Duke Energy®

BRUCE H HAMILTON
Vice President
Oconee Nuclear Station

Duke Energy Corporation
ON01VP / 7800 Rochester Highway
Seneca, SC 29672

864 885 3487
864 885 4208 fax
bhhamilton@duke-energy.com

April 26, 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Subject: Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC
Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
License Amendment Request to Reconcile 10 CFR 50 and 10 CFR 72 Criticality Requirements for Loading and Unloading Dry Spent Fuel Storage Canisters in the Spent Fuel Pool – Duke Response to NRC Request For Additional Information
License Amendment Request (LAR) No. 2005-009

Reference: NRC Regulatory Issue Summary 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," dated March 23, 2005.

In accordance with 10 CFR 50.90, Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke) submitted an amendment to Renewed Facility Operating Licenses Nos. DPR-38, DPR-47, and DPR-55 on March 1, 2006. If granted, this amendment request will allow spent fuel loading, unloading, and handling operations in the Oconee Nuclear Site (Oconee) Spent Fuel Pools (SFP) that support spent fuel transfer to an Independent Spent Fuel Storage Installation (ISFSI) licensed under 10 CFR 72.

In a meeting with the Nuclear Regulatory Commission (NRC) on March 22, 2006, Duke provided an overview of the amendment request and discussed NRC Staff's initial concerns. Duke also restated its commitment to respond to those concerns expeditiously in order to facilitate approval of the amendment request by June 1, 2006. On April 3, 2006, a request for additional information (RAI) was discussed in a conference call between the Staff and Duke. Duke received the RAI on April 5, 2006 and this document is in response to the Staff's request. There are no new commitments being made as a result of this document.

A001

Enclosures 3 and 4 contain the RAI responses. Enclosure 4 contains information proprietary to Transnuclear, Inc. and Areva NP. The RAI responses in Enclosure 4 have been reproduced in their entirety for ease of review. Affidavits from Transnuclear, Inc. and Areva NP are included in Enclosure 2. The affidavits set forth the basis on which the information may be withheld from public disclosure by the NRC pursuant to 10 CFR 2.790.

Inquiries on this amendment request should be directed to Reene' Gambrell of the Oconee Regulatory Compliance Group at (864) 885-3364.

Sincerely,

Bruce Hamilton

B. H. Hamilton, Vice President
Oconee Nuclear Site

Enclosures:

1. Notarized Affidavit
2. Affidavits for Transnuclear, Inc. and Areva NP
3. Duke Response to NRC Request for Additional Information – Non Proprietary
4. Duke Response to NRC Request for Additional Information - Proprietary

bc w/enclosures and attachments:

Mr. W. D. Travers, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. L. N. Olshan, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-14 H25
Washington, D. C. 20555

Mr. M. C. Shannon
Senior Resident Inspector
Oconee Nuclear Site

Mr. Henry Porter, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health & Environmental Control
2600 Bull Street
Columbia, SC 29201

bcc w/enclosures and attachments:

B. G. Davenport
R. M. Glover
S. P. Nesbit
G. R. Walden
J. P. Coletta
W. J. Murphy
C. D. Fago
S. J. Perrero
S. C. Newman
R. V. Gambrell
L. F. Vaughn
S. D. Capps
T. P. Gillespie
R. L. Gill – NRI&IA
R. D. Hart – CNS
C. J. Thomas - MNS
NSRB, EC05N
ELL, EC050
File - T.S. Working
ONS Document Management

ENCLOSURE 1

NOTARIZED AFFIDAVIT

AFFIDAVIT

B. H. Hamilton, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Energy Carolinas, LLC that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Bruce Hamilton
B. H. Hamilton, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 26 day of April, 2006

Sheila A Smith
Notary Public

My Commission Expires:

6/12/2013
Date

SEAL

ENCLOSURE 2

AFFIDAVITS FOR TRANSNUCLEAR, INC. AND AREVA NP

AFFIDAVIT

STATE OF MARYLAND)
)
)
)
COUNTY OF HOWARD)

Before me, the undersigned authority, personally appeared Tara J. Neider who, being by me duly sworn according to law, deposes and says that she is authorized to execute this Affidavit on behalf of Transnuclear, Inc. and that the averments of fact set forth in this Affidavit are true and correct to the best of her knowledge, information, and belief:

Tara J. Neider
TARA J. NEIDER

Sworn to and subscribed
before me this 4th day
of April, 2006.

