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ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649-0001 • 716 546-2700 www.rge.com

ROBERT C. MECREDY
Vice President
Nuclear Operations

March 8, 2000

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Guy S. Vissing
Project Directorate I-1
Washington, D.C. 20555

Subject: Application for Amendment to Facility Operating License
Credit for Soluble Boron in Spent Fuel Pool
Rochester Gas and Electric Corporation
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Reference: (a) Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Boraflex Degradation", dated March 30, 1998.

(b) Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment No. 72 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant", dated July 30, 1998.

(c) Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Application for Amendment to Facility Operating License Date Change for Boraflex Degradation Temporary Measures", dated October 20, 1999.

(d) Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment Regarding a Change from December 31, 1999, to June 30, 2001, Specified in the Technical Specifications (TS) 4.3.1.1.b Note Associated with Maintaining Spent Fuel Pool Boron Concentration >2300 Parts per Million (PPM) at all Times Until a Permanent Resolution to Current Criticality Concerns are Implemented", dated December 21, 1999.

Dear Mr. Vissing:

The enclosed License Amendment Request (LAR) proposes to revise the Ginna Station Improved Technical Specifications (ITS) associated with the Spent Fuel Pool (SFP) Storage (LCO 3.7.13) and Design Features Fuel Storage (4.3).

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In Reference (a), RG&E notified the NRC that testing of boraflex panels contained within Region 2 of the spent fuel pool (SFP) indicated degradation such that certain portions of the criticality analysis provided to the NRC may no longer be conservative. This included the ability of Region 2 to maintain a $k_{eff} \leq 0.95$ if flooded with unborated water. The letter described interim compensatory actions taken by RG&E until a permanent solution with respect to boraflex degradation could be engineered. By Reference (b) the NRC provided approval of the interim measures and the addition of a footnote to the ITS associated with the Design Features Fuel Storage Specification 4.3.1.1.b which required that 2300 ppm boron be maintained in the SFP until December 31, 1999. By Reference (c), RG&E requested a revision to the footnote date for the interim measures, and by Reference (d) the NRC provided approval until June 30, 2001.

This request provides a permanent solution with respect to the boraflex degradation concern through a revision to the storage configuration requirements within the existing storage racks and taking credit for a limited amount of soluble boron, based on a revision of the criticality safety analyses. This would allow resolution of this issue without requiring a physical modification to the storage racks.

RG&E requests that this amendment be approved by May 31, 2001. The requested approval date is based on the date when the authorization for interim measures expires. RG&E requests that upon NRC approval, this LAR should be effective immediately and implemented within 30 days, to allow time for required documentation changes.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy
Vice President
Nuclear Operations Group

Attachments:

I. License Amendment Request
II. No Significant Hazards Consideration Determination
III. Environmental Impact Consideration Determination
IV. Marked up Copy of R.E. Ginna Nuclear Power Plant Improved Technical Specifications
V. Proposed Revised R.E. Ginna Nuclear Power Plant Improved Technical Specifications

Enclosures:

1. R.E. Ginna Spent Fuel Pool Boron Dilution Analysis, January 2000
2. R.E. Ginna Nuclear Power Plant Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit, February 2000

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

U.S. NRC Ginna Senior Resident Inspector

Mr. F. William Valentino, President
New York State Energy, Research, and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Rochester Gas and Electric Corporation	)	Docket No. 50-244
(R.E. Ginna Nuclear Power Plant)	)	

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the regulations of the U.S. Nuclear Regulatory Commission (the "Commission"), Rochester Gas and Electric Corporation ("RG&E"), holder of Facility Operating License No. DPR-18, hereby requests that the Improved Technical Specifications set forth in Appendix A to that license be amended. This request for change in Improved Technical Specifications is to revise the storage requirements contained within the Spent Fuel Pool (SFP) Storage (LCO 3.7.13) and Design Features Fuel Storage (4.3) specifications.

A description of the amendment request, necessary background information, and justification of the requested change are provided in Attachment I. The no significant hazards consideration determination is provided as Attachment II. The environmental impact consideration determination is provided as Attachment III. A marked up copy of the current Ginna Station Improved Technical Specifications which shows the requested change is set forth in Attachment IV. The proposed revised Improved Technical Specifications are provided in Attachment V.

The evaluation set forth in Attachment I and III demonstrates that the proposed change does not involve a significant change in the types or a significant increase in the amounts of effluents or any change in the authorized power level of the facility. The proposed change also does not involve a significant hazards consideration, as documented in Attachment II.

WHEREFORE, Applicant respectfully requests that Appendix A to Facility Operating License No. DPR-18 be amended in the form attached hereto as Attachment V.

Rochester Gas and Electric Corporation

By _____
Robert C. Mecredy
Vice President
Nuclear Operations Group

Subscribed and sworn to before me
on this 8th day of March, 2000.

_____
Notary Public

SHARON P. SORTINO
Notary Public, State of New York
Registration No. 01SO6017755
Monroe County
Commission Expires December 21, 20__

Attachment I
R.E. Ginna Nuclear Power Plant

License Amendment Request
Credit for Soluble Boron in Spent Fuel Pool

This attachment provides a description of the amendment request and necessary justification for the proposed changes. The attachment is divided into four sections as follows. Section A identifies all changes to the current Ginna Station Improved Technical Specifications (ITS) while Section B provides the background and history associated with the changes being requested. Section C provides detailed justification for the proposed changes. Section D lists all references used in Attachments I, II, and III.

A. DESCRIPTION OF AMENDMENT REQUEST

This License Amendment Request (LAR) proposes to revise Ginna Station Improved Technical Specifications to provide a permanent solution with respect to the boraflex degradation concern in the spent fuel pool through a revision to the storage configuration requirements within the existing storage racks and taking credit for a limited amount of soluble boron. The change is summarized below and shown in Attachment IV.

LCO 3.7.13

1. The requirements for storage of fuel assemblies in Region 1 and Region 2 of the storage racks are revised to delete the K-infinity requirement and to provide a single statement as to the storage restrictions based on a combination of initial enrichment, accumulated burnup, and decay times as identified in Figures 3.7.13-1 through 3.7.13-11.

2. The LCO 3.7.13 ACTION is revised to delete the explicit reference to either region.

3. Surveillance SR 3.7.13.1 is revised to:

   a. delete the K-infinity requirement,

   b. add decay time as a parameter,

   c. add reference to new Figures 3.7.13-2 through 3.7.13-11,

   d. clarify that the surveillance is applicable when moving a fuel assembly within the spent fuel pool, and

   e. delete reference to Region 1.

4. Surveillance SR 3.7.13.2 is being deleted, as it has been incorporated into SR 3.7.13.1.

5. Figure 3.7.13-1 which provides the initial nominal enrichment and burnup restrictions for storage in specified locations in Region 1 is revised to incorporate a new figure. There are no technical changes to the figure.

6. Previous Figure 3.7.13-2 which provided restrictions on acceptable storage locations for Region 2 based on initial nominal enrichment and burnup has been replaced by new Figures 3.7.13-2 through 3.7.13-11. The new figures provide restrictions based on initial nominal enrichment, burnup, decay time, and storage cell type.

DESIGN FEATURES 4.3

1. Specification 4.3.1.1(a) is revised to specify the maximum U-235 weight percent of a fuel assembly as a "nominal" value. The current specified value includes the standard manufacturing enrichment tolerance.

2. Specification 4.3.1.1(b) is revised to change the k_{eff} limit to <1.0 when fully flooded with unborated water.

3. A new Specification 4.3.1.1(c) is added which has a requirement for a k_{eff} limit of $\leq$0.95 which allows credit for water borated to $\geq$975 ppm.

4. Specification 4.3.1.1(c) is relabeled as 4.3.1.1(d).

5. The footnote to Specification 4.3.1.1(b) for temporary measures is deleted.

6. Specification 4.3.1.2(a) is revised to specify the maximum U-235 weight percent of a fuel assembly as a "nominal" value. The current specified value includes the standard manufacturing enrichment tolerance.

B. BACKGROUND

1. History

The Ginna Station spent fuel pool (SFP) contains two regions with four rack types. The SFP and the fuel storage rack types and orientation are illustrated in Figure 3.1-1 of Reference 6. In addition to the cask loading and the fuel elevator areas, the pool is divided into Regions 1 and 2. Region 1 consists of five Type 3 storage racks nominally employing borated stainless steel panels in each storage cell as neutron absorbers. The Type 3 racks are designed to store fresh and burned fuel assemblies in a checkerboard array; the fresh assemblies are placed in the cells with the lead-in funnels and the burned assemblies are placed in the alternate cells. In addition, five storage locations adjacent to the cask area, as illustrated in Figure 3.1-1, are available with enlarged internal dimensions for storage of damaged fuel assemblies.

Region 2 consists of the following three types of storage modules for burned fuel assemblies:

1. Six Type 1 modules which were originally fabricated with Boraflex absorber regions within two of the walls of each storage cell,

2. Two Type 2 modules employing borated stainless steel panels and of a similar design as the Region 1, Type 3 storage modules, and

3. Six Type 4 modules employing ten storage cells in a linear array; these cells are of a similar design as those in the Type 2 modules. [Note: Although licensed, these modules are not yet installed.]

In Reference 1, RG&E notified the NRC that testing of boraflex panels contained within Region 2 of the SFP indicated degradation such that certain portions of the criticality safety analysis provided to the NRC may no longer be conservative. This included the ability of Region 2 to maintain a $k_{eff} \leq 0.95$ if flooded with unborated water. The letter described interim compensatory actions taken by RG&E until a permanent solution with respect to boraflex degradation could be engineered. These measures were formally submitted to the NRC (Reference 2) and included a requirement in Improved Technical Specifications (ITS) LCO 3.7.12 to maintain 2300 ppm boron in the SFP with a surveillance frequency of 7 days. By Reference 3 the NRC provided approval of the LCO 3.7.12 changes and the addition of a footnote to the ITS associated with the Design Features Fuel Storage Specification 4.3.1.1(b). The footnote allowed the requirement for $k_{eff} \leq 0.95$ to be met with the SFP borated to $\geq$2300 ppm until December 31, 1999. By Reference 4, RG&E requested a revision to the footnote date, and by Reference 5 the NRC provided approval until June 30, 2001.

The purpose of this license change is to reflect the revision of the criticality safety analysis and the rack utilization schemes for Regions 1 and 2 of the spent fuel racks described in the Ginna Station Updated Final Safety Analysis Report (UFSAR) and the ITS. Due to the observed degradation of the Boraflex panels, the pool criticality safety analysis was reperformed assuming the absence of Boraflex and taking partial credit for the presence of soluble boron in the pool water. This submittal presents the results of the reperformed criticality safety analysis of the Ginna Regions 1 and 2 spent fuel storage racks with partial credit for spent fuel pool soluble boron (from Reference 6). The methodology employed in this analysis for soluble boron credit is analogous to that of Reference 7 and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 8.

The design basis fuel assembly for the fresh fuel storage cells in the Region 1, Type 3 racks was taken to be a conservative representation of the Westinghouse OFA 14 x 14 fuel assembly having a nominal enrichment of 5 wt% U-235, no IFBA loadings, and the instrument tube location replaced by a fuel rod. The design basis fuel assembly for the burned fuel storage cells in both Region 1 and 2 racks was taken to be conservative approximation to the Westinghouse Standard 14 x 14 fuel assembly wherein the RCC guide tubes were represented as zircaloy-4 and the instrument tube was replaced by a fuel rod. This conservative approximation to the burned fuel assembly envelops the characteristics of all burned fuel assemblies, including lead test assemblies, currently stored in the spent fuel pool. The nominal fuel enrichment for the regions is the enrichment of the fuel ordered from the manufacturer. This analysis does not take any credit for the presence of the spent fuel rack Boraflex poison panels.

The Regions 1 and 2 spent fuel rack analysis is based on maintaining $K_{eff} < 1.0$ including uncertainties and tolerances on a 95 percent probability at a 95 percent confidence level (95/95), without the presence of any soluble boron in the storage pool. Soluble boron credit is used to provide safety margin by maintaining $K_{eff} \leq 0.95$ including biases, uncertainties, tolerances, and accident conditions in the presence of spent fuel pool soluble boron. The analyses contained in Reference 6 lead to the conclusion that the total soluble boron concentration required to maintain $K_{eff} \leq 0.95$, after including all biases and uncertainties and assuming the most limiting accident, is 965 ppm (assuming a B-10 atomic fraction equal to .199). The most limiting accident condition was determined to be the misloading of a fresh 5 wt% U-235 (nominal) fuel assembly in the Region 2 Type 1 fuel racks.

This minimum boron concentration is composed of three values:

377 ppm (requirement for $K_{eff} \leq 0.95$),

207 ppm (requirement for reactivity equivalencing methodologies), and

381 ppm (requirement to maintain $K_{eff} \leq 0.95$ for the most limiting accident condition)

Adjusting for a B-10 isotopic fraction equal to 0.197 (to account for potentially recycled boron), the total soluble boron concentration, required to maintain the same concentration of B-10 atoms, would be equal to 975 ppm.

2. Hardware Modifications

There are no hardware related modifications as a result of this proposed change.

C. JUSTIFICATION OF CHANGES

This section provides the justification for all changes described in Section A above and shown on Attachments IV and V. The justifications are organized based on whether the change is: more restrictive (M), less restrictive (L), administrative (A), or the requirement is relocated (R). The justifications listed below are also referenced in the technical specification(s) which are affected (see Attachment IV).

References 6 and 9 (Enclosures 1 and 2) provide the detailed criticality safety analysis and the boron dilution analysis supporting the proposed changes. These analyses are based upon the guidance established in References 7 and 8 and are summarized below to the extent necessary to provide justification for the specific proposed changes to the Improved Technical Specifications (ITS).

C.1 Less Restrictive

L.1 LCO 3.7.13 and DESIGN FEATURES Specification 4.3.1.1(b) are being revised, and DESIGN FEATURES Specification 4.3.1.1(c) added, as the result of the revision of the criticality safety analysis for Regions 1 and 2 of the spent fuel pool (SFP).

LCO 3.7.13 specifies the requirements for storage of fuel assemblies in Regions 1 and 2 of the SFP and is being changed to provide a single statement as to the restrictions based on a combination of initial enrichment, accumulated burnup, and decay times as identified in the new SFP rack utilization Figures 3.7.13-1 through 3.7.13-11. The criticality safety analysis provides the basis for the new figures, which provide additional requirements based on decay time and cell type. The LCO 3.7.13 ACTION is being revised to delete the explicit reference to either region, consistent with the LCO single statement. The surveillances associated with LCO 3.7.13 are also being changed to provide for a single surveillance and add reference to new Figures 3.7.13-1 through 3.7.13-11. The FREQUENCY is also being revised to clarify the requirement that the surveillance be performed any time an assembly is added to, or moved within, the SFP. The requirement for a K-infinity of ≤ 1.458 in Region 1 is no longer required based on the revised criticality safety analysis.

DESIGN FEATURES Specification 4.3.1.1(b) provides the requirements for K_{eff} when the SFP is flooded with unborated water. This requirement is being revised such that the SFP K_{eff} will remain <1.0 (subcritical), at a 95% probability, 95% confidence level, even with no soluble boron in the SFP, as per the guidance of Reference 8. This change is supported by the revised criticality safety analysis. The existing footnote to 4.3.1.1(b) is being deleted based on the new Specification 4.3.1.1(c).

New DESIGN FEATURES Specification 4.3.1.1(c) is being added to provide the 5% subcriticality margin that the NRC has established to meet GDC 62 (Reference 10). The revised criticality safety analysis supports the determination that this requirement will be met with a 95% probability, 95% confidence level, including biases, uncertainties, tolerances, and accident conditions in the presence of SFP soluble boron of ≥ 975 ppm. This amount of soluble boron is significantly less than the amount normally available in the SFP. The margin to criticality is discussed below.

Nuclear Criticality

The methodology employed in the revised criticality safety analysis for soluble boron credit is analogous to that of Reference 7 and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 8.

Soluble boron credit methodology was employed to establish a target K_{eff} value of 0.98051 for the spent fuel pool at zero soluble boron. The allowance for applicable biases, tolerances, and uncertainties was deduced to be 0.01592; thus, the 95/95 upper tolerance limit value of K_{eff} was deduced to be 0.99643.

The final soluble boron requirement is determined from the following summation:

$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm),

$SBC_{95/95}$ = soluble boron requirement for 95/95 $K_{eff} \leq 0.95$ (ppm),

SBC_{RE} = soluble boron required for reactivity equivalencing methodologies (ppm),

SBC_{PA} = soluble boron required for $K_{eff} \leq 0.95$ under accident conditions (ppm).

The total soluble boron requirement for achieving a 95/95 value of $K_{eff} \leq 0.95$ was deduced to be the summation of the following three terms: $SBC_{95/95}$ = 377 ppm, SBC_{RE} = 207 ppm, and SBC_{PA} = 381 ppm for a total of 965 ppm. The soluble boron concentration was increased by 1% due to the difference in the B-10 atom percent used in the analysis (19.9 a/o) and a value typical of recycled boron as previously measured at Ginna (19.7 a/o). This results in a soluble boron concentration equal to 975 ppm. Note that this soluble boron concentration includes an allowance for 5 % burnup uncertainty in the SBC_{RE} term. In addition, all of the burnup values in the burnup versus enrichment storage curves have been increased by 5 %. Therefore, the 5 % burnup uncertainty has been double counted in the determination of the required soluble boron concentration.

The design basis fuel assembly for the fresh fuel storage cells in the Region 1, Type 3 racks was taken to be a conservative representation of the Westinghouse OFA 14 x 14 fuel assembly having a nominal enrichment of 5 wt% U-235, no IFBA loadings, and the instrument tube location replaced by a fuel rod. The design basis fuel assembly for the burned fuel storage cells in both Region 1 and 2 racks was taken to be a conservative approximation of the Westinghouse Standard 14 x 14 fuel assembly wherein the RCC guide tubes were represented as zircaloy-4 and the instrument tube was replaced by a fuel rod. This conservative approximation to the burned fuel assembly envelops the characteristics of all burned fuel assemblies, including lead test assemblies, currently stored in the SFP. This design basis burned fuel assembly was represented by an 8-node axial representation of the assembly burnup and applicable fuel and moderator temperatures.

All representations of the Region 2, Type 1 spent fuel storage racks, originally containing boraflex inserts between the L-shaped insert and the storage cell tube wall, were represented in both the infinite cell array and full storage pool analyses as having nominal pool water in place of the boraflex.

Minimum fuel assembly burnup limits versus fuel assembly initial average enrichment were established for Region 2, Type 2 spent fuel storage cells. These limits were established on both a nominal basis and an equivalent dual tier approach for 0, 5, 10, 15, and 20 years of Pu-241 decay so as to provide more efficient utilization of the available spent fuel storage capacity of the storage racks.

It was demonstrated that the burnup limits established above were applicable to the Region 2, Type 4 storage cells.

It was demonstrated that the existing fuel assembly burnup versus initial enrichment criteria are applicable for Region 1, Type 3 cells. These analyses also demonstrated this objective is easily achieved with fresh fuel enrichments of 5 wt% U-235 (nominal) and no requirements for IFBA credit in the fresh fuel assemblies.

It was further established that a fuel rod consolidation canister is less reactive than a fuel assembly of equivalent burnup when placed in a spent fuel storage cell. Consequently, there are no special restrictions as to placement of fuel rod consolidation canisters in the spent fuel storage cells.

The damaged rod storage basket has been conservatively modeled with fresh 5 wt% U-235 rods and shown to be acceptable in the Region 2, Type 1 cells. Therefore, it can be placed in either Region 1 or Region 2 with no burnup restrictions.

Other items may be stored in the SFP in addition to fresh or discharged fuel assemblies. These items, in general, fall into the category of Non-Special Nuclear Material (SNM). These items are non-multiplying and, in general, are parasitic to the spent fuel rack local reactivity. Some of the items which fall under this category that can be safely stored in the spent fuel pool are: Dummy Canisters containing Non-SNM, Consolidation Hardware, Dummy Fuel Assemblies, Trash Basket containing full length control rods, etc. The general rule for safely storing these types of items is very simple: any non-multiplying and non-fissile item can be safely stored in any cell location. The storing of these components within a water cell does not affect the classification of the cell, i.e., it is still considered a water cell.

Boron Dilution

The SFP is normally maintained with high boron concentrations; consequently, crediting soluble boron to compensate for boraflex degradation will have limited or no impact on direct Ginna Station operation. However, a boron dilution event within the SFP could have different consequences. RG&E has evaluated the SFP configuration and concluded that a boron dilution event which could lead to K_{eff} exceeding 0.95 is not credible.

An analysis of potential scenarios which could dilute the boron concentration in the SFP has been performed. The analysis demonstrates that sufficient time is available to detect and mitigate the dilution prior to exceeding the 0.95 k_{eff} design basis. The potential plant events were quantified to show that sufficient time is available to enable adequate detection and suppression of any dilution event.

The criticality safety analysis demonstrates that a soluble boron concentration of 975 ppm will maintain reactivity within the design basis limit of $k_{eff} \leq 0.95$ (including all biases, tolerances, and uncertainties) with a 95% probability at a 95% confidence level. Deterministic dilution event calculations were performed for Ginna to define the dilution times and volumes necessary to dilute the 213,600 gallon SFP inventory from the minimum 2300 ppm required by ITS LCO 3.7.12 to a soluble boron concentration of 975 ppm. Assuming a well mixed pool, the volume required to dilute the pool from 2300 to 975 ppm was determined to be 183,000 gallons.

The largest possible flow rate for dilution of the pool is 290 gpm due to a through wall crack in the 8-inch fire main. This line does not run above the pool, but it is conservatively assumed that the break flow jet impinges on the pool surface. The fire system draws from the lake, an essentially limitless supply of water. At a flow rate of 290 gpm, over ten hours would be required to dilute the pool from the ITS limit of 2300 ppm to 975 ppm. A rupture of this fire main would result in many alarms which would alert the operators to this potential dilution event within minutes. The expected alarms include auto start of the fire pumps and high SFP level. Ten hours is more than sufficient time for the operators to identify the location of the pipe break and stop the fire pumps or isolate the affected piping.

The only tank large enough to provide the dilution volume without replenishment would be the refueling water storage tank (RWST), which contains water borated to the same concentration as the SFP and thus presents no possibility of dilution. The largest source of unborated water would be the condensate storage tanks, with a combined volume of 160,000 gallons. This is less than the dilution volume of 183,000 gallons and is therefore not capable of diluting the pool to 975 ppm.

Based on the above evaluation, an unplanned or inadvertent event which would reduce the SFP boron concentration from 2300 ppm to 975 ppm is not credible. The large volume of water required for a dilution event, the ITS controls on SFP boron concentration, the plant personnel rounds, and the 7-day sampling interval for the pool concentration would adequately detect a dilution event prior to K_{eff} reaching 0.95 (at 975 ppm).

It should be noted that this boron dilution evaluation was conducted by evaluating the time and water volumes required to dilute the SFP from 2300 ppm to 975 ppm. The 975 ppm end point was utilized to ensure that k_{eff} for the spent fuel racks would remain ≤ 0.95. As part of the Ginna criticality safety analysis, a calculation has been performed on a 95/95 basis to show that the spent fuel rack k_{eff}, including all biases and uncertainties, remains <1.0 with nonborated water in the pool. Thus, even if the SFP were diluted to concentrations approaching zero ppm, the fuel in the racks would remain subcritical and the health and safety of the public would be protected.

C.2 Administrative

A.1 DESIGN FEATURES Specifications 4.3.1.1(a) and 4.3.1.2(a) are being revised to specify the maximum U-235 weight percent of a fuel assembly as a "nominal" value and changing the value from 5.05 weight percent to 5.0 weight percent. The current specified value in 4.3.1.1(a) and 4.3.1.2(a) includes the standard manufacturing tolerance of ±.05 w/o U-235 about the nominal fresh reference enrichment. This is considered an administrative change, in that the actual limit for maximum U-235 enrichment has not changed and is strictly controlled.

There are no relocated (R) or more restrictive (M) changes associated with this LAR.

D. REFERENCES

1. Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Boraflex Degradation", dated March 30, 1998.

2. Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Application for Amendment to Facility Operating License, Revised Spent Fuel Pool Storage Requirements, Revision 1", dated April 27, 1998.

3. Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment No. 72 to Facility Operating License No. DPR-18, R. E. Ginna Nuclear Power Plant", dated July 30, 1998.

4. Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Application for Amendment to Facility Operating License Date Change for Boraflex Degradation Temporary Measures", dated October 20, 1999.

5. Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Issuance of Amendment Regarding a Change from December 31, 1999, to June 30, 2001, Specified in the Technical Specifications (TS) 4.3.1.1.b Note Associated with Maintaining Spent Fuel Pool Boron Concentration >2300 Parts per Million (PPM) at all Times Until a Permanent Resolution to Current Criticality Concerns are Implemented", dated December 21, 1999.

6. ABB Combustion Engineering Nuclear Power, "R.E. Ginna Nuclear Power Plant Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit", Final Report, dated February 2000. (Enclosure 2)

7. Newmyer, W.D., "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", WCAP-14416-NP-A, Revision 1, November 1996.

8. Letter from T.E. Collins, U.S. NRC to T. Greene, WOG, "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC NO. M93254)", October 25, 1996.

9. ABB Combustion Engineering Nuclear Power, "R.E. Ginna Spent Fuel Pool Boron Dilution Analysis", Final Report, dated January 2000. (Enclosure 1)

10. Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 62, "Prevention of Criticality in Fuel Storage and Handling".

Attachment II
R.E. Ginna Nuclear Power Plant

No Significant Hazards Consideration Determination

The proposed changes to the Ginna Station Improved Technical Specifications as identified in Attachment I Section A and justified by Attachment I Section C have been evaluated with respect to 10 CFR 50.92(c) and shown not to involve a significant hazards consideration as described below. This proposed license amendment includes changes which are (1) administrative and (2) provide the criteria for acceptable fuel storage in the spent fuel pool (SFP).

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The administrative change only involves how the maximum initial fuel assembly enrichment is described and has no impact on the probability or consequences of an accident. The remaining change is evaluated below.

   The regions of the SFP and specific storage cell types differ from each other in regards to the specific absorber material within the cells. Administrative controls are used to maintain the specified storage patterns and to assure storage of a fuel assembly in a proper location based on initial U-235 enrichment, burnup, and decay time. Procedures which perform this surveillance will include independent verification provisions.

   There is no significant increase in the probability of an accident concerning the potential insertion of a fuel assembly in an incorrect location in the storage racks. Ginna currently uses administrative controls to move fuel assemblies from location to location within the SFP. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Improved Technical Specification spent fuel rack storage configuration limitations. Fuel movement procedures are planned to include independent verification of fuel handling steps.

   There is no increase in the consequences of the accidental misloading of spent fuel assemblies into the spent fuel pool racks. The criticality safety analysis demonstrate that the pool K_{eff} will remain ≤ 0.95 following an accidental misloading due to the boron concentration of the pool. The existing Improved Technical Specification limitation on soluble boron within the SFP will ensure that an adequate boron concentration is maintained

   Based on the above, it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The administrative change to the Improved Technical Specifications has no impact on plant hardware or operations and therefore cannot create a new or different kind of an accident.

Criticality accidents in the SFP are not new or different types of accidents, they have been analyzed in the Updated Final Safety Analysis Report and in criticality safety analysis reports associated with specific licensing amendments for fuel enrichments up to the nominal 5.0 weight percent U-235 that is assumed for the proposed change.

The current Improved Technical Specifications contain limitations on the minimum SFP boron concentration. The proposed changes to the Improved Technical Specifications to allow credit for soluble boron for a $K_{eff} \leq 0.95$ in the SFP is consistent with the results of the new criticality safety analysis. Since soluble boron has always been maintained in the SFP water, and is currently required by Improved Technical Specifications, the implementation of this new requirement will have no effect on normal SFP operations and maintenance. A dilution of the spent fuel pool soluble boron has always been a possibility, however, it has been shown in the SFP boron dilution analysis that there are no credible dilution events for which the spent fuel pool K_{eff} could increase to >0.95. Therefore, the implementation of crediting soluble boron in the SFP will not result in the possibility of a new kind of accident.

The proposed changes to Improved Technical Specifications LCO 3.7.13 continue to specify the requirements for the spent fuel rack storage configurations. Since the proposed SFP storage configuration limitations will be similar to the current ones, the new limitations will not have any significant effect on normal spent fuel pool operations and maintenance and will not create any possibility of a new or different kind of accident. Verifications will be performed to ensure that the spent fuel pool loading configuration meets specified requirements.

The misloading of a fuel assembly in the required storage configuration has been evaluated. In all cases, the rack K_{eff} remains ≤ 0.95.

Under the proposed amendment, no changes are being made to the racks themselves, any other systems, or to the physical structures of the Auxiliary Building itself. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed administrative change to the Improved Technical Specifications has no impact on any acceptance criteria, plant operations or the actual failure of any systems, components or structure; therefore the change has no impact on the margin of safety.

The spent fuel storage operation limits will provide adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality safety analysis performed in a manner analogous to that of the NRC approved Westinghouse spent fuel rack criticality safety analysis methodology.

While the criticality safety analysis utilized credit for soluble boron, storage configurations have been defined using 95/95 K_{eff} calculations to ensure that the spent fuel rack K_{eff} will be <1.0 with no soluble boron. Soluble boron credit is used to offset uncertainties, tolerances, and off-normal conditions (such as a misplaced assembly) and to provide subcritical margin such that the spent fuel pool K_{eff} is maintained at ≤ 0.95.

The loss of substantial amounts of soluble boron from the spent fuel pool which could lead to K_{eff} exceeding 0.95 has been evaluated and shown to be not credible. An evaluation has been performed which shows that dilution of the SFP boron concentration from 2300 ppm to 975 ppm is not credible. Also, the spent fuel rack K_{eff} will remain <1.0 (with a 95/95 confidence level) with the SFP flooded with unborated water. These analyses demonstrate a level of safety comparable to the conservative criticality safety analysis methodology required by Westinghouse WCAP-14416. Therefore, these changes do not involve a significant reduction in a margin of safety.

Based upon the above information, it has been determined that the proposed changes to the Ginna Station Improved Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of a new or different kind of accident from any accident previously evaluated, and do not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed changes meets the requirements of 10 CFR 50.92(c) and do not involve a significant hazards consideration.

Attachment III
R.E. Ginna Nuclear Power Plant

Environmental Impact Consideration Determination

RG&E has evaluated the proposed changes and determined that:

1. The amendment involves no significant hazards consideration as documented in Attachment II;

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since no specifications related to offsite releases are affected; and

3. There is no significant increase in individual or cumulative occupational radiation exposure since no new or different type of equipment are required to be installed as a result of this LAR.

Accordingly, the proposed changes meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed changes is not required.

Attachment IV
R.E. Ginna Nuclear Power Plant

Marked up Copy of Improved Technical Specifications

Included pages:

3.7-29
3.7-30
3.7-31
3.7-31a
3.7-31b
3.7-31c
3.7-31d
3.7-31e
3.7-31f
3.7-31g
3.7-31h
3.7-31i
4.0-2

B 3.7-86*
B 3.7-87*
B 3.7-88*
B 3.7-89*
B 3.7-90*
B 3.7-90a*
B 3.7-91*
B 3.7-92*
B 3.7-93*
B 3.7-94*
B 3.7-95*
B 3.7-96*

* The bases changes are being provided only for information to show the changes RG&E intends to make following NRC approval of this LAR. The bases are under RG&E control for all changes in accordance with Specification 5.5.13. RG&E requests that the NRC document the acceptance of the changes to these bases in the SER.

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool (SFP) Storage

LCO 3.7.13 Fuel assembly storage in the spent fuel pool shall be maintained as follows:

a. Fuel assemblies in Region 1 shall have a K-infinity of $\leq$ 1.458 and shall have initial enrichment and burnup within the acceptable area of Figure 3.7.13-1; and

b. Fuel assemblies in Region 2 shall have initial enrichment and burnup within the acceptable area of the Figure 3.7.13-2.

