Fort Calhoun Station
P.O. Box 550, Highway 75
Fort Calhoun, NE 68023-0550

March 27, 2006
LIC-06-0023

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

References: 1. Docket Nos. 50-285 and 72-054
2. USNRC Regulatory Issue Summary 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations," March, 23, 2005 (ML043500532) (NRC-05-0038)
3. Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk), "Fort Calhoun Station Unit No. 1 License Amendment Request (LAR) 05-013, "Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool," November 8, 2005 (ML053120421) (LIC-05-0119)
4. Teleconference between OPPD and NRC, Request for Additional Information for Fort Calhoun Station Unit No. 1 License Amendment Request, "Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool," February 22, 2006

SUBJECT: **Response to Requests for Additional Information and Revision of Fort Calhoun Station Unit No. 1 License Amendment Request, "Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool"**

In accordance with Reference 2, the Omaha Public Power District (OPPD) previously submitted the Reference 3 License Amendment Request (LAR) to address criticality control during spent fuel cask loading in the spent fuel pool. Attachment 1 of this letter provides OPPD's response to the Request for Additional Information questions discussed during the Reference 4 teleconference regarding the LAR. Attachment 2 of this letter provides clarifications to editorial items identified by the NRC Staff during their review of Reference 3.

No commitments to the NRC are made in this letter. I declare under penalty of perjury that the foregoing is true and correct. (Executed March 27, 2006)
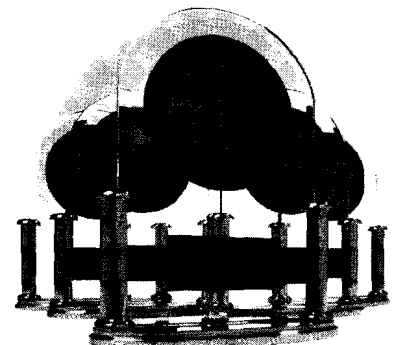
If you have any questions or require additional information, please contact Mr. Thomas C. Matthews at 402-533-6938.

Sincerely,

D. J. Bannister
Plant Manager
Fort Calhoun Station

DJB/rlj

Attachments: 1) Response to Request for Additional Information
2) Clarifications to Omaha Public Power District's Letter LIC-05-0119

Enclosures: 1) NUHOMS®-32PT 50.68 Criticality Analysis for Fort Calhoun
(Revision Number 1)
2) Revised Pages 2 and 5 of Attachment 1 of LIC-05-0119 and Revised 4.0-Page
2 of Clean-Typed Technical Specification

cc: Director of Consumer Health Services, Department of Regulation and Licensure, Nebraska
Health and Human Services, State of Nebraska

Attachment 1
Omaha Public Power District (OPPD) Response to NRC's Request for Additional
Information (RAI) on Fort Calhoun Station, Unit No. 1
License Amendment Request (LAR)
"Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool"

NRC Question 1:

The DSC position in the spent fuel cavity is described as being 3 inches from the spent fuel racks.
Later in the application, the location is described as being 3 centimeters. State the correct
spacing between the fuel racks and the cask and whether the model was conservative in
accounting for neutron coupling in the criticality analysis.

OPPD Response to Question 1:

In the first paragraph of the License Amendment Request, Reference 3, section 4.1.3.6, Dry
Shielded Canister, DSC, Position in the Spent Fuel Pool, SFP, Cavity, (page 12 of Attachment 1
to Reference 2) the unit of "inches" was incorrectly used. The fourth sentence should read, "In
this orientation, the closest the transfer cask can approach the spent fuel racks is approximately
three centimeters." The dimension of three centimeters is consistent with the dimension
provided in section 6.5, DSC Position in the SFP Cavity, on page 23 of 36 of Enclosure 1 of
Reference 2 and is the more conservative distance, i.e., the shortest distance, for neutron
coupling.

NRC Question 2:

TS 4.3.1.2 Design Features states fuel assemblies having a maximum U-235 enrichment of 5.0
weight percent may be stored in the new fuel storage racks. The 70.24 exemption states Ft.
Calhoun is licensed to a limit of 4.5 weight percent new fuel. Provide the LAR acceptance SE
that allows Ft. Calhoun to store up to 5.0 weight percent in the new fuel storage racks.
Additionally, Sections 6.6 and 6.7 of Enclosure 1 of the LAR state a fresh fuel assembly of 4.55
weight percent enrichment is the most reactive fuel and this was the maximum enrichment of fuel
analyzed for the FCS spent fuel cask loading criticality analysis. Provide the technical
justification for not analyzing the most conservative accident condition where a fresh fuel
assembly up to 5 weight percent is misloaded into the cask and $k_{eff} < 1$.

OPPD Response to Question 2:

Even though the new fuel storage racks were originally designed to store fuel with enrichment up
to 5.0 weight percent, Technical Specification 4.3.1.1 restricts the spent fuel storage racks to fuel
with no more than 4.5 weight percent enrichment. Based on this restriction and expected core

a maximum 4.5 weight percent enrichment. This procurement process is under the Fort Calhoun Station Quality Assurance Program, and ensures that no new fuel of greater than 4.5 weight percent enrichment could be received and accidentally loaded into the spent fuel cask. If future core designs include fuel with greater than 4.5 weight percent enrichment, OPPD will need to obtain NRC approval through appropriate licensing actions.

NRC Question 3:

Provide a table with the numerical values and parameters accounted for in the axial bias uncertainty applied to fuel burn-up greater than 30,000 MWD/MTU. How was +0.013 Δk calculated and what burnup profile was used? (Farley used DOE axial burnup shape) Was this consistent with the burnup credit in the SFP criticality analysis previously performed?

OPPD Response to Question 3:

The +0.013 Δk is consistent with the burnup credit in the SFP criticality analysis previously performed. The values and parameters of the Spent Fuel Racks Criticality Safety Analyses are presented in Tables 1 and 2 below. These tables were submitted as Attachment C of Reference C below in support of OPPD's application, made pursuant to 10 CFR 50.90, to increase the enrichment of the SFP racks approved by the NRC in Reference B. This request was subsequently approved by Reference D. The NRC staff's Safety Evaluation, given in Reference D, concluded that "Based on the review described above, the staff finds the criticality aspects of the proposed increased allowed enrichment limit of fuel stored in the Fort Calhoun spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling."

The +0.013 Δk was taken directly from these OPPD licensing basis documents and is consistent with the burnup credit used in the SFP criticality analysis previously performed for the replacement SFP fuel storage racks. This axial bias is from HOLTEC reports HI-92828 (FC06025) and HI-951400 submitted in References A and C below and is therefore consistent with the burnup credit in the SFP criticality analysis previously performed under 10 CFR 50. As a result, the values and parameters consistent with the axial bias uncertainty pertaining to the 10 CFR Part 72 methodology, as requested by this question, were not re-calculated for that analysis and are not available.

The following Table 1 and Table 2 were submitted as Attachment C of Reference C in support of OPPD's application to increase the enrichment of the SFP racks.

Table 1

Summary of Criticality Safety Analyses

	Region 1	Region 2
Design Basis	4.75% enrichment	4.75 % enrichment at 38.900 MWD/MTU
Temperature for analysis	4°C	4°C
Reference k_{∞} (CASMO-3)	0.9392	0.9016
Uncertainties		
In Bias	± 0.0024	± 0.0024
B-10 loading	± 0.0031	± 0.0032
Boral width	± 0.0008	± 0.0006
Inner box dimension	± 0.0009	± 0.0011
Water gap thickness	± 0.0093	NA
SS thickness	± 0.0004	± 0.0002
Fuel enrichment[1]	± 0.0018	± 0.0018
Fuel density[1]	± 0.0022	± 0.0022
Eccentric position	Negative	Negative
Statistical combination of uncertainties[2]	± 0.0106	± 0.0051
Burnup Uncertainty	NA	± 0.0153
Axial Burnup Distribution	NA	+ 0.0130
Total	0.9392 ± 0.0106	0.9299 ± 0.0051
Maximum Reactivity (k_{∞})	0.9498	0.935

[1] For fuel tolerances, uncertainties in Region 2 assumed to be the same as those for Region 1.
[2] Square root of sum of squares.

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Table 2
Evaluation of the Minimum Burnup Requirements in Region 2

Initial Enrichment	Calculated k_{eff}	Depletion Uncert. Δk	Axial Burnup Dist. Δk	Limiting Burnup
2.0%	0.9232	0.0025	0.0	5,270[1]
2.5%	0.9197	0.0060	0.0	12,100[1]
3.0%	0.9169	0.0088	0.0	18,310[1]
3.5%	0.9148	0.0109	0.0	24,240[1]
4.0%	0.9130	0.0127	0.0	29,810[1]
4.2%	0.9124	0.0133	0.0038	32,000[1]
4.5%	0.9074	0.0143	0.0082	35,600
4.75%	0.9016	0.0153	0.0130	38,890
5.0%	0.8958	0.0161	0.0180	42,290

[1] From initial analysis

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References:

A.   Letter from OPPD (W.G. Gates) to NRC (Document Control Desk), "Application for Amendment of Operating License," December 7, 1992, (LIC-92-304A)

B.   Letter from NRC to OPPD, "FCS Unit No. 1 Amendment No. 155 to Facility Operating License No. DPR-40," (TAC No. M85116) August 12, 1993 (NRC-93-0292)

C.   Letter from OPPD (W.G. Gates) to NRC (Document Control Desk) , "Application for Amendment of Operating License," February 1, 1996, (LIC-96-0010)

D.   Letter from NRC to OPPD, "FCS Unit No. 1 Amendment No. 174 to Facility Operating License No. DPR-40," (TAC No. M94789) July 30, 1996, (NRC-96-126)

NRC Question 4:

Provide the numerical value of the benchmark uncertainty obtained previously for the SFP criticality analysis at Ft. Calhoun to show the new benchmark uncertainty for the DSC criticality analysis is bounding and includes the uncertainties for the canister and the SFP. Provide a table that lists the tolerances, uncertainties and biases accounted for in the SFP/DSC criticality analysis and show these are bounding with respect to the existing SFP criticality analysis that credits burnup in the pool and k_{eff} < 0.95. Include tolerances such as eccentric positioning of fuel assemblies in storage cells since the spacing differs from the SFP racks.

OPPD Response to Question 4:

The benchmark uncertainty utilized in the DSC burnup credit criticality analysis is 0.01549 in Δk_{eff} units. The benchmark uncertainty utilized in the spent fuel pool rack burnup credit criticality analysis is listed in HOLTEC report HI-951400 Table 1 (as noted in response to Question 3). The statistical combination of all uncertainty is given as 0.0106 Δk_{eff} units for fuel in Region 1. A comparison of these two values indicates that the benchmark uncertainty utilized in the DSC burnup credit criticality analysis is conservative.

A table that lists the tolerances, uncertainties and biases with respect to the SFP/DSC criticality analysis has not been provided in this response because such a table does not exist in the criticality analysis. However, the underlying methodology in the construction of the design basis criticality analysis model renders such a table unnecessary. The design basis model is constructed utilizing the worst case geometric and material tolerances and results in the most conservative calculation of the system reactivity. A description of the methodology utilized to determine the design basis criticality model is given below:

The final k_{eff} value for the criticality analysis is expressed as follows:

$$k_{final} = k_{keno} + 2\sigma_{keno} + \Delta k_{method} + \Delta k_{geometry} + \Delta k_{fuel} < \text{Safety Limits}$$

where
  k_{final} is the final effective multiplication factor to be compared to the safety limits,
  k_{keno} is the calculated k_{eff} from the criticality safety code (KENO V.a),
  σ_{keno} is the uncertainty in k_{keno},
  Δk_{method} is the bias due to the criticality analysis methodology and is the same as the benchmark uncertainty described above,
  $\Delta k_{geometry}$ is applied as a bias to the calculated k_{keno} based on the various geometrical/material tolerances pertaining to the Dry Shielded Canister -transfer cask, DSC / TC, and
  Δk_{fuel} is applied as a bias to the calculated k_{keno} due to the various geometrical/material tolerances pertaining to the representation of fuel in the criticality analysis model. Note that this also includes the uncertainty in the fuel manufacturing, depletion parameters and fuel representation (axial burnup bias).

The $\Delta k_{geometry}$ is not explicitly calculated in the DSC criticality analysis. Rather, this is conservatively built into the criticality analysis model of the DSC / TC. The KENO model of the DSC / TC utilized in the criticality analysis is directly obtained from the bounding criticality analysis model (Chapter M.6, Criticality Evaluation for Standardized NUHOMS® FSAR, Revision 9, USNRC COC 72-1004) utilized in the part 72 criticality analysis. First, a series of evaluations that evaluate the reactivity effects of geometry and material tolerances of the various DSC / Cask components are performed. The design basis criticality model is then constructed by combining (not statistically combining) the worst reactivity effects of these parameters. For the NUHOMS® -32PT DSC, the design basis criticality model is obtained by utilizing the nominal poison/aluminum plate thicknesses, minimum basket structure thickness, minimum fuel compartment width and minimum assembly to assembly pitch.

In addition, the most reactive placement of the fuel assemblies is evaluated in Table 4-6 (Page 18 of 36) of the criticality analysis, Enclosure of this letter. This evaluation also bounds the consideration of effects due to eccentric positioning of fuel assemblies.

In summary, the DSC/TC criticality analysis model combines the worst reactivity effects of the various geometry/material tolerances important to criticality while the SFP Rack criticality analysis results in a statistical combination of the uncertainties arising from the various geometry/material tolerances important to criticality. Therefore, the treatment of geometry and material tolerances in the DSC/TC analysis is bounding.