[signature]
Notary Public

My Commission Expires 10 / 14 / 2008

(1) I am President and Chief Operating Officer of Transnuclear, Inc. and my responsibilities include reviewing the proprietary information sought to be withheld from public disclosure in connection with the licensing of spent fuel transport cask systems or spent fuel storage cask systems. I am authorized to apply for its withholding on behalf of Transnuclear, Inc.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the commission's regulations and in conjunction with the Transnuclear application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Transnuclear in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) The following information is furnished pursuant to the provisions of paragraph 10 CFR 2.390(b)(4) to determine whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Transnuclear.

(ii) The information is of a type customarily held in confidence by Transnuclear, is not customarily disclosed to the public and is transmitted to the commission in confidence.

(iii) The information sought to be protected is not now available in public sources to the best of our knowledge and belief and the release of such information might result in a loss of competitive advantage as follows:

(a) It reveals the distinguishing aspects of a storage system where prevention of its use by any of Transnuclear's competitors without license from Transnuclear constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including analytical models, relative to a component or material, the application of which secures a competitive economic or technical advantage.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(5) The information is being transmitted to the commission in confidence and, under the provision of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(6) The information sought to be protected is not available in public sources to the best of our knowledge and belief.

(7) The proprietary information, as shown, sought to be withheld is information contained in Amendment 6 of the TN NUHOMS-24P CoC 72-1004, as referenced in 10CFR72 Section 72.214.

(8) This information should be held in confidence because it provides details of analytical methods that were developed at significant expense. This information has substantial commercial value to Transnuclear in connecting with competition with other vendors for contracts.

The subject information could only be duplicated by competitors if they were to invest time and effort equivalent to that invested by Transnuclear provided they have the requisite talent and experience.

Public disclosure of this information is likely to cause substantial harm to the competitive position of Transnuclear, because it would simplify design and evaluation tasks without requiring a commensurate investment of time and effort.

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing in Regulatory Affairs, for AREVA NP, and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the attributes listed in Attachment A and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure.

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 6th
day of April, 2006.

Brenda C. Maddox
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 7/31/07

ENCLOSURE 3

DUKE RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION – NON PROPRIETARY

Enclosure 3
Duke Response to NRC Request for Additional Information – Non-proprietary

Question 1

Provide a description of the benchmark analysis and results used to determine the SCALE 4.4/KENO V.a bias and uncertainty.

Question 2

Provide a description of the benchmark analysis and results used to determine the CASMO-3/SIMULATE-3 bias and uncertainty.

Response to Questions 1 and 2

Table 1 lists the 58 specific critical experiments (from References 1 to 3) that were analyzed for benchmarking purposes with SCALE 4.4/KENO V.a. The calculated k_{eff} values for the SCALE 4.4/KENO V.a models of these experiments are also provided in Table 1. To determine the SCALE 4.4/KENO V.a method bias and uncertainty to be applied to the NUHOMS®-24P/24PHB DSC analysis, the following equations from Reference 4 are used:

Average k_{eff} $$k_{avg} = \frac{\sum_1^N \frac{k_i}{\sigma_i^2}}{\sum_1^N \frac{1}{\sigma_i^2}}$$

Average Variance $$VAR_{avg}^2 = \frac{\sum_1^N (\sigma_i^2 \times NG_i)}{\sum_1^N NG_i}$$

Method Bias $$Bias_{Method} = (\frac{1}{N}) \times \sum_1^N (K_i - k_i)$$

Method Uncertainty $$Unc_{Method} = f_{95/95} \times \sqrt{\left(\frac{N \times \sum_1^N \frac{(k_i - k_{avg})^2}{\sigma_i^2}}{(N-1) \times \sum_1^N \frac{1}{\sigma_i^2}} - VAR_{avg}^2 \right)}$$