L.1

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met for either region.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

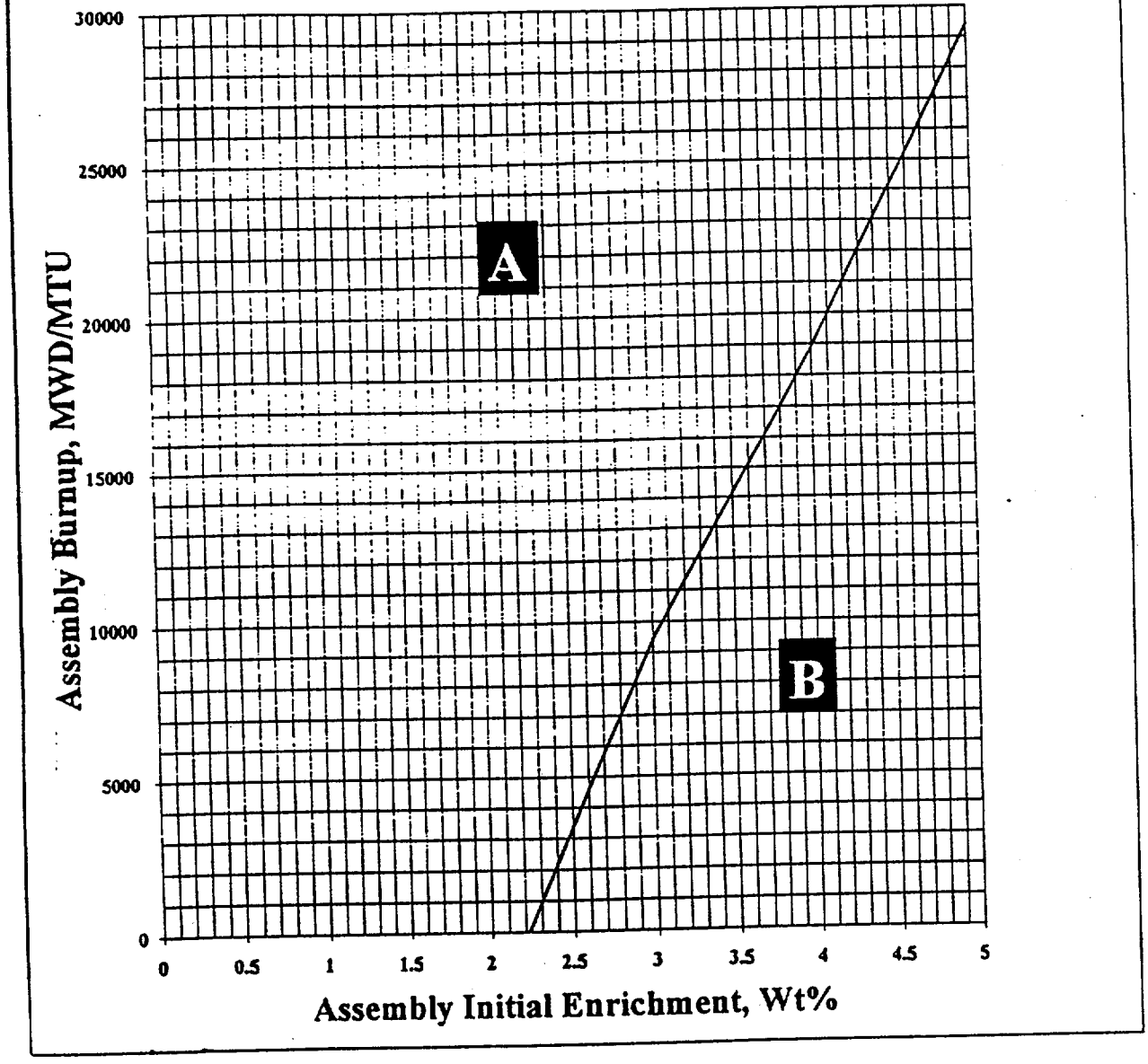
L.1

The combination of initial enrichment and burnup values, with appropriate decay times, of each fuel assembly stored in the spent fuel pool shall be within the acceptable burnup domain of the applicable Figures 3.7.13-1 through 3.7.13-11, based on region and cell type.

SURVEILLANCE REQUIREMENTS

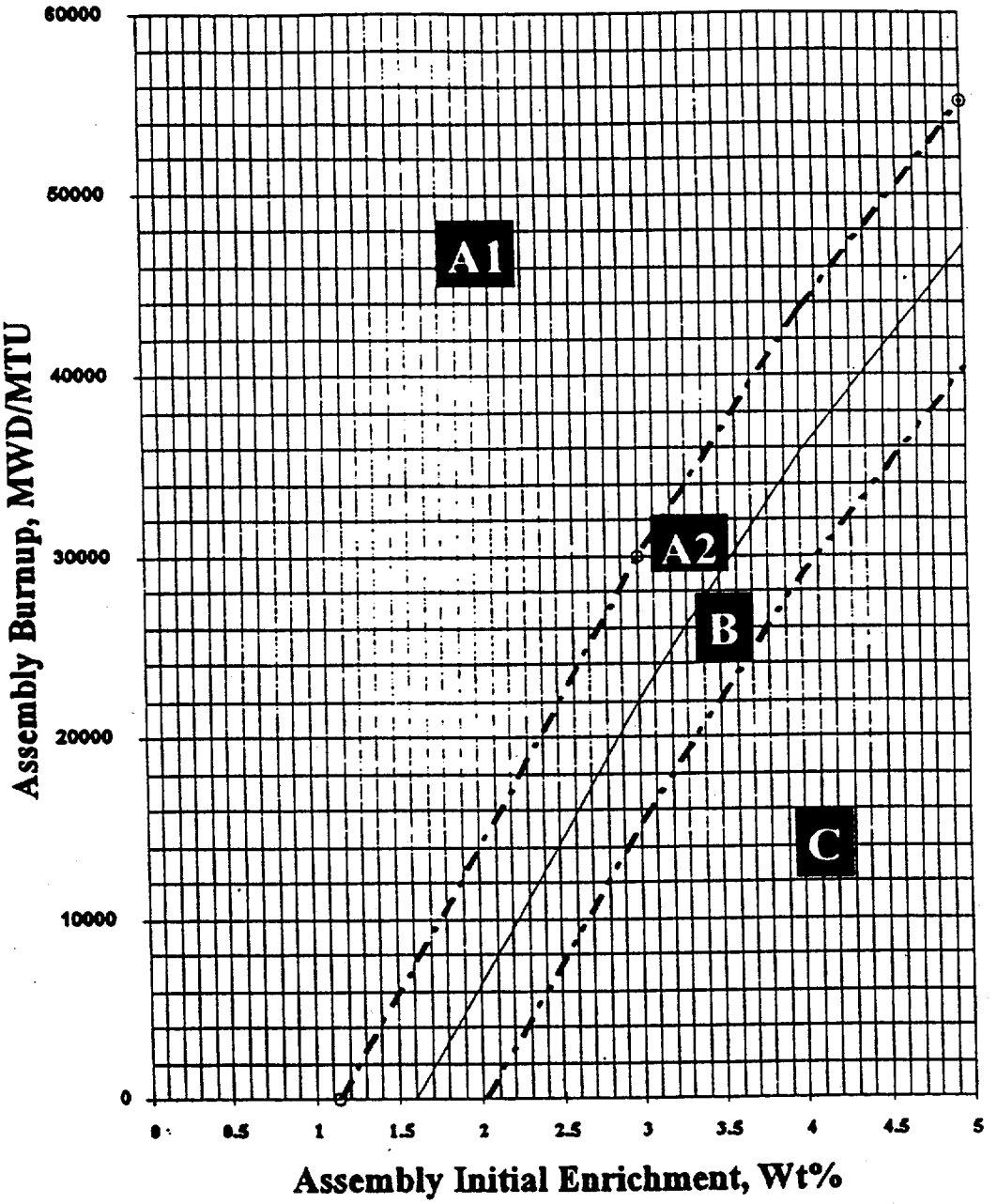
SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify by administrative means the K-infinity of the fuel assembly is ≤ 1.458 and that the initial enrichment and burnup and decay time of the fuel assembly is in accordance with the applicable Figure 3.7.13-1 through 3.7.13-11.	Prior to storing or moving the fuel assembly in Region 1 the spent fuel pool
SR 3.7.13.2	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-2.	Prior to storing the fuel assembly in Region 2

L.1



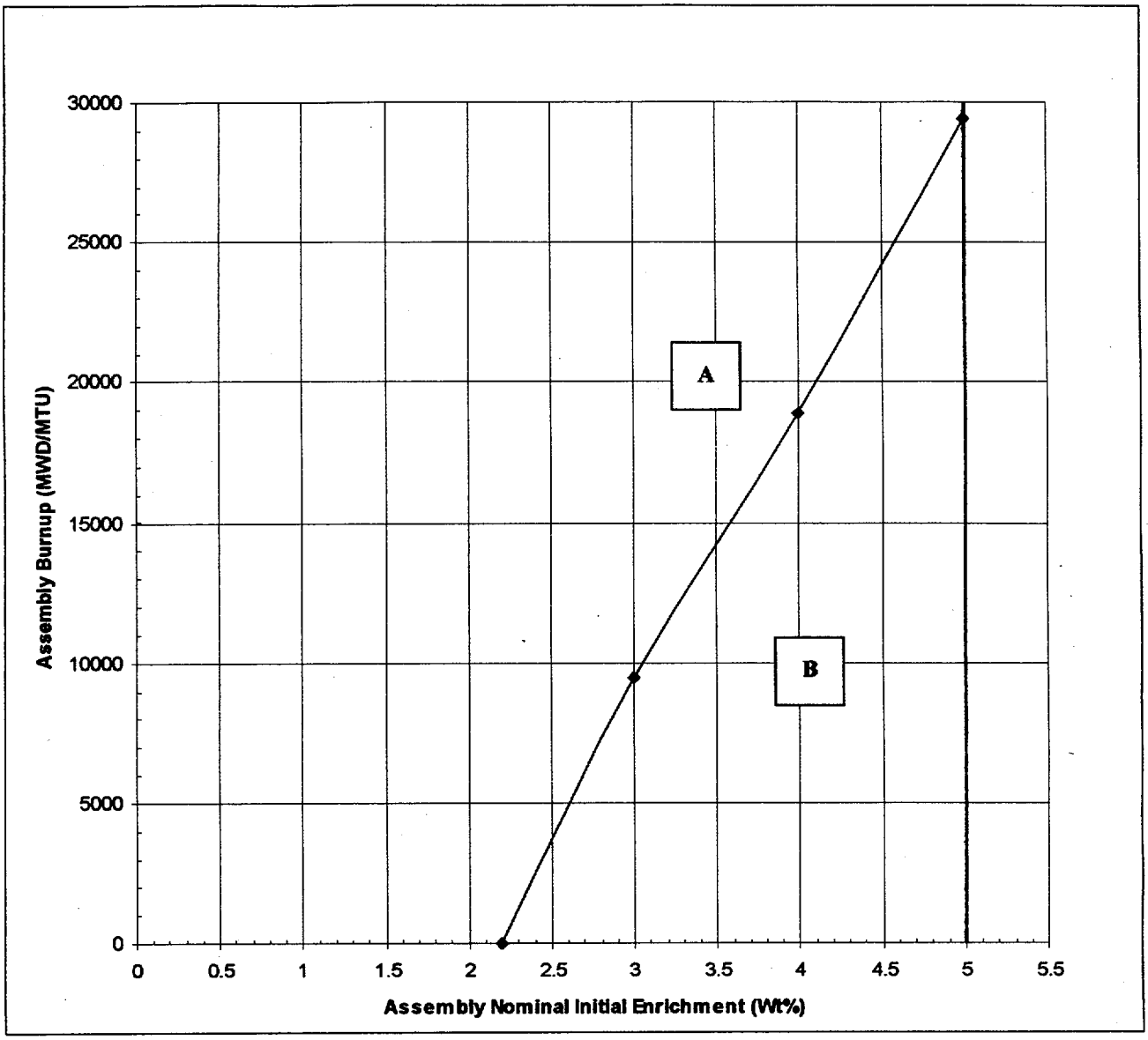
A - Acceptable burnup domain for storage in any location within Region 1.
B - Acceptable burnup domain for storage in cells with lead-in funnels only.

Figure 3.7.13-1
Fuel Assembly Burnup Limits in Region 1



A1 - Acceptable burnup domain for storage in any location within Region 2.
A2 - Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell.
B - Acceptable burnup domain for storage face-adjacent to a Type A1 assembly or a water cell.
C - Acceptable burnup domain for storage face-adjacent to a water cell only.

Figure 3.7.13-2
Fuel Assembly Burnup Limits in Region 2

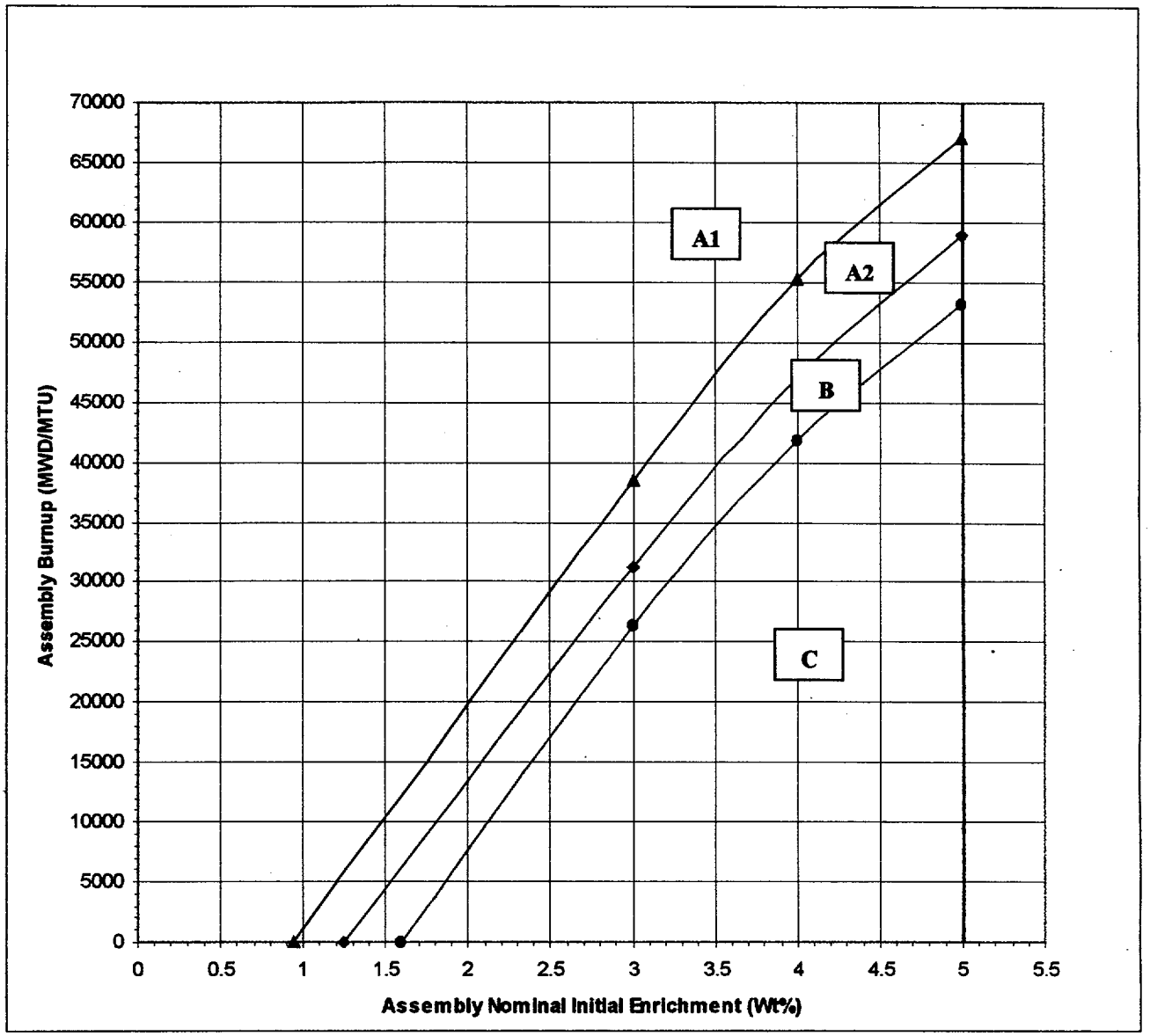


A Acceptable burnup domain for storage in any location within Region 1

B Acceptable burnup domain for storage in cells with lead-in funnels only

Figure 3.7.13-1

Burnup Vs Enrichment Curve for Region 1 Type 3 Cells
(Not Pu-241 Decay Dependent)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

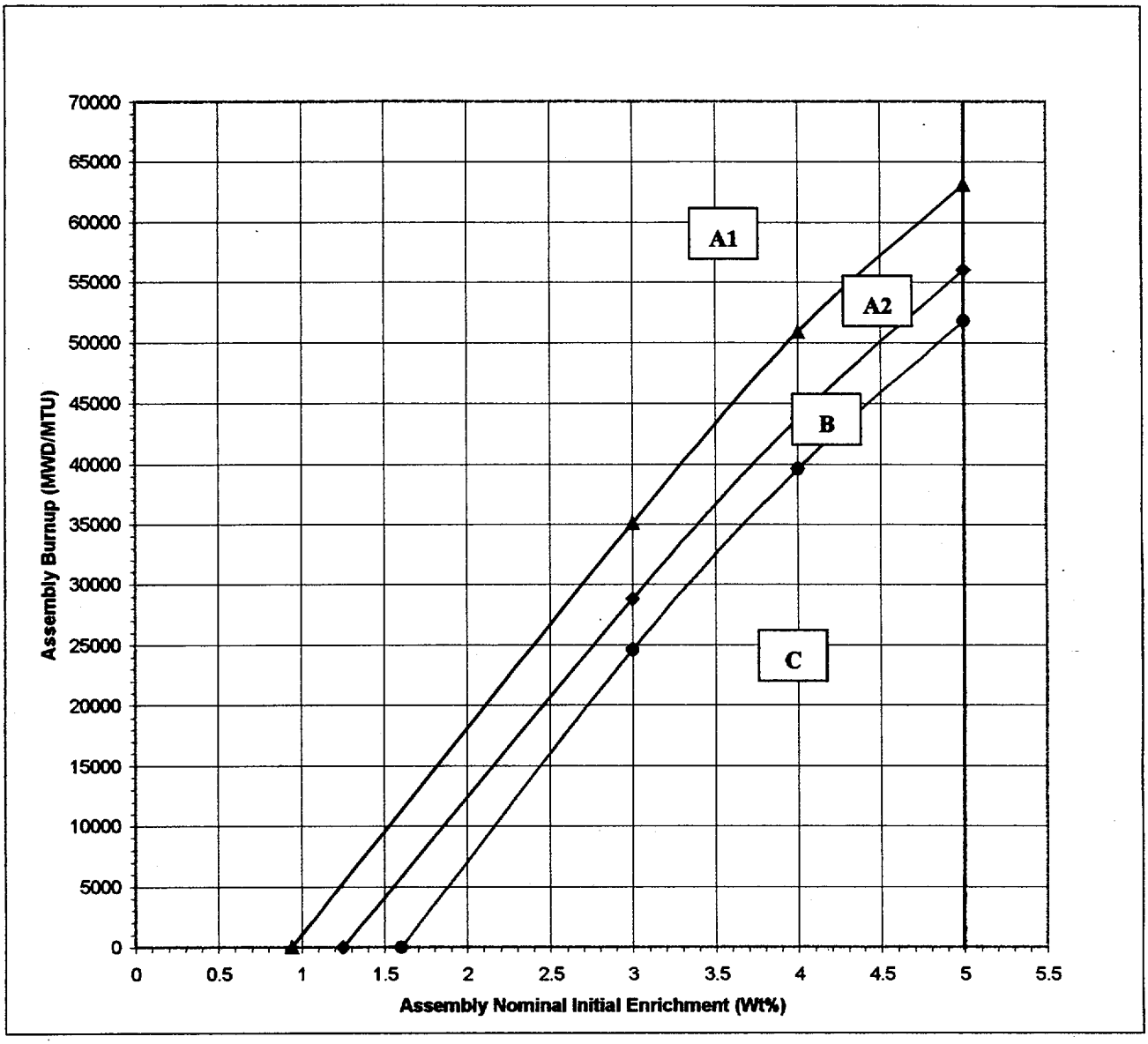
B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-2

Burnup Vs Enrichment Curves for Region 2 Type 1 Cells

(No Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

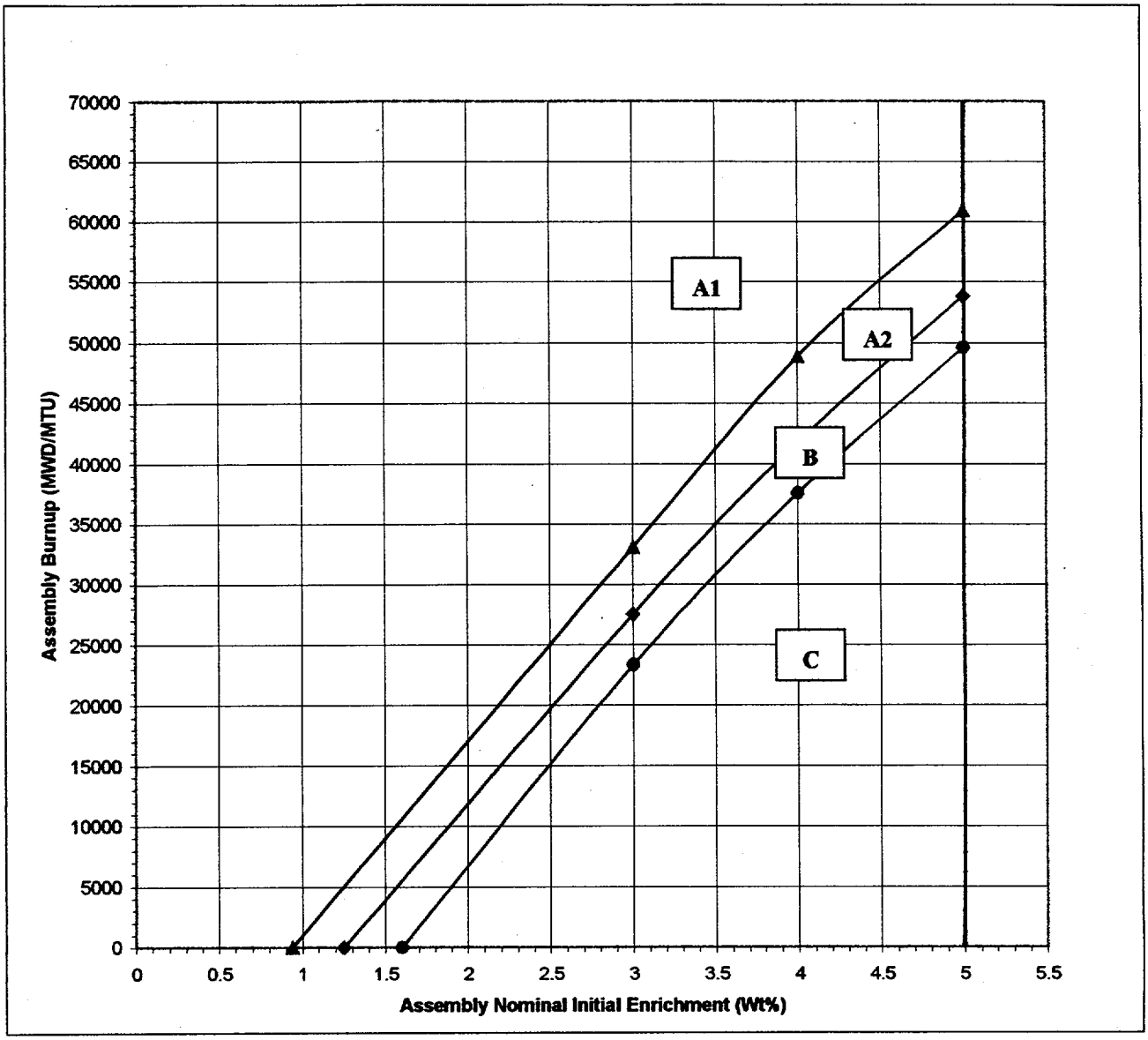
C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-3

Burnup Vs Enrichment Curves for Region 2 Type 1 Cells

(5-Year Pu-241 Decay)

3.7-31a



A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

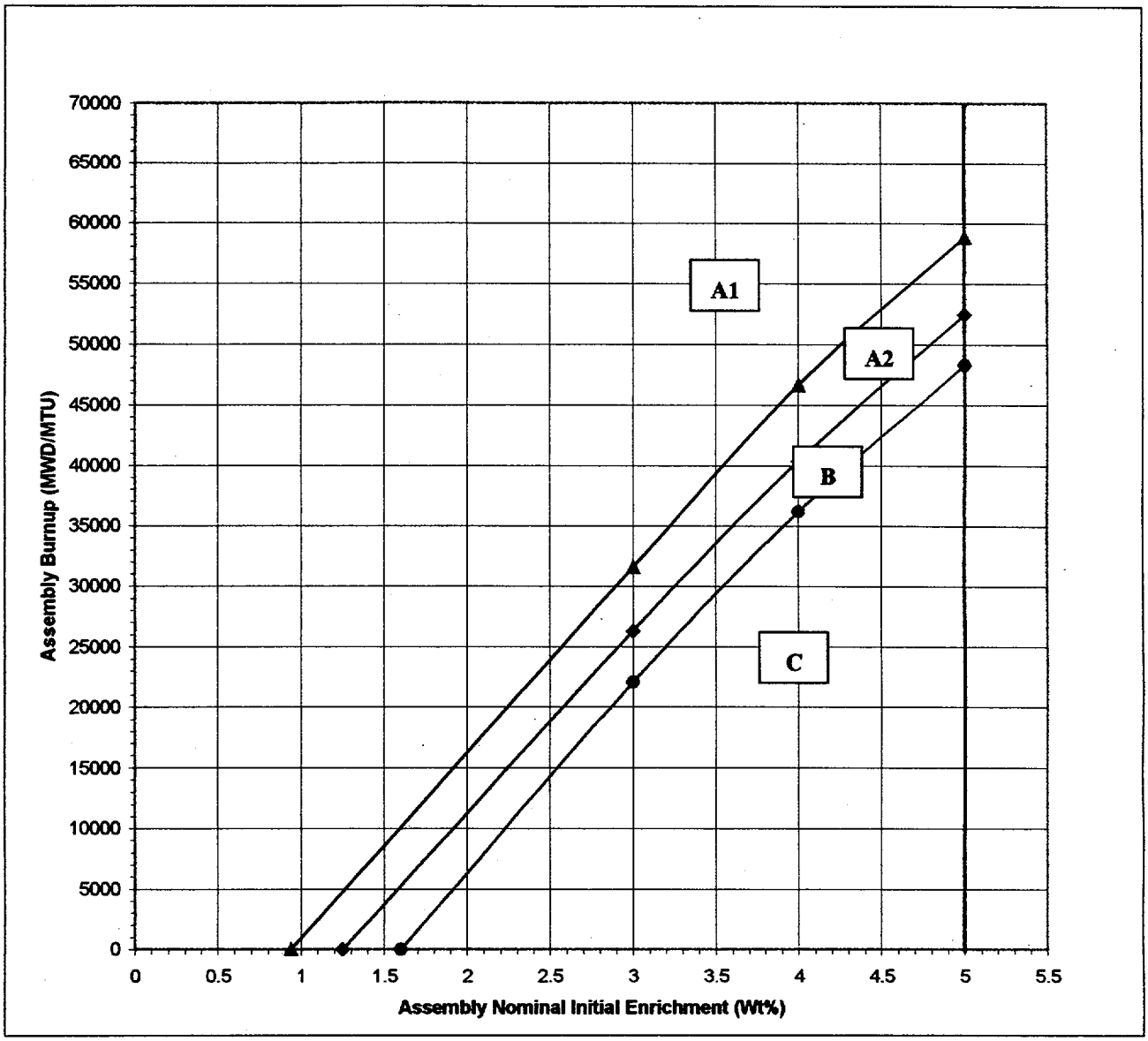
A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-4

Burnup Vs Enrichment Curves for Region 2 Type 1 Cells
(10-Year Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

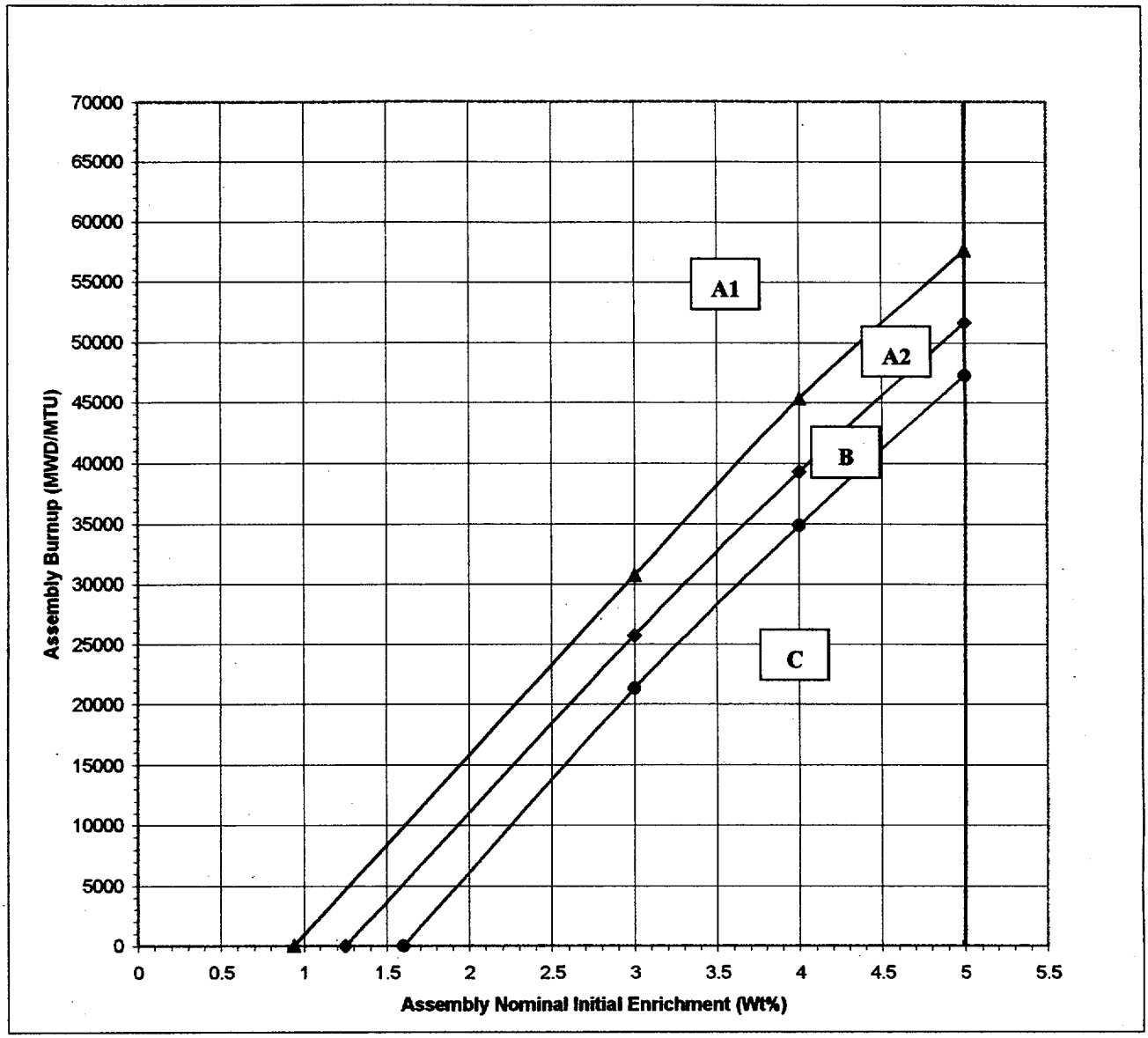
A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-5

Burnup Vs Enrichment Curves for Region 2 Type 1 Cells
(15-Year Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