The Δk_{bias} is not explicitly calculated in the DSC criticality analysis. Rather, it is partly included in the CASMO model that provides the final burned fuel isotopic concentrations utilized in the criticality analysis and the remaining is included as an axial burnup bias (discussed in page 17 of 36).

Two sets of benchmarks are discussed in Page 16 of 36 – reference [8.12] that is a generic CASMO benchmark and reference [8.13] that specifically benchmarks the Fort Calhoun reactor operations for Cycles 17 through 20. The results of reference [8.13] benchmarks are briefly discussed in page 17 of 36 and demonstrate that CASMO calculations tend to overpredict the system reactivity for burnup credit. Since the CASMO predicted isotopic concentrations are directly input to the KENO V.a models, no additional bias due to manufacturing / operation is included in the final results. However, the axial burnup bias is included in the final k_{eff} value to overcome the non-conservatism due to the utilization of an axially uniform burnup profile within the fuel assembly.

NRC Question 5:

Clarify the discrepancy between Summary bullet 1 and page 21 of 36.

OPPD Response to Question 5:

The criticality analysis, Page 21 of 36, enclosed with LIC-05-0119 provided the cask assembly for a case with 500 ppm of soluble boron. The k_{eff} value listed did not include the 0.013 Δk_{eff}. This inadvertent omission is corrected below and the results continue to demonstrate the cask assembly k_{eff} is below 0.95.

Enrichment $^w/_o$ U-235	Burn-Up (MWD/MTU)	KENO k_{eff}+2σ	k_{eff} + 2σ + Δk_{eff}[tt]
3.5	24,110	0.89224	0.92073

[tt] k_{eff} + 2σ + Δk_{eff} = 0.89224 + 0.01549 + 0.013 = 0.92073

The first bullet of the summary has been changed to reflect the correction discussed above and is now consistent Table 6-1:

- The most reactive DSC cask configuration (normal case) has a k_{eff}< 1 (i.e., 0.99713) with unborated water and a k_{eff} < 0.95 (i.e., 0.92073) when flooded with borated water at a concentration of 500 ppm. Both cases apply burnup credit.

The correction above warranted further review of the report and a typographical error was found in the Table 6-1. Namely, the KENO k_{eff}+2σ results for the 4.55 $^w/_o$ U-235 case contained a typographical error which is corrected below. The correction is also applied to Figure 6-1.

Table 6-1
Maximum Enrichment and Burn-up Results for Type "A" and "B" Transfer Cask

Enrichment $^w/_o$ U-235	Burn-Up (MWD/MTU)	KENO k_{eff}+2σ	k_{eff} + 2σ + Δk_{eff}
1.65	0	0.95759	0.97308
2.50	12,180	0.97825	0.99374
3.00	18,340	0.97968	0.99517
3.50	**24,110**	**0.98164**	**0.99713**
3.90	28,670	0.97949	0.99498
4.55	38,220	0.96115	0.98964

Bounding K-eff Results
SFP Peripheral Cells Empty
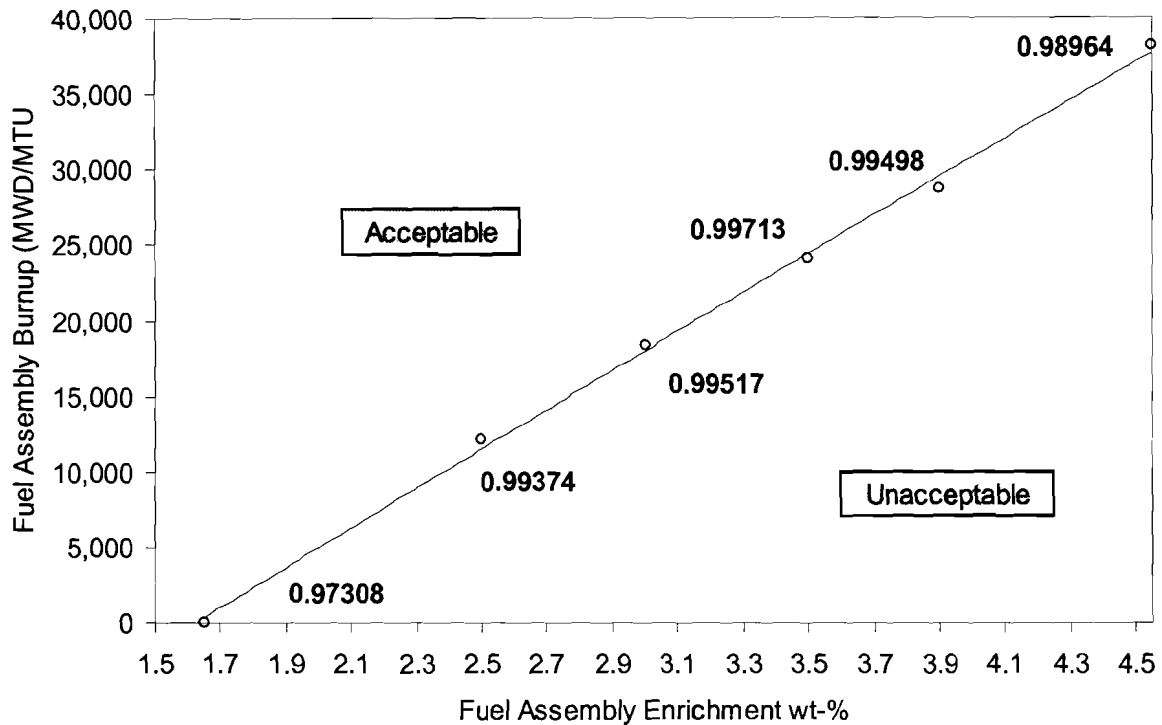
Figure 6-1

Enclosure 1 of this letter contains the revised Calculation Summary "NUHOMS-32PT 50.68 Criticality Analysis for Fort Calhoun," incorporating corrections as noted above.

NRC Question 6:

Clarify where the manufacturing, SFP parameters, and plant operation uncertainties are included in the calculations. Page 16 of 36 towards the bottom states they are not included as part of the benchmark. How were these values calculated? Include them on the table of Question 4.

OPPD Response to Question 6:

The clarifications for the location of the depletion related uncertainties are provided as part of the response to Question 4. However, the relevant portion is repeated here:

Two sets of benchmarks are discussed in Page 16 of 36 of the criticality analysis, Attachment 2 of this letter, – reference [8.12] that is a generic CASMO benchmark and reference [8.13] that specifically benchmarks the Fort Calhoun reactor operations for Cycles 17 through 20. The results of reference [8.13] benchmarks are briefly discussed in page 17 of 36 and demonstrate that CASMO calculations tend to overpredict the system reactivity for burnup credit. Since the CASMO predicted isotopic concentrations are directly input to the KENO V.a models, no additional bias due to manufacturing / operation is included in the final results.

The discussion regarding the inclusion of a Table containing the various uncertainties related to depletion is, therefore, not necessary since the benchmark calculations result in an over-prediction of k_{eff}, thereby rendering the CASMO predicted isotopic concentrations conservative.

NRC Question 7:

In the LAR, Ft. Calhoun states it included a 5 percent burnup uncertainty to the 4.55 enriched fuel on Table 6-1 of Enclosure 1. There is no indication this uncertainty was also added to the less enriched fuel provided in Table 6-1. Section 5.A.5.d of the August 19, 1998, NRC guidance document, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," states the following: "A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculatingly uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption." Please provide the technical justification why a 5 percent reactivity decrement to the burnup was not included in the calculations for burnup less than 30,000 MWD/MTU in accordance with the NRC guidance document referenced in the LAR. Provide the numerical value if a reactivity uncertainty due to uncertainty in the fuel depletion calculations was developed and combined with the other uncertainties for each enriched fuel modeled in the analysis and provided in Table 6-1.

OPPD Response to Question 7:

The reactivity decrement uncertainty due to burnup was incorporated by shifting the burnup curve upwards by 5% uniformly for all burnups and not just for the 4.55 weight percent enriched fuel as Table 6-1 of Enclosure 1 appears to indicate. (Note: Enclosure 1 of LIC-05-0119 has been updated and is included in this letter as Enclosure 1.) The text below the Table 6-1 has been modified to clarify application of this uncertainty to be consistent with the August 19, 1998, NRC guidance document.

Attachment 2
Clarifications to Omaha Public Power District's Letter LIC-05-0119

During the NRC Staff's review of OPPD letter LIC-05-0119 (Reference 3 of this letter), three typographical items were questioned which are clarified as follows:

1. In Attachment 3, "Clean-Typed Technical Specification Pages," TS 4.3.1.3.b should state that "$k_{eff} < 1.0$ if fully flooded with unborated water, ..." The "$\leq$" sign should be a "$<$" sign. Enclosure 2 of this letter contains the corrected page of the Clean-Typed Technical Specification Pages.

2. In Enclosure 1, "Framatome ANP Criticality Analysis," Section 6.1 should have read, "The CASMO results consist of the isotopic inventory used in KENO and these are documented in Reference 8.7 in Attachment A." The reference to "Reference 8.6 Attachment 1" was incorrect. This reference error has been corrected and other clarifications associated with the RAI have been made in the revised criticality analysis being submitted as the Enclosure 1 to this letter.

3. Throughout Attachment 1, the acronym of "DSC" should be referred to as "Dry Shielded Canister." The term "Dry Storage Canister," is synonymous to "Dry Shielded Canister," however, "Dry Shielded Canister" is the correct name which should be applied throughout Reference 3. Enclosure 2 of this letter contains corrected pages 2 and 5 (the only affected pages) of Attachment 1 of Reference 3.

LIC-06-0023
Enclosure 1

NUHOMS®-32PT 50.68 Criticality Analysis for Fort Calhoun
(Revision Number 1)

A TRANSNUCLEAR AN AREVA COMPANY	**Form 3.2-1** **Calculation Cover Sheet**	**Calc. No.:**	1121-0600		
		Rev. No.:	1		
Calculation Title:		**Page:**	1	**of**	37
NUHOMS® -32PT 50.68 Criticality Analysis for Fort Calhoun		**Project No.:**	1121		
		DCR No.:	1121-008		

Project Name: NUHOMS® -32PT for OPPD

Number of CDs attached: 0

If original issue, is Licensing Review per TIP 3.5 required? Not Applicable since it is a Revision

☐ No (explain)	☐ Yes	Licensing Review No._____

Software utilized: N/A since this is a documentation of editorial corrections / clarifications.

Calculation is complete

Originator's Signature: Prakash Narayanan *A·Prakash*	Date:03/24/2006

Calculation has been checked for consistency, completeness, and correctness

Checker Signature:	Date: 3/24/06

Calculation is approved for use

Project Engineer Signature:	Date: 3/24/06

Form 3.2-1, Revision 1

Revision Summary

| Rev. 0 | This calculation is prepared to incorporate OPPD comments and clarifications to the Framatome-ANP calculation, 86-9003453-000, "Fort Calhoun NUHOMS - 32PT Criticality Analysis." The TN file number for the Framatome-ANP calculation is 1121-0090. Framatome-ANP is an authorized supplier/vendor under the Transnuclear QA (TIPS) program and therefore an independent review of the Framatome calculation is not performed here.

The main body of this calculation is exactly the same as the Framatome ANP-calculation except for the changes as noted below. These changes would be incorporated with revision bars such that the nature of the change is clear and unambiguous.

    1) Title of Table 6-1 changed to "Maximum Enrichment and Burn-up Results for Type "A" and Type "B" Transfer Cask" in the List of Tables to be consistent with the actual table title.

    2) Adjust the table (page 9) after Table 4-1 (before Section 4.2) such that the last row does not spill over to the next page.

    3) Table 6-4 (page 26), added "0°" to the last entry in column 1 as it was missing.

    4) Section 7.0 (page 29), third bullet, 4.5 w/o was replaced with 4.55 w/o.

    5) Section 7.0 (page 29), added an additional bullet after the third bullet that describes a polynomial fit of the fuel burnup as a function of initial enrichment based on the results shown in Table 6-1 and Figure 6-5. |
|---|---|

Rev. 1	This calculation is revised to incorporate the changes as a result of a revision to the AREVA/FANP calculation, 86-9003453. The current revision to the calculation is revision 1. The revision was necessitated to provide responses to the NRC in order to correct typographical errors and clarify the treatment of burnup uncertainty in the criticality analysis.

The main body of the calculation is exactly the same as the AREVA/FANP calculation except for the editorial enhancements detailed above for revision 0 of this calculation*. It may be noted that the total number of pages in the document is increased by 1 due to the inclusion of this revision summary. However, the revision summary page numbering has been changed to a different format so that the page numbering in the main document is consistent with that of the referenced AREVA/FANP calculation. The footer section in the main document has been modified to include the total number of pages in the AREVA/FANP document and the TN document for clarity.

For additional clarity, the description of the changes to the AREVA/FANP document is detailed below. As in the previous revision, these changes would be incorporated with revision bars such that the nature of the change is clear and unambiguous.

1) A typographical error in Table 6-1 in page 21 has been corrected for the 4.55 wt. % U-235 case.

2) The note below Table 6-1 in page 21 has been replaced with language that specifies the burnup uncertainty applied.

3) A new note is added in Section 6.3, page 21 that describes the calculation of the k_{eff} value and thus includes the burnup uncertainty in the calculated k_{eff}.

4) Figure 6-1 in page 22 has been updated to reflect the changes to the results shown in Table 6-1.

5) Update the summary in page 29 to reflect the changes to the results as shown in page 21. Specifically, the first bullet is modified to include these results.

In addition to these changes that are directly incorporated from the AREVA/FANP calculation, the following are the changes that are identified by AREVA/FANP and TN and are editorial in nature.