Where:

k_i	=	KENO V.a calculated k_{eff} for critical experiment *i*
σ_i	=	KENO V.a standard deviation for critical experiment *i*
NG_i	=	number of neutron generations used in KENO V.a analysis for critical experiment *i* *(400 for all experiments modeled in Table 1)*
N	=	number of KENO V.a critical experiments *(58)*
K_i	=	measured value of k_{eff} for critical experiment *i* *(1.000 for each of 58 experiments)*
$f_{95/95}$	=	95/95 one-sided tolerance factor *(2.03 for 58 experiments per Reference 5)*

Table 2 lists the 10 benchmark critical experiments from Reference 8 that were evaluated with CASMO-3/SIMULATE-3. The CASMO-3/SIMULATE-3 calculated k_{eff} values and experimentally measured k_{eff} values are included in Table 2. Because CASMO-3/SIMULATE-3 calculations yield deterministic solutions, the method bias and uncertainty calculations simplify to the following:

Method Bias
$$Bias_{Method} = (\frac{1}{N}) \times \sum_{1}^{N} (K_i - k_i)$$

Method Uncertainty
$$Unc_{Method} = f_{95/95} \times \sqrt{\frac{\sum_{1}^{N} (K_i - k_i - Bias_{Method})^2}{(N-1)}}$$

Where:

k_i	=	CASMO-3/SIMULATE-3 calculated k_{eff} for critical experiment *i*
N	=	number of CASMO-3/SIMULATE-3 critical experiments *(10)*
K_i	=	measured value of k_{eff} for critical experiment *i* *(see Table 2)*
$f_{95/95}$	=	95/95 one-sided tolerance factor *(2.911 for 10 experiments per Reference 5)*

Note that the Reference 7 submittal for the Oconee spent fuel pool storage racks employed the same critical experiments for its criticality code benchmarking. Because the SCALE and SIMULATE versions used in Reference 7 have since been updated, the method biases and uncertainties resulting from analysis of these critical experiments are slightly different. Note also that the Reference 10 submittal employed many of the critical experiments in Tables 1 and 2 for its code benchmarking.

The fuel design parameters, storage cell spacing, and SFP conditions associated with loading fuel assemblies into the NUHOMS®-24P/24PHB DSCs are quite similar to those associated with storage of assemblies in the Oconee SFP racks. The applicability of the set of critical experiments used in Reference 7 to the conditions in the Oconee SFPs thus extends to the loading of the NUHOMS®-24P/24PHB DSCs.

Table 3 lists a set of important criticality parameters and the range of values for these parameters in the evaluated SCALE 4.4/KENO V.a and CASMO-3/SIMULATE-3 critical experiments. Included for comparison are the values of these parameters for the NUHOMS®-24P/24PHB DSCs (simplified infinite-array model described in Reference 6). This table shows that the selected benchmark critical experiments are appropriate for application to the NUHOMS®-24P/24PHB DSC model.

Table 1. Calculated k_{eff} Values for the SCALE 4.4/KENO V.a Benchmark Critical Experiments