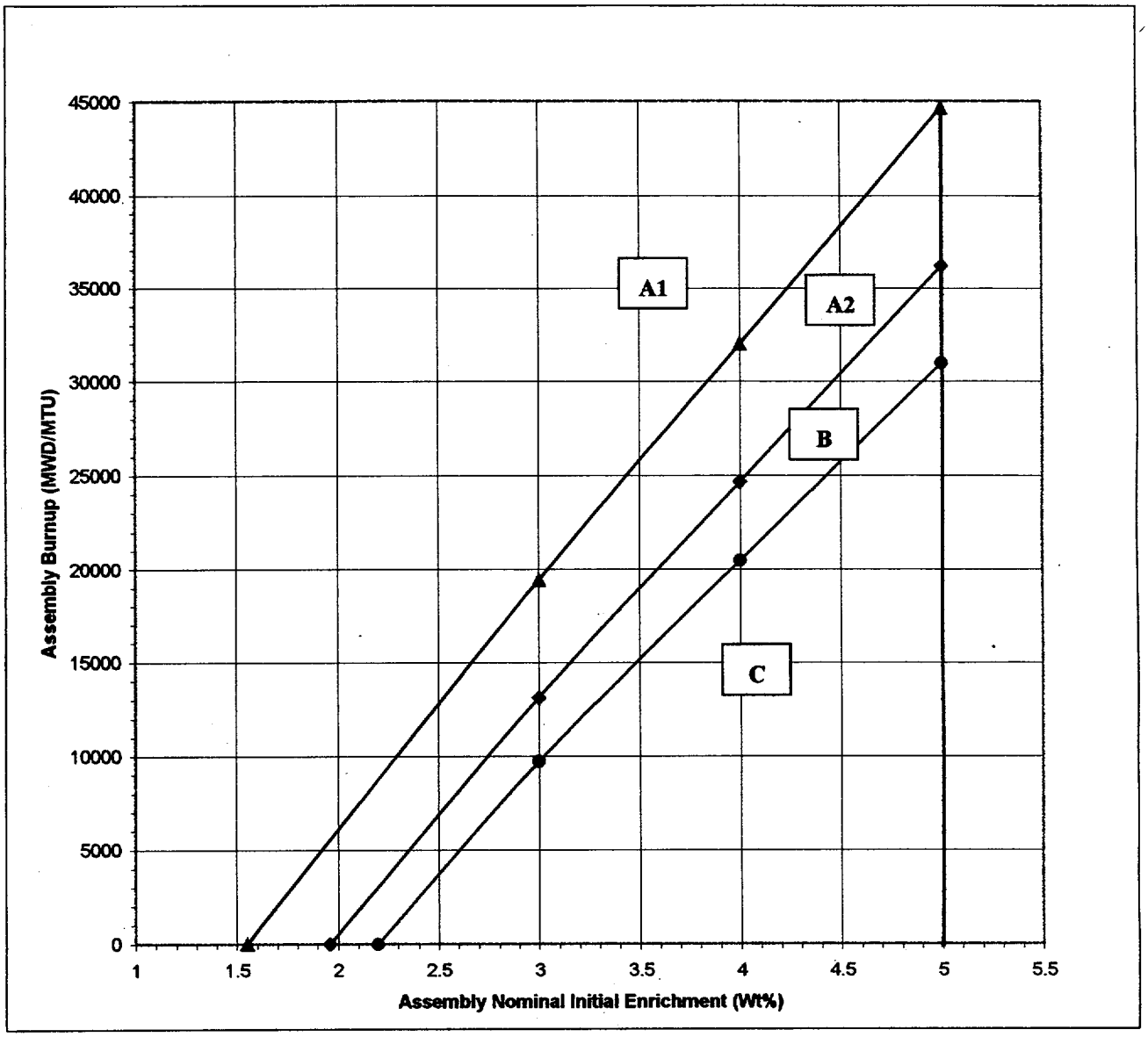
B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-6

Burnup Vs Enrichment Curves for Region 2 Type 1 Cells

(20-Year Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

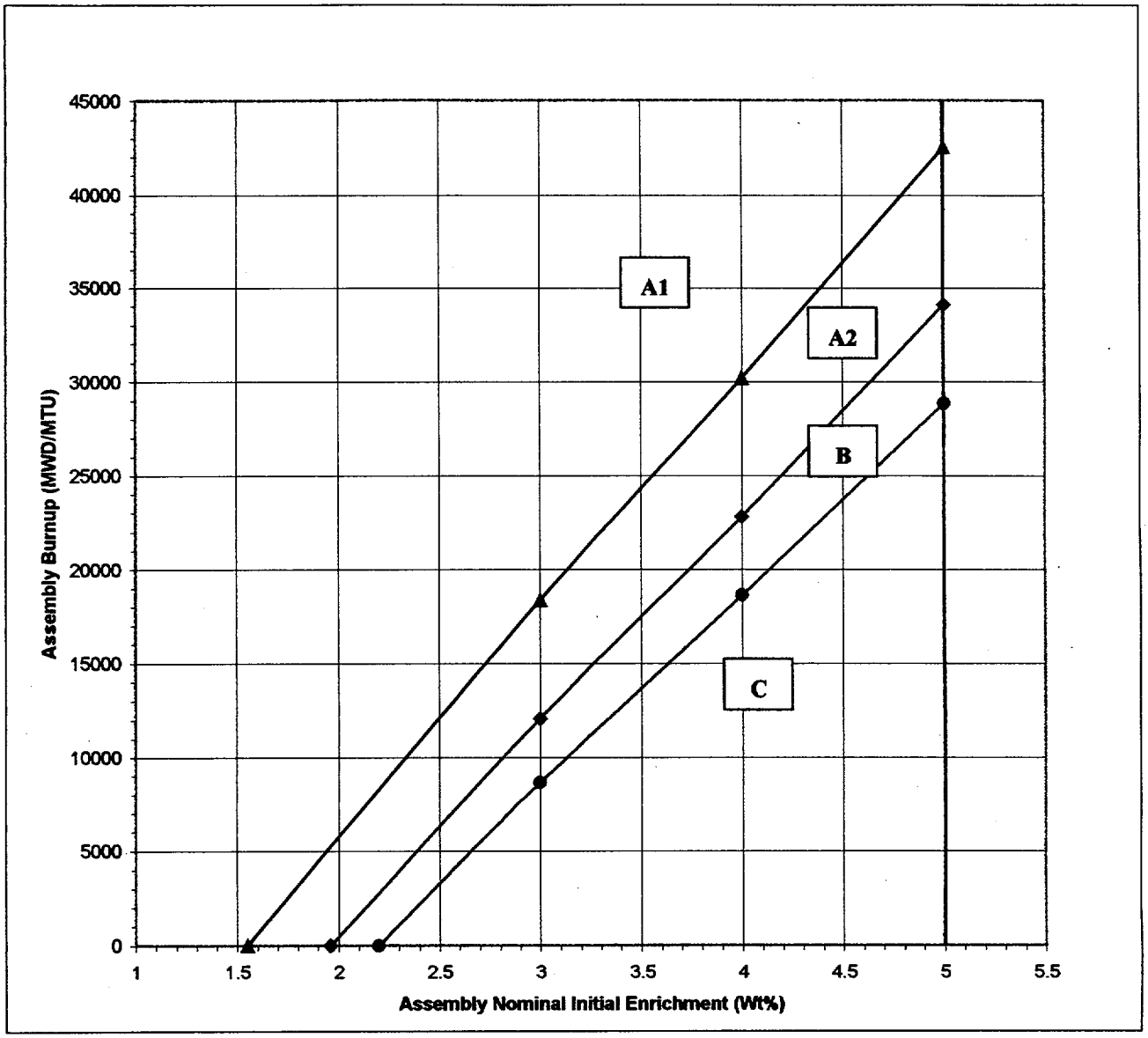
B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-7

Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells

(No Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

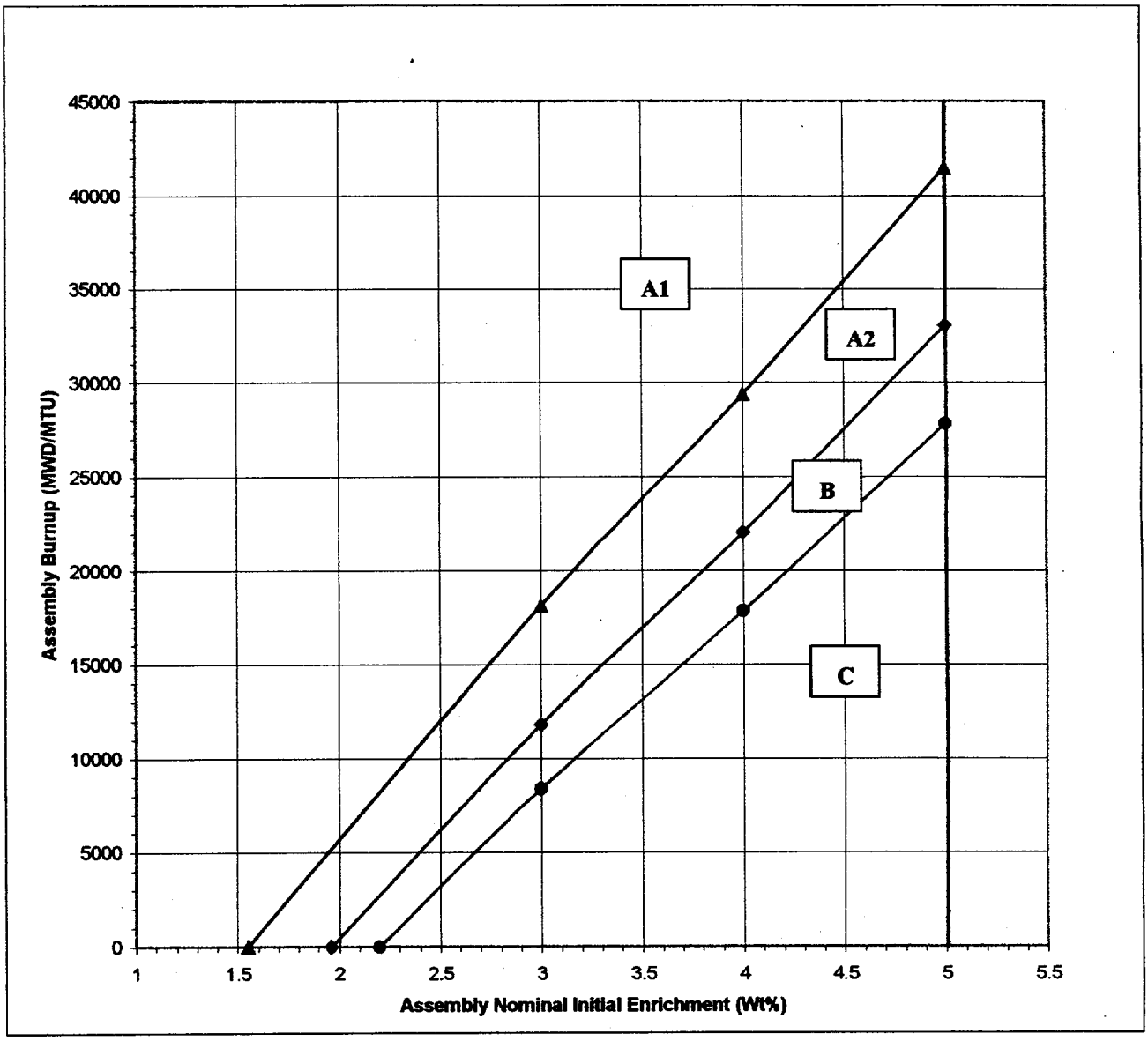
B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-8

Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells

(5-Year Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

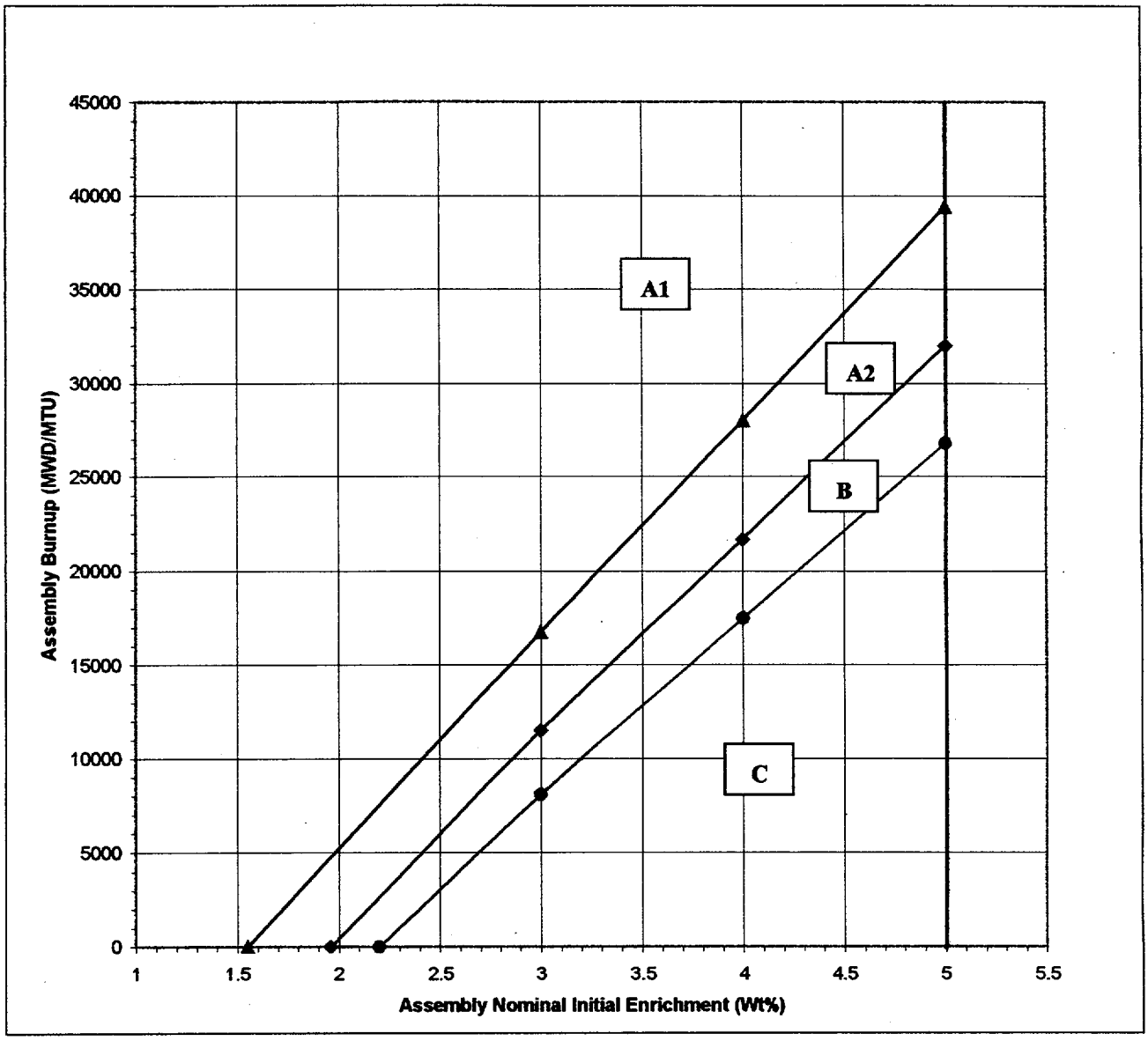
B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-9

Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells

(10-Year Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

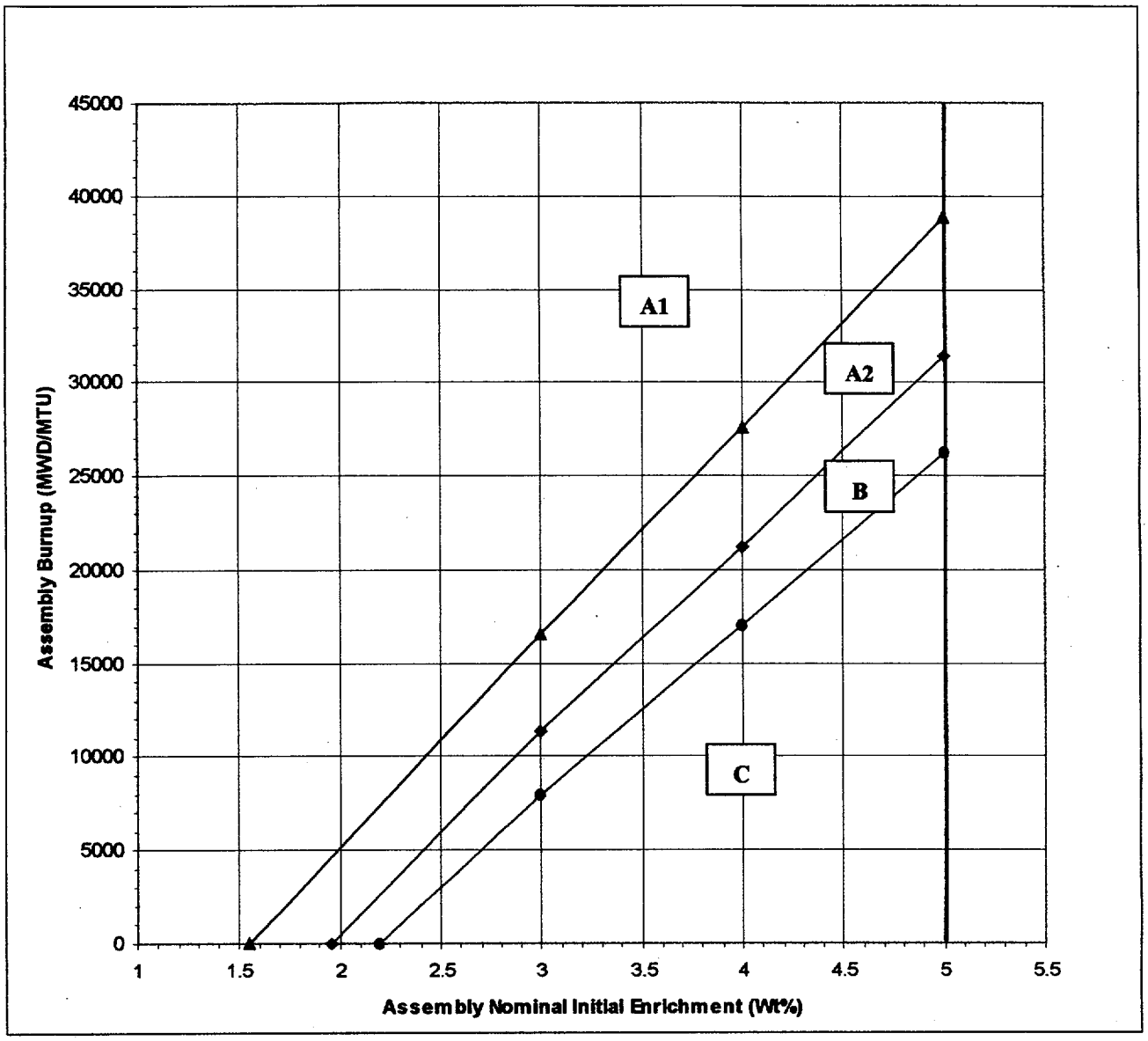
B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-10

Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells

(15-Year Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-11
Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells
(20-Year Pu-241 Decay)

4.0 DESIGN FEATURES

4.2 Reactor Core (continued)

4.2.2 Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.05 weight percent;

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

c. Consolidated rod storage canisters may be stored in the spent fuel storage racks provided that the fuel assemblies from which the rods were removed meet all the requirements of LCO 3.7.13 for the region in which the canister is to be stored. The average decay heat of the fuel assembly from which the rods were removed for all consolidated fuel assemblies must also be ≤ 2150 BTU/hr.

c. $K_{eff} \leq 0.95$ if fully flooded with water borated to ≥ 975 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

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

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.05 weight percent;

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

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

* Until June 30, 2001, the spent fuel storage racks shall be maintained with a k_{eff} ≤ 0.95 when flooded with water containing ≥ 2300 ppm soluble boron.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool (SFP) Boron Concentration

BASES

BACKGROUND

The water in the spent fuel pool (SFP) normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both SFP regions is based on the use of unborated water such that the SFP design and configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded. The Region 2 SFP design uses boraflex material that is secured to the rack structure. Testing has demonstrated that boraflex degradation is occurring such that boron must be credited to maintain $k_{eff} \leq 0.95$ at all times until a long-term solution is reached (Reference 5).

Insert 1

The double contingency principle discussed in ANSI N-16.1-1975 (Ref. 1) and Reference 2 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.13, "Spent Fuel Pool (SFP) Storage" and by maintaining the minimum boron concentration required to address boraflex degradation. Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or 3.7.13.2 as applicable.

Insert 2

(continued)

BASES (continued)

1, 2, 5, and 6

APPLICABLE SAFETY ANALYSES

The postulated accidents in the SFP can be divided into two three basic categories (Ref. 3 and 4). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by a combination of voiding (which would result in the addition of negative reactivity), the SFP geometry, and the high boron concentration in the SFP. The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time.

The SFP criticality safety analysis encompasses a temperature range of 50-212°F.

Insert 3

The concentration of dissolved boron in the SFP satisfies Criterion 2 of the NRC Policy Statement.

LCO

Insert 4

The SFP boron concentration is required to be ≥ 2300 ppm. With the assumed boraflex available per the original design of the SFP storage racks, this value is 450 ppm. The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 3 and 4 (i.e., a fuel handling accident) and the boraflex degradation issue described in Reference 5. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage.

(continued)

BASES (continued)

APPLICABILITY This LCO applies whenever fuel assemblies are stored in the SFP to compensate for boraflex degradation to ensure the SFP k_{eff} remains $\leq$ 0.95 at all times. This is expected to be corrected by December 31, 1999 per Specification 4.3.1.1.b.

ACTIONS A.1 and A.2

When the concentration of boron in the SFP is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. The initiation of actions to restore concentration of boron is simultaneous with suspending movement of fuel assemblies. This is necessary to compensate for any boraflex degradation within the SFP racks.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

This SR verifies that the concentration of boron in the SFP is within the limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate since the boron is credited with maintaining the SFP subcritical due to boraflex degradation. Also, the volume and boron concentration in the pool is normally stable and all water level changes and boron concentration changes are controlled by plant procedures.

This SR is required to be performed prior to fuel assembly movement into Region 1 or Region 2 and must continue to be performed until the necessary SFP verification is accomplished (i.e., SR 3.7.13.1 and 3.7.13.2).

REFERENCES

1. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

2. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

(1.) (3.) Framatome Technologies, Inc., "R.E. Ginna Nuclear Power Plant, Spent Fuel Pool Re-racking Licensing Report," Section 4, February 1997.

(2.) (4.) UFSAR, Section 15.7.3.

5. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, Subject: "Boraflex Degradation," dated March 30, 1998.

Insert 5 →

B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool (SFP) Storage

BASES

BACKGROUND

The spent fuel pool (SFP) is divided into two separate and distinct regions (see Figure B 3.7.13-1) which, for the purpose of criticality considerations, are considered as separate pools (Ref. 1). Region 1, with 294 storage positions, is designed to accommodate new or spent fuel utilizing a checkerboard arrangement. All fuel assemblies stored in Region 1 must have a k-infinity that is ≤ 1.458. The existing design uses Integral Fuel Burnable Absorbers (IFBAs) as the poison for fuel assemblies with enrichments > 4.05 wt% to help achieve this k-infinity limit. IFBAs consist of neutron absorbing material which provides equivalencing reactivity holddown (i.e., neutron poison) that allows storage of higher enrichment fuel. The neutron absorbing material is a non-removable or integral part of the fuel assembly once it is applied. The infinite multiplication factor, K-infinity, is a reference criticality point of each fuel assembly that if maintained ≤ 1.458, will result in a k_{eff} ≤ 0.95 for Region 1. The K-infinity limit is derived for constant conditions of normal reactor core configuration (i.e., typical geometry of fuel assemblies in vertical position arranged in an infinite array) at cold conditions (i.e., 68°F and 14.7 psia). Fuel assemblies with minimum burnups above the curve in Figure 3.7.13-1 (area A) may be stored at any location within Region 1. Fuel assemblies with minimum burnups below the curve in Figure 3.7.13-1 (area B) may be stored in cells with lead-in funnels only.

This will result in a Keff ≤ .95 (with soluble boron credit) for Region 1.

New fuel assemblies are limited to a maximum nominal initial enrichment of 5.0 wt% U-235. The nominal enrichment does not include the standard manufacturing tolerance of ±.05 wt% about the nominal fresh reference enrichment. The tolerance is taken into account in the criticality analysis.

(continued)

BASES (continued)

BACKGROUND (continued)

Region 2, with 1075 storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domains according to Figure 3.7.13-2, in the accompanying LCO. Fuel assemblies with initial enrichments and burnups within domain A1 of Figure 3.7.13-2 may be stored in any location in Region 2. Fuel assemblies with initial enrichments and burnups within domain A2 of Figure 3.7.13-2 shall be stored face-adjacent to a Type A1 of A2 assembly, or a water cell (empty cell). Fuel assemblies with initial enrichments and burnups within domain B of Figure 3.7.13-2 shall be stored face-adjacent to a Type A1 assembly or a water cell (empty cell). Fuel assemblies with initial enrichments and burnups within domain C of Figure 3.7.13-2 shall be stored face-adjacent to a water cell (empty cell) only. The word "face-adjacent" on Figure 3.7.13-2 is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the fuel assembly in the next cell. The storage of fuel assemblies which are within the acceptable ranges of Figure 3.7.13-2 in Region 2 ensures a $K_{eff} \leq 0.95$ in this region.

Insert 6

Consolidated rod storage canisters can also be stored in either region in the SFP provided that the minimum burnup of Figure 3.7.13-1 and Figure 3.7.13-2 through Figure 3.7.13-11 are met (Ref. 2). The canisters are stainless steel containers which contain the fuel rods of a maximum of two fuel assemblies (i.e., 358 rods). All bowed, broken, or otherwise failed fuel rods are first stored in a stainless steel tube of 0.75 inch outer diameter before being placed in a canister. Each canister will accommodate 110 failed fuel rod tubes.

In addition, failed fuel rods may be stored in the consolidated rod storage canisters.

Insert 7

(continued)

BASES (continued)

BACKGROUND (continued)

Insert 8

The water in the SFP normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water such that the SFP design and configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded. The SFP design uses boraflex material that is secured to the rack structure. Testing has demonstrated that boraflex degradation is ocurring such that boron must be temporarily credited to maintain $k_{eff} \leq 0.95$ at all times until a long-term solution is reached (Reference 6).

Insert 9

The double contingency principle discussed in ANSI N16.1-1975 (Ref. 3) and Reference 4 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with this LCO and by maintaining the minimum boron concentration required to address boraflex degradation per LCO 3.7.12. Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or 3.7.13.2 as applicable.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Refs. 2 and 5). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by a combination of voiding (which would result in the addition of negative reactivity), the SFP geometry, and the high boron concentration in the SFP. The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly), and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time.

The configuration of fuel assemblies in the spent fuel pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the SFP ensure the k_{eff} of the SFP will always remain < 0.95. For fuel assemblies stored in Region 1, each assembly must have a K-infinity of ≤ 1.458 with initial enrichment and burnup within the acceptable area of Figure 3.7.13-1. For fuel assemblies stored in Region 2, initial enrichment and burnup shall be within the acceptable area of Figure 3.7.13-2. The word "face-adjacent" on Figure 3.7.13-2 is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the assembly in the next cell.

The x-axis of both figures is the nominal U-235 enrichment wt% which does not include the ± 0.05 wt% tolerance that is allowed for fuel manufacturing and listed in Specification 4.3.1.1.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the SFP.

ACTIONS

A.1

When the configuration of fuel assemblies stored in either Region 1 or Region 2 of the SFP is not within the LCO limits, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Specification 4.3.1.1. This compliance can be made by relocating the fuel assembly to a different region or to an acceptable new location within the same region.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Insert 12

This SR verifies by administrative means that the K-infinity of each fuel assembly is $\leq$ 1.458 prior to storage in Region 1 and that the initial enrichment and burnup is in accordance with Figure 3.7.13-1. If the initial enrichment of a fuel assembly is $\leq$ 4.05 wt%, a K-infinity of $\leq$ 1.458 is always maintained. For fuel assemblies with enrichment > 4.05 wt%, a minimum number of IFBAs must be present in each fuel assembly such that k-infinity $\leq$ 1.458 prior to storage in Region 1. This verification is only required once for each fuel assembly since the burnable poisons, if required, are an integral part of the fuel assembly and will not be removed. The initial enrichment of each assembly will also not change (i.e., increase) while partially burned assemblies are less reactive than when they were new (i.e., fresh). Performance of this SR ensures compliance with Specification 4.3.1.1.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1 (continued)

Though not required for this LCO, this SR must also be performed after completion of fuel movement within Region 1 to exit the Applicability of LCO 3.7.12, "SFP Boron Concentration."

SR 3.7.13.2

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-2 in the accompanying LCO prior to storage in Region 2. Once a fuel assembly has been verified to be within the acceptable range of Figure 3.7.13-1, further verifications are no longer required since the initial enrichment or burnup will not adversely change. For fuel assemblies in the unacceptable range of Figure 3.7.13-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

Though not required for this LCO, this SR must also be performed after completion of fuel movement within Region 2 to exit the Applicability of LCO 3.7.12.

REFERENCES

1. UFSAR, Section 9.1.2.
2. Framatome Technologies, Inc., "R.E. Ginna Nuclear Power Plant, Spent Fuel Pool Re-racking Licensing Report," Section 4, February 1997.
3. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

(continued)

BASES (continued)

REFERENCES (continued)

4. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

3. 5. UFSAR, Section 15.7.3.

6. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, Subject: "Boraflex Degradation," dated March 30, 1998.

Insert 13 →

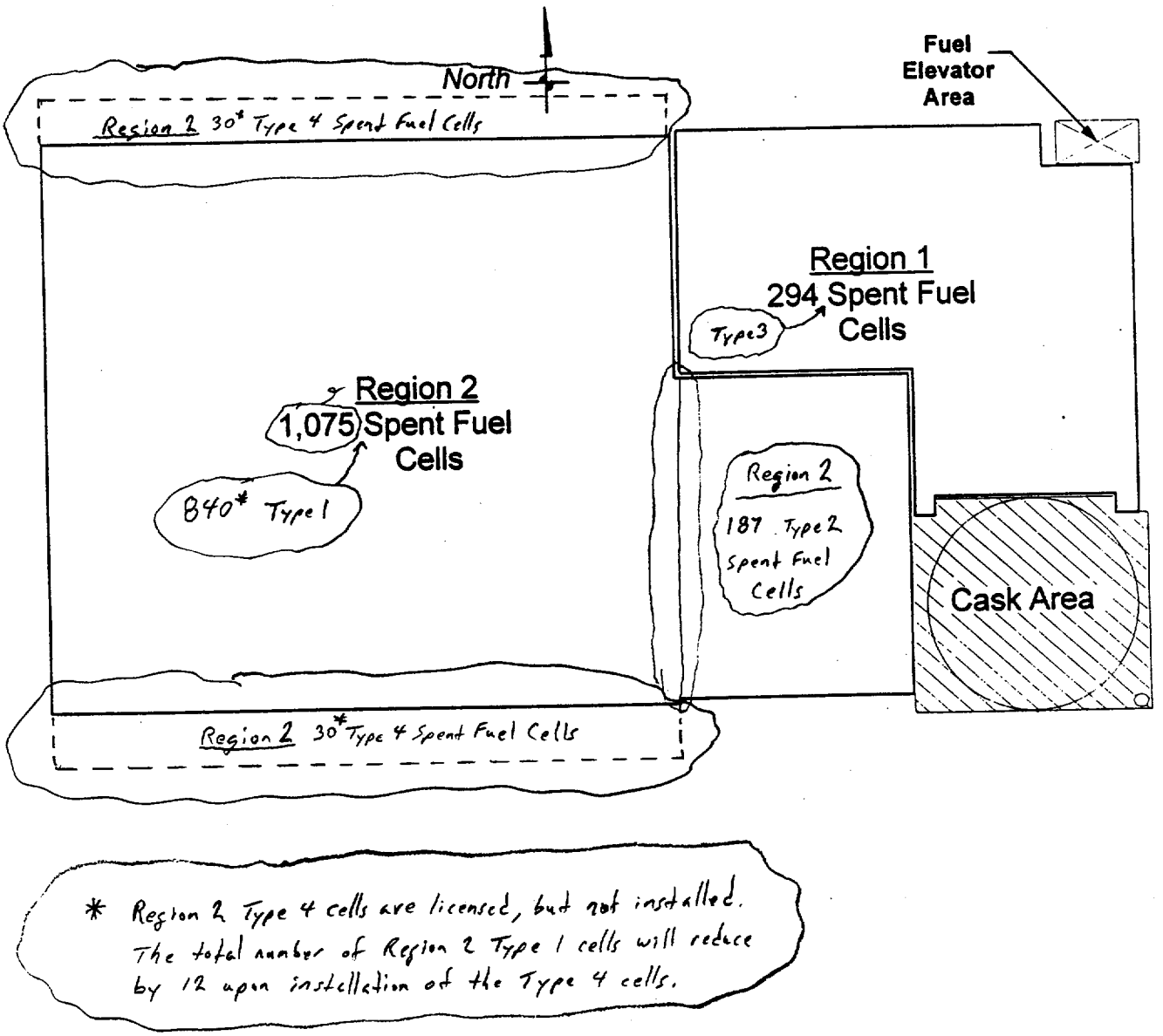


Figure B 3.7.13-1
Spent Fuel Pool

IMPROVED TECHNICAL SPECIFICATION BASES INSERTS

Insert 1

To maintain a 5% subcritical margin ($K_{eff} \leq 0.95$), a total of 975 ppm of soluble boron is required. This total is composed of four components; 377 ppm for $K_{eff} \leq 0.95$, 207 ppm to account for reactivity equivalencing methodologies, 381 ppm for the most limiting fuel mishandling event, and 10 ppm to account for B-10 depletion from recycled boron. In the absence of all soluble boron, the spent fuel pool is subcritical ($K_{eff} < 1.00$), not accounting for a fuel mishandling event. These two conditions (established in References 3 and 4) are met without crediting any of the Boraflex that was originally installed in Region 2.