1) A typographical error in Section 6.1, Page 21 to replace "Reference 8.7 and in Attachment 1" with "Reference 8.~~X~~ and in Attachment A."
   8.7

2) Correction of a typographical error in the Revision 0 summary page of the calculation – "86-90003453" to "86-9003453".

RLJ 3/27/06

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1.0   OBJECTIVE

The purpose of this report is to summarize the results from the criticality analysis that Framatome ANP (Advanced Nuclear Power) performed for loading operations with the Transnuclear, Inc. (TN) NUHOMS®-32PT Dry Shielded Canister (DSC) and transfer cask OS197L designs in the Fort Calhoun spent fuel pool. The license to operate the reactor includes a criticality safety evaluation of the spent fuel pool. The appropriate limits are noted in the plant Technical Specifications [8.1]. However, the Omaha Public Power District (OPPD) plans to improve the operation of Fort Calhoun with an Independent Spent Fuel Storage Installation (ISFSI). This ISFSI will incorporate Transnuclear Inc. (TN) NUHOMS®-32PT DSC to store the spent fuel. TN has an NRC Safety Evaluation Report (SER) to support the licensing of the NUHOMS®-32PT DSC in the Independent Spent Fuel Storage Installation[8.2]. While the Fort Calhoun Technical Specifications and the Transnuclear SER were considered sufficient licensing documents, in March of 2005 the NRC issued Regulatory Issue Summary (RIS) 2005-05 [8.3]. To comply with this RIS, the criticality safety analysis summarized in this report was required.

2.0   SUMMARY OF RESULTS

The TN criticality safety calculations of the NUHOMS®-32PT DSC have been based on bounding "worst case" conditions [8.4, 8.5]. That is, every parameter whose uncertainty could result in an increase in reactivity of the DSC was combined in the criticality model to represent the maximum increase. These bounding modeling conditions were carried over into the Framatome ANP (FANP) analysis of the DSC and transfer cask in the Fort Calhoun spent fuel pool. Moreover, the FANP calculations were benchmarked to previous TN results to show that FANP modeling would produce TN results for the cask criticality evaluations.

The existing Technical Specifications (TS) for the Fort Calhoun spent fuel pool (SFP) allow burned uranium fuel in Region 2 as shown by TS Figure 2-10 [8.1]. FANP used a bounding modeling approach to duplicate the TS Figure 2-10 results. This benchmark demonstrated that the FANP modeling approach was consistent with the criticality safety analysis results that are the basis for the Technical Specifications. That is, using bounding conditions for every parameter whose uncertainty could result in a reactivity increase, and bounding the uncertainty bias from the model benchmarks, the FANP criticality safety analysis results confirmed the validity of the Fort Calhoun spent fuel pool Technical Specifications. Consequently, the FANP modeling approach, with bounding parameters that produce the maximum k_{eff} (effective neutron multiplication factor), is consistent with the licensing of the spent fuel pool, and the licensing of the TN NUHOMS®-32PT DSC.

The FANP modeling of the NUHOMS®-32PT DSC in the spent fuel pool with bounding conditions was used to perform the criticality safety analysis for the canister loading operation[8.6]. The analysis considered normal and accident conditions. The normal condition analysis was performed with burnup credit in lieu of credit for the soluble boron in the spent fuel pool. The analysis determined the required fuel assembly burnup as a function of initial enrichment and equilibrium decay that maintains $k_{eff} < 1$ for the NUHOMS®-32PT Transportable DSC assembly. The accident events considered were the misloading and dropping of a fresh fuel assembly of the highest enrichment permitted by facility Technical Specifications. The accident analysis determined the soluble boron concentration in the spent fuel pool of 800 ppm is required to maintain $k_{eff} < 0.95$ for the DSC assembly. All the cases apply burnup credit.

3.0   BACKGROUND

The Nuclear Regulatory Commission issued Regulatory Issue Summary (RIS) 2005-05 on March 23, 2005 regarding criticality analyses for spent fuel pools and independent spent fuel storage installations[8.3]. The NRC identified regulatory inconsistencies in licensee methodologies for criticality analyses and concluded that dry cask operations performed in the spent fuel pool (SFP) must meet both 10 CFR Part 72 and Part 50 requirements.

Transnuclear, Inc., an AREVA and Siemens Company, contracted with Framatome ANP, also an AREVA and Siemens company to perform the 10 CFR Part 50 criticality analyses for the NUHOMS®-32PT transportable dry shielded canister (DSC) and transfer cask OS197L in the OPPD Fort Calhoun SFP. The criticality analysis for the NUHOMS®-32PT transportable DSC and transfer cask OS197L is documented in Framatome-ANP calculations 32-9003495-000 [8.6] and 32-9001685-00 [8.7]. This report summarizes the reference calculations.

4.0   METHODOLOGY

The methodology applied in the criticality analysis for the NUHOMS®-32PT transportable DSC and transfer cask OS197L documented in References 8.6 and 8.7 is the same methodology that was previously applied for the Fort Calhoun spent fuel pool criticality analysis [8.8, 8.9] and is based on the CASMO-3 and KENO V.a codes. This analysis is also consistent with previous Framatome-ANP analyses through the use of methods and benchmarks that have been previously reviewed and approved by the NRC for other utilities. The most recent instances where the Framatome ANP methodologies have been submitted for review include:

ADAMS Accession ID	Date	Plant/Docket	Description
ML052510504	08/31/05	Shearon Harris/	Framatome-ANP Report 77-5069740-NP-00
ML052510502	09/01/05	50-400	License Amendment Request

Other previous submittals include the following:

USNRC Docket No. 50-305, "Kewaunee Fresh Storage and Spent Fuel Storage Pool".
USNRC Docket No. 50-346, "Davis Besse Fresh Storage and Spent Fuel Storage Pool".
USNRC Docket No. 50-302, "Crystal River 3 Spent Fuel Storage Pool"
USNRC Docket No. 50-244. 'Ginna Spent Fuel Storage Pool."

The NUHOMS®-32PT DSC assembly KENO V.a criticality analysis is documented in Reference 8.6. The CASMO-3 calculations are documented in Reference 8.7. The following sections of this report summarize these calculations.

4.1   CASMO-3 Calculations

The CASMO-3 (CASMO) calculations provided the isotopic atom densities for the burned 14x14 fuel assembly used in the Fort Calhoun reactor [8.7]. The fuel enrichment evaluated ranged from 2.5 wt% U-235 to 4.75 wt%. The lumped fission product number densities in the CASMO depletion cases can not be used directly by the KENOV.a code because they are not available in the KENOV.a materials library. The lumped fission products 401 and 402 in CASMO have been appropriately modeled in the KENOV.a calculation so that reactivity effects of the lumped fission products are preserved. The method used to convert the CASMO generated isotopic inventory to number densities for use in the KENOV.a calculations was previously developed in Reference 8.8 and demonstrated to be conservative in Reference 8.7.

The CASMO model for the Fort Calhoun fuel was developed using the fuel assembly geometrical information and sample code listings from Reference 8.8. This information is summarized in Table 4-1. The CE type 14x14 fuel assembly was loaded with 96% theoretical density fuel that was reduced to 10.3171 g/cm^3 to account for dishing of the pellets. For conservatism, the fuel assemblies had no axial blankets. CASMO hot full power (HFP) depletions were performed and included the effects of non-uniform axial burnup as well as control rod insertion during operation. The depletion covered specific burnup points where the fuel isotopic inventory was needed for the subsequent KENOV.a calculations.

The fuel assemblies contained sixteen gadolinia bearing fuel rods with 4.0 wt% Gd for the 2.5 and 3.0 wt% enriched fuel. All the higher enrichment fuel assemblies contained sixteen Gadolinia bearing fuel rods with 8.0 wt% Gad rods.

Restart files were saved at the different burnup points where the isotopic inventory was to be calculated. The restart CASMO cases were calculated at the different burnup points of interest at cold conditions with zero ppm soluble boron and no control rods inserted. The control assembly and gadolinia assumptions conservatively overestimate the reactivity associated with operation in the spent fuel assembly.

Calculations were performed to effectively model the reactivity effects of the fission products. Also, the short-lived isotopes were appropriately decayed.

Table 4-1

CE14x14 Fuel Dimensions for CASMO Runs

Dimension	inches	cm
Pellet Diameter	0.377	0.95758
Clad Inner Diameter	0.384	0.97540
Clad Outer Diameter	0.440	1.11760
Pitch	0.580	1.4732
GT/IT Inner Diameter	1.0350	2.62890
GT/IT Outer Diameter	1.1150	2.83210
Array Width based upon pitch	8.12	20.6248
Assembly pitch with Water Channel	8.18	20.7772
Total Fuel Length	128	325.12
Total Fuel Rods in Assembly	176	----
%TD of Stack with Dishing Factor	$94.1341 - 10.3171$ g/cm^3	
Temperature Information		
HFP Moderator Temperature.	566.35 °F	570 °K
HFP Fuel Temperature	1077.53 °F	854 °K
HFP Moderator Pressure	2100 psia	144.79 bars
HFP Boron	500	---
Cold Moderator Temperature	38.95 °F	277 °K
Cold Fuel Temperature	38.95 °F	277 °K
Cold Moderator Pressure	14.696 psia	1.01325 bars
Cold Boron	0	---

The cold restart case sets several of the short lived isotopes to zero to account for total decay using the CASMO multiplication option on the "CNU" card. The isotopes that are set to zero are the following:

Isotope	CASMO ID
Rh-105	45105
I-135	53135
Xe-135	54135
Pm-148	61148
Pm-148m	61248

4.2   KENO V.a Calculations

4.2.1   KENO V.a Benchmark Cases

The initial KENO models for the Type "A" DSC were provided by TN to FANP along with results from TN calculations [8.4, 8.5]. The FANP KENO is part of the SCALE version 4.4a code package operating on the Linux operating system platform. The TN KENO cases were from a SCALE 4.4 application on a Windows based PC platform. The differences between SCALE 4.4 and SCALE 4.4a have been previously documented in the open literature such as the SCALE newsletter. Nevertheless, a benchmark exercise was performed in order to qualitatively assess the difference, if any, between the two KENO applications. The exact same cases were run at FANP and the resulting $k_{eff} \pm 2\sigma$ was compared to the TN $k_{eff} \pm 2\sigma$. The minor differences observed is due to a combination of the differences in histories and the differences between SCALE version 4.4 and 4.4a. Results of the benchmark are presented in Table 4-2.

Table 4-2
Comparison of TN SCALE 4.4 results with FANP SCALE 4.4a

DESCRIPTION	k_{Keno}^{TN}	$\pm \sigma_{Keno}^{TN}$	k_{Keno}^{FANP}	$\pm \sigma_{Keno}^{FANP}$
3.8 wt% U-235 0.40% IMD	0.91459	0.00093	0.91131	0.00089
3.8 wt% U-235 0.50% IMD	0.92611	0.00081	0.92333	0.0008
3.8 wt% U-235 0.55% IMD	0.92694	0.00081	0.92518	0.00082
3.8 wt% U-235 0.60% IMD	0.92766	0.00083	0.92529	0.00078
3.8 wt% U-235 0.65% IMD	0.92629	0.00093	0.92373	0.00107
3.8 wt% U-235 0.70% IMD	0.92276	0.00091	0.92106	0.00082
3.8 wt% U-235 0.75% IMD	0.91469	0.00088	0.91569	0.00084
3.8 wt% U-235 0.80% IMD	0.90956	0.00087	0.90726	0.00085
3.8 wt% U-235 0.85% IMD	0.90205	0.00090	0.90048	0.00095
3.8 wt% U-235 0.90% IMD	0.89505	0.00084	0.89488	0.0009
3.8 wt% U-235 0.95% IMD	0.88713	0.00087	0.88886	0.00088
3.8 wt% U-235 1.00% IMD	0.87992	0.00093	0.87915	0.00085
3.9 wt% U-235 0.40% IMD	0.92068	0.00087	0.91855	0.0009
3.9 wt% U-235 0.50% IMD	0.93149	0.00083	0.93124	0.00084
3.9 wt% U-235 0.55% IMD	0.93556	0.00092	0.93239	0.00101
3.9 wt% U-235 0.60% IMD	0.93569	0.00089	0.93202	0.00085
3.9 wt% U-235 0.65% IMD	0.93180	0.00090	0.93178	0.00082
3.9 wt% U-235 0.70% IMD	0.92844	0.00091	0.92741	0.00082
3.9 wt% U-235 0.75% IMD	0.92429	0.00074	0.92308	0.00091
3.9 wt% U-235 0.80% IMD	0.91900	0.00081	0.91682	0.00089
3.9 wt% U-235 0.85% IMD	0.91043	0.00091	0.90821	0.00093
3.9 wt% U-235 0.90% IMD	0.90370	0.00103	0.90325	0.0008
3.9 wt% U-235 0.95% IMD	0.89521	0.00081	0.89543	0.00097
3.9 wt% U-235 1.00% IMD	0.88938	0.00095	0.88645	0.0009

4.2.2 Comparison of FANP SCALE versions 4.4a, 5.0, and 5.0.2

The use of SCALE 4.4a was determined by a number of factors, including consistency with the previous calculation [8.8]. The KENO calculation [8.6] showed that no significant change would have been introduced by the use of either SCALE version 5.0 or 5.0.2. A major difference in SCALE 5.0.2 is the code fix for an error related to cylindrical holes. The error may occur where the boundary of a cylindrical hole in a KENO model overlays the surrounding boundary. Because this is fixed in SCALE version 5.0.2, and there are no significant changes in results then it can be inferred that, even though holes are used in the TN model the usage does not encounter this error. It is also noted that no cylindrical holes occurred in the TN input files. The results from the comparison is presented in Table 4-3.