Report	Experiment Number	Calculated k_{eff}	σ (k_{eff})	Report	Experiment Number	Calculated k_{eff}	σ (k_{eff})
PNL-3314	043	0.99799	0.00198	PNL-3314	085	0.99430	0.00203
PNL-3314	045	0.99613	0.00181	PNL-3314	094	0.99733	0.00199
PNL-3314	046	0.99185	0.00161	PNL-3314	095	0.99723	0.00198
PNL-3314	047	0.99937	0.00204	PNL-3314	096	0.99669	0.00198
PNL-3314	048	0.99728	0.00193	PNL-3314	097	0.99767	0.00194
PNL-3314	04c	0.99604	0.00178	PNL-3314	098	0.99657	0.00204
PNL-3314	051	0.98920	0.00200	PNL-3314	100	0.99292	0.00198
PNL-3314	053	0.98302	0.00225	PNL-3314	101	0.99493	0.00213
PNL-3314	055	0.99403	0.00186	PNL-3314	105	0.99548	0.00195
PNL-3314	056	0.99130	0.00218	PNL-3314	106	0.99325	0.00206
PNL-3314	057	0.98979	0.00201	PNL-3314	107	0.99696	0.00214
PNL-3314	058	0.99355	0.00188	PNL-3314	131	0.99050	0.00170
PNL-3314	059	0.99184	0.00185	PNL-3314	996	0.98675	0.00173
PNL-3314	060	0.99099	0.00179	PNL-3314	997	0.98970	0.00187
PNL-3314	061	0.99213	0.00202	PNL-2438	005	0.99298	0.00151
PNL-3314	062	0.99537	0.00206	PNL-2438	014	0.99163	0.00174
PNL-3314	064	0.99351	0.00226	PNL-2438	015	0.99359	0.00174
PNL-3314	065	0.99185	0.00195	PNL-2438	021	0.99123	0.00182
PNL-3314	066	0.99018	0.00225	PNL-2438	026	0.99216	0.00164
PNL-3314	067	0.98951	0.00207	PNL-2438	027	0.98934	0.00155
PNL-3314	068	0.99025	0.00199	PNL-2438	028	0.99260	0.00148
PNL-3314	069	0.99716	0.00193	PNL-2438	029	0.99524	0.00175
PNL-3314	06d	1.00418	0.00161	PNL-2438	034	0.99118	0.00194
PNL-3314	070	0.98758	0.00184	PNL-2438	035	0.98978	0.00173
PNL-3314	071	0.99521	0.00181	PNL-6205	214	0.99190	0.00241
PNL-3314	072	0.99304	0.00181	PNL-6205	223	1.00122	0.00192
PNL-3314	073	0.98938	0.00176	PNL-6205	224	0.99256	0.00219
PNL-3314	083	0.99749	0.00178	PNL-6205	229	0.99829	0.00170
PNL-3314	084	0.99680	0.00269	PNL-6205	230	0.99744	0.00193

Average Calculated k_{eff} = 0.9936
SCALE 4.4/KENO V.a Method Bias = **+0.0064 Δk** (average)
SCALE 4.4/KENO V.a Method Uncertainty = **±0.0066 Δk**

Table 2. Calculated k_{eff} Values for the CASMO-3/SIMULATE-3 Benchmark Critical Experiments

BAW-1484-7 Critical Experiment Core Number	Experimentally Measured k_{eff}	SIMULATE-3 Calculated k_{eff}	Δk (Measured k_{eff} minus Calculated k_{eff})
2	1.0001	1.00274	(0.00264)
3B	1.0000	1.00320	(0.00320)
9	1.0030	0.99905	0.00395
10	1.0001	0.99793	0.00217
11	1.0000	1.00497	(0.00497)
13B	1.0000	1.00926	(0.00926)
14	1.0001	1.00461	(0.00451)
15	0.9988	0.99611	0.00269
17	1.0000	0.99891	0.00109
19	1.0002	1.00003	0.00017
		Average →	(0.00145)
		Deviation →	0.00416

CASMO-3/SIMULATE-3 Method Bias = **–0.0015 Δk** (average)
CASMO-3/SIMULATE-3 Method Uncertainty = 2.911*0.00416 = **±0.0121 Δk**

Table 3. Important NUHOMS®-24P/24PHB DSC Criticality Analysis Parameters and their Values for Selected Benchmark Critical Experiments

Parameter	Range of Values in Reference 6 simplified infinite-array DSC model	Range of Values in Table 1 SCALE 4.4/KENO V.a Critical Experiments	Range of Values in Table 2 CASMO-3/ SIMULATE-3 Critical Experiments
Lattice water-to-fuel volume ratio	1.62 – 1.66	1.60 (48) and 2.92 (10)	1.84
U-235 Enrichment (wt % U-235)	1.60 - 5.00	2.35 - 4.31	2.46
Separation between Rod Arrays (cm)	4.47	0 – 19.81	0 - 6.54
Soluble Boron Concentration (ppm)	0 - 630	0	0 - 1037

Question 3

This LAR lists two mechanical uncertainties, one with unborated water, the other with water borated to 430 ppm of boron. Each mechanical uncertainty encompasses all manufacturing tolerances and uncertainties for the fuel and the cask.

a. Provide a list of the mechanical uncertainties associated with the fuel assemblies used to determine the two uncertainties listed in the LAR. Explain why these uncertainties are appropriate and how they bound all of the fuel designs listed in Table 2 of the LAR.
b. Provide a list of the mechanical uncertainties associated with the DSC used to determine the two uncertainties listed in the LAR. Explain why these uncertainties are appropriate.
c. Explain the difference between the two mechanical uncertainties, one with unborated water, the other with water borated to 430 ppm of boron, used in the LAR.
d. Explain the method used to combine all of the above uncertainties into the two mechanical uncertainties listed in the LAR.