Insert 2

The soluble boron concentration (975 ppm), required to maintain $K_{eff} \leq 0.95$ also addresses the single most limiting reactivity insertion accident in the spent fuel pool, a fuel mishandling event in a Region 2 Type 1 cell. Therefore, the 975 ppm (determined from Reference 5) of soluble boron will maintain $K_{eff} \leq 0.95$ assuming the most limiting fuel mishandling event. This was established without crediting any Boraflex in the Region 2 Type 1 cells.

Insert 3

The third category consists of boron reduction events for which an analysis of potential scenarios which could dilute the boron concentration in the SFP has been performed (Reference 6). The analysis demonstrates that sufficient time is available to detect and mitigate the dilution prior to exceeding the 0.95 k_{eff} design basis. The potential plant events were quantified to show that sufficient time is available to enable adequate detection and mitigation of any postulated dilution event. Deterministic dilution event calculations were performed to define the dilution times and volumes necessary to dilute the 213,600 gallon SFP water inventory from the minimum required 2300 ppm to a soluble boron concentration of 975 ppm. Assuming a well mixed pool, the volume required to dilute the pool from 2300 to 975 ppm was determined to be 183,000 gallons. Based on the above evaluation, an unplanned or inadvertent event which would reduce the SFP boron concentration from 2300 ppm to 975 ppm is not credible.

Insert 4

The total soluble boron required to maintain $K_{eff} \leq 0.95$ (with soluble boron credit) is determined to be 975 ppm. The specified boron concentration also addresses the single fuel mishandling accident and is the minimum required concentration for fuel assembly storage.

Insert 5

3. Newmeyer, W.D., "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", WCAP-14416-NP-A, Revision 1, November 1996.

4. Letter from T.E. Collins, U.S. NRC to T. Greene, WOG, "Acceptance for Referencing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)", October 25, 1996.

5. ABB Combustion Engineering Nuclear Power, "R. E. Ginna Nuclear Power Plant Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit", February 2000.

6. ABB Combustion Engineering Nuclear Power, "R. E. Ginna Spent Fuel Pool Boron Dilution Analysis", January 2000.

Insert 6

Region 2, with 1075 storage positions, is designed to accommodate fuel of various initial enrichments which have various accumulated minimum burnups and decay times. Decay time refers to the time period for which the fuel assembly has been residing since irradiation from power operation in the reactor. Region 2 is described by three types of cells; Type 1, Type 2 and Type 4 cells.

For the storage of fuel assemblies in Type 1 cells, the acceptable combination of initial enrichment, burnups and decay times are according to Figures 3.7.13-2 through 3.7.13-6. Fuel assemblies with initial enrichments, burnups and decay times, within domain A1 of Figures 3.7.13-2 through 3.7.13-6 may be stored in any location in Region 2 Type 1 cells. Fuel assemblies with initial enrichments, burnups and decay times, within domain A2 of Figures 3.7.13-2 through 3.7.13-6 shall be stored face-adjacent to a Type A1 or A2 assembly, or a water cell in Region 2 Type 1 cells. Fuel assemblies with initial enrichments, burnups and decay times, within domain B of Figures 3.7.13-2 through 3.7.13-6 shall be stored face-adjacent to a Type A1 assembly, or a water cell in Region 2 Type 1 cells. Fuel assemblies with initial enrichments, burnups and decay times, within domain C of Figures 3.7.13-2 through 3.7.13-6 shall be stored face-adjacent to a water cell only, in Region 2 Type 1 cells.

(Insert 6 cont.)

For the storage of fuel assemblies in Type 2 and Type 4 cells, the acceptable combination of initial enrichment, burnups and decay times are according to Figures 3.7.13-7 through 3.7.13-11. Fuel assemblies with initial enrichments, burnups and decay times, within domain A1 of Figures 3.7.13-7 through 3.7.13-11 may be stored in any location in Region 2 Type 2 and Type 4 cells. Fuel assemblies with initial enrichments, burnups and decay times, within domain A2 of Figures 3.7.13-7 through 3.7.13-11 shall be stored face-adjacent to a Type A1 or A2 assembly, or a water cell in Region 2 Type 2 and Type 4 cells. Fuel assemblies with initial enrichments, burnups and decay times, within domain B of Figures 3.7.13-7 through 3.7.13-11 shall be stored face-adjacent to a Type A1 assembly, or a water cell in Region 2 Type 2 and Type 4 cells. Fuel assemblies with initial enrichments, burnups and decay times, within domain C of Figures 3.7.13-7 through 3.7.13-11 shall be stored face-adjacent to a water cell only, in Region 2 Type 2 and Type 4 cells.

The word "face-adjacent" is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the fuel assembly in the next cell. The storage of fuel assemblies which are within the acceptable ranges of Figures 3.7.13-2 through 3.7.13-6 and 3.7.13-7 through 3.7.13-11, in Region 2 ensures a $K_{eff} \leq 0.95$ (with soluble boron credit) in this region.

Insert 7

The failed fuel storage basket has been explicitly modeled with up to nominal 5.0 wt% enriched fuel rods in each location in the basket. It has been shown to be acceptable in any location in the pool regardless of the burnup of the rods contained within it (A1 status for Region 2).

Other items may be stored in the SFP in addition to fresh or discharged fuel assemblies. These items, in general, fall into the category of Non-Special Nuclear Material(SNM). These items are non-multiplying and, in general, are parasitic to the spent fuel rack local reactivity. Some of the items which fall under this category that can be safely stored in the spent fuel pool are: Dummy Canisters containing Non-SNM, Consolidation Hardware, Dummy Fuel Assemblies, Trash Basket containing full length control rods, etc. The general rule for safely storing these types of items is very simple: any non-multiplying and non-fissile item can be safely stored in any cell location. The storing of these components within a water cell does not affect the classification of the cell, i.e., it is still considered a water cell.

Insert 8

To maintain a 5% subcritical margin ($K_{eff} \leq 0.95$), a total of 975 ppm of soluble boron is required. This total is composed of four components; 377 ppm for $K_{eff} \leq 0.95$, 207 ppm to account for reactivity equivalencing methodologies, 381 ppm for the most limiting fuel mishandling event, and 10 ppm to account for B-10 depletion from recycled boron. In the absence of all soluble boron, the spent fuel pool is subcritical ($K_{eff} < 1.00$), not accounting for a fuel mishandling event. These two conditions (established in References 4 and 5) are met without crediting any of the Boraflex that was originally installed in Region 2.

Insert 9

The soluble boron concentration (975 ppm), required to maintain $K_{eff} \leq 0.95$ also addresses the single most limiting reactivity insertion accident in the spent fuel pool, a fuel mishandling event in a Region 2 Type 1 cell. Therefore, the 975 ppm (determined from Reference 6) of soluble boron will maintain $K_{eff} \leq 0.95$ assuming the most limiting fuel mishandling event. This was established without crediting any Boraflex in the Region 2 Type 1 cells.

Insert 10

The third category consists of boron reduction events for which an analysis of potential scenarios which could dilute the boron concentration in the SFP has been performed (Reference 7). The analysis demonstrates that sufficient time is available to detect and mitigate the dilution prior to exceeding the 0.95 k_{eff} design basis. The potential plant events were quantified to show that sufficient time is available to enable adequate detection and mitigation of any potential dilution event. Deterministic dilution event calculations were performed to define the dilution times and volumes necessary to dilute the 213,600 gallon SFP water inventory from the minimum required 2300 ppm to a soluble boron concentration of 975 ppm. Assuming a well mixed pool, the volume required to dilute the pool from 2300 to 975 ppm was determined to be 183,000 gallons. Based on the above evaluation, an unplanned or inadvertent event which would reduce the SFP boron concentration from 2300 ppm to 975 ppm is not credible.

Insert 11

The restrictions on the placement of fuel assemblies within the SFP ensure the K_{eff} of the SFP will always remain ≤ 0.95 (with soluble boron credit). For fuel assemblies stored in Region 1, the maximum nominal U-235 enrichment of the fuel assembly shall not be greater than 5.0 wt% and the initial enrichment and burnup values are within the acceptable area of Figure 3.7.13-1. For fuel assemblies stored in Region 2 Type 1 cells, the initial enrichment and burnup values for various decay times shall be within the acceptable area of Figures 3.7.13-2 through 3.7.13-6. For fuel assemblies stored in Region 2 Type 2 and Type 4 cells, the initial enrichment and burnup values for various decay times shall be within the acceptable area of Figures 3.7.13-7 through 3.7.13-11. The word "face-adjacent" is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the assembly in the next cell.

Insert 12

This SR verifies by administrative means that the maximum nominal initial enrichment of each fuel assembly is ≤ 5.0 wt% U-235 and that the initial enrichment and burnup values of the fuel assemblies, with various decay times, are in accordance with Figures 3.7.13-1 through 3.7.13-11 prior to storage or movement in the SFP. Once a fuel assembly has been verified to be within the acceptable range of Figures 3.7.13-1 through 3.7.13-11 for its correct location, further verifications are no longer required since the initial enrichment of each assembly will not change (i.e., increase) while partially burned fuel assemblies are less reactive than when they were new . Performance of this SR ensures compliance with Specification 4.3.1.1.

Insert 13

4. Newmeyer, W.D., "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", WCAP-14416-NP-A, Revision 1, November 1996.

5. Letter from T.E. Collins, U.S. NRC to T. Greene, WOG, "Acceptance for Referencing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)", October 25, 1996.

6. ABB Combustion Engineering Nuclear Power, "R. E. Ginna Nuclear Power Plant Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit", February 2000.

7. ABB Combustion Engineering Nuclear Power, "R. E. Ginna Spent Fuel Pool Boron Dilution Analysis", January 2000.

Attachment V
R.E. Ginna Nuclear Power Plant

Proposed Revised Improved Technical Specifications

Included pages:

3.7-29
3.7-30
3.7-31
3.7-31a
3.7-31b
3.7-31c
3.7-31d
3.7-31e
3.7-31f
3.7-31g
3.7-31h
3.7-31i
4.0-2
4.0-3

3.7 PLANT SYSTEMS

3.7.13 Spent Fuel Pool (SFP) Storage

LCO 3.7.13 The combination of initial enrichment and burnup values, with appropriate decay times, of each fuel assembly stored in the spent fuel pool shall be within the acceptable burnup domain of the applicable Figures 3.7-1 through 3.7-11, based on region and cell type.

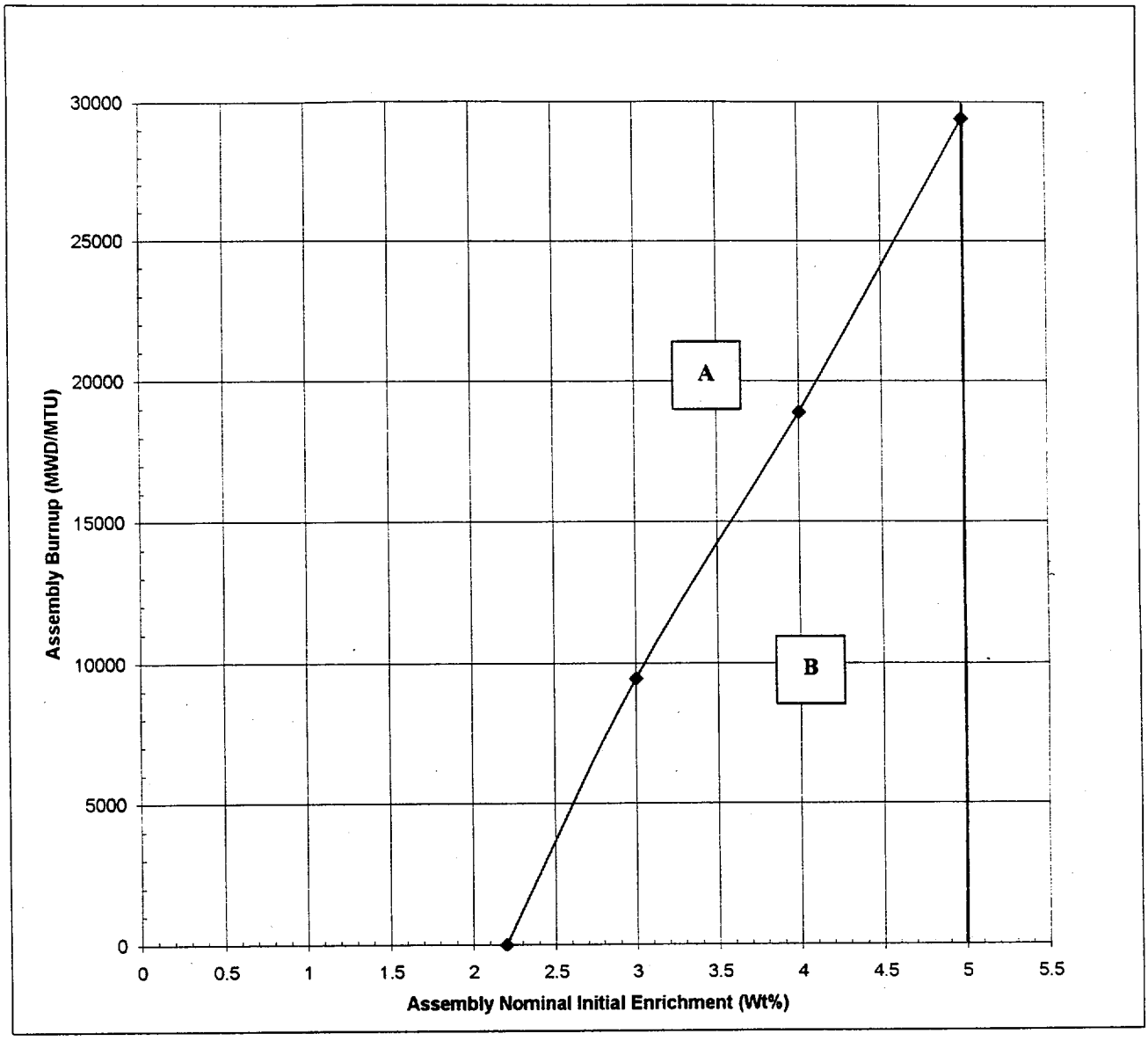
APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE-------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

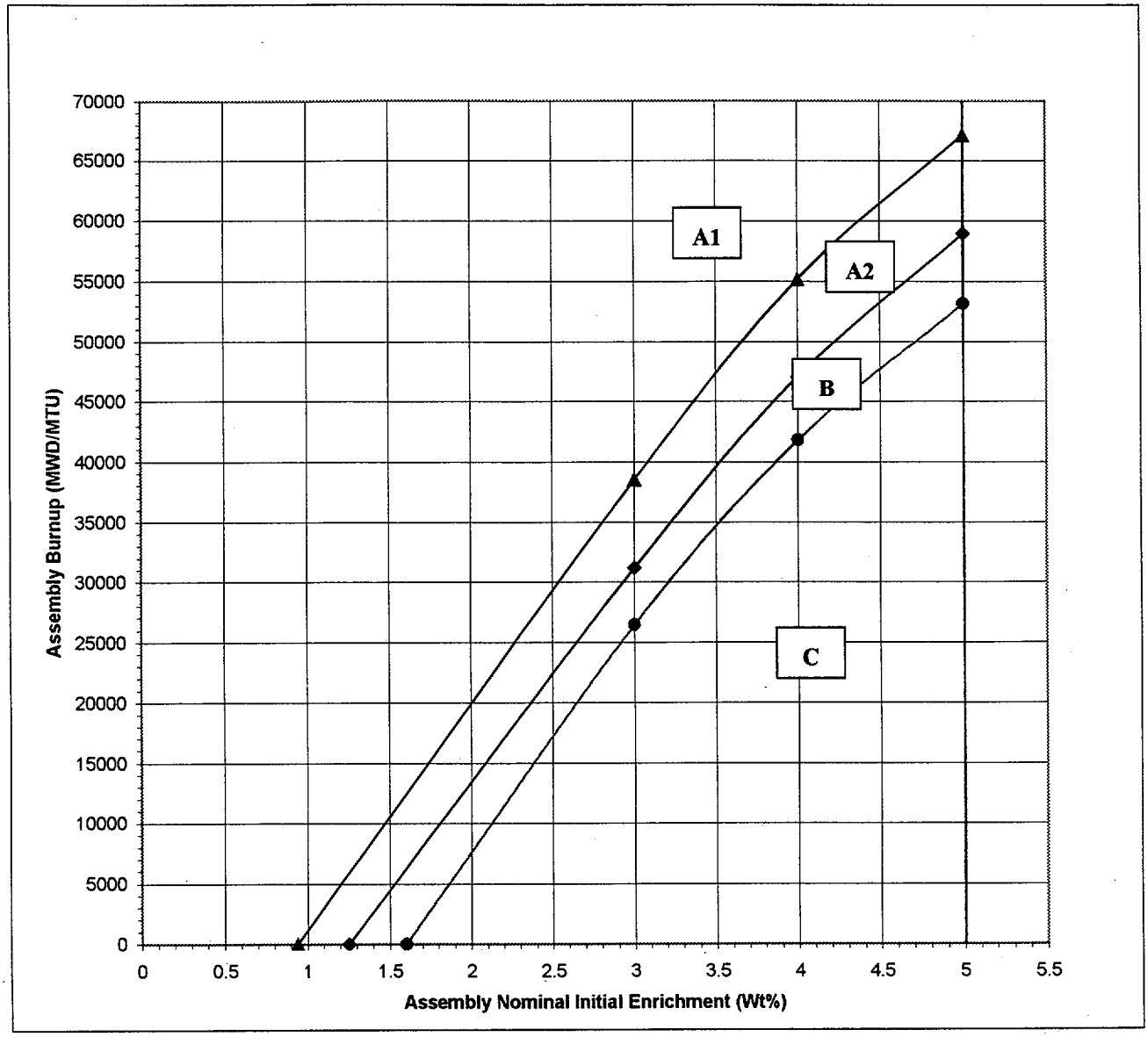
SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Verify by administrative means the initial enrichment, burnup, and decay time of the fuel assembly is in accordance with the applicable Figures 3.7.13-1 through 3.7.11.	Prior to storing, or moving, the fuel assembly in the spent fuel pool



A Acceptable burnup domain for storage in any location within Region 1

B Acceptable burnup domain for storage in cells with lead-in funnels only

Figure 3.7.13-1
Burnup Vs Enrichment Curve for Region 1 Type 3 Cells
(Not Pu-241 Decay Dependent)



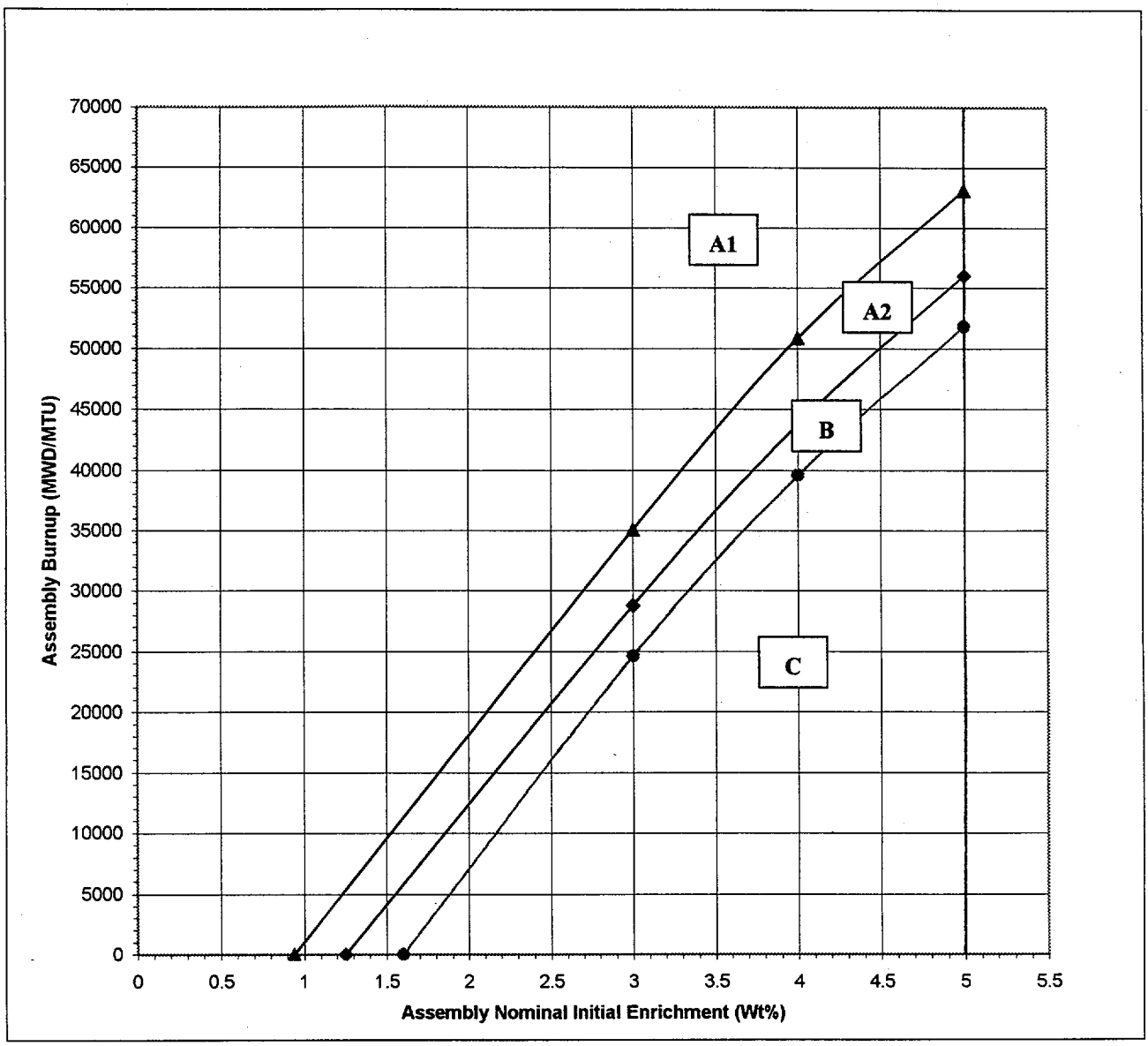
A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-2
Burnup Vs Enrichment Curves for Region 2 Type 1 Cells
(No Pu-241 Decay)



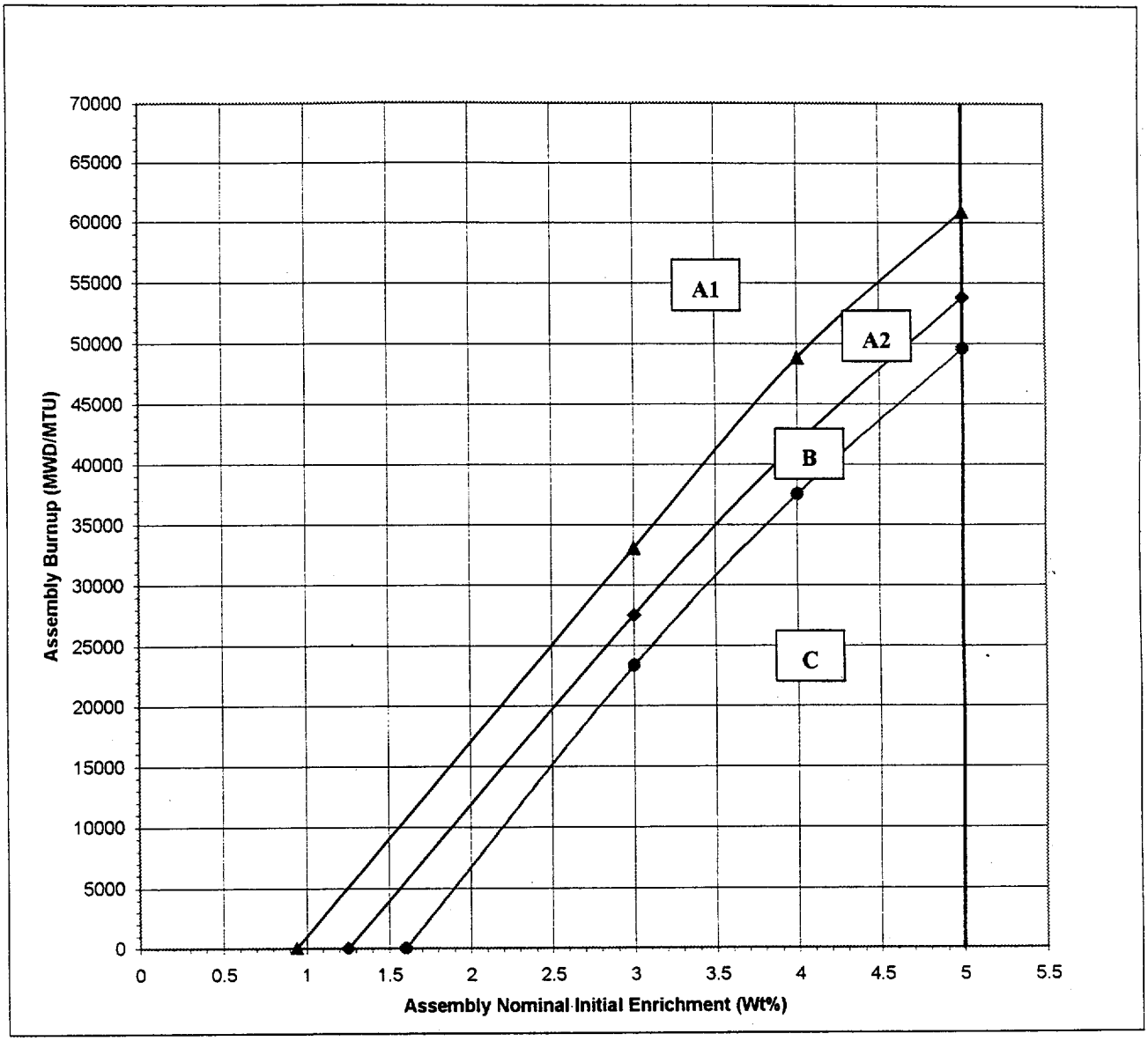
A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-3
Burnup Vs Enrichment Curves for Region 2 Type 1 Cells
(5-Year Pu-241 Decay)



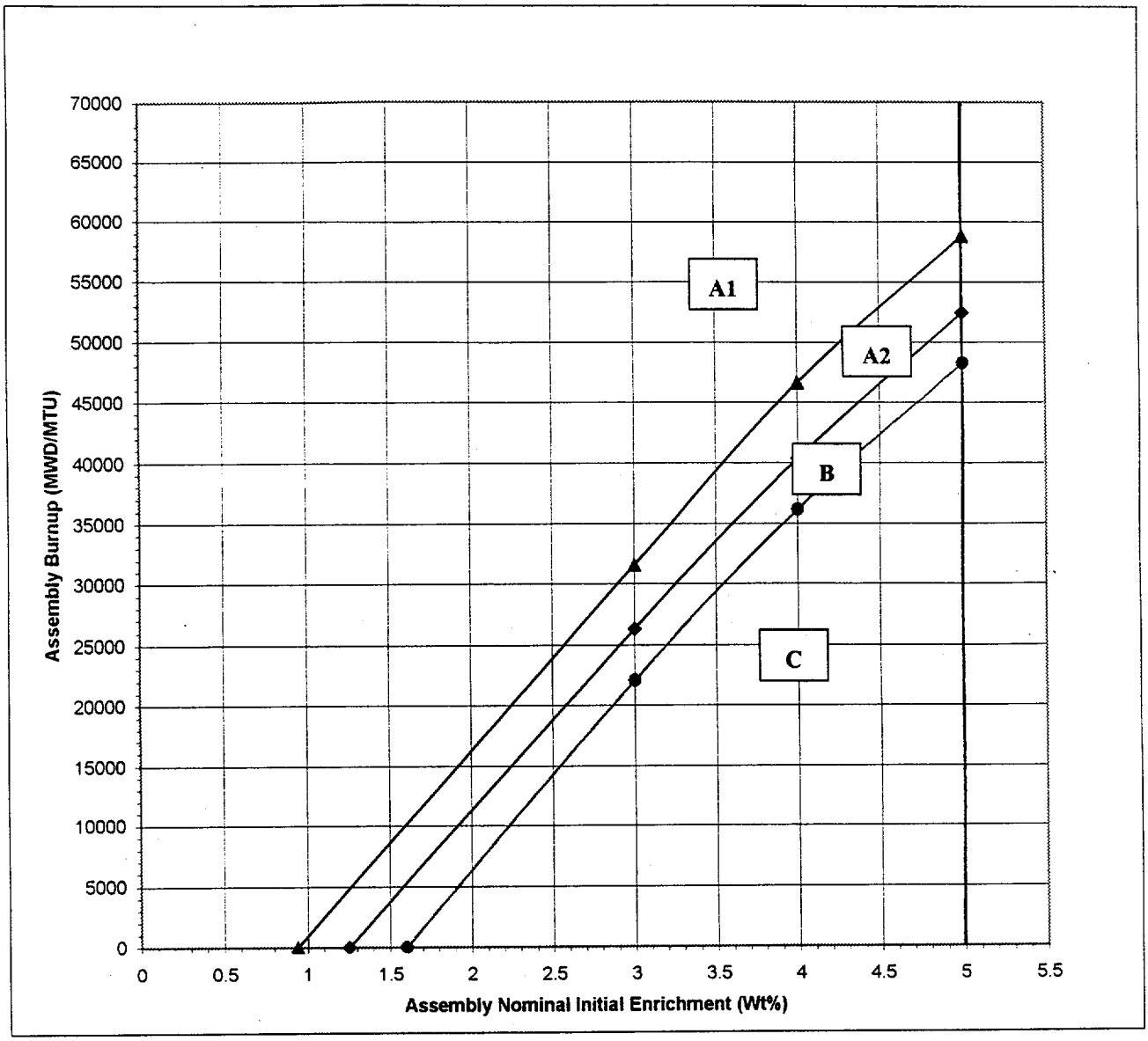
A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-4
Burnup Vs Enrichment Curves for Region 2 Type 1 Cells
(10-Year Pu-241 Decay)



