AREVA

December 13, 2012
NRC:12:064

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

Response to U.S. EPR Design Certification Application RAI 513, Supplement 9

Ref. 1: E-mail, Getachew Tesfaye (NRC) to Dennis Williford, et al (AREVA NP Inc.), "U.S. EPR Design Certification Application RAI No. 513 (5971,5040), FSAR Ch. 9," September 30, 2011.

Ref. 2: E-mail, Dennis Williford (AREVA NP Inc.) to Getachew Tesfaye (NRC), "Response to U.S. EPR Design Certification Application RAI No. 513 (5971,5040), FSAR Ch. 9, Supplement 8," August 31, 2012.

Ref. 3: Letter, Sandra M. Sloan (AREVA NP Inc.) to Document Control Desk (NRC), "Submittal of AREVA Transnuclear Inc. Report TN-Rack.0101 –U.S. EPR New and Spent Fuel Storage Rack Technical Report," NRC:09:117, December 8, 2009.

In Reference 1, the NRC provided a request for additional information (RAI) regarding the U.S. EPR design certification application. A revised schedule for two of the remaining six questions and a history of the prior supplemental responses was provided in Supplement 8 (Reference 2) on August 31, 2012.

The enclosure to this letter provides a technically correct and complete final response to two of the remaining six questions. The enclosure also includes a red line-strikeout format of the affected pages of the U.S. EPR FSAR and AREVA Technical Report TN-Rack.0101 (submitted in Reference 3), which support the response to Questions 09.01.01-53 and 09.01.01-54.

AREVA Transnuclear, Inc. has determined some of the material contained in the enclosed response is proprietary. An affidavit is enclosed, as required by 10 CFR 2.390(b), to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions of the enclosure to this letter are provided.

The following table indicates the respective pages in the enclosed response that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 513 — 09.01.01-53	2	5
RAI 513 — 09.01.01-54	6	6

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NRO

AREVA NP INC.

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Tel.: 434 832-3000 - www.areva.com

The schedule for a technically correct and complete final response to the remaining four questions is unchanged as shown in the following table:

Question #	Response Date
RAI 513 — 09.01.01-58	June 28, 2013
RAI 513 — 09.01.01-61	June 28, 2013
RAI 513 — 09.01.01-62	June 28, 2013
RAI 513 — 09.01.04-20	July 30, 2013

If you have any questions related to this information, please contact Dennis Williford by telephone at (704) 805-2223 or by e-mail at Dennis.Williford@areva.com.

Sincerely,

Pedro Salas, Director
Regulatory Affairs
AREVA NP Inc.

Enclosures:

1. Proprietary Response to U.S. EPR Design Certification Application RAI 513, Supplement 9
2. Non-Proprietary Response to U.S. EPR Design Certification Application RAI 513, Supplement 9
3. Notarized Affidavit

cc: A. M. Snyder
Docket 52-020

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

Transnuclear, Inc.)
State of Maryland) SS.
County of Howard)

I, Glenn Guerra, depose and say that I am a Vice President of Transnuclear, Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is listed below:

Portions of Transnuclear, Inc. document TN-Rack.0101, "U.S. EPR New and Spent Fuel Storage Rack Technical Report," Revision 1.

This document has been appropriately designated as proprietary.

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

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure involves details and analyses related to Transnuclear, Inc.'s design for new and spent fuel storage racks, which are owned and have been held in confidence by Transnuclear, Inc.

2) The information is of a type customarily held in confidence by Transnuclear, Inc. and not customarily disclosed to the public. Transnuclear, Inc. has a rational basis for determining the types of information customarily held in confidence by it.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear, Inc. because the information consists of details and analyses related to Transnuclear, Inc.'s design for new and spent fuel storage racks, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Transnuclear, Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear, Inc.'s product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

Glenn Guerra
Glenn Guerra
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this 13th day of December, 2012.

Lauren McKee
Notary Public

My Commission Expires ____/

Lauren McKee
NOTARY PUBLIC
Anne Arundel County, Maryland
My Commission Expires 2/12/2015

LAUREN MCKEE NOTARY PUBLIC ANNE ARUNDEL COUNTY, MD

Response to

Request for Additional Information No. 513(5971, 6040), Supplement 9
9/30/2011

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 09.01.01 - Criticality Safety of Fresh and Spent Fuel Storage and Handling
SRP Section: 09.01.04 - Light Load Handling System (Related to Refueling)

Application Section: 9.1 and Technical Report TN-Rack.0101, Rev 0

QUESTIONS for Reactor System, Nuclear Performance and Code Review (SRSB)
QUESTIONS for Balance of Plant Branch 1 (AP1000/EPR Projects) (SBPA)

Question 09.01.01-53:

OPEN ITEM

Follow-up to RAI 402, Question 402, Question 09.01.01-48 (Issue #8)

The response to Part a) of Issue #8 in the Response to RAI Letter No. 402 dated June 10, 2011 explicitly states that fuel rod pitch uncertainty is considered in the Region 2 evaluation; however, Table 5-26 of the technical report does not include fuel rod pitch uncertainty. Additionally, in Issue #8, justification was also requested for not considering fuel enrichment uncertainty in both Region 1 and Region 2 evaluations. Provide the fuel rod pitch uncertainty for Table 5-26 of the technical report and the fuel enrichment uncertainty for Tables 5-22 and 5-26, or justify not considering these uncertainties in performing the safety analyses.

Response to Question 09.01.01-53:

The reactivity effect of the various uncertainties associated with the fuel assembly and spent fuel rack geometry design parameters are evaluated in AREVA Technical Report TN-Rack.0101, Tables 5-22 and 5-26.

The fuel rod pitch is controlled by the following two parameters:

- The grid side length of the fuel assembly.
- The number of fuel rods across the width of the fuel assembly.

The nominal pitch is determined as 0.496 in. The fuel rod pitch uncertainty analysis is performed assuming a uniform uncertainty of 0.004 in on the fuel rod pitch.

AREVA Technical Report TN-Rack.0101, Sections 5.4.3 and 5.4.4, will be revised to incorporate information about the criticality analyses associated with fuel rod pitch uncertainty. In addition, Section 5.4.1 (USL determination) and Section 5.4.2 (New Fuel Storage Vault – Results) will be revised as indirect consequences to the consideration of 0.004 in uncertainty on the fuel rod pitch.