Response to Question 3a

Table 4 lists the parameters associated with fuel assemblies whose tolerances were observed to acquire mechanical uncertainty factors at both 0 and 430 ppm soluble boron. In addition, their individual contributions to the final results are provided.

With the exception of the guide tube parameters, each of these fuel assembly parameters were considered in the Reference 7 analysis of the Oconee spent fuel pools. Furthermore, the parameters considered here are consistent with those outlined in Reference 9.

In determining the mechanical uncertainty factors at 0 and 430 ppm soluble boron, the reactivity effects of the parameters in Table 4 were observed for each of the fuel designs listed in Table 2 of Enclosure 3 of Reference 6. The maximum resulting total mechanical uncertainty at 0 and 430 ppm soluble boron was chosen to conservatively bound all three of the fuel designs.

Table 4. Mechanical Uncertainty Factors at 0 and 430 ppm Soluble Boron, Fuel Assembly-Related Parameters

Parameter	Tolerance	Reactivity Effect	
		0 ppm boron	**430 ppm boron**
Fuel Enrichment	[ ]	0.00194Δk	0.0022Δk
Fuel Pellet Dish Volume	Varies by type.	0.00048Δk	0.00052Δk
Fuel Theoretical Density	[ ]	0.0009Δk	0.00173Δk
Fuel Pellet Outer Diameter	[ ] [ ]	0.00023Δk	0.00047Δk
Fuel Clad Outer Diameter	[ ]	0.00177Δk	0.00108Δk
Guide Tube Inner Diameter	[ ]	0.00017Δk	0.00012Δk
Guide Tube Outer Diameter	[ ]	0.00018Δk	0.00013Δk
Fuel Eccentricity (location in cell)	± 0.192" (x and y coord.)	0.0077Δk	0.00571Δk

<u>**Response to Question 3b**</u>

Table 5 lists the parameters associated with the DSC whose tolerances were observed to acquire mechanical uncertainty factors at both 0 and 430 ppm soluble boron. In addition, their individual contributions to the final results are provided.

These parameters in Table 5 were observed in the Reference 7 analysis, but for *spent fuel pool* storage cells (as opposed to DSC storage cells). The methodology is unchanged and remains valid as DSC loading conditions are similar to storage conditions in the Oconee spent fuel storage racks. In addition, the parameters in Table 5 are the only structural characteristics of the DSC that are considered in the homogeneous DSC model (Section 6.3 of Enclosure 3 of Reference 6).

Table 5. Mechanical Uncertainty Factors at 0 and 430 ppm Soluble Boron, DSC-Related Parameters

Parameter	Tolerance	Reactivity Effect	
		0 ppm boron	**430 ppm boron**
[DSC] Cell Inside Dimension	[ ]	0.01004Δk	0.0095Δk
[DSC] Cell Wall Thickness	[ ]	0.00298Δk	0.00153Δk
[DSC] Cell Center-to-Center Spacing	[ ]	0.02457Δk	0.02806Δk

Response to Question 3c

The two mechanical uncertainties used in the Reference 6 analysis, 0.0280Δk and 0.0304Δk, were calculated at 0 and 430 ppm soluble boron, respectively. The same mechanical parameters in Tables 4 and 5 were observed in both cases; however, the presence (or lack) of soluble boron changes the reactivity impact of a given mechanical tolerance. Thus, the statistically-combined overall mechanical uncertainty factor will vary with soluble boron concentration based on the variations of its individual contributors.

Of the mechanical parameters listed in Tables 4 and 5, the DSC cell center-to-center spacing provided the largest reactivity effects at both 0 and 430 ppm soluble boron. This parameter in particular was analyzed in a very conservative fashion, with the pitch of the infinite homogeneous array model being increased/decreased by the [ ] tolerance from Table 5. The total statistically-combined mechanical uncertainty factors at 0 and 430 ppm, respectively, from Enclosure 3 of Reference 6 are 0.0280Δk and 0.0304Δk. This disparity in mechanical uncertainty values stems primarily from the increased reactivity worth of the borated water, which is displaced by the reduction in the DSC cell center-to-center spacing. Consulting the reactivity effect values in Tables 4 and 5 confirms that, indeed, the center-to-center spacing parameter is impacted the most by the addition of 430 ppm soluble boron.