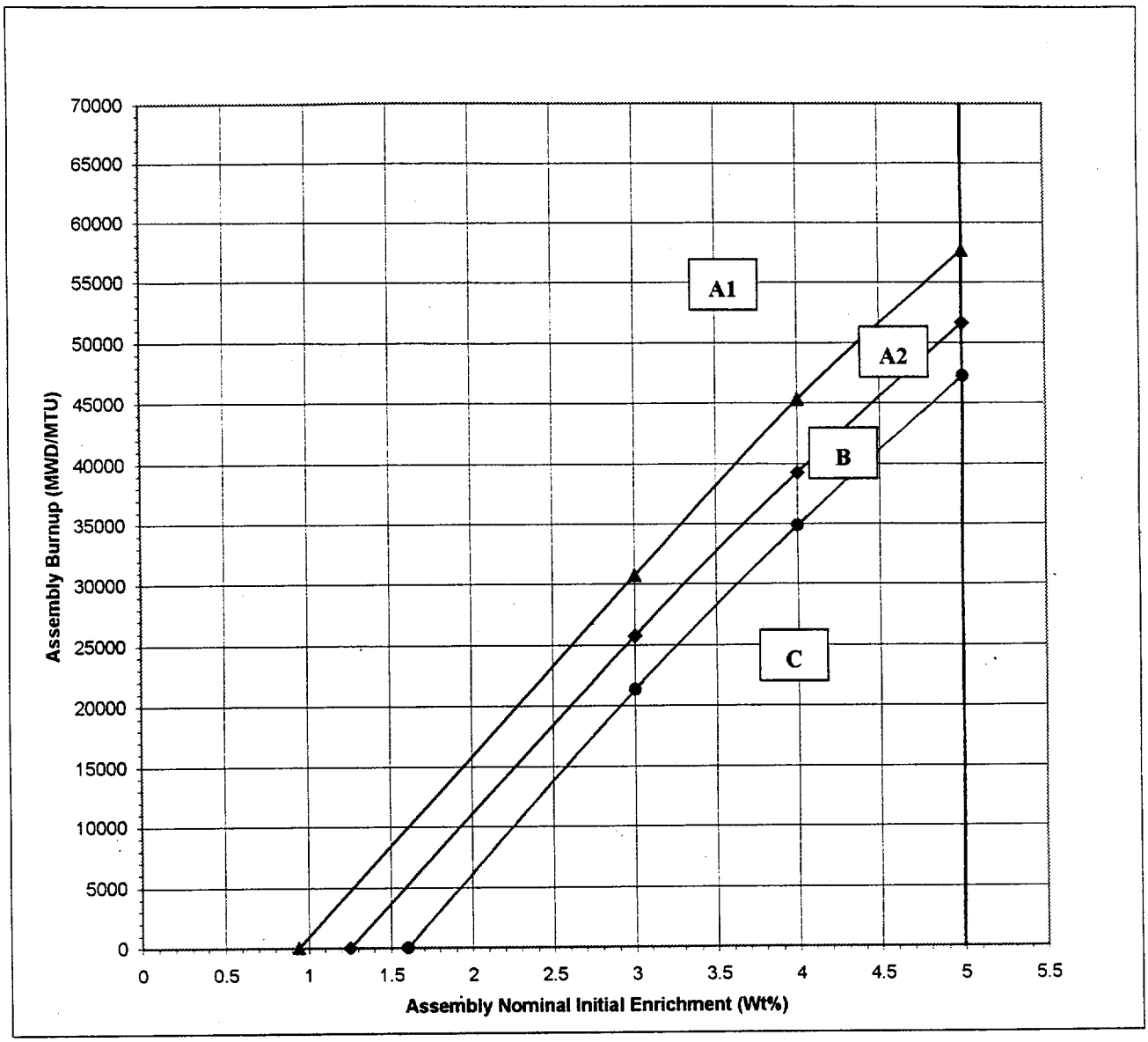
A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-5
Burnup Vs Enrichment Curves for Region 2 Type 1 Cells
(15-Year Pu-241 Decay)



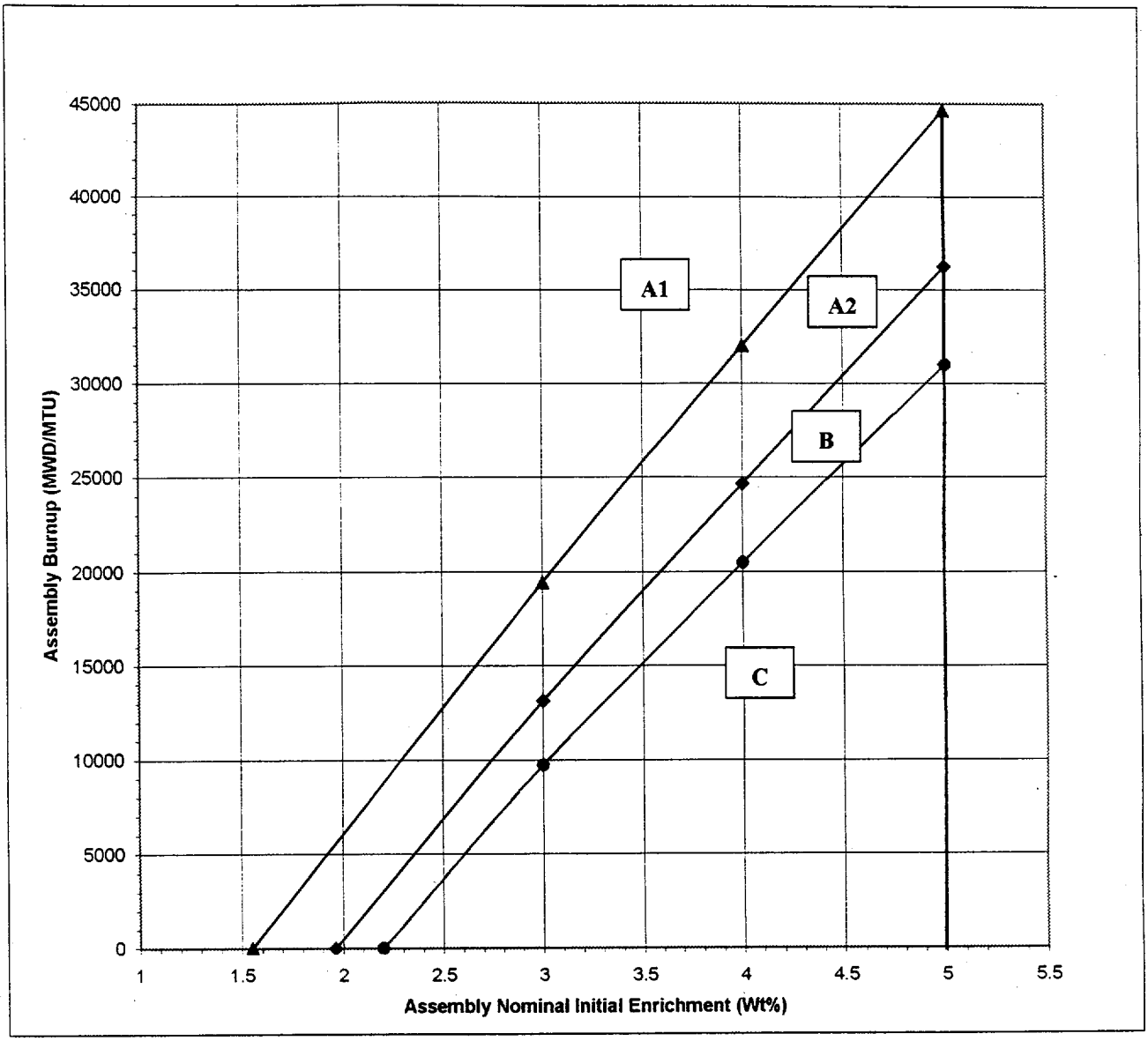
A1 Acceptable burnup domain for storage in any location within Region 2 Type 1 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 1 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 1 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 1 Cells

Figure 3.7.13-6
Burnup Vs Enrichment Curves for Region 2 Type 1 Cells
(20-Year Pu-241 Decay)



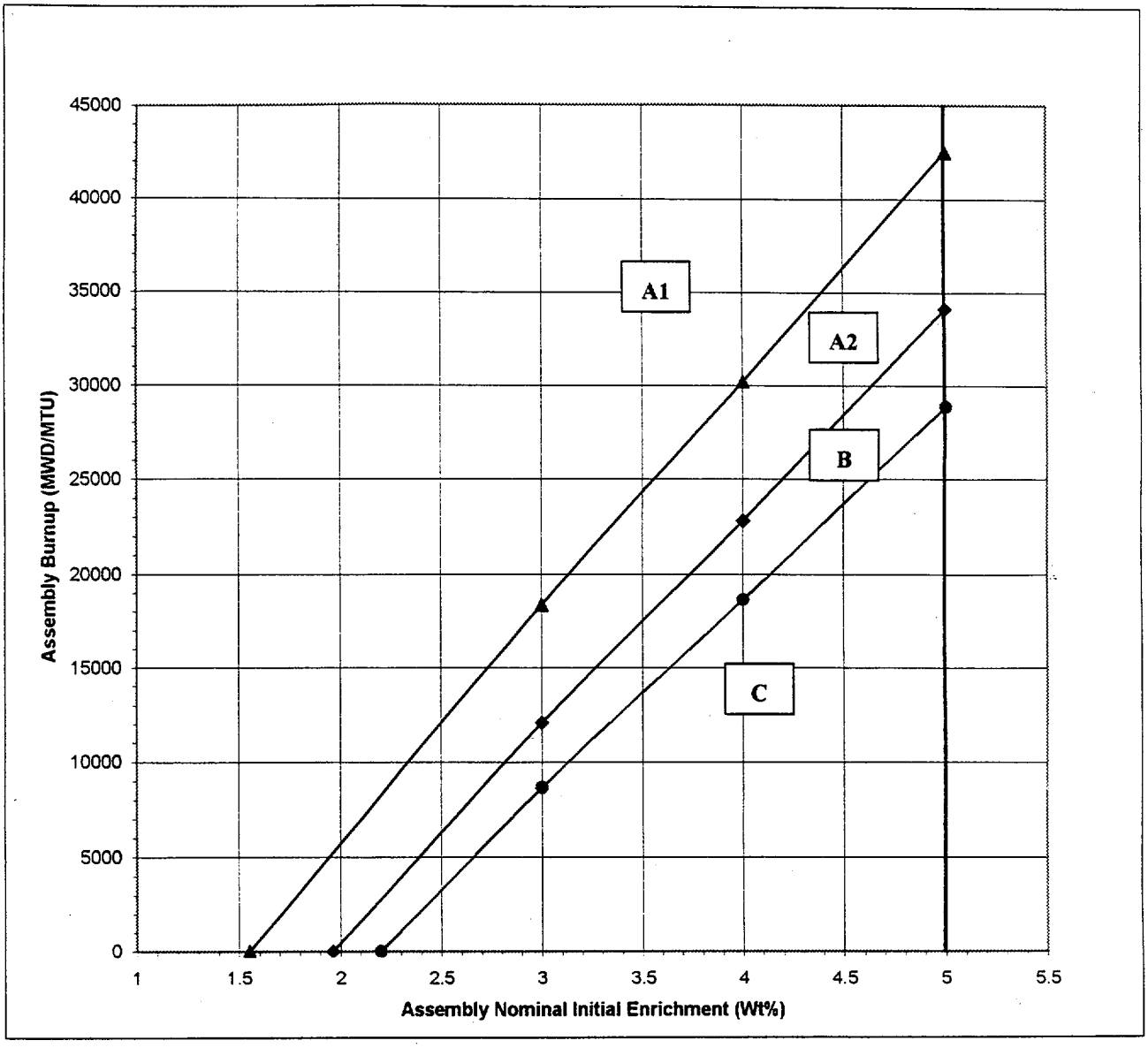
A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-7
Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells
(No Pu-241 Decay)



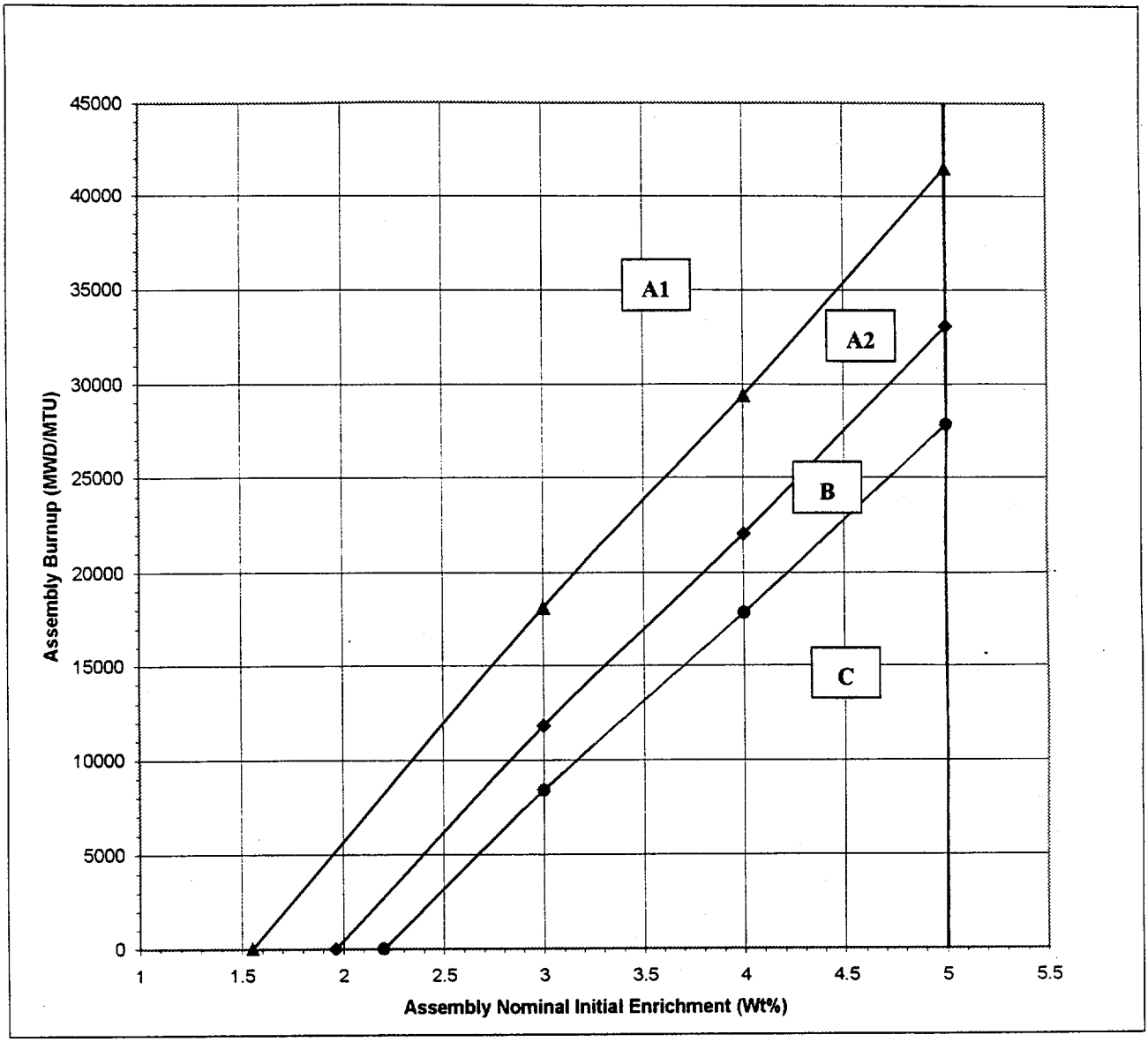
A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-8
Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells
(5-Year Pu-241 Decay)



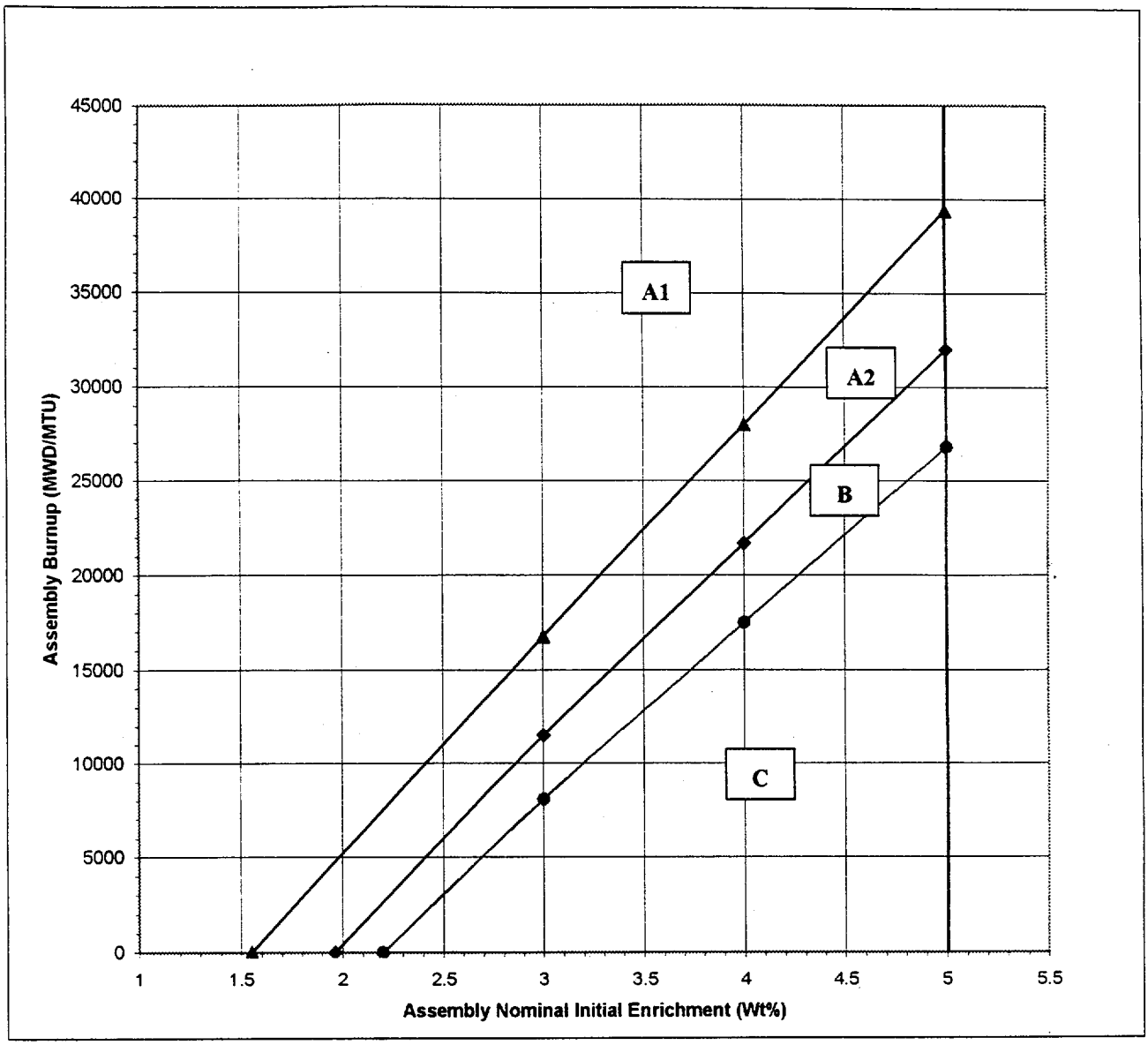
A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-9
Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells
(10-Year Pu-241 Decay)



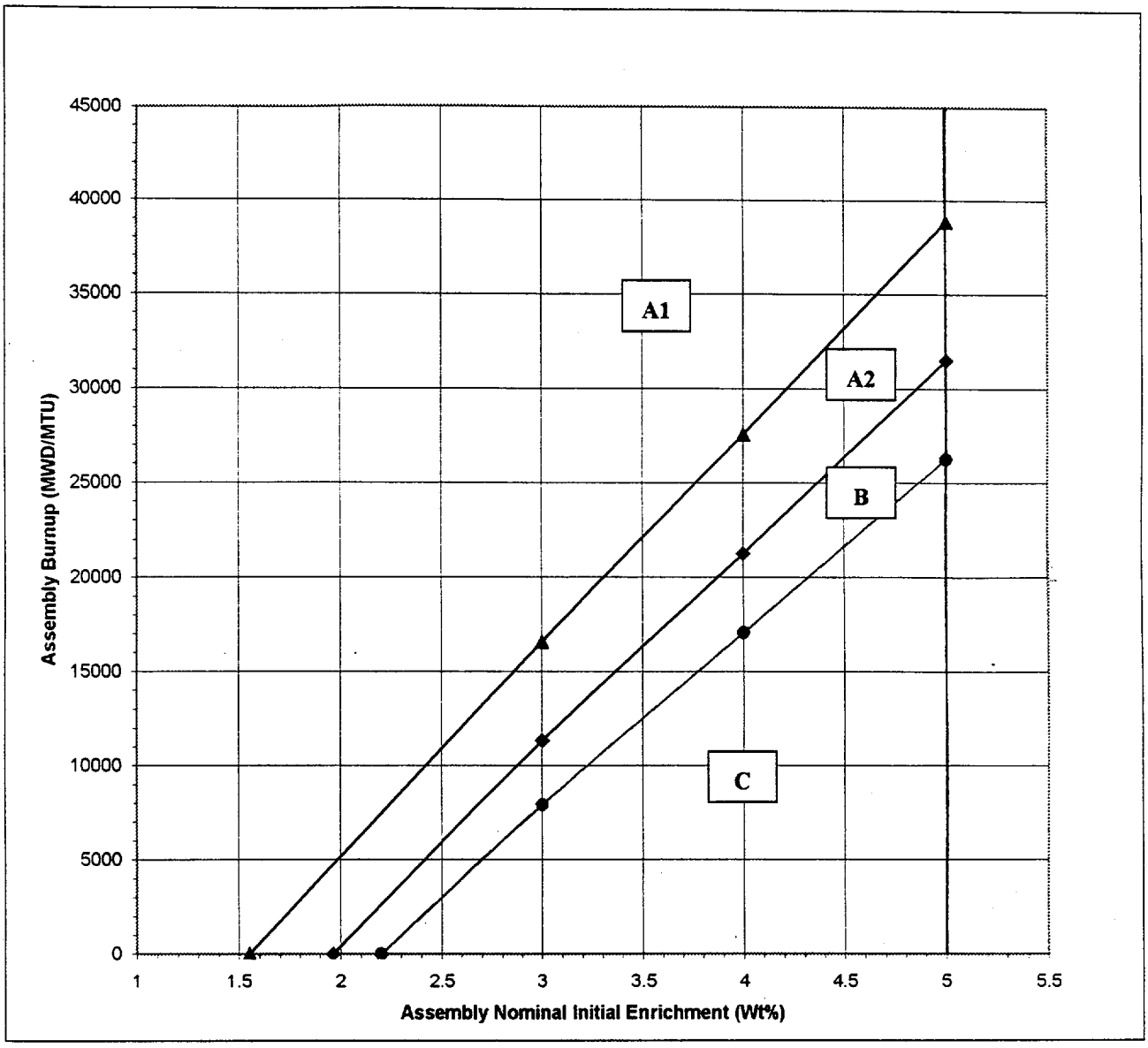
A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-10
Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells
(15-Year Pu-241 Decay)



A1 Acceptable burnup domain for storage in any location within Region 2 Type 2 and Type 4 Cells

A2 Acceptable burnup domain for storage face-adjacent to a Type A1 or A2 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

B Acceptable burnup domain for storage face-adjacent to a Type A1 assembly, or a water cell within Region 2 Type 2 and Type 4 Cells

C Acceptable burnup domain for storage face-adjacent to a water cell only, within Region 2 Type 2 and Type 4 Cells

Figure 3.7.13-11
Burnup Vs Enrichment Curves for Region 2 Type 2 and Type 4 Cells
(20-Year Pu-241 Decay)

4.0 DESIGN FEATURES

4.2 Reactor Core (continued)

4.2.2 Control Rod Assemblies

The reactor core shall contain 29 control rod assemblies. The control material shall be silver indium cadmium.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

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

b. $k_{eff} < 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to $\geq$ 975 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. Consolidated rod storage canisters may be stored in the spent fuel storage racks provided that the fuel assemblies from which the rods were removed meet all the requirements of LCO 3.7.13 for the region in which the canister is to be stored. The average decay heat of the fuel assembly from which the rods were removed for all consolidated fuel assemblies must also be $\leq$ 2150 BTU/hr.

(continued)

4.0 DESIGN FEATURES

4.3.1 Criticality (continued)

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

a. Fuel assemblies having a maximum nominal 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 Section 9.1 of the UFSAR; and

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

4.3.2 Drainage

The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 257'0" (mean sea level).

4.3.3 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1879 fuel assemblies and 1369 storage locations.

Enclosure 1
R.E. Ginna Nuclear Power Plant

Enclosure 1 to RGE-99-0009 Rev. 01

Final Report

R.E. Ginna Spent Fuel Pool Boron Dilution Analysis

January 2000

Verification Status: Complete			
The safety related design information contained in this document has been verified to be acceptable for the intended use by means of Design Review using the applicable portions of QP 3.10 of QPM-101 Revision 3.			
	Printed Name	**Signature**	**Date**
Cognizant Engineers	G. Max		1/10/2000
Independent Reviewers	S. Cybert		1/10/2000
Management Approval	M. Greene		1/10/2000

ABB Combustion Engineering Nuclear Power

VENDOR DRAWING/TECHNICAL DOCUMENT REVIEW

☒ Approved - Mfg. may proceed (If applicable)

☐ Approved - Submit final dwg. - Mfg. may proceed. (If applicable)

☐ Approved except as noted. Make changes and submit final dwg/technical document. Mfg. may proceed as approved.

☐ Review not required - Mfg. may proceed.

Approval of this drawing/technical document does not relieve supplier from full compliance with contract or purchase order requirements.

By PJ Bamford Date 3/3/00

ROCHESTER GAS & ELECTRIC CORP
ROCHESTER, NY

R. E. GINNA NUCLEAR POWER STATION

SPENT FUEL POOL BORON DILUTION ANALYSIS

Prepared by
ABB Combustion Engineering Nuclear Power, Inc.
for
Rochester Gas and Electric Company

Table of Contents

1.0 INTRODUCTION

A boron dilution analysis has been completed for crediting boron in the R. E. Ginna spent fuel rack criticality analysis. The boron dilution analysis includes an evaluation of the following plant specific features:

- Dilution sources
- Boration sources
- Instrumentation
- Administrative procedures
- Piping
- Boron dilution initiating events
- Boron dilution times and volumes

The boron dilution analysis was completed to ensure that sufficient time is available to detect and mitigate the dilution before the design basis limit on the effective multiplication factor (k_{eff} = 0.95) is reached.

2.0 SPENT FUEL POOL AND RELATED SYSTEM FEATURES

This section provides background information on the spent fuel pool and related systems.

2.1 Spent Fuel Pool

The purpose of the spent fuel pool is to safely store the irradiated fuel assemblies removed from the reactor during refueling operations. The pool is a reinforced concrete structure with a watertight welded stainless steel liner. The pool is approximately 38' 2" long, 22' 3" wide, and 43' 0" deep. The pool is adjacent to the fuel transfer canal, with a weir and gate separating the pool and the canal. The normally dry canal is flooded and the gate removed during refueling operations, allowing the fuel assemblies to remain submerged as they are moved between the canal and the pool. When refueling operations are completed, the gate is reinstalled and the transfer canal is drained.

The spent fuel pool (SFP) is filled with borated water. The SFP cooling system transfers the decay heat generated by the stored assemblies to the service water system. The cooling system consists of two pumps and two heat exchangers, along with the necessary piping and instrumentation, to maintain the pool temperature within specified limits. The system also includes a purification demineralizer and filter to maintain the water quality. A separate skimmer pump system is provided to clean the surface of the pool water to enhance visual clarity. A skid mounted pump and a skid mounted heat exchanger can be aligned through temporary hose connections to provide a third train of cooling as required for backup. Connections to the cooling system allow make-up from the reactor make-up water system, the chemical and volume control (CVCS) holdup tanks, and the (borated) refueling water storage tank (RWST). Connections to the reactor coolant drain tank (RCDT) and the demineralized condensate system are also provided, but not normally used. The SFP cooling system also includes connections to the liquid waste system and provisions for recharging the demineralizer resin and the filter element.

2.2 Spent Fuel Pool Storage Racks

The spent fuel storage racks support and protect the spent fuel assemblies in the pool. The storage racks will accommodate 1879 assemblies. The racks displace approximately 613 cubic feet, and each spent fuel assembly displaces approximately 2.31 cubic feet. The volume of water displaced by the racks and the stored assemblies is thus 4,954 cubic feet, or just over 37,000 gallons (conservatively assuming the maximum capacity of 1879 assemblies).

2.3 Spent Fuel Pool Instrumentation

The spent fuel pool is provided with level and temperature instruments which provide annunciated alarms in the main control room on high level, low level, and high temperature. The "B" pump automatically trips on low pool level, and low flow from the "B" pump also results in an alarm in the control room. Radiation monitors (R20A and R20B) on the SFP cooling system heat exchanger shell side outlet piping provide a control room alarm (indicative of leakage from the pool coolant to the service water system through the heat exchanger tubes). Local flow, temperature, and pressure indicators are installed throughout the cooling system as appropriate.

If the pool level were raised from the low level alarm point to the high level alarm point, a dilution of approximately 4,235 gallons could occur before an alarm would be received in the control room. If the spent fuel pool boron concentration was initially at the Technical Specification lower limit of 2300 ppm, this dilution would reduce the concentration by approximately 45 ppm.

2.4 Spent Fuel Pool Administrative Procedures

Technical Specification 3.7.12 requires the pool boron concentration to be greater than or equal to 2300 ppm. This is verified by sampling with a surveillance interval of seven days (Surveillance Requirement SR 3.7.12.1).

In accordance with procedures, plant operators are expected to perform rounds during each 4 hours they are on shift (that is, once during the first four hour of the shift, once during the second four hours, and, for 12 hour shifts, once during the final four hours). These rounds include log readings of the spent fuel pool temperature and an "area check" of the SFP as part of the auxiliary building rounds. The maximum interval between entries into the SFP area would occur if the operator checked the area at the beginning of the first four hours, and again at the end of the second four hours. A maximum of 8 hours could therefore elapse between entries into the area (conservatively neglecting time for shift turnover, etc.).

As discussed in section 4 below, some of the potential dilution paths are subject to administrative controls (e.g., locked valves).

2.5 Boration Sources

The normal source of borated water to the spent fuel pool is from the refueling water storage tank (RWST). The boron concentration in the RWST is maintained above 2300 ppm in accordance with Technical Specification SR 3.5.4.2.

As an alternative, borated water from the CVCS can be provided via the connection to the CVCS holdup tanks.

It would also be possible to borate the spent fuel pool by addition of dry boric acid directly to the pool surface. The dry boric acid would dissolve and be mixed throughout the pool by the action of the SFP cooling system and by the thermal convection induced by the decay heat generated in the fuel assemblies.

3.0 SPENT FUEL POOL DILUTION EVENT

3.1 Calculation of Boron Dilution Times and Volumes

The boron concentration in the spent fuel pool is maintained greater than or equal to the Technical Specification lower limit of 2300 ppm. Based on the criticality analysis, a soluble boron concentration of 975 ppm will maintain reactivity within the design basis limit of $k_{eff} \leq 0.95$ (including all biases and uncertainties) with a 95% probability at a 95% confidence level.

Dilution may occur by two different processes. In the first type, the dilution source increases the total volume (for example, the source fills the pool from its initial level to a higher level). The change in concentration for this type of dilution can be determined from a simple weighted average of the initial pool concentration and the dilution source concentration. The second type of dilution is a "feed and bleed" process in which the pool volume remains constant (for example, with the pool overflowing to match the dilution rate). The change in concentration for a "feed and bleed" dilution can be calculated from a simple differential equation as described below. For a given dilution source, the "feed and bleed" process always results in a lower concentration.

The "feed and bleed" type of dilution is modeled as follows:

Rate of change of mass of boron in pool	=	rate boron enters pool	-	rate boron leaves pool
dB/dt	=	Win * Fin	-	Wout * Fout

Where:

B	=	mass of boron in spent fuel pool (lb)
Win	=	mass flow rate from dilution source (lb/sec)
Wout	=	mass flow rate leaving pool (lb/sec)
Fin	=	weight fraction of boron in dilution source (lb boron/lb fluid)
Fout	=	weight fraction of boron leaving pool (lb boron/lb fluid)

With the conservative assumption that the dilution source is unborated, we set Fin = 0. If the pool volume is V (ft^3), and its fluid density is ρ (lb/ ft^3), then the weight fraction of boron in the pool is equal to $B/\rho V$. We call this weight fraction "F," and note that with perfect mixing, Fout = F. The equation is then:

$$dF/dt = -(Wout/\rho V) * F$$

Finally, $Wout/\rho$ is the volumetric (ft^3/sec) dilution flowrate which we call "Q." Then:

$$dF/dt = - (Q/V) * F$$

The solution of this differential equation is

$$F(t) = F_0 * e^{-(Q/V)t}$$

Where F_0 is the initial weight fraction of boron in the pool. Noting that the weight fraction is proportional to the concentration in ppm, we can express the relation as:

$$C(t) = C_0 * e^{-(Q/V)t}$$

Where C_0 is the initial concentration in ppm.

The pool volume at the low level alarm, less the volume displaced by the racks and stored fuel, is greater than 213,600 gallons. Using this volume, we can determine the total dilution volume needed to dilute the pool from its initial 2300 ppm concentration to the minimum acceptable 975 ppm by solving the above equation for "Q t."

$$Q\,t = V \ln (C_0/C) = (213{,}600) \ln (2300/975)$$

The result is 183,000 gallons. Any dilution source not capable of supplying 183,000 gallons of unborated water will not be capable of diluting the pool to 975 ppm.