The initial enrichment employed in the criticality analyses is the maximal planar nominal enrichment plus a manufacturing uncertainty of +0.05 wt. % U-235. AREVA Technical Report TN-Rack.0101, Section 5.1.3, will be revised to indicate that the loading curve for the Region 2 racks (Figure 5-6) is based on adding 0.05 wt. % U-235 manufacturing uncertainty to the nominal enrichment results. In addition, U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification Figure 3.7.16-1 and Technical Specification Bases for 3.7.16 will also be revised to indicate that enrichments shown in Figure 3.7.16-1 are nominal enrichments plus a manufacturing uncertainty of 0.05 wt. % U-235.

Technical Report Impact:

AREVA Technical Report TN-Rack.0101, Sections 5.1.3, 5.4.1, 5.4.2, 5.4.3 and 5.4.4 will be revised as described in the response and indicated on the enclosed markup.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification Figure 3.7.16-1 and Technical Specification Bases for 3.7.16, will be revised as described in the response and indicated on the enclosed markup.

Supplemental Questions Related to RAI 09.01.01-53:

1. The updated Region 2 uncertainties (which now incorporate rod pitch uncertainty), when applied to Case G2 in Table 5-34, "Criticality Results for the Evaluation of More Than One Misload Fuel Assembly," cause the USL to be exceeded (i.e., Δk_{sys} went from 0.0157 to 0.0219 which leads to a max k_{eff} of ~0.9349 and is >0.9295 by ~540 pcm). This affects conclusions drawn in the Response to Question 9.1.1-59. Why is there no consideration of tolerances and tolerance uncertainties in Table 5-28 (Part 3 of 3), Table 5-33, and Table 5-34?

2. In the response to Question 9.1.1-59, it is stated that "the accident analyses presented in AREVA Technical Report TN-Rack.0101, Section 5.4.4 will be revised to include the possibility of a fuel assembly located in the new fuel elevator as an initial condition of the postulated accidents" and that "Section 5.6 will be revised to include a reference for the SCALE 5 computer code used for the criticality analyses." RAI 402, Question 9.1.1-31 was resolved since the same software (SCALE 4.4) and nuclear data was used for all criticality analyses (except for the CRCs). If SCALE 5 is being introduced at this point, the applicability of the validation becomes questionable for that subset of the safety analysis that uses SCALE5; this affects results in Table 5-28 (Part 3 of 3), Table 5-33, and Table 5-34.

3. Note 6 added to Table 5-26 indicates that the keff from Case C5A will be used for max k_{eff} – k_{eff} from Case C5 is used instead. Also in Table 5-22, the notation for the tolerance uncertainty was modified in footnote (2). For consistency, modify the notation of footnote (3) in Table 5-26.

Response to Supplemental Questions Related to RAI 09.01.01-53:

1. The effect of tolerances and uncertainties, as applied to Case G2 in Table 5-34 of the AREVA Technical Report TN-Rack.0101, will be evaluated as part of a revised supplemental Response to RAI 513, Question 9.1.1-59. This evaluation will include a discussion on the use of an appropriate bias and uncertainty term associated with manufacturing tolerances for the fuel handling accidents and the adequacy of the sub-critical margins for the results shown in Table 5-28 (Part 3 of 3), Table 5-33, and Table 5-34 of AREVA Technical Report TN-Rack.0101.

2. SCALE 4.4 will be used to re-run the appropriate fuel handling analyses. Table 5-28 (Part 3 of 3), Table 5-33, and Table 5-34 of AREVA Technical Report TN-Rack.0101 will be revised to include the new results in a revised supplemental Response to RAI 513, Question 9.1.1-59.

3. Table 5-26 and the corresponding summary table in Section 5.1.3 of AREVA Technical Report TN-Rack.0101 will be revised to update the results for the 3.00 wt. % U-235, 18 GWD/MTU case to be consistent with Note 6. In addition, Table 5-26 of the Technical Report will also be revised to update the equation in Note 3 for consistency.

Technical Report Impact:

AREVA Technical Report TN-Rack.0101, Section 5.1.3 and Table 5-26 will be revised as described in the response and indicated on the enclosed markup.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of these supplemental questions.

Question 09.01.01-54:

OPEN ITEM

Follow-up to RAI 402, Question 402, Question 09.01.01-33

In RAI Letter No. 402 dated July 26, Question 9.1.1-33 requested a description for capturing the treatment of uncertainty in the burnup records. This uncertainty is typically captured in licensee controlled documentation, such as plant procedures, or it can be addressed in the criticality safety analysis. In the Response to RAI Letter No. 402 dated June 10, 2011, the response to Question 9.1.1-33 explains that the Chapter 16 TS Bases will be updated to indicate that no burnup uncertainty term in the plant records is included in Chapter 16 Figure 3.7.16-1 depicting the spent fuel loading curve. Furthermore, it is specified that it will be up to the COL applicant to determine the uncertainty associated with final assembly burnup values. Is the burnup record uncertainty going to be handled as a COL information item? If so, update the FSAR accordingly.

Response to Question 09.01.01-54:

Burnup uncertainty is typically captured in licensee-controlled documentation or addressed in the criticality safety analyses. Since most plant Technical Specifications have requirements for the local power uncertainty to be 5% or lower, it is reasonable to assume that burnup uncertainty is bounded by 5%. The minimum burnup requirements determined in the criticality safety analysis are increased by 5% in the Region 2 spent fuel pool loading curve in order to account for uncertainty associated with burnup.

AREVA Technical Report TN-Rack.0101, Section 5.1.3 and Figure 5-6, will be revised to indicate that the loading curve for the Region 2 racks includes 5% uncertainty added to the minimum burnup requirements (as a function of initial enrichments) determined in the criticality safety analysis. In addition, U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification Figure 3.7.16-1 and --Technical Specification Bases for 3.7.16 will also be revised to indicate that Figure 3.7.16-1 includes a 5% uncertainty on minimum burnup requirements.

Technical Report Impact:

AREVA Technical Report TN-Rack.0101, Sections 5.1.3 and Figure 5-6, will be revised as described in the response and indicated on the enclosed markup.