Response to Question 3d

Reference 9 states that the reactivity effects of tolerance variations may be combined statistically if they are independent. The following equations were used to statistically combine the independent tolerance variations into an overall mechanical uncertainty, which was then applied to the calculated nominal multiplication factor:

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

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

for KENO Va.

or,

$$\Delta k_{MechUnc} = \sqrt{\sum_i (\Delta k_i)^2}$$

$$\Delta k_i = k_i - k_{nom}$$

for CASMO-3

where,

Δk_i = uncertainty due to tolerance (i),
k_{nom} = CASMO-3 or KENO V.a Calculated k_{eff} for nominal mechanical parameters,
σ_{nom} = KENO V.a one-sigma uncertainty for nominal mechanical parameters,
k_i = CASMO-3 or KENO V.a Calculated k_{eff} for mechanical tolerance (i),
σ_i = KENO V.a one-sigma uncertainty for mechanical tolerance (i),
f_{95} = 95^{th} percentile one-sided tolerance factor (1.727 for 1000 generations)

Both the upper and lower tolerances of each mechanical parameter were evaluated to determine which had a greater effect on the multiplication factor. Only the direction of the tolerance (upper or lower) which produced the largest positive difference of $\{k_i - k_{nom}\}$ was used to determine the uncertainty in k_{eff} due to that particular parameter.

Question 4

Please explain the effect of placing burnable poison rod assemblies (BPRA) into fuel assemblies loaded into the DSC while in the SFP. Include any limitations on quantity or location of BPRAs.

Response to Question 4

In order to determine whether the presence of BPRA components produces a net increase in reactivity with the displacement of borated water, a CASMO-3 job was executed to analyze the infinite homogeneous DSC model with fresh (unirradiated) 5.00 wt% U-235 mbl fuel at 630 ppm soluble boron. Fully depleted (i.e. 100% Al_2O_3) BPRA components were conservatively placed in each assembly of the infinite model. The maximum 95/95 k_{eff} from this case was then compared with the same model executed with no BPRA components present. As this comparison was performed with unirradiated, maximum enrichment (5.00 wt% U-235) fuel assemblies with fully depleted BPRA components infinitely modeled and with the maximum soluble boron concentration credited in Reference 6, this analysis fully bounds all DSC loading conditions.

The maximum 95/95 k_{eff} for unirradiated 5.00 wt% U-235 mbl fuel in the infinite homogeneous DSC model with 630 ppm soluble boron and depleted BPRA components is 1.15765, while the maximum 95/95 k_{eff} for the same model with no BPRA components modeled is 1.16650. The presence of depleted BPRAs led to a net decrease in reactivity of **0.00885Δk**, indicating that, at the maximum (630 ppm) soluble boron concentration credited in Reference 6, it remains conservative and bounding to assume that no BPRA components are present.

Question 5

The LAR briefly describes the accident analysis associated with the misloading of a single unirradiated 5.0 w/o U-235 fuel assembly in the DSC. The LAR states this is the most limiting accident. With respect to the accident analysis provide the following:

a. A description of the controls that limit misloading event to one assembly.
b. A description of the analysis and results that led to the conclusion that the remaining accidents are bounded by the misloading accident.

Response to Question 5a

Page 9 of Attachment 3 in the Reference 7 submittal states the following, with regard to misloading fuel assemblies in the Oconee SFP storage racks:

> *"Reference 4 [Kopp letter] requires that only a single fuel assembly misload be analyzed unless there are circumstances that make multiple loading errors credible. Redundant checks and procedural verifications of each fuel assembly movement within the Oconee spent fuel pools preclude the occurrence of multiple fuel assembly loading errors in any storage region."*

The same procedural verifications and redundant checks that are used with spent fuel movements are also in effect during NUHOMS®-24P/24PHB DSC loading operations. These include:

1. Fuel movement instructions prepared and independently reviewed by qualified engineers in accordance with approved technical procedures.
2. Bridge positioning for fuel movement performed and independently verified by qualified fuel handlers in accordance with approved technical procedures and the fuel movement instructions. Each fuel move is independent of any other fuel move.

