows that the moderator temperature coefficient is negative. A sensitivity analysis of the CASMO model of Region 2 shows that, at 0 ppm boron and 248 °F, the voiding coefficient is negative. Therefore, the voiding condition does not need to be considered.

Question 21

As stated in 10 CFR 50.68(b)(8), licensees are required to update their Final Safety Analysis Report (FSAR) to indicate compliance with 10 CFR 50.68. The licensee's amendment request does not contain a description of the proposed changes to Section 9.1, "Fuel Storage and Handling," of the FSAR. Since the licensee's proposed TSs (4.3.1.1a, 4.3.1.1b, 4.3.1.2b and 4.3.1.2c) reference Section 9.1 of the FSAR as containing a description of the allowance for uncertainties in its analysis, please provide a copy of the revised portions of FSAR, Section 9.1. However, if the licensee's response to Question 7 is thorough in providing the requested information, the licensee may reference that response and state that the information provided therein will be incorporated into the revision for Section 9.1.

Response

The information provided in response to Question 7 will be incorporated into the UFSAR in accordance with the regulatory requirements of 10CFR50.71(e).

Question 22

The licensee's proposed TS Limiting Condition of Operation 3.7.15b states "New or irradiated fuel which has decayed at least 16 days may be stored in Region 2 of the spent fuel pool in accordance with these limits:". For background information that would expedite the NRC staff's review, please describe the basis for the 16-day limit prior to storing irradiated assemblies in the Region 2 racks. Specifically, since a description of the basis for the 16-day limits is not provided in the corresponding TS bases, state whether this limit is developed from its criticality analysis or this limit is based on dose considerations. Additionally, describe how it was confirmed that the 16-day limit is still conservative in light of the proposed changes to the spent fuel storage requirements such as fuel enrichment, burnup, and cooling times.

Response

A tornado missile event could damage fuel assemblies stored in Region 2 of the spent fuel pool. The radiological consequences of this event are mitigated by limiting the age of the fuel discharged to Region 2 of the spent fuel pool to fuel that has decayed at least 16 days. Accordingly, the 16-day limit is based on dose considerations. By Duke Letters dated February 17, 1984 and March 20, 1984, information regarding this event was provided to the NRC in support of the proposed Technical Specification Amendment to allow for expansion of the spent fuel storage capacity from 500 to 1463 spaces. Approval of this Technical Specification Amendment request was provided by NRC Letter dated September 24, 1984. The fuel parameters (burnup, initial enrichment, cooling time, and fuel assembly type) as proposed by this License Amendment Request (LAR) are bounded by the maximum spent fuel assembly fission product inventories assumed for this event.

ATTACHMENT 2

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

d. A nominal 10.4 inch center to center distance between fuel assemblies placed in Regions 1A and 1B; and

e. A nominal 9.125 inch center to center distance between fuel assemblies placed in Regions 2A and 2B.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

f. Neutron absorber (Boral) installed between fuel assemblies in the Region 1 racks

a. Fuel assemblies having a maximum nominal U-235 enrichment of 4.75 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745 ft.-7 in.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1463 fuel assemblies (286 total spaces in Regions 1A and 1B and 1177 total spaces in Regions 2A and 2B).