FSAR Impact:

U.S. EPR FSAR Tier 2, Chapter 16, Technical Specification Figure 3.7.16-1 and Technical Specification Bases for 3.7.16 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

RAI 513,
Q. 09.01.01-53 & 54

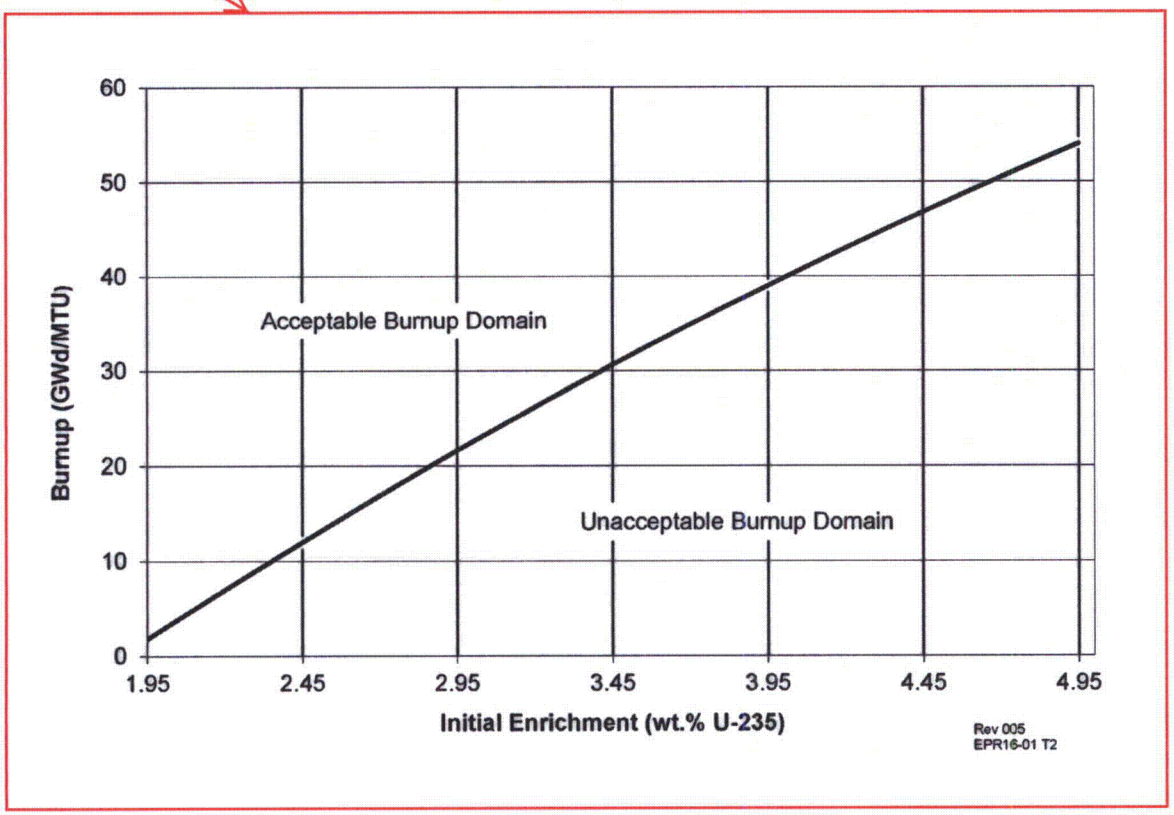

Figure 3.7.16-1
Fuel Assembly Burnup Versus Initial Enrichment for Region 2 Storage

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage

BASES

BACKGROUND

The spent fuel storage pool as described in Ref. 1 consists of 17 modular racks of various sizes. The spent fuel storage pool is divided into two separate and distinct regions. Region 1, with 382 storage locations, is designed to accommodate new fuel assemblies with a maximum ~~nominal~~ enrichment (including uncertainty) of 5.0 wt% U-235 or spent fuel assemblies regardless of the discharge burnup. Region 2, with a maximum of 865 storage locations is designed to accommodate spent fuel assemblies in all locations which comply with the combination of initial enrichment and burnup specified in Figure 3.7.16-1. ~~No burnup or enrichment uncertainty is included in~~ Figure 3.7.16-1 includes an initial enrichment uncertainty of 0.05 wt.% U-235 and a burnup uncertainty of 5%. Enrichment uncertainty is fuel batch dependent. Burnup uncertainty is controlled by plant administrative procedures via either a reduction in the calculated assembly burnup or an increase in the burnup limits determined by analysis. Additionally, new or irradiated fuel assemblies with a combination of burnup and initial enrichment that are not in the acceptable range of Figure 3.7.16-1 may be stored in Region 2 in a checkerboard arrangement (rack location is surrounded on all four adjacent faces by empty rack cells or other non-reactive materials). The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.16.

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The uncertainty for assembly burnup can be based on the $F_{\Delta H}$ uncertainty, which is derived from the maximum pin power uncertainty at any time during the cycle. The $F_{\Delta H}$ uncertainty can be applied to reduce the assembly burnup throughout the cycle to provide a conservative low estimate of the burnup for the spent fuel pool. This method assumes that the maximum pin power uncertainty difference is maintained in the same assembly during the entire cycle, which is not realistic and results in overly conservative fuel assembly burnups.

If more accuracy is desired, measured to predicted comparisons of burnup (time integrated power) using the Aeroball Measurement System (AMS) can be used to define a more accurate uncertainty. Time integrated calculated power and plant measured power can be compared to obtain a calculated burnup uncertainty. Using multiple maps throughout the cycles, measured assembly burnups are accumulated using the measured power distribution. Using the same burnup steps, calculated burnups are also derived using the calculated power distribution. The observed difference between calculated and measured burnups is considered to be an adequate measure of the burnup

Technical Report TN-Rack.0101 Markups

5.1.3 Summary of Criticality Evaluations

The results of the criticality analysis are summarized as follows:

- The maximum k_{eff} with full density moderation remains below the applicable limit of 0.95 for fuel storage in the NFSV.
- The maximum k_{eff} with optimum moderation remains below the applicable limit of 0.98 for fuel storage in the NFSV.
- The maximum k_{eff} with full density moderation in unborated water remains below the applicable limit of 1.0 for fuel storage under normal conditions in the Region 1 racks.
- The maximum k_{eff} with full density moderation in unborated water remains below the applicable limit of 1.0 for fuel storage under normal conditions in the Region 2 racks.
- The maximum k_{eff} with full density moderation in 600~~500~~ ppm (natural boron) borated water remains below the applicable limit of 0.95 for fuel storage under normal conditions in the Region 1 or Region 2 racks.