The General Office Spent Fuel Management group generates the list of fuel assemblies to be loaded into the DSC. This list is created and independently verified through the use of an approved procedure that follows the fuel selection process as set forth in the ISFSI Technical Specifications. This list is formally transmitted to the Reactor Engineering group at ONS.

The Reactor Engineering group uses this list to generate the fuel move sheets that are provided to the fuel handling group. The fuel move sheets are created and independently verified using an approved procedure.

NUHOMS Certificate of Compliance (72-1004) Technical Specification 1.2.1 states..... "Immediately, before insertion of a spent fuel assembly into a DSC, the identity of each fuel assembly shall be independently verified and documented." The controlling procedure for loading fuel into the DSC requires this as a blind verification performed independently by two individuals. This verification is made using an underwater camera. Additionally all fuel moves are made using a blind verification technique. The fuel bridge operator has a copy of the fuel move sheet showing the step number, fuel assembly ID, withdraw location, and insert location. The fuel bridge operator is directed to perform a step number by the procedure controller. When the fuel bridge is over the location called for by the move sheet, a fuel bridge spotter verifies the fuel bridge is in the proper location. During this process the step number is the only information that is verbally communicated. The fuel bridge spotter does not have access to the fuel move sheets, and does not know where the fuel bridge is supposed to be. The procedure controller verifies that the spotter has called out the proper location, and then directs the fuel bridge operator to withdraw, or insert the fuel assembly.

These barriers, in and of themselves, are sufficient to preclude a single fuel assembly misload and render multiple loading errors non-credible.

<u>Response to Question 5b</u>

From Section 6.5 of Enclosure 3 of Reference 6, a misloaded MkB11 fuel assembly requires 630 ppm soluble boron credit in order to maintain DSC system k_{eff} under 0.95. The following description of the remaining credible accident scenarios demonstrates that the misloading accident requires the greatest quantity of soluble boron to remain under regulatory limits.

- **Seismic Events**

Per Reference 9, the analysis must "consider the effect on criticality of natural events (e.g. earthquakes) that may deform, and change the relative position of, the storage racks and fuel in the spent fuel pool." The mechanical uncertainty calculation performed in support of the Reference 6 analysis considered the placement of fuel assemblies into the most optimum possible pitch (consistent with the Reference 10 analysis); that is, each assembly is positioned as close to one another as possible within their respective storage cells). The maximum reactivity impact of such a transient (from Table 4) is **0.0077Δk** for 0 ppm soluble boron credit. From Section 6.5 of Enclosure 3 of Reference 6, the maximum 95/95 keff for the DSC system with 430 ppm soluble boron credit is 0.9264. With the aforementioned reactivity increase, the maximum 95/95 keff for the DSC system during a seismic transient is 0.9341 with 430 ppm soluble boron credit, resulting in a **0.0159Δk** margin from the regulatory maximum of 0.95. Thus, this accident scenario is bounded by the misload presented in Enclosure 3 of Reference 6.

- **Abnormal Water Temperatures**

Reference 9 states that "abnormal temperatures (above those normally expected) and the reactivity consequences of void formation (boiling) should be evaluated." The criticality analysis determined that a water temperature of 150°F is more reactive than the lower nominal temperature limit, 68°F. Thus, a water temperature of 150°F is assumed when calculating the minimum burnup requirements for DSC loading. In order to fully analyze all criticality consequences for abnormal temperature conditions, temperatures both above and below those considered by the nominal analysis were evaluated as well as voiding effects at higher temperatures. The maximum 95/95 k_{eff} of 0.93365 occurs at 212°F with 0 percent water voiding with credit taken for 430 ppm soluble boron (i.e. the partial boron credit assumed in the nominal analysis), resulting in a **0.01635Δk** margin from the regulatory maximum of 0.95. Clearly, this accident scenario is bounded by the misload presented in Enclosure 3 of Reference 6.