For dilution sources with automatic make-up, the capacity is essentially infinite. Should one of these sources begin adding water to the pool, the pool level would rise to the high level alarm setpoint. If no actions were taken, the pool would eventually fill to the curb and then begin overflowing. The overflow would be directed to the waste holdup tank (WHT), either directly through floor drains, or via the auxiliary building sump and sump pumps. Automatic operation of the sump pumps produces an alarm in the control room. The rising level in the waste holdup tank would eventually result in a high level alarm in the control room. Finally, the effects of this overflow would also be apparent to the plant operators performing their rounds. Taking 8 hours as a conservatively long interval between operator rounds (as discussed above), and assuming an additional 2 hours are required to diagnose the source and terminate the dilution, we can determine the dilution flow rate necessary to challenge the 975 ppm limit. The result is approximately 305 gpm. Any dilution source supplying less than 305 gpm of unborated water will not be capable of diluting the pool to 975 ppm before the plant operators will detect and mitigate the dilution – even without taking credit for the many alarms which would notify the control room operators that the pool was overflowing.

Another possible scenario would be leakage into the pool at a very low flowrate not detectable by any effects on the leaking system (for example, routine makeup to the system could mask a slow leak to the pool, such that the operators would not detect that leakage and dilution were taking place). Unless there was a corresponding leak somewhere in the SFP cooling system, the pool level would rise to the high level alarm. If the alarm failed, the level would continue to rise to the top of the curb and then begin overflowing. This would result in the alarms described above, or be detected by the operators on their rounds. In the unlikely event that there was a second leak just big enough to maintain a constant pool level, the weekly surveillance of pool

boron concentration would detect such a dilution before the total dilution volume reached the 183,000 gallon limit. In order for the total dilution volume to reach the 183,000 gallon limit prior to performance of the weekly concentration surveillance, the undetected leak rate would have to exceed approximately 14 gpm, or in excess of 20,000 gallons per day, for the entire week.

Leakage from the pool across the SFP heat exchanger tubes into the service water system would likely be identified by the radiation monitors downstream of the heat exchangers, due the normal activity in the pool.

4.0 DILUTION SOURCE PATH EVALUATION

This section evaluates the potential for dilution of the spent fuel pool via the SFP cooling system. The cooling system includes connections from the reactor make-up water system, the CVCS holdup tanks and reactor coolant drain tank, the RWST, and the demineralized water / condensate system. The SFP cooling system also includes connections to the liquid waste system and provisions for recharging the demineralizer resin and the filter element. A cross-tie to and from the letdown portion of the CVCS is provided. The SFP heat exchangers use service water as a cooling medium, so leakage across the heat exchanger tubes is evaluated. The inflatable seals in the transfer gate and in the transfer tube provide an interface between the SFP and the instrument air system.

4.1 Reactor Make-Up Water

The SFP cooling system design includes a connection to the reactor makeup water tank to allow normal make-up to the system. The tank can be aligned to supply water to the system via the reactor makeup water pumps, tying into the system in the purification loop, between the demineralizer and the filter. Depending on which system(s) the makeup system is servicing, one or more normally closed manual valves would have to leak by or be mispositioned in order to dilute the SFP cooling system. The maximum dilution rate (with one reactor makeup water pump operating) would be 60 gpm, well below the 305 gpm limit, ensuring that the operators would observe the overflowing pool before the pool was diluted to 975 ppm. In the unlikely event that leakage or valve misalignment are undetected, depletion of the entire reactor makeup water tank volume of 75,000 gallons would not challenge the 183,000 gallon dilution limit.

4.2 CVCS Holdup Tanks and Reactor Coolant Drain Tank

The system design includes a connection to the CVCS holdup tanks and the discharge of the reactor coolant drain tank pumps. This connection is normally isolated by a closed manual valve. In the unlikely event that this valve is inadvertently open, two potential dilution sources exist. First, a dilution path could exist during operation of the CVCS holdup tanks recirculation pump. The return line from this pump to the holdup tanks includes a branch line which ties into the RCDT pump discharge line. With the normally closed valve open, operation of the recirculation pump could direct the CVCS holdup tank inventory into the SFP cooling system. Under normal conditions, only one of the three CVCS holdup tanks is in service, with the other two tanks isolated. The volume of each CVCS holdup tanks is 31,156 gallons. In the unlikely event that all three tanks are mis-aligned and available to dilute the SFP, the total combined volume of approximately 93,500 gallons is well below the dilution limit of 183,000 gallons. The second dilution path would be from the reactor coolant drain tank, via the RCDT pumps. The 350 gallon volume of the RCDT is insignificant in comparison with the required dilution volume of 183,000 gallons. The RCDT is constantly being replenished by the small leakages it is designed to collect, but the weekly sampling of the SFP for boron (per Technical Specification SR 3.7.12.1) would identify that dilution was occurring before the accumulated leakage would reach 183,000 gallons.

Additional sources for backleakage, connected to the RCDT pump discharge lines are evaluated as follows:

The lines to the residual heat removal (RHR) pump discharge headers would have to leak back through a normally closed valve and a check valve, then back through the locked closed valve.

The line from the auxiliary building sump pumps would have to leak back through two normally closed valves and a check valve. In addition, the dilution rate would be limited by the capacity of the auxiliary building sump pumps (50 gpm per pump), which is well below the 305 gpm limit, ensuring that the operators would observe the overflowing pool before the pool was diluted to 975 ppm.

The line to the containment spray system piping connects to the return line to the RWST through a normally closed valve. This does not represent a dilution path, since the spray pumps take suction from the borated RWST.

The line to the waste holdup tank would have to leak back through the locked closed valve. In the unlikely event that this were to occur, the dilution would be limited by the 21,000 gallon volume of the waste holdup tank, which is insignificant in comparison to the 183,000 gallon limit.

A second interface with the RCDT is a line between the bottom of the fuel transfer canal portion of the spent fuel pool and the suction piping of the Reactor Coolant Drain Tank pumps. This line allows the RCDT pumps to drain the transfer canal. Under normal conditions, the transfer canal is dry and the drain line is isolated by a locked closed manual valve. In the unlikely event that this valve leaks or is inadvertently open, and if the drain line were pressurized, backleakage would enter the transfer canal and be prevented from mixing with the SFP liquid until the leakage filled the transfer canal and began overflowing the gate into the pool. The transfer canal volume is approximately 44,000 gallons. Leakage of this magnitude would almost certainly be observed if the plant were not refueling and the canal was expected to be dry. If the unit was refueling (with the canal flooded), the most likely consequence of leakage through the drain valve would be to drain the transfer canal and this does not affect the pool boron concentration. If the drain line were pressurized, then the total amount of leakage would be limited by the volumes of the potential sources of the backleakage. The RCDT pump suction piping connects to the RCDT, with an additional connection to the RHR system. As discussed above, the 350 gallon volume of the RCDT is insignificant in comparison with the required dilution volume of 183,000 gallons, and the weekly sampling of the SFP for boron (per Technical Specification SR 3.7.12.1) would identify that dilution was occurring due to the small continuous flow into the RCDT before the accumulated leakage would reach 183,000 gallons. Leakage from the RHR system could occur only if the RHR system were operating, but in that case, the leakage would result in decreasing reactor cooling system (RCS) inventory and the operators would identify the loss of inventory well before it approached the 183,000 gallon dilution limit.

4.3 Refueling Water Storage Tank

The SFP cooling system design includes a connection to the refueling water storage tank (RWST) to allow normal make-up to the system. The RWST can be aligned to supply water to the system via the discharge of the refueling water purification pump. This pump takes suction from the RWST and discharges to the SFP cooling system via the demineralizer and filter. The

filter discharge can also be recirculated back to the RWST (allowing the SFP demineralizer and filter to purify the RWST contents). Make-up to the SFP cooling system from the RWST cannot result in a dilution below 2300 ppm, since the RWST boron concentration is controlled by Technical Specification SR 3.5.4.2 which ensures at least 2300 ppm in the RWST.

4.4 Demineralized Water / Condensate

The design includes a connection to the Demineralized Water / Condensate system to allow make-up to the system, although this line is not normally used for make-up. The connection is made via a normally locked closed valve. The condensate system includes the condensate storage tanks (total volume of 160,000 gallons), and a makeup capacity of up to 160 gpm from the primary water treatment system (or 240 gpm if the temporary, truck-mounted reverse osmosis system is in service). Due to the elevation difference between the condensate storage tank (CST) and the connection to the SFP cooling system, the CST inventory cannot inadvertently dilute the SFP, further, the normally open valve aligning the treated water to the CST and the condenser hotwell would have to be throttled closed in order for the treated water to reach the connection to the SFP cooling system. In the unlikely event that the locked valve is mispositioned and the normally open valve is closed, the maximum dilution rate through this path would be 240 gpm, which does not challenge the 305 gpm limit, ensuring that the operators would observe the overflowing pool before the pool was diluted to 975 ppm. A low level in the condensate storage tanks would result in an alarm in the control room.

4.5 Waste Holdup Tank Line

Ion exchangers in the letdown portion of the CVCS, in the holdup tank portion of the CVCS, and in the SFP cooling system include provisions for backflushing to the waste holdup tank, through a common line. The SFP filter and the "A" heat exchanger drain may also be aligned to the WHT. Dilution of the SFP through these connections would require both the normally closed manual valve on the SFP demineralizer or filter to be leaking or inadvertently open, along with the backflush valve on one of the other ion exchangers to also be open or leaking. Also, unless the normally open valve in the line to the waste holdup tank is inadvertently closed, any leakage would likely be to the WHT, rather than into the SFP cooling system. Such leakage from the letdown ion exchangers would be identified as RCS leakage, and thus would be limited to 1 gpm with the plant in Modes 1 – 4. Leakage from the CVCS holdup tank recycle process ion exchangers would be limited to the 93,500 gallon capacity of the three tanks, at the processing rate of 12.5 gpm. This is well below the 183,000 gallon limit, ensuring the dilution would not reach 975 ppm. It is also within the 14 gpm limit, ensuring detection by the weekly surveillance of pool concentration.

4.6 Resin Fill

Reactor make-up water could be added to the system through the piping designed to fill the SFP demineralizer with new resin. A reactor make-up water line connects to the outlet of the resin batch hopper to create a resin slurry which can be aligned to the demineralizer. This alignment is made through a quick disconnect coupling which allows the single batch hopper to service a number of demineralizers. When the SFP demineralizer resin is recharged, the reactor make-up component of the resin slurry will be added to the SFP cooling system inventory. The small

volume of water involved in the resin addition process (about 1,000 gallons) is insignificant in comparison to the dilution volume limit of 183,000 gallons. Inadvertent dilution by continued addition of water after the resin addition is completed is unlikely, since the resin fill process is a closely controlled evolution performed in accordance with procedures, which ensure that the quick disconnect coupling would not remain connected to the demineralizer following completion of the resin fill. In the unlikely event that reactor make-up water remained aligned to the SFP demineralizer, depletion of the entire reactor make-up water tank volume of 75,000 gallons would not challenge the 183,000 gallon dilution limit.

4.7 Inadvertently Placing Resin Bed in Service Without Prior Boron Saturation

When placing a demineralizer bed into service with a fluid containing dissolved boric acid it is possible to initiate a boron dilution as the anion resin in the bed removes boron until it becomes "saturated." This event is typically more significant when it relates to the reactor coolant system since small changes in boron concentration have an immediate impact on a critical reactor. For borated spent fuel pool systems the effect is typically not significant since the pool volume is larger than the reactor coolant system volume and small changes in boron concentration do not have a significant reactivity effect.

For the Ginna spent fuel pool cooling system any postulated dilution from this mechanism is ultimately limited by the saturation of the resin bed. After saturation boron is no longer removed from the system. This potential dilution is limited to the equivalent of approximately 400 gallons of pure water being added to the spent fuel pool in a feed and bleed process over a period of approximately seven minutes. This volume is negligible when compared to the pool dilution limit of 183,000 gallons.

4.8 Spent Resin Tank

The SFP cooling system design includes a piping connection from the SFP demineralizer to the spent resin tank, allowing discharge of the spent resin from the SFP demineralizer. The ion exchangers in the letdown and holdup tank portions of the CVCS share the common line to the spent resin tank, a configuration similar to the common line to the waste holdup tank described in Section 4.5, above. Dilution through this path would require mispositioning or leakage by at least two valves. Leakage from the letdown ion exchangers would be identified as RCS leakage, and thus would be limited to 1 gpm with the plant in Modes 1 – 4. Leakage from the CVCS holdup tank ion exchangers would be limited to the 93,500 gallon capacity of the three tanks.

4.9 Letdown System

The system design includes two cross-ties from the letdown line (just upstream of the non-regenerative heat exchanger) to the suction and discharge of the refueling water purification pump. These lines are normally isolated by closed manual valves (two closed valves in the cross-tie to the purification pump discharge line and one valve in the cross-tie to the pump suction line). Leakage through one of these lines could result in dilution since the RCS boron concentration is normally less than the 2300 ppm SFP concentration.

In the unlikely event that the normally closed isolation valves leak or are inadvertently opened, a dilution rate as low as 1 gpm would be readily detected with the plant in Modes 1 - 4, since the

operators are required to monitor for RCS leakage of this magnitude. This is far less than the 305 gpm limit calculated in Section 3, ensuring that the operators would observe the overflowing pool before the pool was diluted to 975 ppm. While the 1 gpm Technical Specification limit on RCS leakage does not apply in Modes 5 and 6, the operators continue to monitor for changes in RCS inventory in these lower modes. A loss of inventory would soon be apparent by decreasing pressurizer or RCS level. Leakage from the RCS through the letdown line would be limited by the letdown orifices. The orifices are sized to limit the letdown flow to 140 gpm (all three orifices open) with the RCS at normal operating pressure. The flow through the orifices in the lower modes would be less than 140 gpm, well below the 305 gpm limit. In Mode 6, dilution from the RCS would not reduce the SFP concentration below the Technical Specification minimum required refueling boron concentration of 2300 ppm.

4.10 Service Water System

The service water system provides coolant to the shell side of the SFP heat exchangers. The heat exchanger tubes form a physical barrier between the service water and SFP cooling systems. The maximum pressure differential across the tubes would occur if the SFP cooling system (tube side) were depressurized while the service water system (shell side) was pressurized. If a tube leak were to occur under these conditions, the service water would enter the SFP cooling system, diluting the pool. In the unlikely event of a heat exchanger tube rupture, the break flow would be less than approximately 230 gpm (assuming a guillotine break of a 3/4-inch tube and a differential pressure of 80 psid). This well below the 305 gpm limit calculated in Section 3, ensuring that the operators would observe the overflowing pool before the pool was diluted to 975 ppm.

4.11 Instrument Air System

The instrument air system supplies the weir gate separating the transfer canal from the main portion of the SFP. The instrument air and the backup nitrogen systems are normally dry and would not be expected to be a source of dilution water. Any postulated condition in which water is introduced into the instrument air system (e.g., dryer malfunction, or inadvertent cross-tying of a liquid system to the air system) would involve a dilution volume far below the dilution limit of 183,000 gallons and a flow rate much less than the 305 gpm limit.

4.12 Auxiliary Building Sumps

The design includes a 3/4 inch line which routes any packing leakage from the Spent Fuel Pool Recirculation pumps to the auxiliary building sump. The sump is located below the floor at the 219 foot elevation, well below the spent fuel pool. This line cannot introduce dilution flow into the spent fuel pool.

4.13 Filling the Transfer Canal

Another possible dilution scenario would involve operator error when filling the fuel transfer canal. If the additional inventory necessary to fill the canal were inadvertently unborated water rather than borated water, a dilution event could result. The transfer canal volume is approximately 44,000 gallons, which is insufficient to challenge the 183,000 gallon dilution limit.

5.0 PIPE BREAKS AND LEAKS

A walkdown was performed to identify piping in the spent fuel pool area in order to evaluate the potential for dilution by water flowing into the pool from a leaking or ruptured line. Each line (or set of related lines) is addressed below.

5.1 VCT Sample and Return Lines

Sample and return lines from the volume control tank (VCT) pass through the SFP area on the north / northeast side of the pool. Sampling is performed intermittently rather than continuously, and the lines are isolated from the VCT when sampling is not in progress. In the unlikely event of leakage or a rupture of the lines, there would be no flow into the SFP area until the system was lined up for sampling, and the leak flow would be terminated when the sample line was restored to its normally isolated condition. Should the operators fail to isolate the line at the conclusion of sampling, they would detect the continued inventory loss from the VCT, since VCT level is closely monitored as an indication of RCS leakage. Since the Technical Specifications permit only very limited unidentified RCS leakage (1 gpm with the plant in Modes 1 - 4), continued loss of inventory from the RCS would be detected, investigated, and isolated long before any discernable dilution of the spent fuel pool occurred. A dilution flow of 1 gpm is well below the 14 gpm limit, ensuring that the dilution would be identified by the weekly sampling of the pool boron concentration. While the 1 gpm Technical Specification limit on RCS leakage does not apply in Modes 5 and 6, the operators continue to monitor for changes in RCS inventory in these lower modes. A loss of inventory would soon be apparent by decreasing pressurizer or RCS level. Leakage from a ruptured line in the SFP area would be observed by the operators performing their rounds. In Mode 6, dilution from the RCS would not reduce the SFP concentration below the refueling boron concentration of 2300 ppm.

5.2 Containment Sump Sample and Return Line

Sample and return lines from the containment sump are provided with flushing water which allows the lines to be purged with reactor makeup water (RMW). The lines supplying this RMW to the sample system pass through the SFP area on the north / northeast side of the pool. Sampling and flushing is performed intermittently rather than continuously, and the lines are isolated from the RMW system when sampling is not in progress. In the unlikely event of leakage or a rupture of the lines, there would be no flow into the SFP area until the system was lined up for flushing, and the leak flow would be terminated when the flushing line was restored to its normally isolated condition. Should the operators fail to isolate the line at the conclusion of sampling, the dilution flow would be limited to 1 gpm. A dilution flow of 1 gpm is well below the 14 gpm limit, ensuring that the dilution would be identified by the weekly sampling of the pool boron concentration. Leakage from a ruptured line in the SFP area would be observed by the operators performing their rounds.

5.3 PASS Waste Return

The post accident sample system (PASS) return line to containment passes through the SFP area on the north / northeast side of the pool. Sampling is performed intermittently rather than continuously, and the lines are depressurized when sampling is not in progress. In the unlikely

event of leakage or a rupture of the lines, there would be no flow into the SFP area until the system was lined up for sampling, and the leak flow would be terminated when the PASS was restored to its normal standby condition. With the system operating, the dilution flow would be limited to 1 gpm. A dilution flow of 1 gpm is well below the 14 gpm limit, ensuring that the dilution would be identified by the weekly sampling of the pool boron concentration. Leakage from a ruptured line in the SFP area would be observed by the operators performing their rounds.

5.4 Domestic Heating System

The SFP area is provided with a number of domestic heating units and the associated house heating steam and condensate return lines pass through the area on the north, east, and south sides of the pool. The house heating process steam lines supplying the boric acid evaporators also pass through the area on the north / northeast side. In the event that these lines leak or rupture, the maximum flow rate is not expected to exceed 160 gpm from the primary water treatment system (or 240 gpm if the temporary, truck-mounted reverse osmosis system is in service). In either case, the flow rate is well below the 305 gpm limit ensuring detection.

5.5 Fire Protection System

An 8-inch fire main enters the SFP area on the north / northeast side of the pool, running east and then vertically downwards to a hose station. The main fire protection system piping is normally pressurized by the head tank. In the event of a leak or rupture of the fire main the system pressure would decrease, and the electric driven fire pump would automatically start on low pressure. If the pressure decrease is sufficiently rapid, the diesel driven fire pump may also automatically start. Each pump has a capacity of 2,000 gpm. The pumps take suction from the lake. The fire protection system is thus a high flow capacity, infinite source of water. A walkdown of the fire protection system piping in the SFP area shows that it is highly unlikely that a break would provide a dilution source, since the break would have to occur in specific portions of the piping, and the resulting jet would have to be directed through a specific narrow angle in order to impinge on the pool. If this were to occur, the operators would be aware that the fire pump had started (since auto start of the pump is annunciated in the main control room).

Since the maximum system operating temperature and pressure are below 200 °F and 275 psig, this line may be evaluated as a moderate energy line in accordance with Branch Technical Positions (BTP) SPLB 3-1 and MEB 3-1. In accordance with BTP MEB 3-1, a through wall crack is postulated with the leakage based on a circular crack area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width. For the 8-inch fire main, the resulting leak flow is less than 290 gallons per minute. Assuming the entire 290 gpm leak flow is diluting the pool, more than 10 hours would be required to reach the dilution volume of 183,000 gallons. This time is conservative, since the pool level would rise to the top of the curb and begin overflowing, and the fire system water raining down on the pool surface would then be expected to immediately overflow. The 183,000 gallon dilution volume, however, conservatively assumes perfect mixing in the pool. Ten hours is considered more than sufficient time for the operators to identify the break and stop the pumps or isolate the affected piping. Control room alarms would indicate an automatic start of the fire pump and high level in the SFP as the pool level rises. The 290 gpm flow rate is also below the 305 gpm limit, ensuring that the operators would observe the overflowing pool before the pool was diluted to 975 ppm.

5.6 Component Cooling Water System

Component cooling water (CCW) supply and return lines to sample coolers in the Intermediate Building run through the SFP area along the east / southeast side of the pool. In the unlikely event of a leak or pipe break in the CCW piping which acts to dilute the SFP, the volume which could dilute the pool is limited to approximately 2,000 gallons (based on the maximum volume of the surge tank). Decreasing surge tank level would result in an alarm in the main control room.

It is unlikely that intermittent routine makeup to the CCW system could mask a slow leak to the pool, such that the operators would not detect that leakage and dilution were taking place. This is unlikely because the CCW piping does not run above the pool, so slow leakage would drip to the floor, not into the pool. Also, makeup to the system is manual, not automatic. Since the CCW system is chromated to inhibit biological fouling and corrosion, the operators would question the need for makeup to the system and investigate for leaks.

5.7 Waste Water System

Waste water release piping and waste water transfer piping run through the SFP area along the south and east sides of the pool. They provide for transfer of potentially contaminated liquid wastes (from the laundry, hot showers, and chemical drains) to the monitor tanks, and from the monitor tanks to the discharge for releases. These lines are used only infrequently during normal power operation, and intermittently during shutdowns. A leak or break in these lines could drain the monitor tank into the area. In the unlikely event that the entire inventory of both monitor tanks (7,500 gallons each) drained into the pool, the 183,000 dilution limit would not be challenged.

5.8 Caustic Line

The caustic line runs through the SFP area along the east / southeast side of the pool. This line is abandoned in place, drained, isolated, and the tank has been removed. The line poses no threat of dilution.

5.9 Condensate System

Demineralized Water / Condensate system lines run through the SFP area along the south, east, and west sides of the pool. This includes 2 and 3-inch piping associated with the makeup connection discussed above in section 4, as well as hot water supply outlets and other portions of the system. The condensate system includes the condensate storage tanks (total volume of 160,000 gallons), and a makeup capacity of up to 160 gpm from the primary water treatment system (or 240 gpm if the temporary, truck-mounted reverse osmosis system is in service). Due to the elevation difference between the CST and the connection to the SFP cooling system, leakage or a break in this piping could not drain the CST inventory into the SFP area, further, the normally open valve aligning the treated water to the CST and the condenser hotwell would have to be throttled closed in order for the treated water to reach the SFP area. In the unlikely event that leakage or a pipe break occurs, and the normally open valve is closed, and the leak / break flow is directed into the pool, the maximum dilution rate would be 240 gpm, which does not challenge the 305 gpm limit.

5.10 Roof Drains

The auxiliary building roof is provided with drains to remove rainwater and snow melt. A portion of the drain piping passes through the SFP area along the south wall. The drain piping is not located above the pool, so any leakage from the drain line would drip to the floor, not into the pool. In the unlikely event that the roof leaks into the pool, or the drain line breaks and somehow discharges into the pool, dilution could occur. The auxiliary building high roof is provided with three drains. The building parapet does not enclose the entire high roof area, and a backup scupper system is available if the roof drains fail. In the unlikely event that all three drains clog, and the scuppers fail to drain the roof, the total impounded volume would be approximately 25,000 gallons. Sudden release of this inventory into the pool would not challenge the 183,000 gallon dilution limit. Assuming that the entire roof surface drains into the pool (i.e., that two drains are clogged, scuppers fail, and with the third drain line ruptured and discharging to the pool), a rainfall of greater than 44 inches would be required in order to challenge the 183,000 gallon dilution limit. This is not considered to be credible, in light of available historical rainfall data which indicates that the maximum 24-hour rainfall event at Ginna Station is 29.4 inches.

6.0 CONCLUSIONS

An analysis of potential scenarios which dilute the boron concentration in the spent fuel pool has been performed. The analysis demonstrates that sufficient time is available to detect and mitigate the dilution prior to exceeding the 0.95 k_{eff} design basis. The potential plant events were quantified to show that sufficient time is available to enable adequate detection and suppression of any dilution event.

The criticality analysis demonstrates that a soluble boron concentration of 975 ppm will maintain reactivity within the design basis limit of $k_{eff} \leq 0.95$ (including all biases and uncertainties) with a 95% probability at a 95% confidence level. Deterministic dilution event calculations were performed for the R. E. Ginna plant to define the dilution times and volumes necessary to dilute the 213,600 gallon SFP inventory from the minimum 2300 ppm required by the plant Technical Specifications to a soluble boron concentration of 975 ppm. Assuming a well mixed pool, the volume required to dilute the pool from 2300 to 975 ppm was determined to be 183,000 gallons.

The largest possible flow rate for dilution of the pool is 290 gpm due to a through wall crack in the 8-inch fire main. This line does not run above the pool, but it is conservatively assumed that the break flow jet impinges on the pool surface. The fire system draws from the lake, an essentially limitless supply of water. At a flow rate of 290 gpm, over ten hours would be required to dilute the pool from the Technical Specification limit of 2300 ppm to 975 ppm. A rupture of this fire main would result in many alarms which would alert the operators to this potential dilution event within minutes. The expected alarms include auto start of the fire pumps and high spent fuel pool level. Ten hours is more than sufficient time for the operators to identify the location of the pipe break and stop the fire pumps or isolate the affected piping.

The only tank large enough to provide the dilution volume without replenishment would be the refueling water storage tank (RWST), which contains water borated to the same concentration as the SFP and thus poses no possibility of dilution. The largest source of unborated water would be the condensate storage tanks, with a combined volume of 160,000 gallons. This is less than the dilution volume of 183,000 gallons and is therefore not capable of diluting the pool to 975 ppm.

Based on the above evaluation, an unplanned or inadvertent event which would reduce the SFP boron concentration from 2300 ppm to 975 ppm is not credible for R. E. Ginna. The large volume of water required for a dilution event, the Technical Specification controls on SFP boron concentration, the plant personnel rounds, and the 7-day sampling interval for the pool concentration would adequately detect a dilution event prior to K_{eff} reaching 0.95 (at 975 ppm).

Finally, the criticality analysis for the spent fuel pool shows that the stored assemblies would remain subcritical even if the pool were completely filled with unborated water.

Enclosure 2
R.E. Ginna Nuclear Power Plant

VENDOR DRAWING/TECHNICAL DOCUMENT REVIEW

☒ Approved - Mfg. may proceed (if applicable)

☐ Approved - Submit final dwg. - Mfg. may proceed. (if applicable)

☐ Approved except as noted. Make changes and submit final dwg/technical document. Mfg. may proceed as approved

☐ Review not required - Mfg. may proceed.

Approval of this drawing/technical document does not relieve supplier from full compliance with contract or purchase order requirements.

By PJ Bamford Date 3/3/00

ROCHESTER GAS & ELECTRIC CORP.
ROCHESTER, NY

Enclosure 2 to RGE-99-0009 Rev. 02

Final Report

R.E. Ginna Nuclear Power Plant Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit

February 2000

Verification Status: Complete

The safety related design information contained in this document has been verified to be acceptable for the intended use by means of Design Review using the applicable portions of QP 3.10 of QPM-101 Revision 3.

	Printed Name	Signature	Date
Cognizant Engineers	P. Narayanan	A Prakash	02/29/2000
Independent Reviewers	P. O'Donnell	Paul. O'Donnell	02/29/2000
Management Approval	M. Greene	Mel A Gr	2/29/2000

Revision 01 replaced Page 1 only, to document the removal of contingencies associated with Revision 00.

Revision 02 replaces Pages 1, 38 (to clarify text), and 119 (to update Reference 2).

ABB Combustion Engineering Nuclear Power

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1.0 Introduction

This report presents the results of an updated criticality analysis for the R. E. Ginna Nuclear Power Plant spent fuel storage pool. The physical description of the spent fuel storage racks is unchanged from that in the Spent Fuel Pool Re-racking Licensing Report, Reference 1. The primary objectives of this updated analysis are as follows:

1. to employ Soluble Boron Credit in establishing the design/storage basis for the spent fuel pool,

2. to establish the design basis fuel assembly for each cell type,

3. to establish assembly burnup versus initial enrichment limits assuming total loss of the Boraflex originally incorporated in the Region 2, Type 1 spent fuel storage cells,

4. to establish assembly burnup versus initial enrichment limits for Region 2, Type 2 spent fuel storage cells,

5. to demonstrate the assembly burnup versus initial enrichment limits for Region 2, Type 2 cells are applicable to the Region 2, Type 4 storage cells, and

6. to demonstrate the existing fuel assembly burnup versus initial enrichment criteria are applicable for Region 1, Type 3 cells.

Please note that for the Region 1, Type 3 storage racks, reactivity control is achieved by means of a checkerboard of burned and fresh fuel assemblies having initial enrichments of up to 5.0 wt% ^{235}U(nominal); no Integrated Fuel Burnable Absorber (IFBA) credit for fresh fuel assemblies with nominal enrichments above 4.0 wt% ^{235}U is required. Region 2 accommodates burned fuel assemblies having initial enrichments up to 5 wt% ^{235}U(nominal) at prescribed minimum burnups.

The methodology employed in this analysis for soluble boron credit is analogous to that of Reference 2 and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 3.

1.1 Design Criteria

The design criteria are consistent with GDC 62, Reference 4, and NRC guidance to all Power Reactor Licensees, Reference 5. Section 2.0 describes the analysis methods including a description of the computer codes used to perform the criticality safety analysis. A brief summary of the analysis approach and criteria follows.

1. Determine the fresh and spent fuel storage configuration of the spent fuel pool using no soluble boron conditions such that the 95/95 upper tolerance limit value of K_{eff} for the storage pool, including applicable biases and uncertainties, is less than unity.

2. Next, using the resulting storage configuration from the previous step, calculate the spent fuel rack effective multiplication factor with the chosen concentration of spent fuel pool soluble boron present. Then calculate the sum of: (a) the latter multiplication factor, (b) the reactivity uncertainty associated with fuel assembly and storage rack tolerances, and (c) the biases and other uncertainties required to determine the final 95/95 confidence level effective multiplication factor and show that at the chosen concentration of soluble boron, the system maintains the overall effective multiplication factor less than or equal to 0.95.

3. Use reactivity equivalencing methodologies to determine the minimum fuel assembly burnup for fuel assembly enrichments higher than allowed in Step 1, above. As a function of time after discharge and burnup, calculate the reactivity credit due to actinides for each fuel assembly.

4. Determine the increase in reactivity caused by postulated accidents and the corresponding additional amount of soluble boron needed to offset these reactivity increases.

An alternative form of expressing the soluble boron requirements is given in Reference 3. The final soluble boron requirement is determined from the following summation:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm),

$SBC_{95/95}$ = soluble boron requirement for 95/95 $K_{eff} \leq 0.95$ (ppm),

SBC_{RE} = soluble boron required for reactivity equivalencing methodologies (ppm),

SBC_{PA} = soluble boron required for $K_{eff} \leq 0.95$ under accident conditions (ppm).

For purposes of the analyses contained herein, minimum burnup limits established for fuel assemblies to be stored in the different types of storage racks do include burnup credit established in a manner which takes into account conservative approximations to the operating history of the fuel assemblies. Variables such as the axial burnup profile as well as the axial profile of moderator and fuel temperatures have been factored into the analyses.

1.2 Analysis Results

The primary objectives of this analysis were accomplished; a summary of the results is as follows.