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- The maximum k_{eff} with full density moderation in 1100 ppm (natural boron) borated water remains below the applicable limit of 0.950 for fuel storage under all accident conditions in the Region 1 or Region 2 Racks.
- Sensitivity calculations with burnup credit demonstrate that the methodology is conservative and results in additional margin to the calculated k_{eff}.

A summary of the criticality analysis results for each of the rack design along with the source are presented in the following tables:

Rack Design – Region 1 or NFSV	K_{EFF}	USL
NFSV – unborated water – full moderator density- water reflected – 5.00 wt. % U-235	0.9373 ~~0.9287~~	0.9414 ~~0.9320~~
NFSV – unborated water – optimum moderator density- water reflected – 5.00 wt. % U-235	0.9373 ~~0.9287~~	0.9595 ~~0.9620~~
Array of Region 1 racks – unborated water 5.00 wt. % U-235	0.9789 ~~0.9585~~	0.9795 ~~0.9820~~
Array of Region 1 racks – ~~200~~ 600 ppm soluble boron (natural), 5.00 wt. % U-235	0.9270 ~~0.9265~~	0.9295 ~~0.9320~~

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Rack Design – Region 2	K_{EFF}	USL
	0.9777 ~~0.9802~~	0.9795 ~~0.9820~~
	0.9189 ~~0.9295~~	0.9~~7~~295 ~~0.9320~~
	0.9115 ~~0.9262~~	0.9~~7~~295 ~~0.9320~~
	0.9780 ~~0.9710~~	0.9795 ~~0.9820~~
	0.9759 ~~0.9788~~	0.9795 ~~0.9820~~
	0.9783 ~~0.9734~~	0.9795 ~~0.9820~~
	0.9762 ~~0.9634~~	0.9795 ~~0.9820~~
	0.9676 ~~0.9633~~	0.9795 ~~0.9820~~
	0.9576 ~~0.9533~~	0.9795 ~~0.9820~~

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Based on the results shown in the table above for burned fuel in Region 2 racks, a loading curve~~, Figure 5-6,~~ for spent fuel qualified to be placed in the Region 2 racks, is developed. [

]

The initial enrichment employed in the criticality analysis includes a manufacturing uncertainty which is bounded by +0.05 wt. % U-235.

The following table provides the allowable enrichment / burnup combinations (loading curve) for storage in Region 2. Loading curves with and without the consideration of uncertainties are shown in this table.

The Region 2 loading curve shown in Figure 5-6, is developed based on those burnup/enrichment combinations that include the enrichment and burnup uncertainties shown in the following table.

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Rack Design	Enrichment (wt% U-235)	Burnup (GWd/tU)
[ ]		

Spent fuel assemblies with initial enrichment and burnup values below the curve are not acceptable for all-cell storage in Region 2 racks, however they are acceptable for storage in a [ ] in Region 2 or Region 1 racks if needed. The best fit second degree polynomial function and its curve are located on the graph.

The criticality analysis is valid for all-cell storage of fuel assemblies in Region 2 for blanketed fuel assemblies with a minimum blanket height of 4.6 inches and a maximum initial blanket enrichment (as a function of assembly average initial enrichment) as shown in Table 5-24. In addition, this analysis is valid for an average hot leg moderator temperature of up to 615 degrees K and the limiting axial burnup distribution (as a function of assembly average burnup) as shown in Table 5-3.

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5.4 Criticality Calculation

5.4.1 USL Determination

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The criticality evaluation used the same cross section set, fuel materials and similar material/geometry options that were used in the ~~142~~ 267 benchmark calculations. The parameters utilized in the USL determination must be specifically calculated for the burnup credit analysis for the racks. Once these specific parameters are calculated, they can be used with the equations shown in Table 5-15 to determine the USL for the racks. **[**

] The EALF is a parameter that is included in each CSAS25 output. The fuel assembly parameters are shown in Table 5-1 and rack dimensions are obtained from design drawings. The EALF (Energy of the Average Lethargy for Fission) values are extracted from the burnup credit criticality (CSAS25) runs. The limiting values are shown in Table 5-16 along with the USL-1 for the limiting parameter. The modeling techniques and the applicable parameters listed in Table 5-16 for the actual criticality evaluations fall within the range of those addressed by the benchmarks. The limiting USL value is **0.~~9820~~ 9795** and is the most conservative USL-1 **[**

] shown in Table 5-16.

The USL functions applied to the NFSV are evaluated considering only fresh fuels critical experiments, the equations are shown in Table 5-15. The USL for the NFSV criticality analysis is determined as **0.9414**, as shown in Table 5-16.

For the Region 1 calculations when soluble boron is not credited, the limiting USL of **0.~~9820~~ 9795** is employed. This is conservative because the Region 1 calculations do not consider burnup credit and therefore, the results of the **[** **]** are not required to be included in the benchmark evaluations.

For both the Region 1 and Region 2 calculations when soluble boron in the spent fuel pool is credited, the USL is reduced to **0.~~9320~~ 9295** (0.~~9820~~ 9795 - 0.05) to ensure that a subcritical margin of at least 0.005 Δk_{eff} is included in these calculations.

5.4.2 New Fuel Storage Vault – Results

The NFSV is filled with 5.0 wt % fresh fuel, **[** **]** and water reflection in all directions. A void is assumed in the gap between the fuel pellet OD and the clad ID. For Region 1 and Region 2 analyses, **[** **]** water is conservatively assumed in the pellet/clad gap. Table 5-25 shows the conservatism in this modeling assumption. The results of the criticality calculations for the NFSV are shown in Table 5-22, case N1. Since the rack design and authorized contents are identical, the tolerance / uncertainty results from Region 1 calculations are directly applied. All calculations are carried out with moderation by **[** **]** water which is also the optimum density for the NFSV. Calculations at various moderator densities are carried out for the NFSV racks and are shown in Table 5-22, cases N2 through N6).

[

].