Table 4-3
Comparison of FANP KENO Output at 3.9 w/o Enrichment

IMD	SCALE4.4a k_{KENO}	σ	SCALE 5.0 k_{KENO}	σ	SCALE 5.0.2 k_{KENO}	σ
100%	0.88645	0.00090	0.88761	0.00078	0.88632	0.00091
95%	0.89543	0.00097	0.89618	0.00085	0.89603	0.00076
90%	0.90325	0.00080	0.90276	0.00089	0.90252	0.00079
85%	0.90821	0.00093	0.90849	0.00086	0.91102	0.00090
80%	0.91682	0.00089	0.91648	0.00080	0.91609	0.00089
75%	0.92308	0.00091	0.92235	0.00123	0.92080	0.00081
70%	0.92741	0.00082	0.92682	0.00089	0.92758	0.00080
65%	0.93178	0.00082	0.93019	0.00082	0.93235	0.00082
60%	0.93202	0.00085	0.93459	0.00095	0.93181	0.00099
55%	0.93239	0.00101	0.93407	0.00083	0.93216	0.00079
50%	0.93124	0.00084	0.93081	0.00087	0.93085	0.00103
40%	0.91855	0.00090	0.91776	0.00091	0.91766	0.00084

4.2.3 Changes to the TN KENO Model

The changes made to the KENO model involved the number of histories selected, the replacement of the lead shield in the transfer cask with stainless steel and removing the soluble boron. These changes are discussed in the following sections.

4.2.3.1 Number of Histories

Illustrated in Table 4-4 are results with 500k, 1000k, 2000k, and 4000k neutron histories. The aforementioned table shows that results are statistically equal for all cases, except for a couple of cases that slightly exceed 2σ. Therefore, cases with ~1000k histories are sufficient to provide acceptable results. For conservatism, FANP elected to use 2500k histories (eg., gen =2600, npg=1000, and nsk=100) in the continuing input decks.

Table 4-4
Comparison of FANP KENO Results for Different Histories

Enrichment	IMD	Original $keff_{KENO}$	σ	1 million $keff_{KENO}$	σ	2 million $keff_{KENO}$	σ	4 million $keff_{KENO}$	σ
3.8 w/o	40%	0.91131	0.00093	0.91149	0.00056	0.91123	0.00040	0.91154	0.00033
3.8 w/o	50%	0.92333	0.00084	0.92445	0.00061	0.92497	0.00044	0.92437	0.00034
3.8 w/o	55%	0.92518	0.00081	0.92646	0.00064	0.92591	0.00042	0.92603	0.00031
3.8 w/o	60%	0.92529	0.00091	0.92554	0.00056	0.92588	0.00041	0.92566	0.00033
3.8 w/o	65%	0.92373	0.00079	0.92349	0.00069	0.92352	0.00041	0.92319	0.00030
3.8 w/o	70%	0.92106	0.00093	0.91981	0.00062	0.91945	0.00043	0.91979	0.00034
3.8 w/o	75%	0.91569	0.00077	0.91584	0.00067	0.91519	0.00044	0.91520	0.00031
3.8 w/o	80%	0.90726	0.00076	0.90861	0.00064	0.90888	0.00043	0.90869	0.00033
3.8 w/o	85%	0.90048	0.00077	0.90159	0.00072	0.90175	0.00048	0.90250	0.00034
3.8 w/o	90%	0.89488	0.00095	0.89456	0.00062	0.89472	0.00042	0.89500	0.00030
3.8 w/o	95%	0.88886	0.00079	0.88736	0.00069	0.88768	0.00047	0.88679	0.00032
3.8 w/o	100%	0.87915	0.00096	0.87932	0.00068	0.87901	0.00047	0.87948	0.00035
3.9 w/o	40%	0.91855	0.00091	0.91865	0.00059	0.91867	0.00043	0.91821	0.00034
3.9 w/o	50%	0.93124	0.00087	0.93149	0.00062	0.93201	0.00043	0.93133	0.00030
3.9 w/o	55%	0.93239	0.00083	0.93314	0.00066	0.93367	0.00045	0.93310	0.00032
3.9 w/o	60%	0.93202	0.00095	0.93225	0.00062	0.93215	0.00043	0.93320	0.00032
3.9 w/o	65%	0.93178	0.00082	0.93148	0.00065	0.93096	0.00043	0.93139	0.00033
3.9 w/o	70%	0.92741	0.00089	0.92719	0.00066	0.92706	0.00045	0.92729	0.00031
3.9 w/o	75%	0.92308	0.00123	0.92271	0.00063	0.92246	0.00042	0.92244	0.00030
3.9 w/o	80%	0.91682	0.00080	0.91727	0.00064	0.91720	0.00043	0.91734	0.00032
3.9 w/o	85%	0.90821	0.00086	0.90951	0.00063	0.90992	0.00044	0.91053	0.00031
3.9 w/o	90%	0.90325	0.00089	0.90319	0.00061	0.90302	0.00043	0.90292	0.00033
3.9 w/o	95%	0.89543	0.00085	0.89585	0.00064	0.89596	0.00042	0.89535	0.00031
3.9 w/o	100%	0.88645	0.00078	0.88610	0.00062	0.88577	0.00046	0.88709	0.00031

4.2.3.2 Removal of Soluble Boron and Gamma Shield

The KENO criticality calculation [8.6] utilized burnup credit in lieu of boron credit to satisfy the criticality safety criterion. This required that with no soluble boron the system remains subcritical (k_{eff} < 1). The TN KENO model was altered to remove the soluble boron and the interspersed moderator density (IMD) was adjusted from 5% to 100% to determine the most reactive case. All calculations used 2.5 million histories as previously discussed. Based on the Fort Calhoun Technical Specifications, a fresh fuel assembly with an enrichment of $1.65^{w}/_o$ was used as a starting point.

The other modification to the model was the replacement of the lead (Pb) gamma shield in the transfer cask with stainless steel 304 (SS304) as presented by TN for the 'light' transfer cask OS197L [8.10]. Table 4-5 summarizes the changes in results due to the change in shield materials. Figure 4-1 shows the KENO k_{eff} as a function of IMD values from 5% to 100% to determine the most reactive case.

Table 4-5
1.65 w/o Enrichment, Fresh Fuel, Pb and SS304 Shield

IMD	k_{Keno} w/ Pb	σ	k_{Keno} w/ SS304	σ
5%	0.46079	0.00024	0.45935	0.00022
10%	0.54373	0.00028	0.54150	0.00028
15%	0.61306	0.00033	0.61092	0.00030
20%	0.67031	0.00032	0.66978	0.00035
25%	0.71969	0.00035	0.71736	0.00039
30%	0.75922	0.00035	0.75836	0.00035
35%	0.79308	0.00037	0.79205	0.00038
40%	0.82207	0.00039	0.82088	0.00038
45%	0.84652	0.00038	0.84579	0.00040
50%	0.86772	0.00038	0.86672	0.00042
55%	0.88447	0.00044	0.88488	0.00042
60%	0.90022	0.00038	0.89965	0.00041
65%	0.91340	0.00039	0.91278	0.00040
70%	0.92400	0.00042	0.92382	0.00040
75%	0.93297	0.00041	0.93457	0.00038
80%	0.94205	0.00038	0.94159	0.00040
85%	0.94868	0.00042	0.94859	0.00042
90%	0.95430	0.00039	0.95463	0.00040
95%	0.95997	0.00040	0.95947	0.00041
100%	0.96240	0.00038	0.96259	0.00040

Figure 4-1

4.2.4 Benchmark Uncertainties
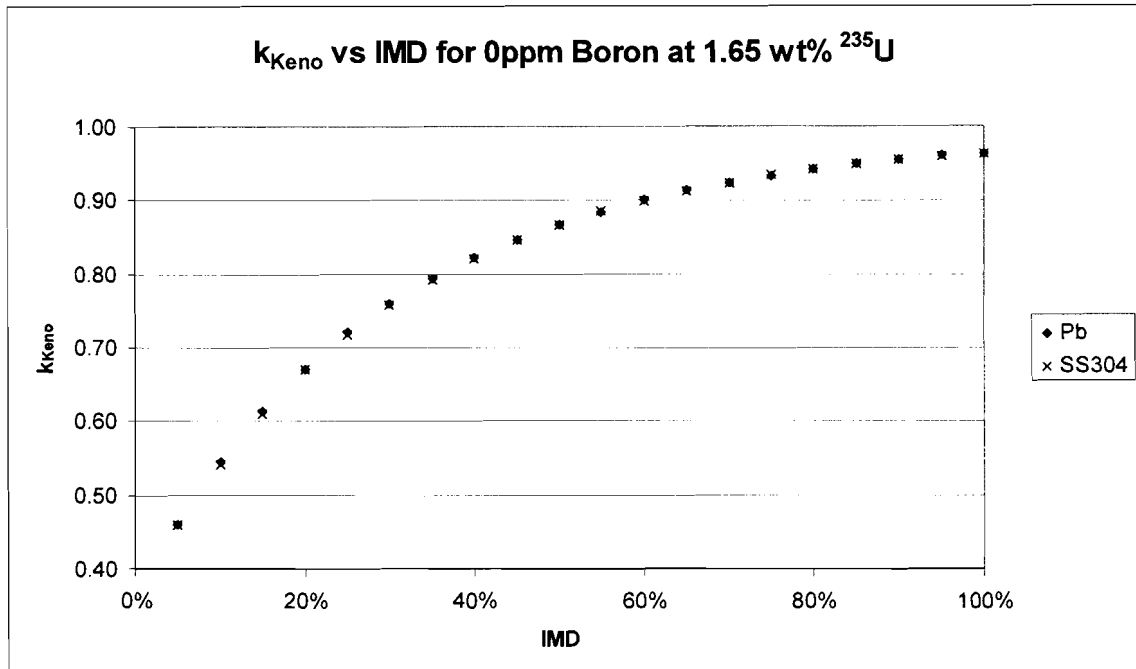
There are four independent benchmarks that FANP has used to establish the appropriate uncertainties for the criticality safety analysis of the Transnuclear NUHOMS®-32PT DSC in the Fort Calhoun spent fuel pool.

(1) The FANP KENO V.a calculations of the NUHOMS®-32PT DSC were compared to those from Transnuclear. The Transnuclear calculations were independently benchmarked to a set of critical experiments; but more importantly, the Transnuclear calculations contained the worst case uncertainties to produce the maximum k_{eff} value.

(2) The FANP KENO V.a calculations were used to model the Fort Calhoun spent fuel pool. While the criticality safety modeling used to support the Technical Specifications incorporated an independent set of uncertainties, the FANP modeling should meet the licensing criteria for the Technical Specifications in order to demonstrate a consistent set of uncertainties.

(3) The FANP KENO V.a calculations were used for benchmark comparisons to a set of critical measurements from cold experiments with the appropriate canister and spent fuel configurations. The cold experiments included plutonium buildup effects.

(4) The FANP CASMO calculations were used for benchmark comparisons of the Fort Calhoun reactor operation during several reload cycles.

The results from the benchmarks demonstrated the maximum reactivity that is associated with the uncertainties in the methods and models provide bounding results for the criticality safety modeling of the NUHOMS®-32PT DSC. The results from the FANP benchmark comparisons to the Transnuclear KENO V.a calculations of the NUHOMS®-32PT DSC demonstrated that the results were equivalent within the statistical uncertainty associated with 500,000 neutron histories.

The transition from the criticality safety modeling of the NUHOMS®-32PT DSC in the Independent Spent Fuel Storage Installation to the loading of the DSC in the spent fuel pool required changing the boron concentration in the water as well as modeling the burned fuel. The DSC in the Independent Spent Fuel Storage Installation had boron concentrations around 2000 ppm (parts per million boron). However, the spent fuel pool criticality safety requirements are based on no boron. Therefore, the Transnuclear optimization of the highest reactivity conditions were repeated for the 0 ppm cases. The results followed those for the spent fuel pool. The highest density (one gram per cubic centimeter) was the most reactive. This included both fresh and burned fuel. In addition, the assembly location in the canister with high boron concentrations continued to have the most reactive conditions with burned fuel and no boron. The other parameters that Transnuclear considered in the modeling of the canister, such as the minimum borated metal loading in the poison plates, remained optimized for the highest reactivity.

The results from the FANP benchmark comparisons with KENO V.a calculations to the Fort Calhoun Technical Specifications for the spent fuel pool demonstrated that the bounding modeling uncertainties produced the Region 2 Technical Specification results in Figure 2-10.[8.1] The bounding modeling uncertainties not only included the uncertainties associated with the spent fuel pool parameters, but also included the uncertainties associated with critical experiment benchmark comparisons and the modeling of plant operation and burnup for several cycles. Thus, while the FANP burned fuel modeling uncertainties in the spent fuel pool would not be expected to be equivalent to those in the safety analysis for the Technical Specifications, they do produce equivalent criticality safety results.

The FANP modeling approach, with bounding parameters that produce the maximum k_{eff}, is consequently consistent with the licensing of the spent fuel pool, and the licensing of the Transnuclear NUHOMS®-32PT DSC. The KENO V.a results discussed in this document include the bounding uncertainties for the canister and spent fuel pool. The only additional uncertainties that need to be considered are those associated with the methods and those associated with the burned fuel isotopic concentrations.