- **Fuel Assembly Drop**

In considering the consequences of a drop of a fuel assembly, several types of drop accidents are postulated. A drop resulting in an assembly residing immediately adjacent to the NUHOMS transfer cask would be essentially neutronically decoupled from the cask, as the thickness of the shielded transfer cask would provide for at least 17 inches of spacing between the dropped assembly and the loaded assemblies.

A drop resulting in an assembly falling on top of the DSC could result in one of several outcomes. The most likely outcome is the fuel assembly coming to rest on top of the DSC upper spacer disk. In such a scenario, a sufficient amount of spacing is present between the active fuel regions of the dropped assembly and those of the loaded assemblies to preclude any neutronic interaction between the dropped and stored fuel. Were a dropped assembly to land on an already-stored assembly, the impact would, at worst, slightly compress the stored assembly; however, the distance between the dropped and loaded assemblies would still remain sufficient to preclude any interaction, and the compression of the stored assembly would, at worst, lend itself to a *slight* reactivity change with the change in the water-to-fuel ratio. The worst-case misload accident – a fresh, maximum enrichment fuel assembly loaded into a DSC storage cell – clearly bounds any fuel drop/misplacement scenario, as a fuel assembly dropped anywhere on/outside the shielded canister would be essentially isolated from the active fuel in the DSC.

- ***Spent Fuel Pool (SFP) Dilution Accident***

A dilution accident concurrent with loading a DSC in the SFP, while highly unlikely, is a credible transient; however, the presence of a DSC or related activities does not create any additional initiating mechanisms for such an event. Thus, the dilution analysis currently supporting the Oconee SFPs (submitted to NRC in Reference 7) also supports cask loading operations in the SFPs. The dilution analysis concluded that, for both spent fuel pools, at least 32.7 hours must pass before the spent fuel pool boron concentration is reduced to the credited quantity of 430 ppm. This amount of time is more than sufficient for site personnel to initiate action to mitigate the situation. Thus, this accident poses no risk to criticality safety.

References

1. **Criticality Experiments with Subcritical Clusters of 2.35 and 4.31 wt % U-235 Enriched UO2 Rods in Water at a Water to Fuel Volume Ratio of 1.6**, PNL-3314, July 1980.

2. **Critical Separation Between Subcritical Clusters of 2.35 wt % U-235 Enriched UO2 Rods in Water with Fixed Neutron Poisons**, PNL-2438, October 1977.

3. **Criticality Experiments to Provide Benchmark Data on Neutron Flux Traps**, PNL-6205, June 1988.

4. **"Validation of YAEC Criticality Safety Methodology,"** D. Napolitano and F. Carpenito, ANS 1988 Annual Meeting Transactions, Vol 56, pp 325-327.

5. **Factors for One-Sided Tolerance Limits and for Variables Sampling Plans**, SCR-607, Sandia Corporation, March 1963.

6. **"Oconee Nuclear Site, Units 1, 2 and 3; Docket Numbers 50-269, 50-270 and 50-287; License Amendment Request to Reconcile 10 CFR 50 and 10 CFR 72 Criticality Requirements for Loading and Unloading Dry Spent Fuel Storage Canisters in the Spent Fuel Pool"** – Transmittal from B. Hamilton (Duke Power) to U.S. NRC, March 1, 2006.

7. **"Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2 and 3; Docket Numbers 50-269, 50-270 and 50-287; Response to Request for Additional Information – Proposed Technical Specification Amendment; Generic Letter 96-04 -- Spent Fuel Storage Racks (TSCR 2000-01)"** – Transmittal from W. McCollum (Duke Power) to U.S. NRC, October 31, 2001. (Technical Specification amendment request approved by NRC via SER dated April 22, 2002)

8. **Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel**, BAW-1484-7, July 1979.

9. **"Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,"** Memorandum from L. Kopp (NRC) to T. Collins (NRC), U.S. Nuclear Regulatory Commission, August 19, 1998.

10. **"Joseph M. Farley Nuclear Plant, Technical Specification Revision, Spent Fuel Cask Loading Requirements"** – Transmitted from L. Stinson (Southern Company) to U.S. NRC, May 17, 2005. (Technical Specification amendment approved by NRC via SER dated June 28, 2005)