The various accident scenarios considered for NFSV evaluations result in no significant change to the k_{eff} of the system. The layout of the NFSV modules ensures that a fuel assembly cannot be accommodated in the space between the modules. This implies that such accidental "drops" are not credible. These scenarios are evaluated as follows:

- The case with the misloaded fuel considered the accidental misloading of a single fuel assembly with the highest allowable enrichment. Since, all the fuel assemblies are evaluated at the highest allowable enrichment of 5.00 wt. % U-235, no separate misload calculation is required.
- The worst case vertical accidental drop of the fuel assembly does not result in a rack geometry configuration that is not encompassed by the normal condition calculations whose results are described above. Therefore, a separate calculation is not required.
- The worst case horizontal accidental drop of the fuel assembly results in the fuel assembly being located at the top of the rack module supported by the top steel grid assembly. This configuration results in the presence of at least [ ] inches of water between the fuel assemblies (at maximum moderation conditions) loaded in the cells and the "accidental" fuel assembly. This separation distance is sufficient to isolate the system and does not result in any increase in reactivity and consequently no calculations were performed.

These results demonstrate that the maximum k_{eff} of the NFSV remains below the applicable regulatory limits. The comparisons to the limits are shown below:

1. Maximum k_{eff} (from Table 5-22) at [ ] density 0.9373~~0.9287~~ < 0.9414~~0.9320~~ (USL)
2. Maximum k_{eff} (from Table 5-22) at [ ] density 0.9373~~0.9287~~ < 0.9800 (USL)

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5.4.3 Region 1 Racks – Results

The results of the criticality calculations for the Region 1 racks are also shown in Table 5-22. The results of the tolerance/uncertainty evaluations are also shown in Table 5-22 (case ID A7, A8). [

]. In addition, the results of the fuel eccentric positioning are also shown in Table 5-22 [("Inward position," case ID A10). For the Region 1 evaluation, uncertainties are considered for fuel pellet diameter, fuel pellet density, fuel rod diameter, rod pitch, storage cell inner dimension and fuel assembly position. The flux trap gap and poison plate thickness is modeled using minimum dimensions ensuring that a separate uncertainty evaluation is not required. Separation distance between fuel assemblies is not separately included since the fuel compartment size, poison plate thickness and flux trap gap

define this dimension. Other uncertainties that are not considered include the dimensions of the guide tubes that do not affect the reactivity of the system. The evaluated ~~A bounding~~ model for eccentric position includes one fuel assembly in the middle of the rack module surrounded by fuel assemblies that are centered in their respective cells. To confirm optimum moderator density for Region 1, calculations at various moderator densities are carried out for the Region 1 racks and are shown in Table 5-22] [ ] The "second" model described in Section 5.3.~~2~~3.~~4~~3, (10x10 array with [ ] is used for cases A1 thru A12. No additional evaluations are performed to determine other (possibly severe) eccentric positions because of the [ ]

[

]

The various accident scenarios considered for Region 1 evaluations result in no significant change to the k_{eff} of the system. The layout of the Region 1 modules in the spent fuel pool ensures that a fuel assembly cannot be accommodated in the space between the modules or the modules and the pool wall. This implies that ~~such accidental "drops" are~~placement of fuel assemblies adjacent to the Region 1 racks is not credible, however, calculations are performed to determine the soluble boron concentration to mitigate the effect of such accidental configurations. These scenarios are evaluated as follows:

- The case with the misloaded fuel within the racks considered the accidental misloading of a single fuel assembly with the highest allowable enrichment. Since, all the fuel assemblies are evaluated at the highest allowable enrichment of 5.00 wt. % U-235, no separate misload calculation within the racks is required.
- The worst case vertical accidental drop of the fuel assembly does not result in a rack geometry configuration that is not encompassed by the normal condition calculations whose results are described above. Therefore, a separate calculation is not required.
- The worst case horizontal accidental drop of the fuel assembly results in the fuel assembly being located at the top of the rack module supported by the top steel grid assembly. This configuration results in the presence of at least [ ] inches of water between the fuel assemblies loaded in the cells and the "accidental" fuel assembly. This separation distance is sufficient to isolate the system and does not result in any increase in reactivity and consequently no calculations were performed.

These results demonstrate that the maximum k_{eff} of the Region 1 remains below the applicable regulatory limits. The maximum k_{eff} with and without the presence of soluble boron are calculated using the second model described in Section 5.3.3.3~~2.4~~ that does not include the poison plates in the periphery of the Region 1 rack modules. The comparisons to the limits are shown below:

1. Maximum k_{eff} (from Table 5-22) with no soluble boron 0.9789~~9585~~ < 0.9795~~9820~~ (USL)
2. Maximum k_{eff} (from Table 5-22) with ~~200~~ 600 ppm soluble boron 0.9270~~9265~~ < 0.9295~~9320~~ (USL)

Note that results for additional calculations involving both Region 1 and Region 2 cells are included in Section 5.4.4. The soluble boron worth is higher for fresh fuel compared to burned fuel, therefore, the soluble boron requirements calculated from the whole pool models are bounding for fresh fuel. The additional calculations include the following:

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- The case with the accidental placement of a single fuel assembly adjacent to the Region 1 racks is evaluated using the whole pool model. The results of this evaluation are shown in Table 5-28 as Case #C20. Soluble boron requirements for the worst case are then determined.

5.4.4 Region 2 Racks – Results

The results of the criticality calculations for the Region 2 racks are shown in Table 5-24. These results demonstrate that the calculational models that include the Region 1 and Region 2 interface [

] Case ID # B1, C1, B2 and C2 in Table 5-23.

[

]

Additionally, Table 5-23 shows the result for the case of the fuel pool with fresh fuel stored in a [ ] Region 2 rack modules, fresh fuel and spent fuel in Region 2 rack modules and fresh fuel in Region 1 rack modules. [

]

Table 5-24 shows the results for the nominal criticality evaluation for burned fuel in the Region 2 racks. The results of the tolerance / uncertainty evaluations are also shown in Table 5-26. The uncertainties are calculated as the tolerance case k_{eff} minus the nominal case k_{eff}. In addition, the results of the fuel eccentric positioning are also shown in Table 5-26 (case ID # C10). [

] Since the Region 2 calculations include burnup credit, these results (including tolerance and uncertainty calculations) are shown as a function of burnup and enrichment. For the Region 2 evaluation, uncertainties are considered for fuel pellet diameter, fuel rod diameter, rod pitch, storage cell inner dimension and fuel assembly position. The poison plate thickness is modeled using minimum dimensions ensuring that a separate uncertainty evaluation is not required. Separation distance between fuel assemblies is not separately included since they are covered by the fuel compartment dimension and poison plate thickness evaluations. Other uncertainties that are not considered include the dimensions of the guide tubes that do not affect the reactivity of the system.