An important part of the guidance that the NRC and ANSI standards provide concerns the benchmark of the methods used for calculating k_{eff}. The information stresses the importance of having the experimental conditions in the benchmarks essentially the same as those for the fuel and fuel cell models. Moreover, the standards note the requirement that the methods used to analyze the spent fuel pool models must be the same as those used to benchmark the experiments. The FANP criticality analysis to model the loading of the NUHOMS®-32PT DSC in the spent fuel pool includes two sets of independent benchmarks in addition to the

CASMO benchmarks. One set of benchmarks includes experiments of specific configurations that are comparable to the Fort Calhoun spent fuel pool and the DSC.[8.11] The second set includes experiments of fuel assembly configurations that are comparable to the Fort Calhoun fuel assemblies.[8.12]

The KENO V.a benchmark calculations [8.15] modeled one hundred critical experiments that are representative of spent fuel pool and canister configurations. Twenty-one of the one hundred experiments were performed by FANP - B & W and are recommended by the NRC in its "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" [8.14]. This set included the effects of burnup by modeling plutonium fuel in addition to uranium fuel. While the standard recommends taking advantage of the statistical nature of the random uncertainties, the approach utilized for the NUHOMS®-32PT DSC criticality safety analysis was to bound all uncertainties associated with the benchmark results. The bounding uncertainty is:

$$\text{Bounding Benchmark Uncertainty} = 0.01549 \ \Delta k_{eff}$$

No calculation of the spent fuel pool - canister experiments gave a higher uncertainty than the above value. Consequently, this value is bounding for subsequent criticality safety evaluations of loading the DSC in the Fort Calhoun spent fuel pool.

The second set of benchmarks is referenced in the "Reactor Analysis System for PWRs" [8.12] report. The fuel design in the first set of benchmarks was not particularly representative of Fort Calhoun, but the second set specifically represented the fuel. Moreover, the first set validated the NITAWL - KENO methods while the second set validated the CASMO methods. The results of the calculations from the second set of benchmarks indicated that the bias and random uncertainties associated with CASMO were smaller than those associated with NITAWL - KENO. That is, no statistically significant bias could be observed and the random deviations were the result of the same type of parameters in the KENO V.a benchmarks. Consequently, no additional unique uncertainty was assigned to the CASMO methods with fuel assemblies of the Fort Calhoun type. Any k_{eff} that is predicted by CASMO includes the bounding uncertainties from the first set of benchmark results (0.01549 Δk_{eff}).

The FANP CASMO calculations were used for benchmark comparisons of the Fort Calhoun reactor operation during several reload cycles. The uncertainties determined by this benchmark comparison are independent of the bounding conditions that are applied to the burned fuel based on the various tolerances for the components. Thus, the overloading of the uranium, replacing the grids with water, decreasing the burnup, increasing the axial burnup effects by modeling the insertion of control rods, overloading the burnable poisons, etc, are not included in this benchmark comparison.

The CASMO benchmark discussed in Reference 8.12 above was based on cold-clean critical experiments. However, the burnup effects of the fuel need to be benchmarked if burnup credit is to be utilized in the spent fuel pool loading of the canister. This set of benchmarks, in the "Ft. Calhoun PRISM Benchmarking Cycles 17-20" [8.13] document, was used to assess CASMO burnup uncertainties. The calculations modeled the fuel in Cycles 17 through 20

and followed the core operation throughout Cycles 17, 18 and 19. The results show a calculation bias that averages -40 ppm boron at the beginning of cycles 17 through 19 and -26 ppm boron at the end. The calculations are more reactive; that is, the calculations require more boron to be critical than is measured. Thus, for applications to criticality safety the CASMO burnup results are considered to be approximately $+.0025 \Delta\rho$ (delta-rho, {delta-k}/k) too reactive. Consequently, they are conservative.

While the uncertainty associated with CASMO burnup benchmarks is conservative, there are uncertainty biases that are applied to ensure bounding criticality safety predictions. The dominant bias is from the axial burnup effects. The burnup credit calculations are performed assuming a uniform burnup profile throughout the active length of the fuel assemblies. The burnup profile for the burned fuel assemblies is not generally uniform due to the axial flux distribution in the core and the neutron leakage from the ends of the fuel assembly. This typically results in a burnup profile that looks resembles a "flattened cosine". The uniform burnup profile assumption results in the over-prediction of burnup at the ends of the fuel assembly and under-prediction of burnup in the fuel mid-region. The difference between the k_{eff} values based on the axial burnup profile and the uniform burnup assumption is what is termed as "axial end-effect" whose magnitude depends on the actual burnup value and the axial burnup profile selected. For the burnup credit calculations, the expected burnup values are such that the assumption of a uniform burnup profile may not be conservative. In other words, an axial end-effect bias needs to be applied to the burnup credit calculations to account for the increase in reactivity due to axial end-effects.

Generic analyses confirm the minor and generally negative reactivity effect of the axially distributed burn-up at values less than about 30,000 MWD/MTU[8.16]. As a result, KENO calculations with less than 30,000 MWD/MTU do not contain an axial bias. The highest burnup evaluated in this effort was 38,200 MWD/MTU. The axial bias uncertainty applied at this burn-up is $+0.013 \Delta k$[8.16].

Another major contributor to the bounding uncertainty is the bias in the assembly burnup. This bias results from inaccurate predictions of fuel assembly power, core power, and cycle lifetime. The modeling follows NRC guidance with a 5% uncertainty at the lower burnups expressed in terms of MWD/MTU (mega-Watt days per metric ton of uranium). When the fuel has been burned for several cycles, the bounding burnup is represented by 1565 MWD/MTU, or an equivalent of 52 EFPDs (effective full power days) in Fort Calhoun. While the application of these biases ensure conservative results with k_{eff} values that are too high, the benchmark of the burned fuel to the Fort Calhoun spent fuel pool Technical Specifications shows that there is an overall consistency with the existing criticality safety analysis.

Because the approach taken with the criticality safety analysis for loading Transnuclear's NUHOMS®-32PT DSC (canister) in the Fort Calhoun spent fuel pool is to treat uncertainties with bounding – biased values, the uncertainty from the benchmarks confirms that the uncertainties are appropriately bounding.

The additional reactivity value that must be applied to the KENO V.a calculations of the loading model k_{eff} is the bounding bias from the benchmark comparisons. As noted above, this value is 0.01549 Δk_{eff}.

4.2.5 Cask Manufacturing and Assembly Tolerances

The bounding conditions for the NUHOMS®-32PT DSC and fuel assemblies were determined by Transnuclear based on various tolerances for the components [8.4, 8.5]. The most reactive system configuration was used for the present criticality evaluation. However, the Transnuclear Part 72 evaluation was performed assuming soluble boron. Since the Part 50 criticality evaluation is performed with fresh water, the present work re-evaluated the moderator density and fuel assembly spacing assumptions to confirm the most reactive configuration was being used.

4.2.6 Fuel Assembly Position Evaluation

The fuel assembly position evaluation for the Type "A" basket was performed by utilizing the most reactive Type A cask configuration and performing three cases; namely, off-set case, the centered case, and a symmetric offset case. The off-set configuration proved to be slightly more reactive. The models and results are shown on the following pages.

The most reactive system water density changed from an IMD of about 0.8 in soluble boron to an IMD of 1.0 in fresh water. The most reactive fuel assembly position is the same as in the soluble boron case. The position corresponds to an off-set toward the center of the cask. These two conditions were combined to perform the criticality analysis. The results are summarized in Table 4-6.

Table 4-6
System Bias Evaluation Results

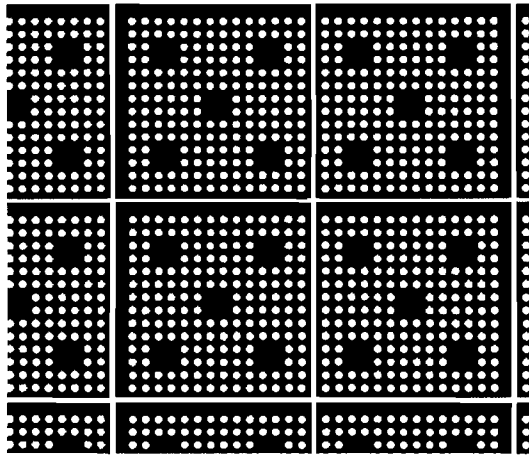| Centered with SS304 no Soluble Boron 4.75wt% at 38.96 GWD/MTU		
IMD	k_{KENO}	S_{KENO}
80%	0.93651	0.00039
90%	0.95552	0.00038
100%	0.96911	0.00041
TN Off-set with SS304 no Soluble Boron 4.75wt% at 38.96 GWD/MTU		
80%	0.93913	0.00039
90%	0.95696	0.00039
100%	**0.97105**	**0.00038**
FANP Offset with SS304 no Soluble Boron 4.75 wt% at 38.96 GWD/MTU		
80%	0.93354	0.00041
90%	0.95148	0.00038
100%	0.96522	0.00042

Figure 4-2
Fuel Assembly Position
Off-Set Inward – Model
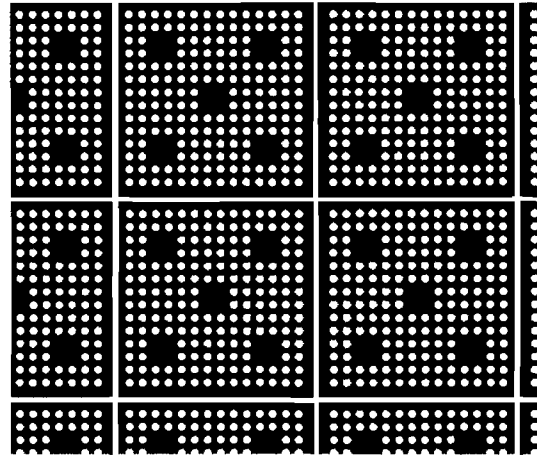
Figure 4-3
Fuel Assembly Position
Centered - Model
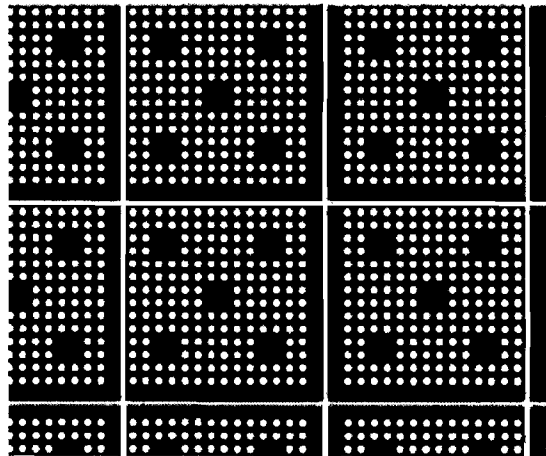
Figure 4-4
Fuel Assembly Position
Off-set Outward Model

5.0   ASSUMPTIONS

5.1   Key Assumptions

A key assumption is any assumption or limitation that must be verified prior to using the the most reactive design.

- NO PRA assemblies are modeled in the DSC.

- The NUHOMS®-32PT Type "A" DSC without PRA bounds the Type "B" DSC.

- Fort Calhoun SFP peripheral cells adjacent to the Cask Pit Area are maintained empty during DSC loading.

- No burnable poisons accounted for in any fuel assembly in the KENO model.

- NO PRA assemblies are modeled in the DSC.

- The transition rails between the basket and the canister shell is modeled as 100% aluminum. Steel and open space in the transition rails reduces reactivity because these materials have much higher absorption cross-sections as compared to the aluminum.

- All stainless steel is modeled as SS304. The small differences in the composition of the various stainless steels have no effect on results of the calculation

- CASMO cases assumed control rod was inserted for part of the depletion to maximize axial effects.

- Water density was at the optimum moderator density of 1.00 gram/cc corresponding to 4°C.

- All fuel rods are filled with fresh water in the pellet/clad gap for both normal and accident conditions.

- All cases assume full DSC reflection in the radial direction and axially with a 20 cm reflector.

- Only 90% credit is taken for the DSC B-10 in the poison plates.

6.0   RESULTS

6.1   CASMO Results

The CASMO results consist of the isotopic inventory used in KENO and these are documented in Reference 8.7 and in Attachment A.

6.2   Normal Conditions KENO Results - Fresh Water

A series of CASMO / KENO runs has produced a curve to coincide in form with the technical specification of the Fort Calhoun Spent Fuel pool. However, the acceptance criteria are different; namely, the spent fuel pool acceptance criterion is for $k_{eff} < 0.95$ whereas the curves in this report have an acceptance criterion of $k_{eff} < 1$. It is understood that the Type "A" basket is limited to 3.9 $^w/_o$. Table 6-1 and Figure 6-1 summarize the results.

Table 6-1
Maximum Enrichment and Burn-up Results for Type "A" and "B" Transfer Cask

Enrichment $^w/_o$ U-235	Burn-Up (MWD/MTU)	KENO $k_{eff}+2\sigma$	$k_{eff} + 2\sigma + \Delta k_{eff}$
1.65	0	0.95759	0.97308
2.50	12,180	0.97825	0.99374
3.00	18,340	0.97968	0.99517
3.50	**24,110**	**0.98164**	**0.99713**
3.90	28,670	0.97949	0.99498
4.55	38,220	0.96115	0.98964

The maximum k_{eff} values in Table 6-1 include an axial bias uncertainty of 0.013 Δk when appropriate and a 5% burnup uncertainty for cases with fuel exposure.

6.3   Normal Conditions KENO Results – Borated Water

The most reactive configuration from the fresh case was analyzed with 500 ppm of soluble boron. The results below demonstrate the cask assembly k_{eff} is below 0.95.

Enrichment $^w/_o$ U-235	Burn-Up (MWD/MTU)	KENO $k_{eff}+2\sigma$	$k_{eff} + 2\sigma + \Delta k_{eff}^{\dagger\dagger}$
3.5	24,110	0.89224	0.92073

†† $k_{eff} + 2\sigma + \Delta k_{eff} = 0.89224 + 0.01549 + 0.013 = 0.92073$

6.4   SFP Region 2 and DSC Type "A" Reactivity Comparison

The reactivity comparison between the DSC and SFP Region 2 was performed by selecting the DSC 3.50 $^W/_o$ result from Reference 8.6 and the SFP Region 2 3.5 $^W/_o$ case from Reference 8.16. The point is on Fort Calhoun Technical Specifications Figure 2-1[8.1]. These results are summarized in Table 6-2.