1) Soluble boron credit methodology was employed to establish a target K_{eff} value of 0.98051 for the spent fuel pool at zero soluble boron. The allowance for applicable biases and uncertainties was deduced to be 0.01592; thus, the 95/95 upper tolerance limit value of K_{eff} was deduced to be 0.99643. The total soluble boron requirement for achieving a 95/95 value of $K_{eff} \leq 0.95$ was deduced to be the summation of the following three terms: $SBC_{95/95}$ = 377 ppm, SBC_{RE} = 207 ppm, and SBC_{PA} = 381 ppm for a total of 965 ppm. The soluble boron concentration was increased by 1% due to the difference in the ^{10}B atom percent used in the analysis (19.9 a/o) and that measured at Ginna (19.7 a/o). This results in a soluble boron concentration equal to 975 ppm. Note that this soluble boron concentration includes an allowance for 5 % burnup uncertainty. In addition, all of the burnup versus enrichment storage curves have been increased by 5 %. Therefore, a 5 % burnup uncertainty has been double counted.

2) The design basis fuel assembly for the fresh fuel storage cells in the Region 1, Type 3 racks was taken to be a conservative representation of the Westinghouse OFA 14 x 14 fuel assembly having a nominal enrichment of 5 wt% ^{235}U, no IFBA loadings, and the instrument tube location replaced by a fuel rod. The design basis fuel assembly for the burned fuel storage cells in both Region 1 and 2 racks was taken to be conservative approximation to the Westinghouse Standard 14 x 14 fuel assembly wherein the RCC guide tubes were represented as zircalloy-4 and the instrument tube was replaced by a fuel rod. This conservative approximation to the burned fuel assembly envelopes the characteristics of all burned fuel assemblies, including lead test assemblies, currently stored in the spent fuel pool. This design basis burned fuel assembly was represented by an 8-node axial representation of the assembly burnup and applicable fuel and moderator temperatures.

3) All representations of the Region 2, Type 1 spent fuel storage racks, originally containing boraflex inserts between the L-shaped insert and the storage cell tube wall, were represented in both the infinite cell array and full storage pool analyses as having nominal pool water in place of the boraflex.

4) Minimum fuel assembly burnup limits versus fuel assembly initial average enrichment were established for Region 2, Type 2 spent fuel storage cells. These limits were established on both a nominal basis and an equivalent dual tier approach for 0, 5, 10, 15, and 20 years of ^{241}Pu decay so as to provide more efficient utilization of the available spent fuel storage capacity of the storage racks.

5) It was demonstrated that the burnup limits established in 4), above, were applicable to the Region 2, Type 4 storage cells.

6) It was demonstrated that the existing fuel assembly burnup versus initial enrichment criteria established in Reference 1 are applicable for Region 1, Type 3 cells. These analyses also demonstrated this objective is easily achieved with fresh fuel enrichments of 5 wt% ^{235}U and no requirements for IFBA credit in the fresh fuel assemblies.

It was further established that either a fuel rod consolidation canister or a damaged rod storage basket is less reactive than a fuel assembly of equivalent burnup when placed in a spent fuel storage cell. Consequently, there are no restrictions as to placement of these storage devices in the spent fuel storage cells.

The analyses contained herein lead to the conclusion that the total soluble boron concentration required to maintain K_{eff} less than 0.95, after including all biases and uncertainties and assuming the most limiting accident, is less than or equal to 965 ppm (assuming a B-10 atomic fraction equal to .199). This latter value is composed of three values:

1) a 377 ppm requirement for $K_{eff} \leq 0.95$,
2) a 207 ppm requirement for reactivity equivalencing methodologies, and
3) a 381 ppm requirement to maintain $K_{eff} \leq 0.95$ for the most limiting accident condition.

The most limiting accident condition was determined to be the misloading of a fresh 5 wt% ^{235}U fuel assembly in the Region 2 Type 1 fuel racks.

Section 10 of this report contains the plots defining the limits on storage of spent fuel assemblies versus assembly burnup, initial enrichment, and years of ^{241}Pu decay.

1.3 Design Approach

The design input employed in this analysis is basically the same as that employed in the Ginna SFP Reracking Licensing Report, Reference 1. However, the current analyses employ a broader scope by implementing the Soluble Boron Credit Methodology, taking credit for the decay of ^{241}Pu, and quantifying the spent fuel storage limits for Region 2, Type 1 and 2 Racks independently. The Soluble Boron Credit Methodology provides additional reactivity margin in the spent fuel storage analyses which may then be used to implement added flexibility in storage criteria and, for example, eliminate the need to

implement the degraded boraflex modeling as well as eliminate credit for IFBA in fresh fuel assemblies with enrichments above 4 wt% ^{235}U.

The selection of design basis fuel assembly types was based on an evaluation of the variety of fuel assemblies employed in the reactor to date and selecting the most reactive type for a given evaluation. The candidate fuel assembly types include the Westinghouse Standard, Westinghouse OFA, Mixed Oxide fuel assemblies and other Lead Test Assemblies, as well as the Consolidated Fuel Assembly Canisters and the damaged fuel rod basket.

The selection of the Westinghouse OFA as the design basis fresh fuel assembly is predicated on the fact that this assembly is an optimized design and is more reactive than the Westinghouse Standard in the fresh fuel condition. This result is consistent with the analyses of Reference 1; the latter analyses also conclude that the Westinghouse Standard assembly becomes more reactive than the OFA assembly beyond burnups greater than about 12,000 MWD/T burnup. Thus, the design basis burned fuel assembly employed for these analyses is taken to be a variant on the Westinghouse Standard fuel assembly because of its burnup characteristics and the fact that it is, in general, more representative of fuel assemblies employed in past operation of the plant. The design basis burned fuel assembly is taken to be a Westinghouse Standard fuel assembly with the instrument tube replaced by a fuel rod and the RCC guide tubes made of zircaloy. These changes were simply added conservatisms to assure enveloping of the variety of fuel assemblies that had passed through the core and presently reside in the spent fuel pool.

The reactivity characteristics of the different rack Types 1, 2, and 3 were evaluated using infinite lattice analyses; this environment was employed in the evaluation of the burnup limits versus initial enrichment for each rack type as well as the evaluation of physical tolerances and uncertainties. The equivalence between rack Types 4 and 2 was established in the full core model. The full core model was also employed to evaluate soluble boron worths and the reactivity worth of postulated accidents.

2.0 Analysis Methods

This section describes the analysis methodology employed to assure the criticality safety of the spent fuel pool and to define limits placed on fresh and spent fuel storage in the Regions 1 and 2 of the spent fuel pool. The analysis methodology employs: (1) SCALE-PC, a personal computer version of the SCALE-4.3 code system, as documented in Reference 6, with the updated SCALE-4.3 version of the 44 group ENDF/B-V neutron cross section library, and (2) the two-dimensional integral transport code DIT, Reference 7, with an ENDF/B-VI neutron cross section library.

SCALE-PC is used for calculations involving infinite arrays of storage cells and checkerboarded storage cells depending on the storage features of individual rack types. In addition, it is employed in a full pool representation of the storage racks to evaluate soluble boron worths and postulated accidents.

SCALE-PC modules employed in both the benchmarking analyses and the spent fuel storage rack analyses include the control module CSAS and the following functional modules: BONAMI, NITAWL-II, and KENOV.a. All references to KENO in the text to follow should be interpreted as referring to the KENO V.a module.

The DIT code is used for simulation of in-reactor fuel assembly depletion. The following sections describe the application of these codes in more detail.

2.1 SCALE-PC

The SCALE system was developed for the Nuclear Regulatory Commission to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system which runs on specific classes of personal computers.

2.1.1 Validation of SCALE-PC

Validation of SCALE-PC for purposes of fuel storage rack analyses is based on the analysis of selected critical experiments from two experimental programs. The first program is the Babcox & Wilcox (B&W) experiments carried out in support of Close Proximity Storage of Power Reactor Fuel, Reference 8. The second program is the Pacific Northwest Laboratory (PNL) Program carried out in support of the design of Fuel Shipping and Storage Configurations; the experiments of current interest to this effort are documented in Reference 9. Reference 10, as well as several of the relevant thermal experiment evaluations in Reference 11, were found to be useful in updating pertinent experimental data for the PNL experiments.

Nineteen experimental configurations were selected from the B&W experimental program; these consisted of the following experimental cores: Core X, the seven measured configurations of Core XI, Cores XII through XXI, and Core XIIIa. These analyses employed measured critical data, rather than the extrapolated configurations to a fixed critical water height reported in Reference 8, so as to avoid introducing possible biases or added uncertainties associated with the extrapolation techniques. In addition to the active fuel region of the core, the full environment of the latter region, including the dry fuel above the critical water height, was represented explicitly in the analyses.

The B&W group of experimental configurations employed variable spacing between individual rod clusters in the nominal 3 x 3 array. In addition, the effects of placing either SS-304 or B/Al plates of different boron contents in the water channels between rod clusters were measured. Table 2.1-1 summarizes the results of these analyses.

Eleven experimental configurations were selected from the PNL experimental program. These experiments included unpoisoned uniform arrays of fuel pins and 2 x 2 arrays of rod clusters with and without interposed SS-304 or B/Al plates of different blacknesses. As in the case of the B&W experiments, the full environment of the active fuel region was represented explicitly. Table 2.1-2 summarizes the results of these analyses.

The approach employed for a determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of Reference 12. For a given KENO calculated value of K_{eff} and associated one sigma uncertainty, the magnitude of $K_{95/95}$ is computed by the following equation; by this definition, there is a 95 percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than $K_{95/95}$.

$$K_{95/95} = K_{KENO} + \Delta K_B + M_{95/95}\left(\sigma_m^2 + \sigma_{KENO}^2\right)^{1/2}$$

where:

K_{KENO} is the KENO multiplication factor of interest,

ΔK_B is the mean calculational method bias,

$M_{95/95}$ is the 95/95 multiplier appropriate to the degrees of freedom for the number of validation analyses,

σ_m^2 is the mean calculational method variance deduced from the validation analyses, and

σ^2_{KENO} is the standard deviation appropriate to the KENO multiplication factor of interest.

The equation for the mean calculational methods bias is as follows.

$$\Delta K_B = \frac{1}{n}\sum_{i=1}^{n}(1-K_i)$$

where:

K_i is the i^{th} value of the multiplication factor for the validation lattices of interest, and

$M_{95/95}$ is obtained from the tables of Reference 13.

The equation for the mean calculational variance of the relevant validating multiplication factors is as follows.

$$(\sigma_m)^2 = \left[\frac{n\sum_1^n (K_i - K^{ave})^2 \sigma_i^{-2}}{(n-1)\sum_1^n (\sigma_i)^{-2}}\right] - \sigma_{ave}^2$$

Where k^{ave} is given by the following equation.

$$K^{ave} = \sum_1^n K_i(\sigma_i)^{-2} \Big/ \sum_1^n (\sigma_i)^{-2}$$

σ^2_{ave} is given by the following equation.

$$\sigma^2_{ave} = \sum_1^n (\sigma_i)^2 G_i \Big/ \sum_1^n G_i$$

Where G_i is the number of generations.

For purposes of this bias evaluation, the data points of Tables 2.1-1 and 2.1-2 are pooled into a single group. With this approach, the mean calculational methods bias, ΔK_B, and the mean calculational variance, $(\sigma_m)^2$, calculated by equations given above, are determined to be 0.00259 and $(0.00288)^2$, respectively. The magnitude of $M_{95/95}$ is deduced from Reference 13 for the total number of pooled data points, 30.

The magnitude of $K_{95/95}$ is given by the following equation for SCALE 4.3 KENO analyses employing the 44 group ENDF/B-V neutron cross section library and for analyses where these experiments are a suitable basis for assessing the methods bias and calculational variance.

$$K_{95/95} = K_{KENO} + 0.00259 + 2.22\,[0.00288^2 + (\sigma_{KENO}{}^2)]^{1/2}$$

Based on the above analyses, the mean calculational bias, the mean calculational variance, and the 95/95 confidence level multiplier are deduced as 0.00259, $(0.00288)^2$, and 2.22, respectively.

2.1.2 Application to Fuel Storage Pool Calculations

As noted above, the CSAS control module was employed to execute the functional modules within SCALE-PC. The CSAS25 control module was used in the majority of the cases to analyze either infinite arrays of single or multiple storage cells or the full spent storage pool.

Standard material compositions were employed in the SCALE-PC analyses consistent with those of Reference 1; these data are listed in Table 2.1-3. For fresh fuel conditions, the fuel nuclide number densities were derived within the CSAS module using input consistent with the data of Table 2.1-3. For burned fuel representations, the fuel isotopics were derived from the DIT code as described below.

2.2 The DIT Code

The DIT (Discrete Integral Transport) code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The neutron transport equations are solved in integral form within each pin cell. The cells retain full heterogeneity throughout the discrete integral transport calculations. The multigroup spectra are coupled between cells through the use of multigroup interface currents. The angular dependence of the neutron flux is approximated at cell boundaries by a pair of second order Legendre polynomials. Anisotropic scattering within the cells, together with the anisotropic current coupling between cells, provide an accurate representation of the flux gradients between dissimilar cells.

The multigroup cross sections are based on the Evaluated Nuclear Data File Version 6 (ENDF/B-VI). Cross sections have been collapsed into an 89 group structure which is used in the assembly spectrum calculation. Following the multigroup spectrum calculation, the region-wise cross sections within each heterogeneous cell are collapsed to a few groups (usually 4 broad groups), for use in the assembly flux calculation. A B1 assembly leakage correction is performed to modify the spectrum according to the

assembly in- or out-leakage. Following the flux calculation, a depletion step is performed to generate a set of region-wise isotopic concentrations at the end of a burnup interval. An extensive set of depletion chains are available, containing 33 actinide nuclides in the thorium, uranium and plutonium chains, 171 fission products, the gadolinium, erbium and boron depletable absorbers, and all structural nuclides. The spectrum-depletion sequence of calculations is repeated over the life of the fuel assembly. Several restart capabilities provide the temperature, density and boron concentration dependencies needed for three dimensional calculations with full thermal-hydraulic feedback effects.

The DIT code and its cross section library are employed in the design of initial and reload cores and have been extensively benchmarked against operating reactor history and test data.

For the purpose of spent fuel pool criticality analysis calculations, the DIT code is used to generate the detailed fuel isotopic concentrations as a function of fuel burnup and initial feed enrichment. Each selected set of fuel isotopics is equivalenced to a reduced set of burned fuel isotopics at specified time points after discharge. The latter burned fuel representation includes the following nuclides: ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{149}Sm, ^{16}O, and ^{10}B. The DIT code lists the Samarium-149 isotopics for ^{149}Sm and ^{149D}Sm (a metastable isomer). Since, ^{149}Sm is a stable isotope, the concentration of this Samarium isotope is the sum of the individual concentration of these two isomers.

The isotopic number densities from the DIT calculation are based upon Cell average values. The input to KENO calculations require that the number densities be specified for the fuel pellet. Therefore, the number densities from the DIT calculations are scaled by the ratio of area of the cell to the area of the fuel pellet for use in the KENO calculations. The concentration of ^{10}B is determined by reactivity equivalencing a given DIT cell calculation with a corresponding KENO cell calculation to within the KENO one sigma uncertainty level.

Table 2.1-1
Summary of Calculational Results for
Cores X Through XXI of the B&W Close Proximity Experiments

Core	Run No.	KENO K_{eff}	Plate Type [a]	Spacing [b]
X	2348	0.99610 ± 0.00084	none	3
XI	2355	1.00049 ± 0.00080	SS-304	1
XI	2359	0.99884 ± 0.00077	SS-304	1
XI	2360	1.00315 ± 0.00081	SS-304	1
XI	2361	0.99831 ± 0.00080	SS-304	1
XI	2362	1.00060 ± 0.00078	SS-304	1
XI	2363	0.99957 ± 0.00078	SS-304	1
XI	2364	1.00246 ± 0.00080	SS-304	1
XII	2370	0.99990 ± 0.00082	SS-304	2
XIII	2378	0.99754 ± 0.00089	B/Al	1
XIIIA	2423	0.99575 ± 0.00087	B/Al	1
XIV	2384	0.99465 ± 0.00086	B/Al	1
XV	2388	0.99158 ± 0.00084	B/Al	1
XVI	2396	0.99230 ± 0.00088	B/Al	2
XVII	2402	0.99478 ± 0.00079	B/Al	1
XVIII	2407	0.99440 ± 0.00083	B/Al	2
XIX	2411	0.99821 ± 0.00081	B/Al	1
XX	2414	0.99498 ± 0.00082	B/Al	2
XXI	2420	0.99318 ± 0.00094	B/Al	3

(a) - metal separating unit assemblies

(b) - spacing between unit assemblies in units of fuel rod pitch

Table 2.1-2
Summary of Calculational Results for Selected Experimental PNL Lattices, Fuel Shipping and Storage Configurations

Exp't. No.	K_{eff}	Comments
043	0.99787 ± 0.00106	Uniform rectangular array, no poison
044	1.00104 ± 0.00102	"
045	0.99955 ± 0.00101	"
046	0.99960 ± 0.00103	"
061	0.99792 ± 0.00099	2 x 2 array of rod clusters, no poison
062	0.99628 ± 0.00096	"
064	0.99696 ± 0.00103	2 x 2 array of rod clusters, 0.302 cm thick SS-304 cross
071	0.99970 ± 0.00101	2 x 2 array of rod clusters, 0.485 cm thick SS-304 cross
079	0.99463 ± 0.00102	2 x 2 array of rod clusters, cross of 0.3666 g boron/cm^2
087	0.99423 ± 0.00099	2 x 2 array of rod clusters, cross of 0.1639 g boron/cm^2
093	0.99787 ± 0.00098	2 x 2 array of rod clusters, cross of 0.1425 g boron/cm^2

Table 2.1-3
Standard Material Compositions Employed in Criticality Analysis for R. E. Ginna Nuclear Power Plant Spent Fuel Storage Rack

Material	Element	Weight Fraction
Stainless Steel 304L	Cr	0.180
Den.= 8.0 g/cc	Mn	0.020
	Fe	0.720
	Ni	0.080
Borated Stainless	Cr	0.180
Steel	Mn	0.020
Den.= 7.73 g/cc	Fe	0.663
	Ni	0.120
	B	0.017
Zircaloy-4	Zr	0.9829
Den.= 6.56 g/cc	Sn	0.0140
	Fe	0.0021
	Cr	0.0010
Water	SCALE Standard Composition Library w/ Den. = 0.9982 @ 293 °K	
Concrete	SCALE Standard Composition Library	
Fresh UO_2	Fraction of Theoretical Density = 0.95*(1- 0.01187)	

3.0 Description of Spent Fuel Pool Storage Rack Configurations

As noted in the Introduction Section, the Ginna spent fuel storage pool configuration and the Region 1 and 2 storage racks as analyzed herein are consistent with Reference 1. This section provides a brief description of the spent fuel storage rack with the objective of establishing a basis for the analytical model employed in the criticality analyses described in Section 4.0.

3.1 Spent Fuel Pool Storage Configuration Description

The spent fuel pool and the fuel storage rack types and orientation are illustrated in Figure 3.1-1. In addition to the cask loading and the fuel elevator areas, the pool is divided into Regions 1 and 2. Region 1 consists of five Type 3 storage racks nominally employing borated stainless steel panels in each storage cell as parasitic absorbers. The Type 3 racks are designed to store fresh and burned fuel assemblies in a checkerboard array; the fresh assemblies are placed in the cells with the lead-in funnels and the burned assemblies are placed in the alternate cells. In addition, five storage locations adjacent to the cask area, as illustrated in Figure 3.1-1, are available with enlarged internal dimensions for storage of damaged fuel assemblies.

Region 2 consists of the following three types of storage modules for burned fuel assemblies:

1. Six Type 1 modules which were originally fabricated with Boraflex absorber regions within two of the walls of each storage cell,

2. Two Type 2 modules employing borated stainless steel panels and of a similar design as the Region 1, Type 3 storage modules, and

3. Six Type 4 modules employing ten storage cells in a linear array; these cells are of a similar design as those in the Type 2 modules. [Note : Although licensed by Reference 1, these modules are not yet installed.]

Changes in the spatial distribution of the Boraflex absorber material under irradiation resulted in the decision to take no credit for the presence of the Boraflex within the six Type 1 storage modules for purposes of this re-analysis.

3.2 Individual Storage Rack Type Descriptions

Subsequent sections describe the different types of Region 1 and 2 storage racks in greater detail.

3.2.1 Region 1, Type 3 Racks.

The Region 1, Type 3 storage racks accommodate fresh and spent fuel assemblies in a checkerboard pattern; these storage racks are free standing and self-supporting. The fuel cells are of two types and are distinguished apart by the presence or absence of square SS funnels to guide the fresh fuel assembly into the storage cells. The fresh fuel storage cells are created from four borated stainless steel (BSS) sheets linked together at the corners and resting on the base plate. Four millimeter high horizontal stainless steel belts maintain the square geometry of the four BSS sheets at eight axial locations and preserve the spacing of 0.76 inches between the latter cell and the adjacent SS tubes.

The alternate storage cells of the checkerboard are intended for burned fuel assemblies and consist of square stainless steel (SS) tubes welded into precut grooves in the baseplate of the module. Four BSS panels, which are linked together at the corners, are positioned vertically within the SS tubes by lower and upper tabs welded to the internal SS tube walls. The four segments of the square SS funnels which serve to guide the fuel assembly into the laterally adjacent fresh fuel storage cells are actually welded to the top of the SS tube encompassing the burned fuel storage locations.

In both types of storage cells along the periphery of the Type 3 racks and adjacent to either the pool wall, fuel elevator area, or the cask area, the BSS panel facing the pool wall, fuel elevator area, or cask area is replaced by a SS panel. In addition, the BSS fresh fuel cells located along the edge of the Type 3 racks incorporate a SS closure plate to maintain the structural integrity of the rack and protect the BSS plate. The stainless steel belts surrounding the BSS cells along the periphery of the rack are modified to reduce the separation between the BSS plates and the closure plate to 0.31 inches. The pertinent dimensions of the constituent materials for the typical Region 1, Type 3 rack storage cells are summarized in Table 3.2-1.

The Region 1 Rack 3E, adjacent to the cask area, contains five "damaged" fuel assembly storage cells along the side of the rack facing the cask area. These storage cells are designed to have a larger internal area to accommodate damaged assemblies and a 20% thicker BSS wall plate as noted in Table 3.2-2, entitled "Region 1, Rack Type 3 Damaged Fuel Cell Dimensions". Since these cells have higher BSS content, the results of the Region 1 Type 3 (regular cells) will also be conservatively applicable to these cells. Therefore, no modeling of these cells is done in this analysis.

3.2.2 Region 2, Type 1 Racks

The Region 2, Type 1 storage racks employ a 10 x 14 array of identical storage cells consisting of a square SS tube with an L-shaped SS insert. The gap between the L-insert and the interior of the square tube wall originally contained a Boraflex poison sheet; degradation of the physical characteristics of this poison material led to a decision to ignore the presence of any of the Boraflex material in this analysis. Replacement of the contents of this region by water is a conservative approximation since much of the Boraflex material remains in this region. Table 3.2-3 lists the pertinent dimensions of the Region 2, Type 1 storage cells. The Type 1 storage racks are oriented in the spent fuel pool so as to position all L-inserts on the north and east faces of the storage cells.

3.2.3 Region 2, Type 2 Racks

The Region 2, Type 2 storage racks consist of a checkerboard array of SS and BSS cells. The SS cells consist of a square tube which is welded into precut grooves in the baseplate of the rack. The SS tubes within the rack are joined together along their length by SS connecting tabs welded to the SS square tube faces. The adjacent BSS cells are composed of four BSS sheets linked together at the corners to form a square; these BSS sheets are positioned vertically by tabs welded to the exterior of the SS tubes and spaced away from the surface of the tube by the thickness of the connecting tabs linking the diagonally adjacent tubes. Table 3.2-4 lists the pertinent dimensions of the Region 2, Type 2 storage cells.

As in the case of the Region 1, Type 3 racks, the BSS cells located either along the edge of the rack or at a corner incorporate SS closure plates similar to those in the Region 1, Type 3 racks. However, in this case the closure plate is closely spaced to the exterior of the BSS storage cell and serves to preserve the spacing of the SS tube along the periphery of the rack as well as protect the exterior of the BSS cell. In the BSS storage cells along the periphery of the Type 2 racks and adjacent to the either the pool wall or the cask area, the BSS panels facing the pool wall or cask area are replaced by a SS panel.

3.2.4 Region 2, Type 4 Racks

The Region 2, Type 4 storage racks employ a linear array of ten storage cells. Individual storage cells consist of SS tubes welded into precut grooves in the baseplate; SS tabs are welded between tube faces to provide structural rigidity to the rack. BSS sheets are inserted between the SS tubes of adjacent storage cells; these BSS sheets rest on the base plate and extend the full height of the fuel assembly. On the sides of the Region 2, Type 4 racks facing the Region 2, Type 1 racks and the pool wall, there are no BSS sheets. The 4B and 4E storage racks each have eleven BSS panels; nine between the ten SS storage cells and one at each end with a supporting SS panel. The other four Type 4 racks have only the nine BSS panels between the ten SS storage cells. Thus, each of the two linear

arrays of the three Type 4 racks contains 30 storage cells with each storage cell separated by at least two SS panels with an intervening BSS panel. At the end of the array facing the pool wall the BSS is absent and at the opposite end, a BSS panel exists in either the Type 2 or 3 racks. The side of the Type 4 racks facing the Type 1 racks employs a minimum water gap of 3.0 cm between the two rack exteriors as a result of the linkage mechanism between the two rack types noted below. Pertinent dimensions of these Type 4 racks are summarized in Table 3.2-5.

The Type 4 racks each have two support legs below the base plate as well as two upper and lower connecting devices which attach the Type 4 rack to the side of a Region 2, Type 1 rack facing the pool wall. Each upper connecting device consists of a square tube inserted into a cell of the existing Type 1 rack; thus, two cells in each Region 2, Type 1 are taken out of service when the six Type 4 racks are installed. Each lower connecting device consists of a locking arm inserted into the cooling flow hole at the base of the Type 1 cell receiving the upper connecting device.

Table 3.2-1
Region 1, Type 3 Storage Rack Cell Dimensions

Description	Design Dimensions	Model Dimensions
Cell Pitch, cm.(in.)	23.45 ± 0.2 (9.2323)	23.45
Cell ID, cm.(in.)	20.68 +0.2/-0.1 (8.1418)	20.68
Wall Thickness, cm.(in.)		
SS304L	0.20 ± 0.018 (0.0787)	0.20
BSS	0.25 +0.05/-0.0 (0.0984)	0.25
Nominal Gap, cm.(in.)	2.07(0.815)/1.95 min.	2.07
Peripheral row BSS support		
Belt plate width, cm.(in.)	0.8 (0.3228)	0.8
SS thickness, cm.(in.)	0.20 ± 0.018 (0.0787)	0.20
BSS Parameters		
BSS density, g/cc	7.73 – 7.78	7.73
Boron content, wt%	1.7 min.	1.7
^{10}B wt% in natural boron	18.14	18.14
Plate length	145.7	144.0

Table 3.2-2
Region 1, Type 3 Rack Damaged Fuel Cell Dimensions

Description	Design Dimensions
Cell Pitch, cm.(in.)	23.45 ± 0.2 (9.2323)
Cell ID, cm.(in.)	22. +0.2/-0.1 (8.701)
Wall Thickness, cm.(in.)	
SS304L	0.2 ± 0.018 (0.0787)
BSS	0.30 +0.05/-0.0 (0.1181)
Nominal Gap, cm.(in.)	
Between damaged cells	0.55 (0.2165) / 0.43 min.
Between damaged/normal cells	1.36 (0.5354) / 1.13 min.
BSS Parameters	
BSS density, g/cc	7.73 – 7.78
Boron content, wt%	1.7 min.
^{10}B wt% in natural boron	18.14
Plate length, in.	145.7

Table 3.2-3
Region 2, Type 1 Rack Cell Dimensions

Cell Pitch	Design Dimensions	Model Dimensions
Cell Pitch, in.	8.43 +0.06/-0.0	8.43
Cell ID (without poisons), in.	8.25 +0.06/-0.0. square	8.25
Wall Thickness, in.	0.09 ± 0.004	0.09
Wall Material	SS-304	SS-304
SS Poison support sheet thickness, in.	0.062 ± 0.003	0.062
Cell ID with poison, in.	8.113	8.113
Boraflex Poison length, in. width, in. thickness, in. ^{10}B Self Shielding Bias Min. ^{10}B content, g/cm^2	 144 ± 1/16 7.625 ± 0.0625 0.075 ± 0.007 +0.0014 0.20	No Boraflex Poison

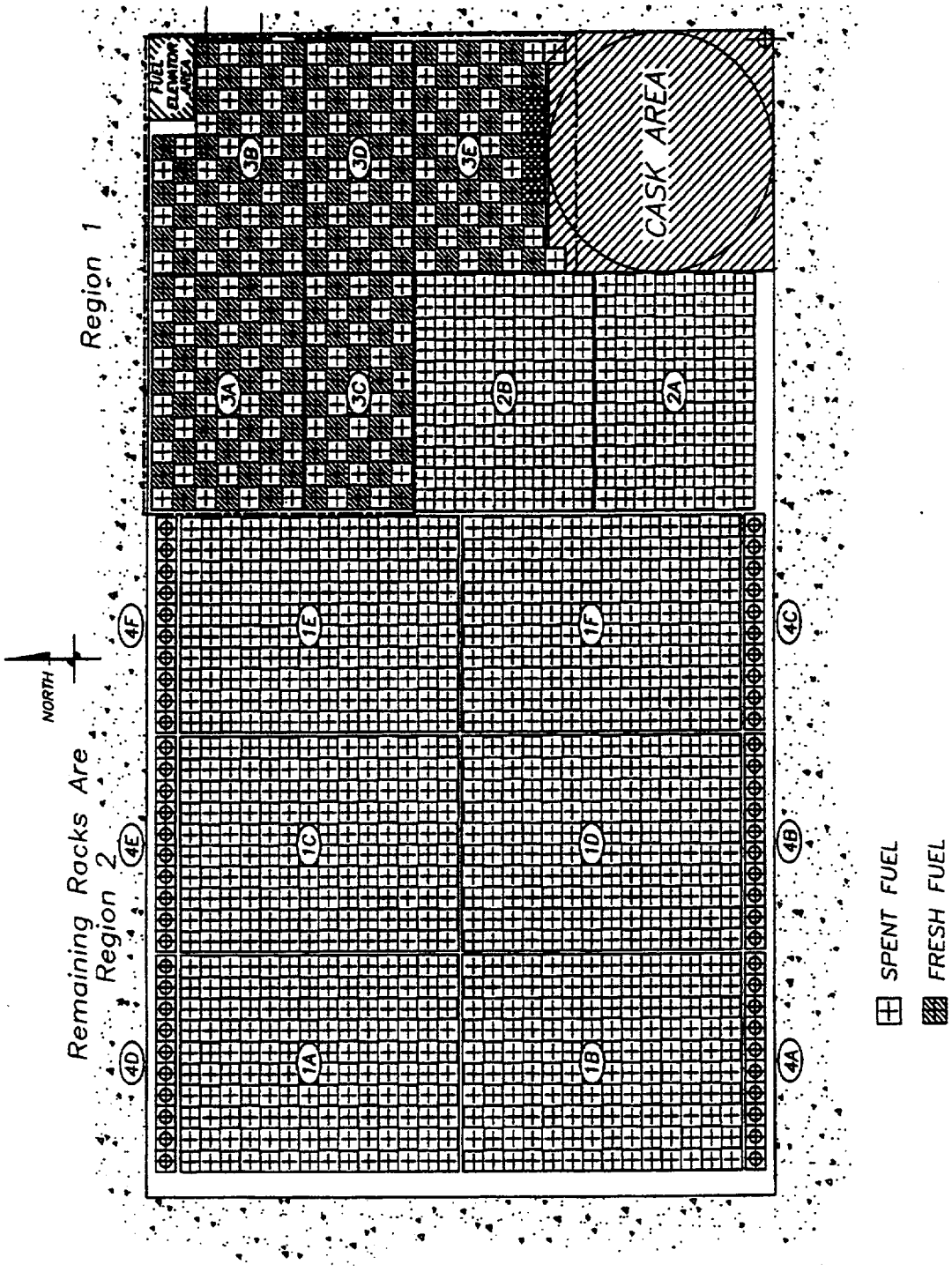
Table 3.2-4
Region 2, Type 2 Rack Cell Dimensions

Description	Design Dimensions	Model Dimensions
Cell Pitch, cm.(in.)	21.412 ± 0.2 (8.43)	21.412
Cell ID, cm.(in.)	20.68 +0.2/ -0.1 (8.1418)	20.68
Wall Thickness, cm.(in.) SS304L BSS	 0.2 ± 0.018 (0.0787) 0.3 +0.05/-0.0 (8.1418)	 0.2 0.3
Nominal Gap, cm.(in.)	0.232 (0.0913)/0.15 min.	0.232
BSS Parameters BSS density, g/cc Boron content, wt% ^{10}B wt% in natural boron Plate length, in.	 7.73 – 7.78 1.7 min. 18.14 145.7	 7.73 1.7 18.14 144.0

Table 3.2-5
Region 2, Type 4 Rack Cell Dimensions

Description	Design Dimensions	Model Dimensions
Cell Pitch, cm.(in.)	21.412 ± 0.2 (8.43)	21.412
Cell ID, cm.(in.)	20.68 +0.2/-0.1 (8.1418)	20.68
Wall Thickness, cm.(in.) SS304L BSS	 0.2 ± 0.018 (0.08) 0.25 +0.05/-0.0 (0.10)	 0.2 0.25
Nominal gap thickness, cm.(in.) Between Type 4 cells (nom./min.) Between Type 4 and Type 1 Mod. Between Type 4 and pool wall	 0.082 (0.03228)/0.03 min. 3.0 (1.18) min. 13.334 (5.25) min.	 0.082 3.0 13.334
BSS Parameters BSS density, g/cc Boron content, wt% ^{10}B wt% in natural boron Plate Length	 7.73 – 7.78 1.7 min. 18.14 145.7	 7.73 1.7 18.14 144.0

Figure 3.1-1
Spent Fuel Pool, General Arrangement



4.0 KENO Models

The purpose of this section is to describe the models erected in KENO to represent either arrays of different types of storage cells or the full spent pool storage configuration. With the exception of the Region 2 Type 4 racks the characteristics of the various rack types are evaluated on the basis of infinite array analyses for the various storage cell types. The Type 4 racks are excluded since they consist of linear arrays of cells with a single cell width.