[ ]

~~Two~~ One additional uncertainty calculation is ~~are also~~ included in the Region 2 burnup credit evaluation to account for uncertainty in the reactivity effect due to depletion ~~and to account for the uncertainty of the spent fuel composition (fission product isotopes), case ID C13 in Table 5-~~

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Table 5-1
Fuel Assembly Specifications for SAS2H Models

Parameter	Nominal Value	Tolerance (± cm)
Rod array	17×17	
Stack density, % of theoretical value of 10.96 g/cm^3	96.5	
Fuel rod pitch, in (cm)	[ ]	[ ]
Number of fuel rods	265	
Number of guide tubes and instrument tube	24	
Fuel rod clad OD, in (cm)		
Clad thickness		
Fuel rod clad ID, in (cm)		
Active length, in (cm)		
Fuel pellet diameter, in (cm)		
Guide tube OD, in (cm)		
Guide tube ID, in (cm)		
Uranium mass, kg		
Maximum enrichment, wt% U-235		
Maximum enrichment of axial blankets, wt% U-235		

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Table 5-15
USLSTATS Results

USL Function for Region 1 and Region 2

Parameter	Range of applicability	Formula for USL-1 (0.005 Δk_{eff} Margin)	
U-235 initial enrichment (wt%)	[ ]	0.9808 + 1.0203E-03 * X	X < 5.5219
		0.9864	X >= 5.5219
PuO_2 enrichment (wt%)	[ ]	0.9890+ 4.6796E-04 * X	X < 2.4666
		0.9902	X >= 2.4666
Fuel rod pitch (cm)	[ ]	0.9785+ 3.8998E-03 * X	X < 2.1722
		0.987	X >= 2.1722
Water-to-fuel volume ratio	[ ]	0.9831+ 7.7845E-04 * X	X > 4.2920
		0.9865	X <= 4.2920
EALF (eV)	[ ]	0.9880 - 6.5396E-03 * X	X > 0.1380
		0.9871	X <= 0.1380
Soluble boron (PPM)	[ ]	0.9795 + 4.9993E-08 * X	
Assembly Separation in cm	[ ]	0.9873 + 3.3309E-04 * X	X < 6.9944
		0.9896	X >= 6.9944

USL Function for NFSV

Parameter	Range of Applicability	Formula for USL-1 (0.005 Δk_{eff} Margin)	
Initial Enrichment (wt. % U-235)	[ ]	0.9406+9.7267E-04*X	X < 3.7186
		0.9442	X ≥ 3.7186
Fuel Pin Pitch (cm)	[ ]	0.9351+5.2323E-03*X	X < 1.7753
		0.9443	X ≥ 1.7753
Water-to-Fuel Volume Ratio	[ ]	0.9416+7.3797E-04*X	X < 2.1094
		0.9431	X ≥ 2.1094
Fuel Assembly Separation (cm)	[ ]	0.9410+4.9375E-04*X	X < 6.9867
		0.9444	X ≥ 6.9867
EALF (eV)	[ ]	0.9434	Entire Range
~~U Enrichment (wt% U-235) (all experiments)~~		~~0.9813 + (8.9565E-04)*X~~ ~~0.9861~~	~~(X < 5.4382)~~ ~~(X >= 5.4382)~~
~~Fuel Rod Pitch (cm) (all experiments)~~		~~0.9721 + (0.0078263)*X~~ ~~0.9869~~	~~(X < 1.8956)~~ ~~(X >= 1.8956)~~
~~Water/Fuel Volume Ratio (all experiments)~~		~~0.9810 + (1.2274E-03)*X~~ ~~0.9850~~	~~(X < 3.2763)~~ ~~(X >= 3.2763)~~

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~~Energy of the Average Lethargy for Fission (EALF) (all experiments)~~		~~0.9865 (X < 0.1921)~~ ~~0.9879 + (7.1038E-03)*X (X >= 0.1921)~~

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Table 5-16
USL Determination for Criticality Analysis

Region 1 and Region 2

Parameter	Value from Limiting Analysis	Bounding USL-1
Enrichment (wt. % U-235) (all benchmarks)	[ ]	0.9828 ~~0.9831~~
PuO_2 Enrichment (wt%)	1.00	0.9893
Pin Pitch (cm) (all benchmarks)	[ ]	0.9834 ~~0.9820~~
Fuel Separation (cm)	**0.781**	0.9876
Water to Fuel Volume Ratio (all benchmarks)	[ ]	0.9849 ~~0.9839~~
Energy of the Average Lethargy for Fission (EALF) (all benchmarks)	[ ]	0.9830 ~~0.9849~~
Soluble Boron (PPM)	[ ]	0.9795

NFSV

Parameter	Value from Limiting Analysis	Bounding USL-1
Initial Enrichment (wt. % U-235) (fresh fuels)	5.00	0.9442
Fuel Pin Pitch (cm)	[ ]	0.9417
Water-to-Fuel Volume Ratio	[ ]	0.9431
Fuel Assembly Separation (cm)	[ ]	0.9414
Energy of Average Lethargy Causing Fission (eV) (fresh fuels)	[ ]	0.9434

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Table 5-22
Criticality Results for NFSV and Region 1