Table 6-2
SFP Region 2 and DSC Reactivity Comparison

Case	Enrichment $^W/_o$ U-235	Burn-Up (MWD/MTU)	KENO k_{eff}
SFP Region 2	3.5	24,240	0.9148
DSC- Type A	3.5	24,110	0.9809

Figure 6-1

**Bounding K-eff Results
SFP Peripheral Cells Empty**
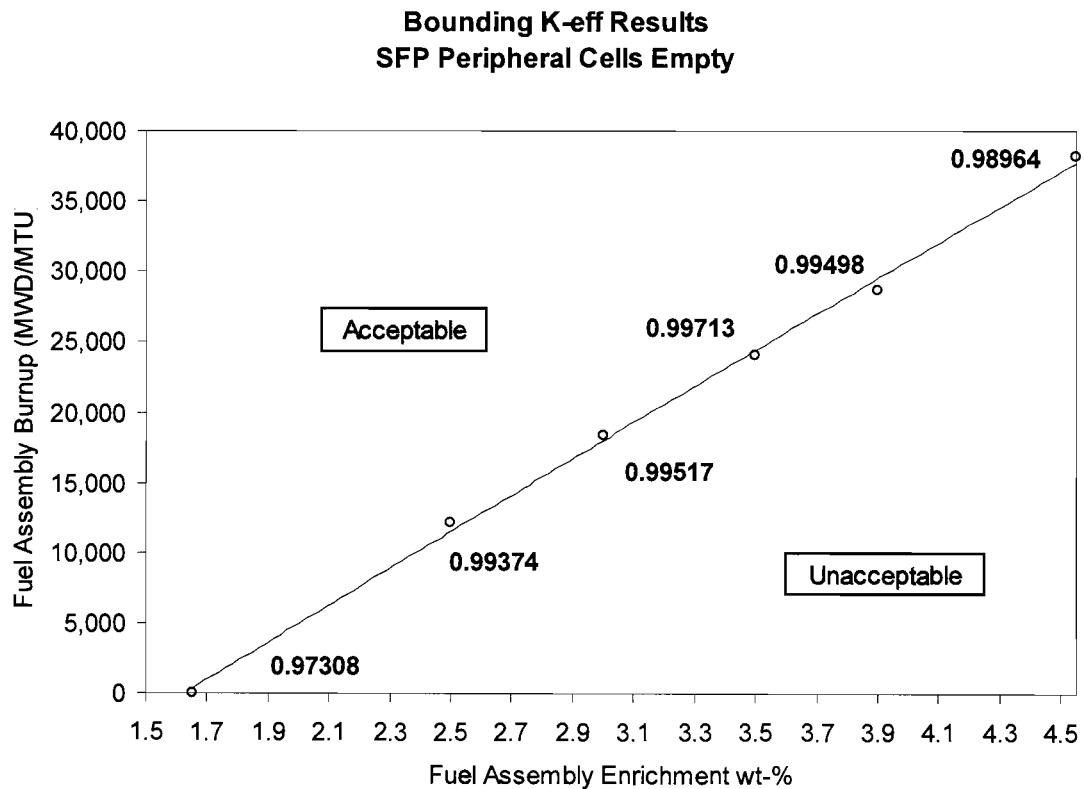
6.5   DSC Position in the SFP Cavity

The DSC and transfer cask assembly is placed diagonally to the SFP fuel racks at approximately 45° due to physical limitations; that is, the cask can not fit in any other configuration. In the configuration shown in Figure 6-2, the closest the DSC transfer cask can approach the SFP rack is proximately 3 centimeters. That proximity is reached when the lifting hooks make physical contact with the SFP rack.

Figure 6-2
DSC and Transfer Cask Position in the SFP Cavity[8.10]
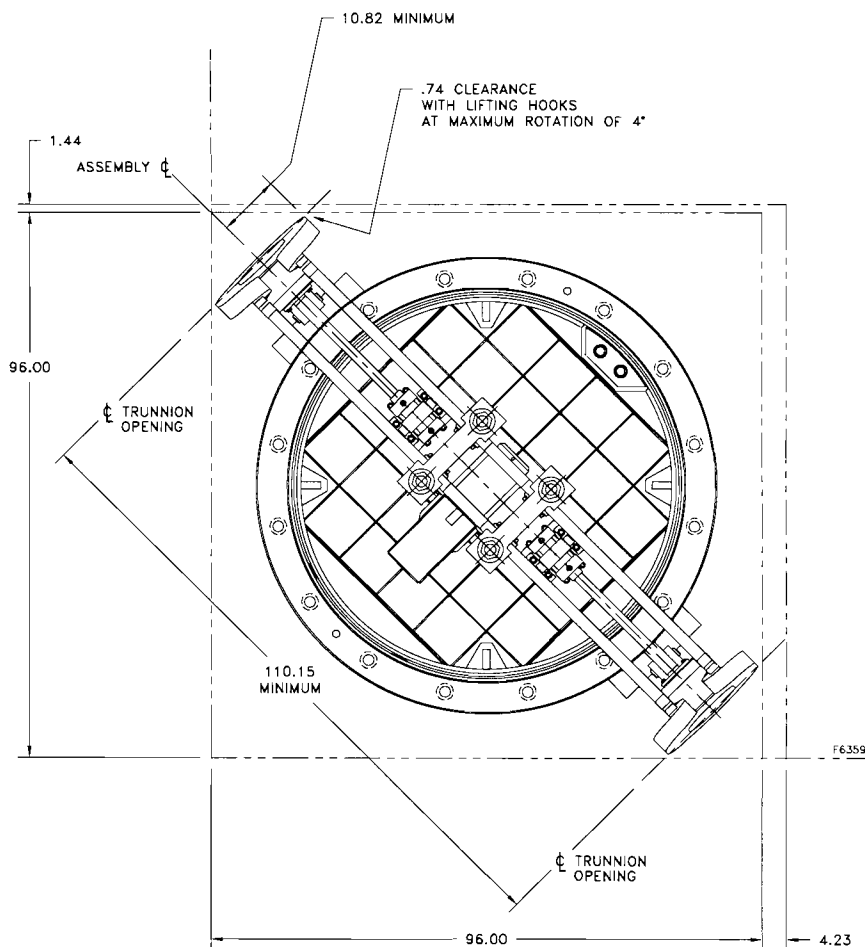
The coupling between the SFP and the DSC was evaluated by assuming the SFP peripheral cells are maintained empty and reflective boundary condition applied to the DSC in the radial direction. The full reflection assumption is in effect a second DSC where the SFP Region 2 is located. This is a conservative approach since the DSC is more reactive than the SFP Region No. 2.

6.6   Accident Condition – Misloaded Fuel

The misloaded fuel assembly accident analysis was performed by assuming the transfer cask is loaded with various enrichments as shown in Table 6-3 and the empty position is loaded with a fresh fuel assembly of 4.5 $^w/_o$ enrichment. Multiple empty locations were assumed in order to locate the most reactive empty cell. The soluble boron concentration values ranged from 500 to 800 ppm. All analysis assumptions from the normal case were also applied to this analysis; namely, all fuel rod gaps are flooded with pure water and the cask is fully reflected. The most reactive fuel in this configuration is the 4.55 $^w/_o$ enrichment. These results show the minimum required soluble boron concentration to maintain $k_{eff} < 0.95$ is 800 ppm with all uncertainties.

Table 6-3
Misloaded Fuel Assembly KENO k_{eff}

Reflected 0 cm Water DSC 3.5 $^w/_o$ with 500 ppm boron			
Bundle Position [†]	k_{eff}	σ	$k_{eff} + 2\sigma + \Delta k_{eff}$
M1	0.92134	0.00045	0.93773
M2	0.94132	0.00053	0.95787
M3	**0.95504**	**0.00055**	0.97163
M4	0.93673	0.00049	0.95320
Reflected 0 cm of Water DSC 3.5 $^w/_o$ with 600 ppm boron			
M3	0.94204	0.00050	0.95853
Reflected 0 cm of Water DSC 3.5 $^w/_o$ with 700 ppm boron			
M3	0.92854	0.00053	0.94509
Reflected 0 cm of Water DSC 3.5 $^w/_o$ with 800 ppm boron			
M3	0.91710	0.00047	0.93353
Reflected 0 cm of Water DSC 3.9 $^w/_o$ with 700 ppm boron			
M3	0.93606	0.00046	0.95247
Reflected 0 cm of Water DSC 3.9 $^w/_o$ with 800 ppm boron			
M3	0.92384	0.00051	0.94035
Reflected 0 cm of Water DSC 4.55 $^w/_o$ with 800 ppm boron			
M3	0.91670	0.00052	0.94623

[†] See next page for the positions

The analysis demonstrated the most reactive cells are toward the periphery of the cask array since these locations do not have boron plates. The misloaded positions are shown in Figure 6-3. The bounding value for the misloading fuel bundle accident at 800 ppm is the following:

$$k_{eff} = k_{eff\text{-}KENO} + 2\sigma_{KENO} + \Delta k_{eff}$$

$$k_{eff} = 0.91670 + 0.00104 + 0.01549 + 0.013 = 0.94623$$

CASE M1
Misload 4.5 w/o Fresh Fuel
0 cm of Water, Reflected

CASE M2
Misload 4.5 w/o Fresh Fuel
0 cm of Water, Reflected

CASE M3
Misload 4.5 w/o Fresh Fuel
0 cm of Water, Reflected

CASE M4
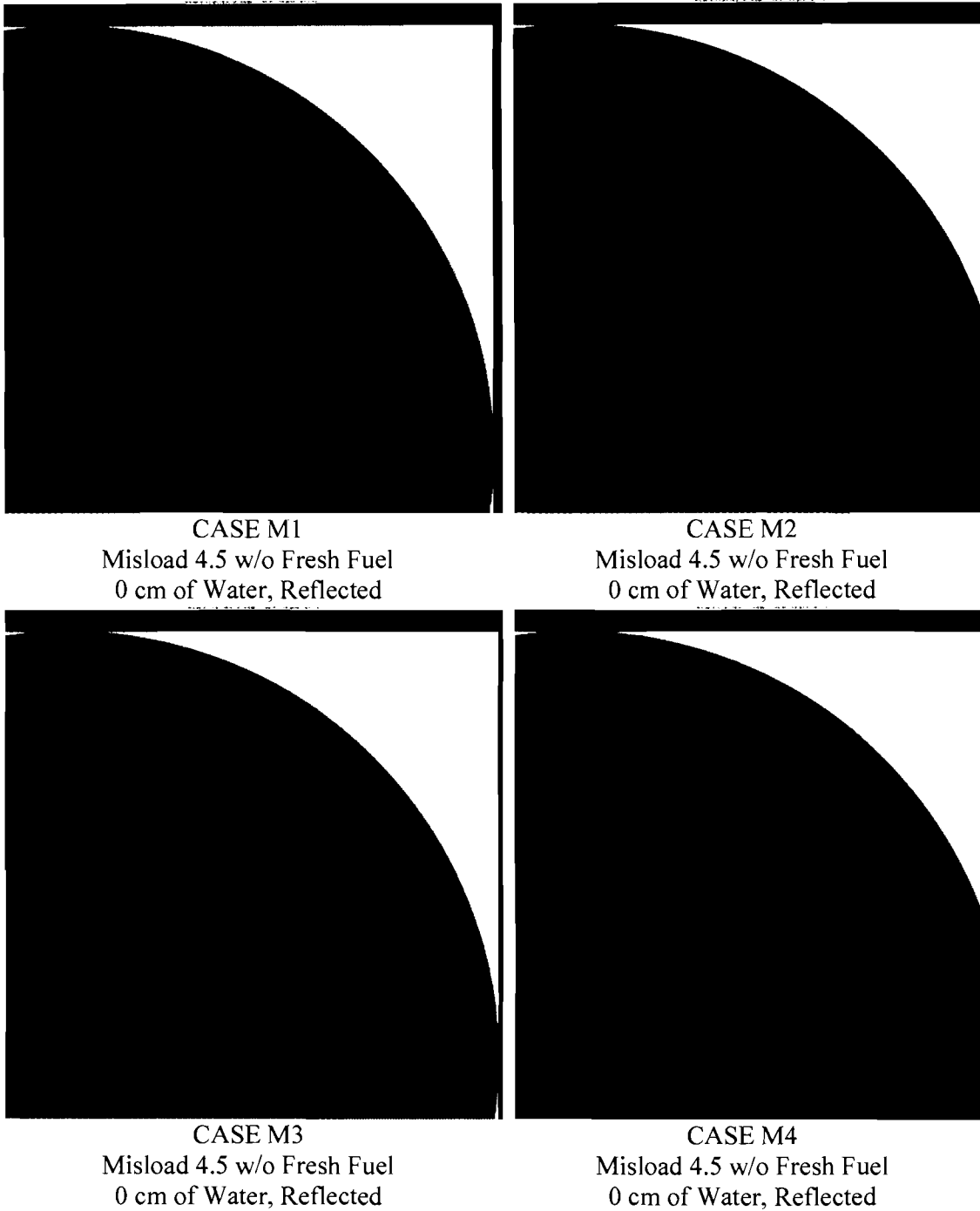Misload 4.5 w/o Fresh Fuel
0 cm of Water, Reflected

Figure 6-3

Misloaded Fresh Fuel Bundle Locations

6.7   Accident Condition - Fuel Drop

The fuel drop accident was evaluated with the same fuel assembly soluble boron assumptions as in the misloading accident. The dropped assembly is placed along the perimeter of the transfer cask aligned longitudinally and evaluated at different azimuthal locations to find the most reactive position. Table 6-4 summarizes the results.