Case ID	Description	k_{KENO}	σ_{KENO}	k_{eff}	Δk_{eff} (1)
N1	NFSV 10x20 array	0.9045	0.0009	0.9063	
[				]	
N2	90% Internal Moderator Density	0.8704	0.0009	0.8722	
N3	85% Internal Moderator Density	0.8495	0.0008	0.8511	
N4	80% Internal Moderator Density	0.8319	0.0009	0.8337	
N5	75% Internal Moderator Density	0.8112	0.0009	0.8130	
N6	70% Internal Moderator Density	0.7924	0.0009	0.7942	
A1	10x10 array – fresh fuel See Figure ~~4~~5.2	0.9342	0.0010	0.9362	
[				]	
[				]	
A2	90% Internal Moderator Density	0.9097	0.0010	0.9117	
A3	85% Internal Moderator Density	0.8951	0.0010	0.8971	
A4	80% Internal Moderator Density	0.8852	0.0010	0.8872	
A5	75% Internal Moderator Density	0.8753	0.0009	0.8771	
A6	70% Internal Moderator Density	0.8678	0.0011	0.8690	
A7	Maximum Fuel Density	[		]	0.0055
A8	Maximum Pellet Diameter				0.0038
A9	Minimum Outer Clad				0.0018
A10	Inward Position				0.0026
A11	Minimum Inward Compartment				0.0003
A12	~~Maximum~~ Pitch Uncertainty (5)				0.0126 ~~0.0038~~
$\sum \Delta k_{eff}$ (positive U_i only) = Δk_{sys}		0.0263 ~~0.0263~~			
σ_{sys} (2)		0.0031 ~~0.0031~~			
Maximum k_{eff} (full moderator density) NFSV (N1)(3)		**0.9373** ~~0.9287~~			
[		]			
[				]	
[				]	
Maximum k_{eff} (full moderator density) Region 1 (AR13)(3)		**0.9270** ~~0.9265~~			

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(1) Δk_{eff} $U_i = k_{eff\,Ui} - k_{eff}$ (A1) U_i = Uncertainties (Case # A7 through # A12)

(2) $\sigma^2_{sys} = \sum(\cancel{\sigma^2_{KENO} +} (\sigma_{KENO}A1)^2 + (\sigma_{KENO}U_i)^2)$ This results in a conservative calculation of σ^2_{sys} ~~$^2\Delta k_{eff\,Ui}$)~~

(3) max $k_{eff} = k_{KENO} + \Delta k_{sys} + \text{sqrt}(4(\sigma^2_{KENO} + \sigma^2_{sys}))$

(4) max $k_{eff} = k_{KENO} + \Delta k_{sys} + \text{sqrt}(4(\sigma^2_{KENO} + \sigma^2_{sys})) + [k_{eff}^{(AR2)} - k_{eff}^{(A1)}]$

[ ]

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Table 5-24
Criticality Results for Region 2 – Nominal Summary

Model Description	k_{KENO}	σ_{KENO}	k_{eff} ($k_{KENO} + 2\ \sigma_{KENO}$)	Case ID
2.00 wt. % U-235 – fresh	0.9408	0.0008	0.9424	C3
2.50 wt. % U-235 –[ ]	0.9572 ~~0.9478~~	0.0008	0.9588 ~~0.9494~~	C4
3.00 wt. % U-235 –[ ]	0.9574 ~~0.9502~~	0.0008 ~~0.0008~~	0.9590 ~~0.9518~~	C5[(3)]
3.00 wt. % U-235 –[ ]	0.9635	0.0009	0.9653	C5A
3.50 wt. % U-235 –[ ]	0.9551 ~~0.9502~~	0.0008 ~~0.0010~~	0.9567 ~~0.9522~~	C6
4.00 wt. % U-235 –[ ]	0.9581 ~~0.9529~~	0.0008	0.9597 ~~0.9545~~	C7
4.50 wt. % U-235 –[ ]	0.9541 ~~0.9506~~	0.0006 ~~0.0009~~	0.9553 ~~0.9524~~	C2
5.00 wt. % U-235 –[ ]	0.9525 ~~0.9507~~	0.0008 ~~0.0009~~	0.9541 ~~0.9525~~	C1
Results with Soluble Boron Letdown				
2.50 wt. % U-235 –[ ]	0.9515	0.0008	0.9531	L4
3.00 wt. % U-235 –[ ]	0.9543	0.0007	0.9557	L5
3.50 wt. % U-235 –[ ]	0.9576	0.0009	0.9594	L6
5.00 wt. % U-235 –[ ]	0.9522	0.0008	0.9538	L1
[ ]				
	0.8686	0.0011	0.8708	CBR1
	0.8455	0.0011	0.8477	CBR2
	0.8227	0.0011	0.8249	CBR3
	0.9131	0.0005	0.9141	CBR4
	0.9440	0.0005	0.9450	CBR5
	0.9313	0.0004	0.9321	CBR6
	0.9152	0.0004	0.9160	CBR7

(1) 2.50% wt. % U-235 in blanket zone
(2) 3.00% wt. % U-235 in blanket zone
(3) Evaluation employed only in the sensitivity and uncertainty evaluations

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Table 5-26
Criticality Results for Region 2 – Uncertainties Evaluation

	2.50 wt% U-235	3.00 wt% U-235	3.50 wt% U-235	4.00 wt% U-235	4.50 wt% U-235	5.00 wt% U-235	2.00 wt% U-235	Case ID
[								]
$\sum U_i$ Δk_{sys} (positive U_i only)	0.0075 ~~0.0118~~	0.0094 ~~0.0102~~	0.0196 ~~0.0191~~	0.0160 ~~0.0227~~	0.0190 ~~0.0165~~	0.0219	0.0133	-
σ_{sys} (3)	0.0012 ~~0.0017~~	0.0015 ~~0.0013~~	0.0016 ~~0.0018~~	0.0016 ~~0.0014~~	0.0013 ~~0.0018~~	0.0016 ~~0.0014~~	0.0016	
k_{KENO}	0.9572 ~~0.9478~~	0.9635 ~~0.9502~~	0.9551 ~~0.9502~~	0.9581 ~~0.9529~~	0.9541 ~~0.9506~~	0.9525 ~~0.9507~~	0.9408	C1-C7
Maximum k_{eff} (full moderator density) (4)	**0.9676** ~~0.9633~~	**0.9762** ~~0.9634~~	**0.9783** ~~0.9734~~	**0.9777** ~~0.9788~~	**0. 9759** ~~0.9710~~	**0.9780** ~~0.9802~~	**0.9576** ~~0.9533~~	-
USL	0.9795 ~~0.9820~~	0.9795 ~~0.9820~~	0.9795 ~~0.9820~~	0.9795 ~~0.9820~~	0.9795 ~~0.9820~~	0.9795 ~~0.9820~~	0.9795 ~~0.9820~~	-