Table 6-4
Fuel Assembly Drop KENO k_{eff}

DSC 3.5 $^{W}/_{o}$ with Soluble Boron 500 ppm			
Bundle Position [†]	k_{eff}	σ	$k_{eff} + 2\sigma + \Delta k$
0°	0.94282	0.00062	0.95955
15°	0.94246	0.00052	0.95899
30°	0.92996	0.00165	0.94875
45°	0.89168	0.00038	0.90793
DSC 3.5 $^{W}/_{o}$ with Soluble Boron 600 ppm			
0°	0.92607	0.00062	0.94280
DSC 3.5 $^{W}/_{o}$ with Soluble Boron 700 ppm			
0°	0.91023	0.00057	0.92686
DSC 3.5 $^{W}/_{o}$ with Soluble Boron 800 ppm			
0°	0.89391	0.00077	0.91094
DSC 3.9 $^{W}/_{o}$ with Soluble Boron 800 ppm			
0°	0.89554	0.00076	0.91255
DSC 4.55 $^{W}/_{o}$ with Soluble Boron 800 ppm			
0°	0.89518	0.00066	0.92499

[†] See next page for the positions

The 0° position is most reactive and the system k_{eff} is bounded by the misloading accident. The bounding k_{eff} value with 800 ppm of soluble boron for the dropped fuel bundle accident is bounded by the misloading accident and is the following:

$$k_{eff} = k_{eff\text{-}KENO} + 2\sigma_{KENO} + \Delta k_{eff}$$

$$k_{eff} = 0.89518 + 0.00132 + 0.01549 + 0.013 = 0.92499$$

Dropped Assembly at 0°	Dropped Assembly at 15°

Dropped Assembly at 30°	Dropped Assembly at 45°
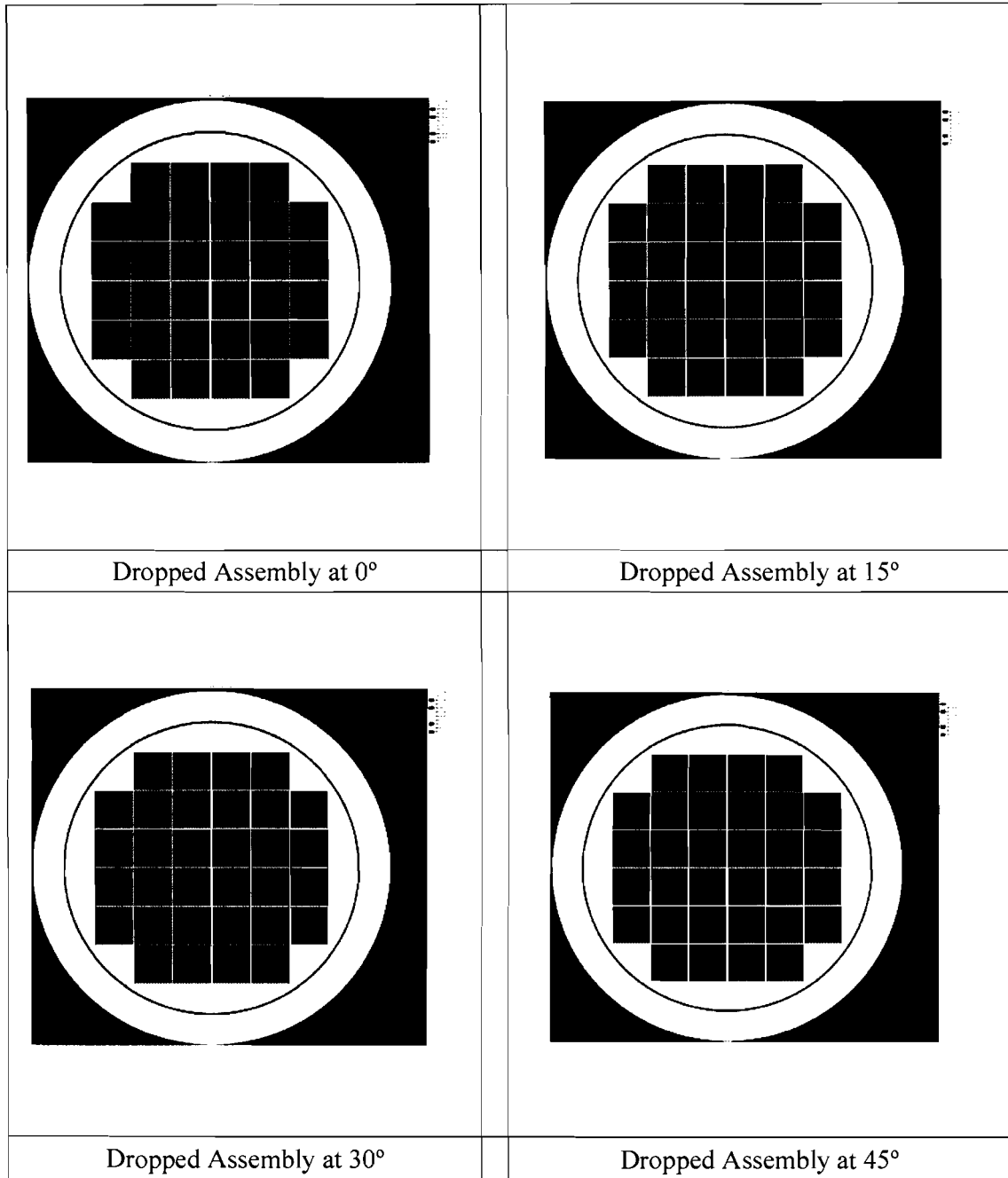
Figure 6-4

Dropped Fuel Bundle KENO Geometry

6.8   DSC Type "A" and Type "B"

The reference KENO calculations were all performed in the most reactive DSC design. It corresponds to the Type "A" basket with no PRA and only 16 poison plates. The basket is limited in enrichment to 3.9 $^W/_o$ by the licensing Certificate of Conformance (CoC) [8.2]. The KENO calculations, however, were extended to 4.55 $^W/_o$ and the results show the Type "A" DSC with 4.55 $^W/_o$ meet the acceptance criterion. Therefore, any fuel cask up to 4.55 $^W/_o$ with a minimum burn-up of 38,220 MWD/MTU can be stored in any of the NUHOMS®32PT DSC basket designs listed in the CoC.

6.9   DSC Type "A" and Type "B" Loading Curve

The KENO results for the fuel enrichment and burn-up combinations listed in Table 6-1 support a loading curve for all the NUHOMS®32PT DSC basket designs listed in the CoC. Figure 6-5 is a composite curve for the Type A and B DSC basket designs.

Figure 6-5

DSC Type "A" and Type "B" Loading Requirements
SFP Peripheral Cells Empty
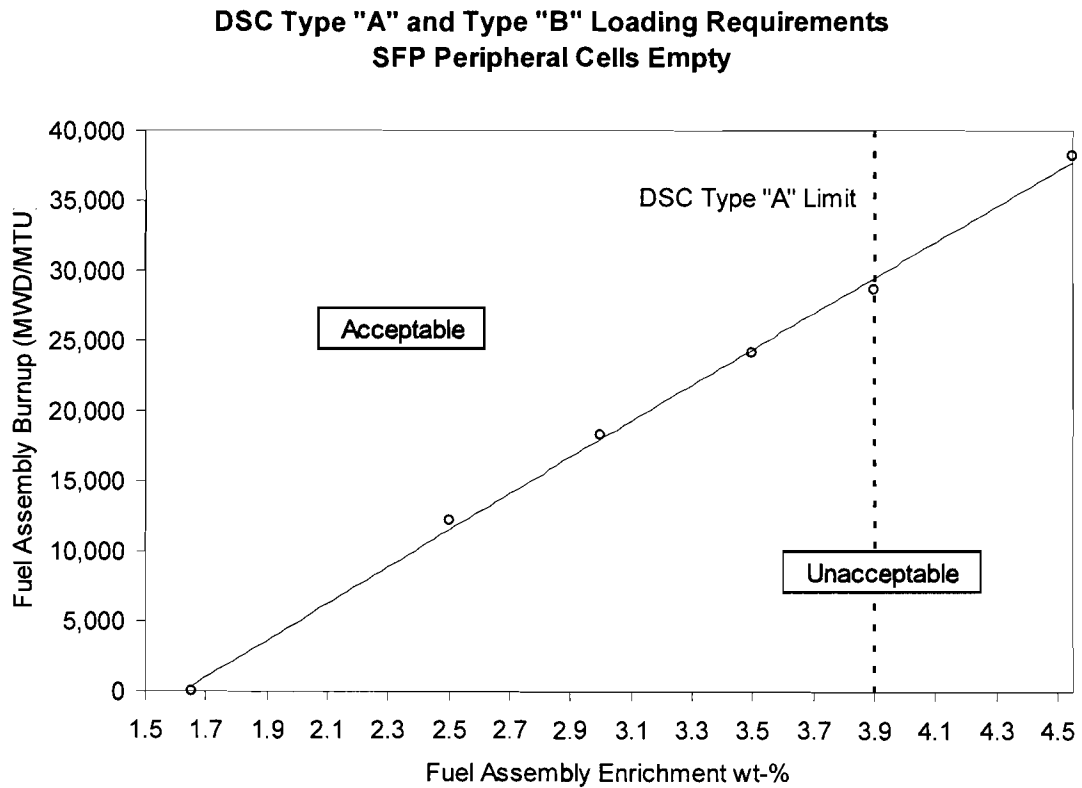
7.0 SUMMARY/CONCLUSIONS

The DSC criticality analysis performed by FANP demonstrates the calculated loading curve complies with §50.68(4) as follows:

- The most reactive DSC cask configuration (normal case) has a $k_{eff} < 1$ (i.e., 0.99713) with unborated water and a $k_{eff} < 0.95$ (i.e., 0.92073) when flooded with borated water at a concentration of 500 ppm. Both cases apply burnup credit.

- The bounding misloaded fuel assembly accident credits soluble boron and the results for 800 ppm is a k_{eff} value of 0.94035. Therefore, the minimum boron concentration required to maintain $k_{eff} < 0.95$ for accident conditions is 800 ppm with burnup credit.

- Any fuel cask up to 4.55 $^{w}/_{o}$ with a minimum burn-up of 38,220 MWD/MTU can be stored in any of the NUHOMS®32PT DSC basket designs listed in the OPPD Certificate of Conformance.

- The minimum required burnup as a function of initial enrichment can be expressed as a third order polynomial as shown below:

$$\textbf{Burnup (MWD/MTU)} = \textbf{A} + \textbf{B1*E} + \textbf{B2*E}^2 + \textbf{B3*E}^3$$

Where

A	= -42324
B1	= 36442
B2	=-7929.3
B3	= 837.1
E	= Initial Enrichment expressed as $^{w}/_{o}$ U-235

- The misloaded fuel bundle accident bounds the dropped assembly accident.

These results require spent fuel pool peripheral cells adjacent to the Cask Pit Area are maintained empty during DSC loading operation.

8.0   REFERENCES

8.1   AREVA/FANP 38-9004068-000, "OPPD Tech Spec for SFP," 1121-0100-23.

8.2   AREVA/FANP 38-9004083-000, TN Transmittal No. 1121-0011-01, Amendment 8A, Chapter M6, Criticality Evaluation".

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

8.4   AREVA/TN Doc. # NUH32PT.0600, "NUHOMS®-32PT Transportable Dry Shielded Canister Criticality Analysis", Transnuclear West Calculational Package, 6/28/2001.

8.5   AREVA/TN Doc. # NUH32PT.0601, "NUHOMS®-32PT with 16 Poison Aluminum Plates", Transnuclear Calculational Package, approved 2/2/2004.

8.6   AREVA/FANP Calculation 32-9003495-000, NUHOMS-32PT Criticality Analysis for Fort Calhoun, October 2005.

8.7   AREVA/FANP Calculation 32-9001685-00, Fort Calhoun Transnuclear CASMO Analysis, October 18, 2005

8.8   AREVA/FANP Calculation 32-5038209-00, Fort Calhoun SFP Criticality Analysis, February 26, 2004.

8.9   AREVA/FANP Calculation 32-5038771-00, Fort Calhoun Spent Fuel Pool CASMO & Uncertainty Analysis, October 19, 2004.

8.10   AREVA/FANP 38-9004097-000, TN Transmittal No. 1121-0011-01, Drawing Sketch 1121-4035, "OS197L Light Yoke Clearences in OPPD Spent Fuel Pool".

8.11   AREVA/FANP Doc. # 32-1245321-00, "KENO V.a SCALE 4.2 Benchmark Calculations," L.A. Hassler, released 3/24/1997.

8.12   AREVA/FANP Doc. # EMF-96-029(P)(A), "Reactor Analysis System for PWRs," S.K. Merk, N.A. Anguiano, C.J. Lewis, R.W. Twitchell, A.H. O'Leary, R.G. Grummer, January, 1997.

8.13   AREVA/FANP Doc. # E-6088-N01-1, Revision 0, "Ft. Calhoun PRISM Benchmarking Cycles 17-20," K.C. Segard and N.A. Anguiano, signed off on 1/25/2002.

8.14   U.S. NRC Memorandum, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," from Laurence Kopp, August 19, 1998.

8.15   AREVA/FANP Doc. # 77-5069740, Revision 0, "Shearon Harris Criticality
       Evaluation" Appendix A - Bias and Bias Uncertainty Evaluation, August 2005.

8.16   AREVA/FANP 32-9004446-000, "Design Input 1121-0100-26 EA-FC-96-001
       Rev 0"

Attachment A
CASMO Isotopic Inventory for Burnup Credit

The number densities for the various isotopes were determined from the running of CASMO in calculation 32-9001685-000 [8.7]. While a significant number of CASMO and KENO runs were performed, the results presented in this attachment are the five sets values that are germane to the technical specification DSC loading curve. The isotopics were calculated in [8.7] with special constraints regarding burnup and burnable poisons, in order to add conservatism to this calculation the burnable poison is not included as a part of the KENO input. The following tables present the isotopic number densities for the specific cases of interest.

Number Densities of Isotopes at 2.5 w/o Enrichment and 11.57 MWd/kg Burnup

Isotope	Number Density
kr-83	1.14978E-06
rh-103	8.77403E-06
xe-131	7.57771E-06
cs-133	1.73247E-05
cs-134	7.74486E-07
cs-135	5.44835E-06
nd-143	1.35117E-05
nd-145	5.60599E-05
pm-147	4.06968E-06
sm-147	6.00616E-07
sm-149	1.05617E-07
sm-150	3.50666E-06
sm-151	2.99131E-07
sm-152	1.74630E-06
eu-153	1.06085E-06
eu-154	1.72340E-07
eu-155	1.35399E-07
u-234	3.78897E-06
u-235	3.50755E-04
u-236	4.17397E-05
u-238	2.21288E-02
np-237	2.55799E-06
pu-238	3.63081E-07
pu-239	9.44123E-05
pu-240	2.16590E-05
pu-241	8.93673E-06
pu-242	1.09121E-06
am-241	1.28453E-07
am-242m	1.38667E-09
am-243	7.29087E-08
cm-242	2.36401E-08
cm-244	6.35366E-09
ag-109	8.57360E-07

Number Densities of Isotopes at 3.0 w/o Enrichment and 17.24 MWd/kg Burnup

Isotope	Number Density
kr-83	1.64262E-06
rh-103	1.25700E-05
xe-131	1.07698E-05
cs-133	2.52894E-05
cs-134	1.50213E-06
cs-135	9.07020E-06
nd-143	1.93952E-05
nd-145	8.26078E-05
pm-147	5.19305E-06
sm-147	1.16650E-06
sm-149	1.21835E-07
sm-150	5.35565E-06
sm-151	3.78887E-07
sm-152	2.46499E-06
eu-153	1.77454E-06
eu-154	3.50312E-07
eu-155	2.21189E-07
u-234	4.21565E-06
u-235	3.63224E-04
u-236	6.07220E-05
u-238	2.19283E-02
np-237	4.31517E-06
pu-238	8.13810E-07
pu-239	1.10581E-04
pu-240	3.07325E-05
pu-241	1.47130E-05
pu-242	2.40265E-06
am-241	3.04944E-07
am-242m	3.73336E-09
am-243	2.31333E-07
cm-242	6.93993E-08
cm-244	2.85659E-08
ag-109	1.34392E-06

Number Densities of Isotopes at 3.5 w/o Enrichment and 22.90 MWd/kg Burnup

Isotope	Number Density
kr-83	2.09699E-06
rh-103	1.60198E-05
xe-131	1.35904E-05
cs-133	3.27658E-05
cs-134	2.39189E-06
cs-135	1.33913E-05
nd-143	2.49432E-05
nd-145	1.08197E-04
pm-147	5.95153E-06
sm-147	1.79783E-06
sm-149	1.35956E-07
sm-150	7.19843E-06
sm-151	4.64319E-07
sm-152	3.07305E-06
eu-153	2.53253E-06
eu-154	5.77729E-07
eu-155	3.29501E-07
u-234	4.54626E-06
u-235	3.78380E-04
u-236	7.94066E-05
u-238	2.16453E-02
np-237	6.46629E-06
pu-238	1.52726E-06
pu-239	1.25355E-04
pu-240	3.89142E-05
pu-241	2.04652E-05
pu-242	4.03443E-06
am-241	5.51327E-07
am-242m	7.32830E-09
am-243	5.06057E-07
cm-242	1.40253E-07
cm-244	8.13660E-08
ag-109	1.83072E-06

Number Densities of Isotopes at 3.9 w/o Enrichment and 27.23 MWd/kg Burnup

Isotope	Number Density
kr-83	2.41887E-06
rh-103	1.82878E-05
xe-131	1.54680E-05
cs-133	3.82667E-05
cs-134	3.18175E-06
cs-135	1.66449E-05
nd-143	2.88363E-05
nd-145	1.27757E-04
pm-147	6.31422E-06
sm-147	2.34653E-06
sm-149	1.42764E-07
sm-150	8.55528E-06
sm-151	5.14663E-07
sm-152	3.45391E-06
eu-153	3.11976E-06
eu-154	7.82885E-07
eu-155	4.25635E-07
u-234	4.85731E-06
u-235	3.93985E-04
u-236	9.29018E-05
u-238	2.15005E-02
np-237	8.15039E-06
pu-238	2.31059E-06
pu-239	1.27526E-04
pu-240	4.21303E-05
pu-241	2.39232E-05
pu-242	5.73574E-06
am-241	7.80913E-07
am-242m	1.10183E-08
am-243	9.00791E-07
cm-242	2.23098E-07
cm-244	1.84831E-07
ag-109	2.17563E-06

Number Densities of Isotopes at 4.55 w/o Enrichment and 34.59 MWd/kg Burnup

Isotope	Number Density
kr-83	3.01079E-06
rh-103	2.24158E-05
xe-131	1.88101E-05
cs-133	4.77601E-05
cs-134	4.45106E-06
cs-135	2.31554E-05
nd-143	3.60785E-05
nd-145	1.60996E-04
pm-147	7.04561E-06
sm-147	3.28527E-06
sm-149	1.57943E-07
sm-150	1.08629E-05
sm-151	6.19866E-07
sm-152	4.14096E-06
eu-153	4.08317E-06
eu-154	1.11946E-06
eu-155	5.80637E-07
u-234	5.20317E-06
u-235	4.04771E-04
u-236	1.19459E-04
u-238	2.12510E-02
np-237	1.14123E-05
pu-238	3.59804E-06
pu-239	1.43440E-04
pu-240	5.17003E-05
pu-241	2.99696E-05
pu-242	7.52720E-06
am-241	1.12934E-06
am-242m	1.66321E-08
am-243	1.32010E-06
cm-242	3.22710E-07
cm-244	2.95961E-07
ag-109	2.69958E-06

LIC-06-0023
Enclosure 2

Revised Pages 2 and 5 of Attachment 1 of LIC-05-0119
Omaha Public Power District's Evaluation for
Amendment of Operating License
"Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool"

Revised 4.0-Page 2 of Clean-Typed Technical Specification Pages

1.0   DESCRIPTION

The Omaha Public Power District (OPPD) is requesting to amend Operating License DPR-40 for Fort Calhoun Station (FCS), Unit No. 1. The proposed changes would revise the FCS Technical Specifications (TS) to add limits and controls for spent fuel cask loading and unloading[1] operations in the spent fuel pool. This License Amendment Request (LAR) is being submitted in response to NRC Regulatory Issue Summary (RIS) 2005-05 (Reference 7.1). An amendment to the FCS operating licensing is required to support dry shielded canister loading operations, scheduled to begin on or about March 1, 2006.

Currently, the FCS TS include limits and controls for storage of unirradiated (fresh) fuel and spent fuel in the FCS spent fuel pool storage racks. Spent fuel cask loading in the spent fuel pool in support of dry shielded canister storage has not previously been performed at FCS. OPPD plans to implement dry spent fuel storage under the general license provisions of 10 CFR 72, Subpart K (Docket No. 72-054) utilizing the Transnuclear Standard NUHOMS® System (10 CFR 72 Certificate of Compliance No. 1004). As a result of our review of RIS 2005-05 and discussions with NRC staff, OPPD has determined that a Part 50 operating license amendment is necessary to support cask loading operations. Because the fuel basket inside the Transnuclear 32PT Dry Shielded Canister (DSC) has a different geometric spacing and neutron poison plate design than the FCS spent fuel storage racks, separate criticality analyses were required to demonstrate compliance with the Part 50 regulations, and corresponding new and revised Part 50 TS were deemed necessary.

Spent fuel cask loading will be performed in the northwest corner of the FCS spent fuel pool in a cask loading area adjacent to existing Region II spent fuel storage racks designated "D," "G2," and "E." The proposed TS changes are consistent with the assumptions and inputs used in the supporting criticality analysis (Enclosure 1). The criticality analysis is consistent with previously accepted methodologies used in licensing actions for the FCS plant and at other nuclear power plants.

Recent similar license amendment requests submitted by Southern Nuclear Operating Company and Entergy Operations (References 7.2 and 7.3), and associated responses to NRC Requests for Additional Information (RAI) (References 7.4 and 7.5) have been reviewed. OPPD has taken into consideration the content of those amendment applications and the issues discussed in the RAIs in developing this LAR, to the extent the information is applicable to FCS, in an attempt to reduce or eliminate any RAIs for this license amendment request.

The following sections include detailed information regarding the proposed changes, background, technical basis, regulatory requirements, no significant hazards, and environmental considerations associated with this license amendment request.

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[1] This LAR addresses any time the spent fuel cask is submerged in the spent fuel pool with one or more fuel assemblies in the cask during loading or unloading operations. Hereafter in this LAR, only loading operations will be discussed for simplicity. The proposed TS are written appropriately to govern both loading and unloading operations.

remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The spent fuel to be loaded into the NUHOMS® System 32PT Dry Shielded Canister (DSC) at FCS cannot be shown by analysis to meet $k_{eff} \leq 0.95$ with unborated water. Therefore, the second set of criteria from 10 CFR 50.68(b)(4) has been used to demonstrate compliance with 10 CFR 50.68(b)(1) for spent fuel cask loading at FCS for this LAR.

In order to demonstrate $k_{eff} < 1.0$ for the spent fuel when flooded with unborated water, the NRC Office of Nuclear Reactor Regulation (NRR) has historically permitted licensees to credit the reduced reactivity of the spent fuel associated with burnup during operation in the Part 50 criticality analysis. The NRC Spent Fuel Project Office (SFPO) has historically required a maximum value of k_{eff} (≤ 0.95) to be demonstrated with all fuel in the spent fuel cask assumed to be fresh fuel at the maximum enrichment allowed by the cask Certificate of Compliance (CoC) as described in NUREG-1536 (Reference 7.9). (Current SFPO review guidance does permit a limited amount of burnup credit to be considered.)

To date, no spent fuel storage systems have been licensed under 10 CFR 72 with burnup credit considered in the criticality analysis. Instead, the criticality analysts have taken credit for the negative reactivity of soluble boron in the spent fuel pool during loading operations for PWR fuel. Thus, Part 72 CoCs require soluble boron credit for certain PWR fuel storage systems to maintain spent fuel in the cask sufficiently subcritical during cask loading operations in the spent fuel pool. In addition, certain Part 50 criticality analyses also incorporate credit soluble boron in the spent fuel pool. However, the minimum soluble boron concentrations in the spent fuel pool required by the Part 50 and Part 72 Technical Specifications are also dependent upon differences in the storage system geometries and the amount of credit taken for neutron poison in the fixed neutron absorber in the spent fuel storage racks and spent fuel cask in the respective criticality analyses. These differences in criticality methodology and acceptance criteria, and the need to comply with both Part 50 and Part 72 during cask loading operations, are described in detail in RIS 2005-05.

FCS Dry Spent Fuel Storage

As part of the long-term spent fuel management strategy at FCS, OPPD has decided to move some of its spent fuel assemblies currently in the spent fuel pool into dry storage at an on-site Independent Spent Fuel Storage Installation (ISFSI) under the general license provisions of 10 CFR 72, Subpart K. ISFSI operations are expected to begin in the first quarter of 2006 and proceed with periodic loading campaigns into the future. OPPD has chosen the Transnuclear Standard NUHOMS® System using the 32PT DSC for dry spent fuel storage. OPPD will load the 32PT DSC under Amendment 8 to the CoC, which is expected to be effective on December 5, 2005 (FR Notice dated 9/20/05). Depending on the type of fuel basket in the 32PT DSC and the enrichment of the fuel to be stored, the Technical Specifications in the NUHOMS® System 10 CFR 72 CoC require anywhere from 1800 to 2500 ppm soluble boron in the DSC for criticality control during wet loading of FCS spent fuel to preserve the assumptions made in the storage system Part 72 design basis criticality analyses.

4.0   DESIGN FEATURES (Continued)

c. A nominal 8.6 inch center to center distance between fuel assemblies placed in Region 2, the high density fuel storage racks,

d. A nominal 9.8 inches (East-West) by 10.3 inches (North South) center to center distances between fuel assemblies placed in Region 1, the low density fuel storage racks,

e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable domain" of Figure 2-10 for "Region 2 Unrestricted" may be allowed unrestricted storage in any of the Region 2 fuel storage racks in compliance with Reference (1),

f. Partially spent fuel assemblies with a discharge burnup between the "acceptable domain" and "Peripheral Cells" of Figure 2-10 may be allowed unrestricted storage in the peripheral cells of the Region 2 fuel storage racks in compliance with Reference (1),

g. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 2-10 will be stored in Region 1 in compliance with Reference (1).

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

a.   Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference (2).

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Reference (2),

d.   A nominal 16 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.1.3   The spent fuel casks are designed and shall be maintained with:

a.   Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent,

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

c. $k_{eff} \leq 0.95$ if fully flooded with borated water ≥ 800 ppm, which includes an allowance for uncertainties as described in Section 9.5 of the USAR,

d.   A nominal 9.075-inch center-to-center distance between fuel assemblies placed in the spent fuel cask,

e.   Spent fuel assemblies with a combination of discharge burnup and initial average assembly enrichment in the "acceptable" range of Figure 2-11.