(1) $\Delta k_{eff} = k_{eff\,Ui}$ ~~–~~ nominal k_{eff}
(2) $U_i = (k_{effUi} - \text{nominal } k_{eff}) \times$ ~~-~~0.05 U_i = Uncertainties, reference [9], conservatively included in addition to depletion methodology bias and uncertainty determined from isotopic assay benchmarks.
(3) $\sigma^2_{sys} = \sum[(\sigma^2_{KENO})^2 + (\sigma_{KENO}\,U_i)^2$ ~~$\Delta k_{eff\,Ui}$~~$)]$.
(4) max $k_{eff} = k_{KENO} + \Delta k_{sys} + \text{sqrt}(4(\sigma^2_{KENO} + \sigma^2_{sys}))$

[

]

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Table 5-27
Criticality Results for Region 2 – Soluble Poison

~~Case~~	Description	k_{KENO}	σ_{KENO}	Δk_{sys} (1)	σ_{sys} (1)	max k_{eff} (2)	USL	Case ID

(1) from Table 5-26

(2) $k_{eff} = k_{KENO} + \Delta k_{sys} + sqrt(4(\sigma^2_{KENO} + \sigma^2_{sys}))$ ~~$k_{KENO} + 2\sigma_{KENO} + 2\sigma_{sys}$~~

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Table 5-28
Criticality Results for Region 2 – Mishandling Accident

(Part 1 of 3~~2~~)

Case	Description	k_{KENO}	σ_{KENO}	Δk_{sys} (1)	σ_{sys} (1)	k_{eff} (2)	USL	Case ID

(1) from Table 5-26

(2) $k_{eff} = k_{KENO} + \Delta k_{sys} + sqrt(4(\sigma^2_{KENO} + \sigma^2_{sys}))$ ~~$k_{KENO} + 2\,\sigma_{KENO} + 2\,\sigma_{sys}$~~

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Table 5-28
Criticality Results for Region 2 – Mishandling Accident

(Part 2 of 3~~2~~)

Case	Description	k_{KENO}	σ_{KENO}	Δk_{sys} (1)	σ_{sys} (1)	k_{eff} (2)	USL	Case ID

(1) from Table 5-26

(2) $k_{eff} = k_{KENO} + \Delta k_{sys} + sqrt(4(\sigma^2_{KENO} + \sigma^2_{sys}))$ ~~$k_{KENO} + 2\sigma_{KENO} + 2\sigma_{tscu}$~~

Additional Results for Mishandling / Misloading Accident

[ ]

Region 2 Parameters	Description	k_{KENO}	σ_{KENO}	k_{eff}	Case ID

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[ ]

X = Initial enrichment (wt. % U-235) and includes an uncertainty of 0.05 wt. % U-235
Y = Burnup (GWd/tU) and includes a burnup uncertainty of 5%

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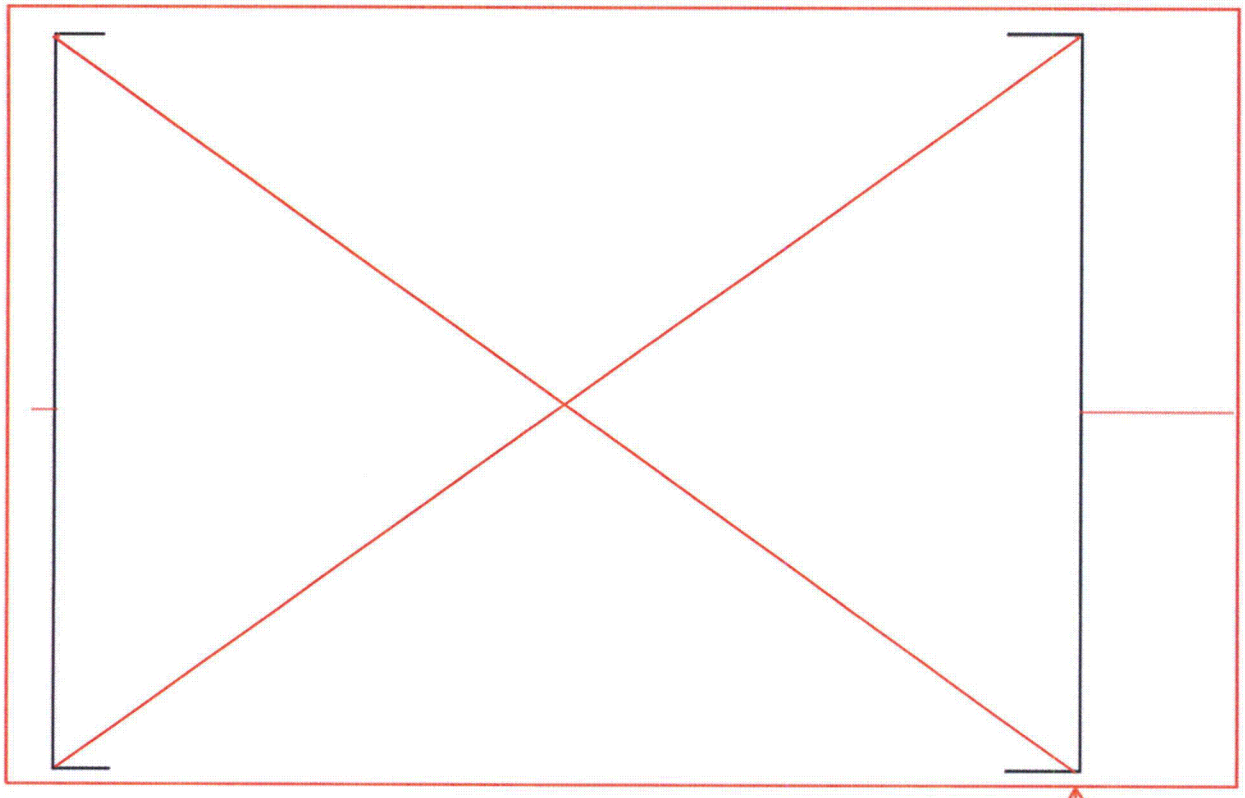

Figure 5-6
Region 2 Spent Fuel Loading Curve

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