4.1 Region 1 Type 3

The KENO model for the Region 1 Type 3 storage racks consists of an infinite checkerboarded array of the fresh and burned fuel storage cells. The dimensions of Table 3.2-1 were employed for each cell type and the basic two by two array was assigned periodic boundary conditions to simulate an infinite array. The fresh fuel cell employed the design basis fresh fuel assembly representation with a 144-inch long uniform active fuel zone with reflecting boundary conditions at each end; the moderator temperature was the ambient moderator temperature for the pool unless off-nominal temperature analyses were being done. The burned fuel cell employed the eight axial zone representation of the design basis spent fuel assembly with the appropriate fuel isotopic concentrations for the specific analysis. Figure 4.1-1 shows a sketch of the Region 1 Type 3 cell geometry employed in the KENO analyses.

4.2 Region 2 Type 1

The Region 2 Type 1 infinite array of storage cells was modeled using the dimensions of Table 3.2-3 except in this case the infinite array is based on a single storage cell consisting of an eight axial zone representation of a design basis spent fuel assembly and periodic boundary conditions on the storage cell. As noted previously, the volume between the L-insert and the SS tube interior wall was filled with water at the ambient temperature and soluble boron concentration for the case being analyzed. Figure 4.2-1 shows a sketch of the geometry employed in the KENO representation of the Region 2 Type 1 storage cell array.

4.3 Region 2 Type 2

The Region 2 Type 2 infinite array of storage cells was modeled using the dimensions of Table 3.2-4 as a checkerboard of SS and BSS adjacent cell types in the global array with periodic boundary conditions in the planar directions. Both cell types employed the eight axial zone model of the design basis spent fuel assembly. Figure 4.3-1 shows a sketch of the geometry employed in the KENO representation of the Region 2 Type 2 cell array.

4.4 Region 2 Type 4

Figure 4.4-1 shows a sketch of the geometry employed in the KENO representation of the Region 2 Type 4 storage cell array. The Region 2 Type 4 linear array of storage cells is modeled consistent with the discussion of Section 3.2.4. Since these cells exist in a linear array along the north and south sides of the array of the Region 2 Type 1 storage racks, their reactivity characteristics are deduced from the full pool KENO model discussed in Section 4.5, below.

4.5 Full Spent Fuel Pool Model

A KENO model of the full spent fuel pool was created to evaluate fuel mishandling accidents and soluble boron requirements. The following are the salient features of the KENO model for the entire RG&E spent fuel pool.

- There were six modules (3 x 2 array) of Type 1 Cells with a capacity to store 840 fuel assemblies. Each module contained a rectangular array of 10 x 14 locations. No inter-module gap was modeled.

- The Type 4 cells were modeled as a single row of cells above and below the Type 1 cells. A water gap of 3 cm between the Type 4 and the Type 1 cells was utilized for this purpose.

- The Type 2 cells were modeled as two arrays of 11 x 9 (module 2B) and 11 x 8 (module 2A) respectively. A minimum gap of 3.6 cm of water was used to separate the two modules. The gap between modules 2B and 3C was 3.175 cm.

- The Type 3 cells were modeled using five modules. Module 3A and 3B are adjacent to each other and were modeled using a 10 x 7 array. In reality, the fuel elevator area occupies part of the top 4 x 2 array portion of module 3B. The Modules 3C and 3D are modeled using a 10 x 5 array. A minimum gap of 3.175 cm was used to model the inter module gap between these four modules. Module 3E is modeled as a 10 x 7 array with the middle eight locations of the

bottom row being simulated as water cells. This is because the Cask Area occupies these locations. The damaged cells are conservatively represented, as regular Type 3 cells, in this model. A gap of 3.65 cm is used between modules 3E and 3D while a gap of 2.65 cm is used between 3E and 2B

- To simplify the description, the term "west half of the pool" shall represent the Type 1 and Type 4 racks and the term " east half of the pool" shall represent the Type 2 and Type 3 racks. The gaps between the west half of the pool and module 2A, module 2B, and modules 3A and 3C is 3.6 cm, 4.445 cm and 6.5 cm, respectively.

- All the racks types are housed in a pool of water. The pool wall is made of 20 cm-thick concrete and completely surrounds the pool from all the four sides. There is a minimum gap of 10 cm from the left half of the pool and the pool wall along the Western wall. The distance between the left half of the pool and the pool wall along the Northern and Southern directions is 10 cm and 18.1 cms respectively. The gap between the module 2A and the pool wall is 16 cm and the gap between the modules 3A and 3B and the pool wall is 4.445 cm. The maximum distance of separation between the eastern pool wall and the right half of the pool is 7.75 cm.

- All the dimensions were based on the as-built dimensions of the Ginna spent fuel pool racks.

Figures 4.5-1 and 4.5-2 are GIF plots from the KENO model of the entire spent fuel pool; Figure 4.5-1 is of the west half of the pool and 4.5-2 is of the east half of the pool.

Figure 4.1-1
Sketch of KENO Model for Infinite Array of Region 1 Type 3 Cells

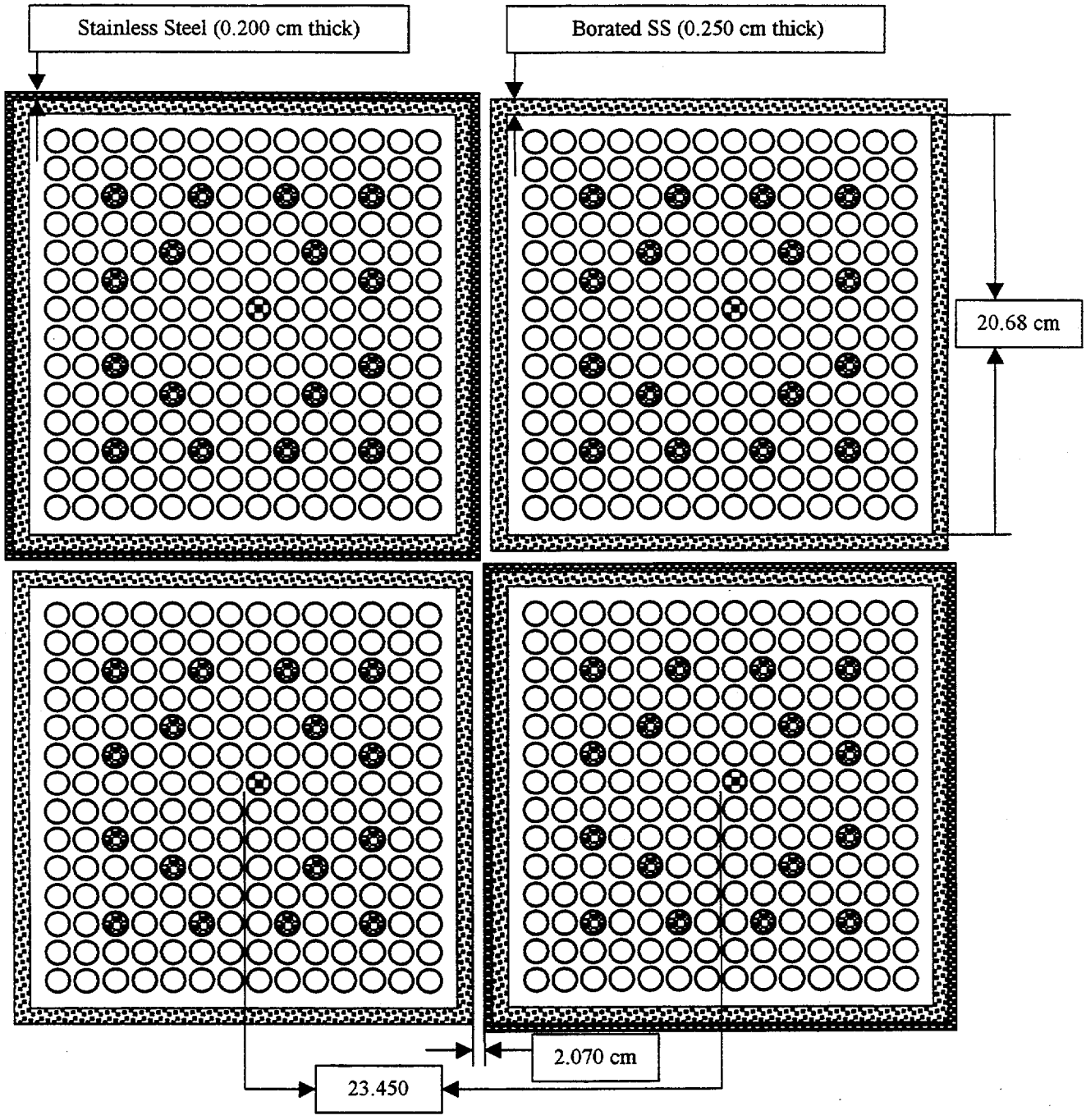


Figure 4.2.1
Sketch of KENO Model for Region 2 Type 1 Cells

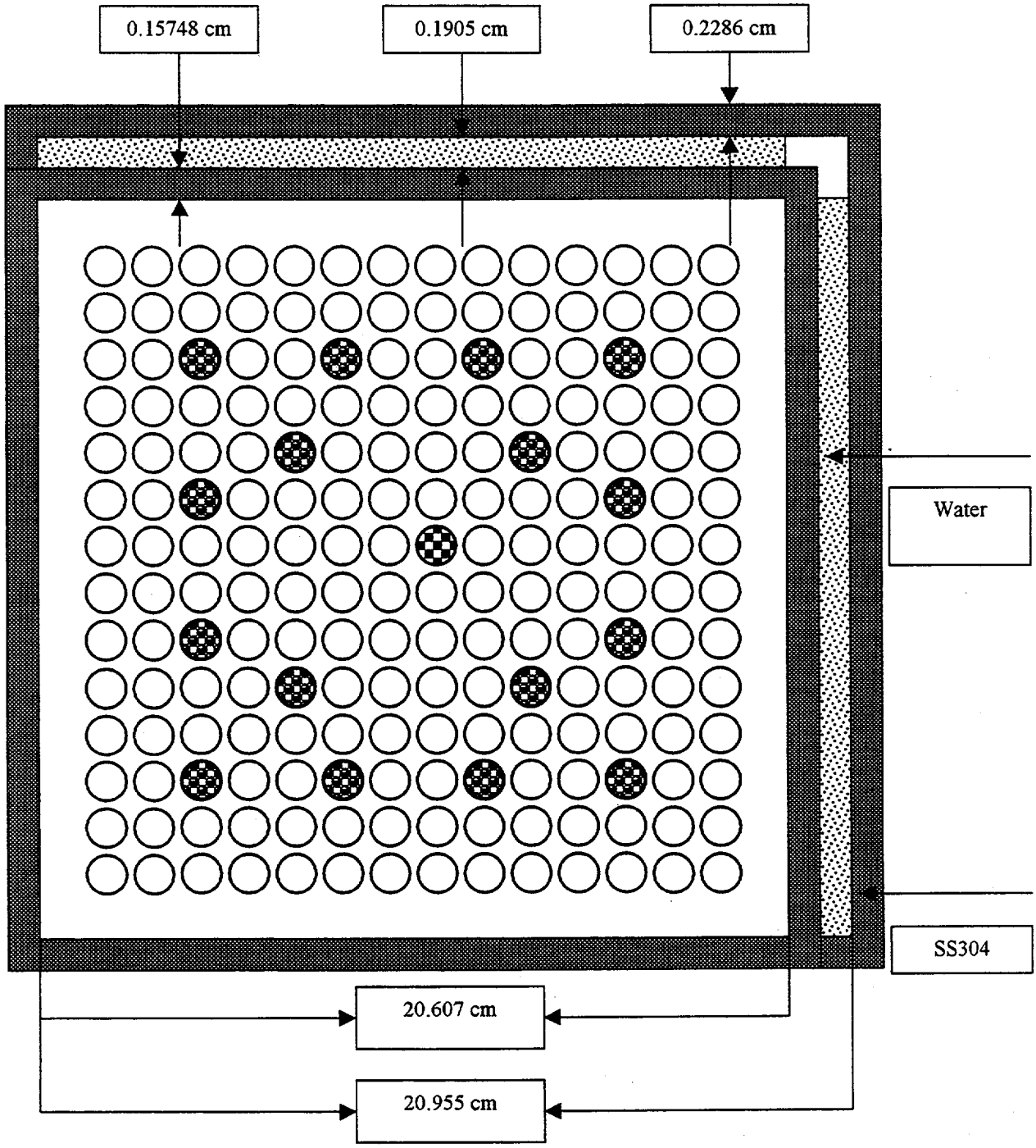


Figure 4.3-1
Sketch of KENO Model for Region 2 Type 2 Cells (BSS)

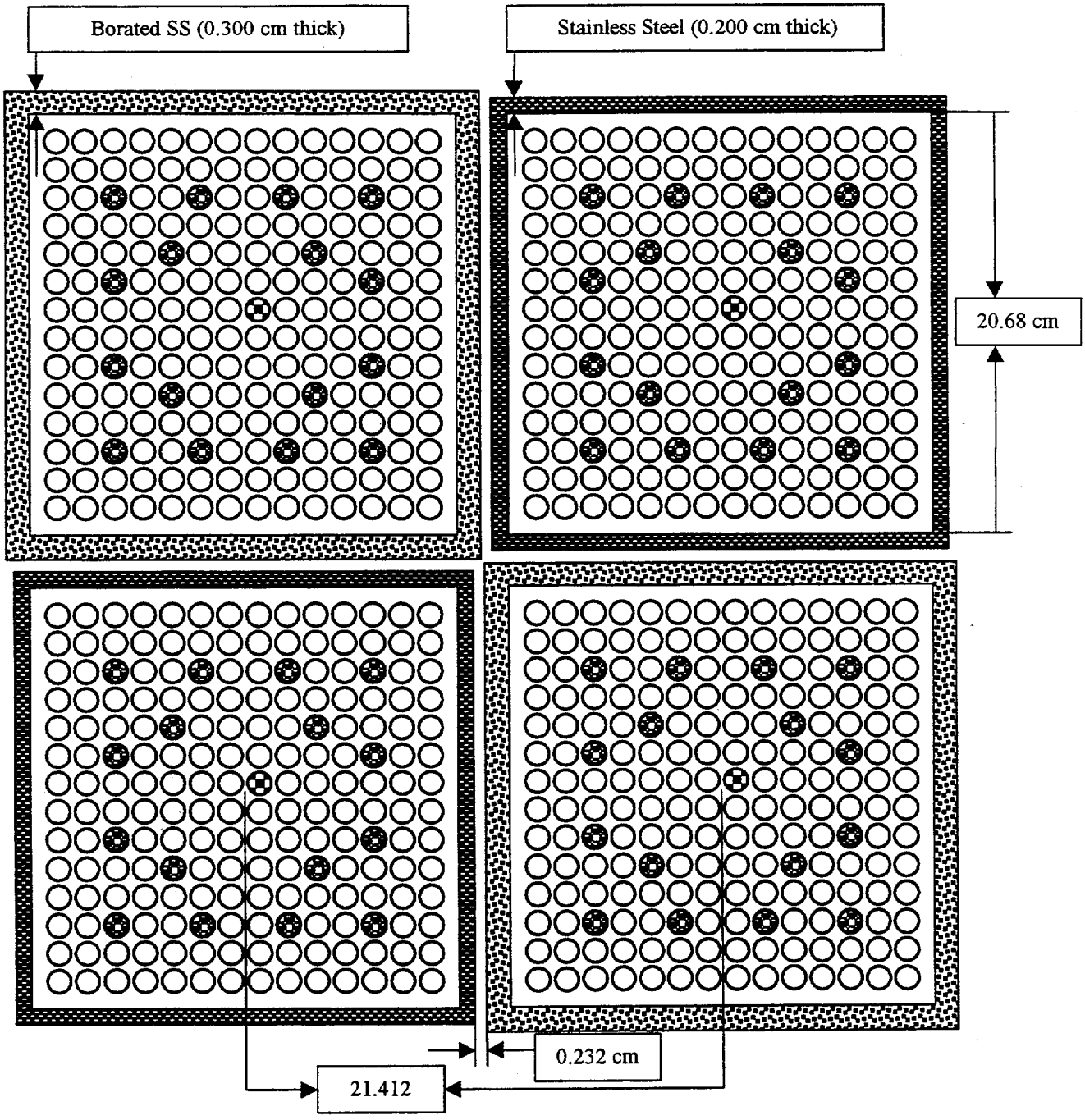


Figure 4.4-1
Sketch of KENO Model for Region 2 Type 4 Cells

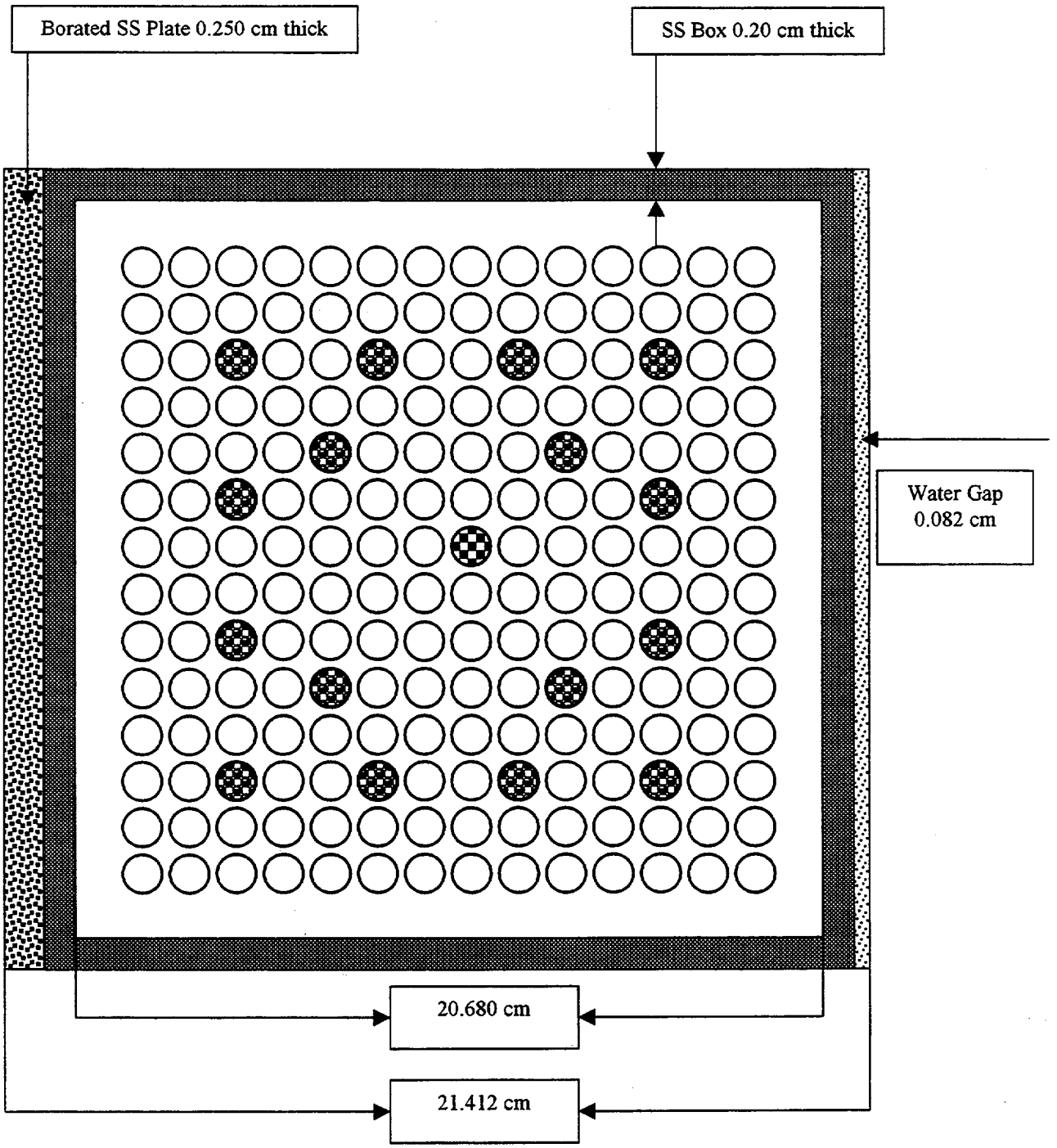


Figure 4.5-1
KENO GIF Plot of the Region 2 Type 1 and 4 Storage Racks

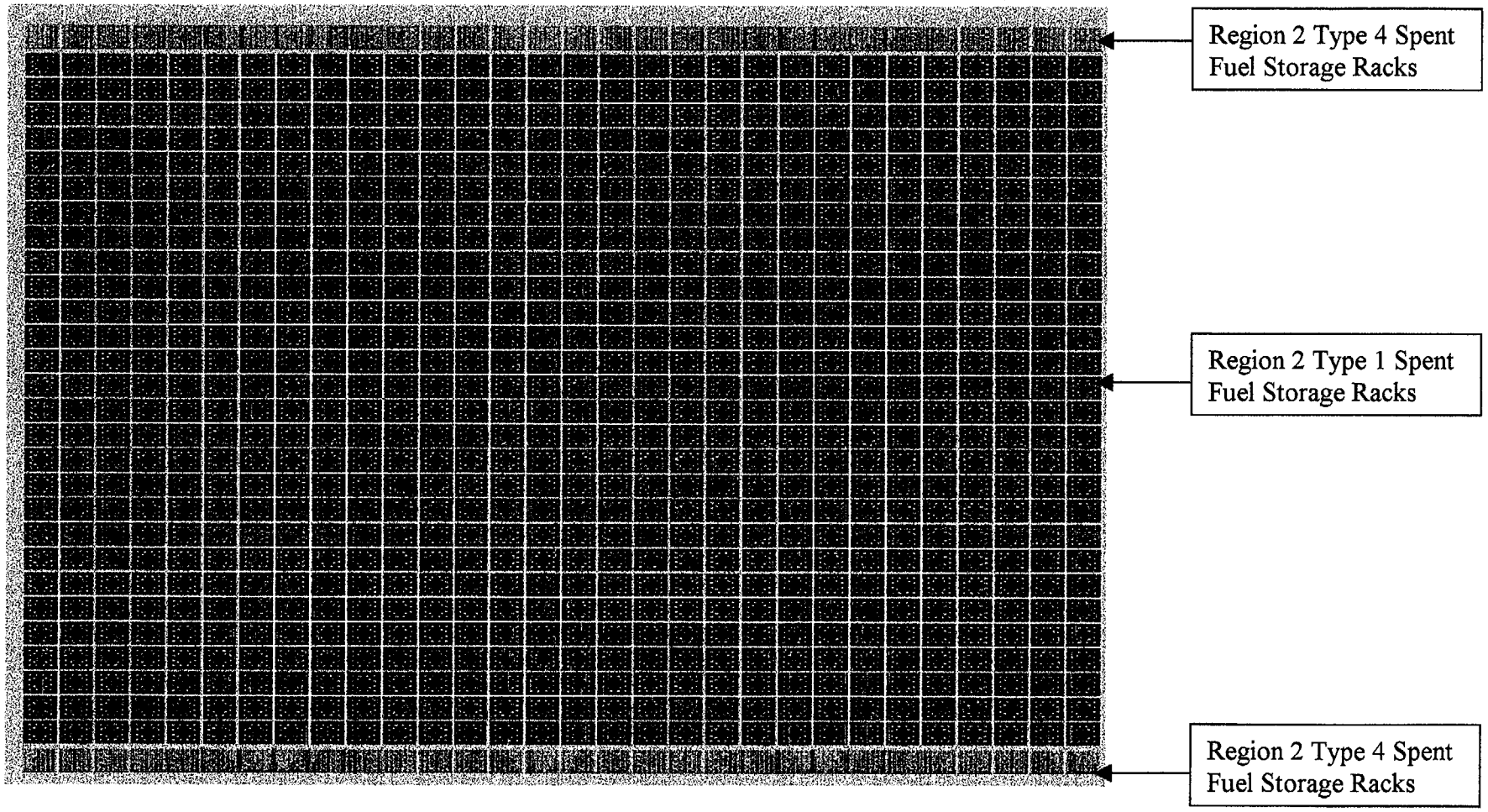
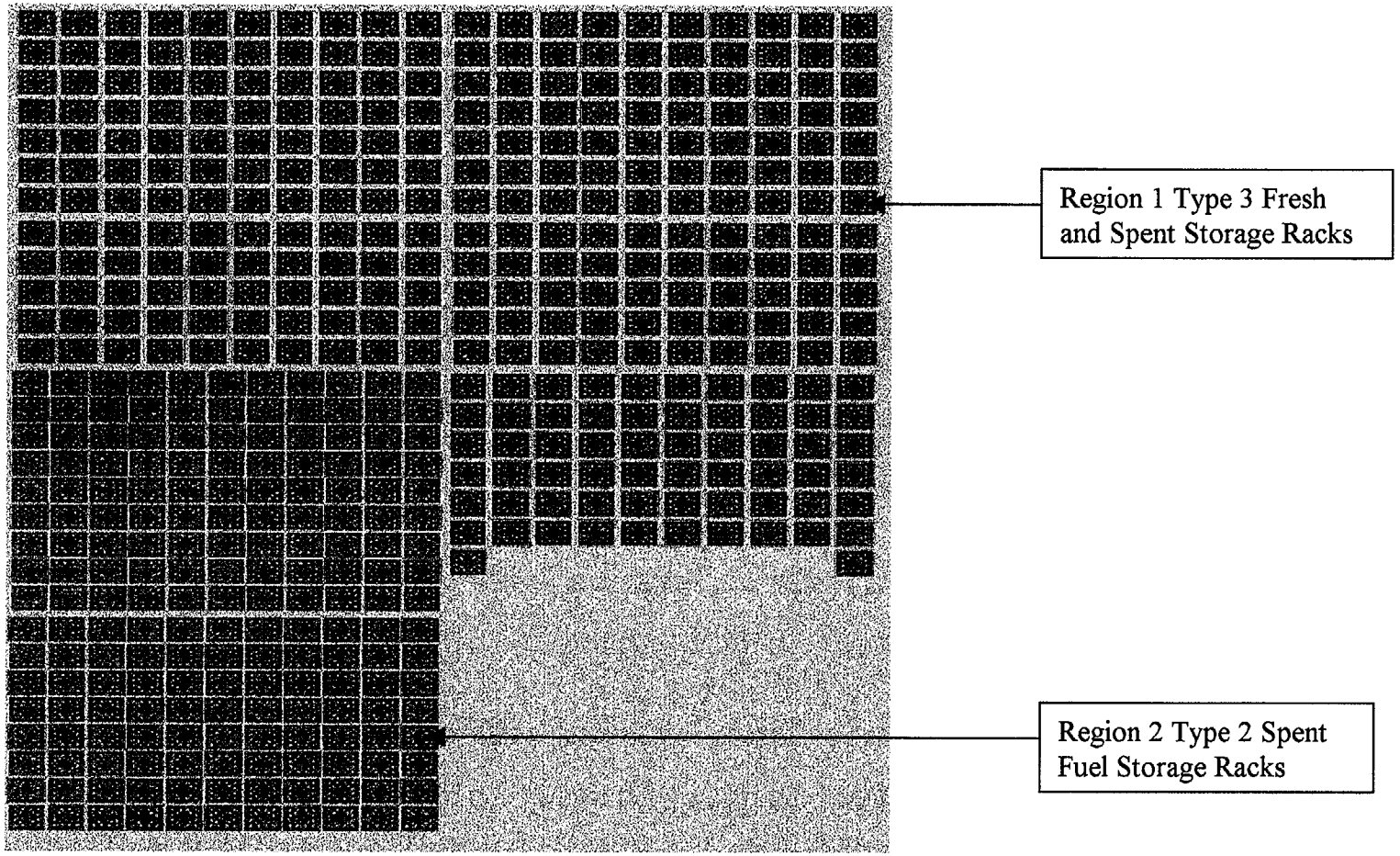


Figure 4.5-2
KENO GIF Plot of the Region 1 Type 2 and Region 2 Type3 Storage Racks



5.0 Design Basis Fuel Assemblies

The R.E. Ginna Nuclear Power Plant has been in operation for more than three decades and during that time interval a variety of lead test assemblies and reload batches containing different fuel assembly designs have been cycled through the reactor. Thus, the criticality safety analysis of the spent fuel pool must take into account possible differences in the reactivity characteristics of the different assembly types. For purposes of this analysis, the different types of fuel assemblies were surveyed so as to define a reference design fuel assembly which would assure conservative results for the analysis.

5.1 Design Basis Fuel Assemblies

Table 5.1-1 summarizes key data on the commercial fuel assembly types presently stored in the spent fuel storage racks. In addition, there are two other types of rod clusters to be considered. These are the damaged fuel rod storage basket and the fuel rod consolidation canisters. These two types will be treated at the end of the Section.

The fuel assemblies listed in Table 5.1-1 are comparable in many respects. However, there is one characteristic in the table that points to the more reactive fuel assembly and that is the normalized grams of ^{235}U/cm at the bottom of the table. The latter parameter was calculated assuming no dishing factor, a fuel density of 95% of theoretical, 179 rods per assembly, and the enrichments listed in the table for the respective assemblies. Using grams ^{235}U/cm as a figure of merit, one would expect the Westinghouse Standard fuel assembly to be the most reactive fuel assembly. One other conclusion is that, to first order, the latter assembly should envelope the reactivity behavior of the lead test assemblies listed in the first four columns.

Table 5.1-2 provides added detail on the Exxon Standard, Westinghouse Standard, and Westinghouse OFA assembly designs for closer examination of their characteristics in the spent fuel pool environment. Note that in these calculations, additional conservative approximations are introduced. As noted in Table 5.1-2, the stainless steel guide tubes are represented as Zirc.-4 in the Westinghouse Standard fuel assembly (a geometrical description of the same is given in Figure 5.1-1), but in addition, the instrument tube is replaced by a fuel rod in all three assembly types for these analyses. These three modified fuel assembly types should provide a conservative basis for evaluating the reactivity characteristics versus burnup for the different types of storage cells in the spent fuel pool.

DIT calculations were carried out to calculate fuel isotopics versus burnup for use in single axial zone KENO models for each of the three fuel assembly types. KENO calculations were performed for three assembly types in the environment of the Type 1, 2, and 3 storage cells. The results of these analyses are summarized in Table 5.1-3. The

KENO results for the fresh fuel assemblies are obtained from Table 2.1.2 of Reference 15. The data listed for the Type 1 storage cells shows the modified Westinghouse Standard fuel assembly design employing 4 wt% ^{235}U fuel to be the same, within the KENO sigma, as the Westinghouse OFA assembly but as the fuel burnup increases, the reactivity difference between these two assemblies increases. Note also, the modified Exxon assembly design is more reactive than the modified Westinghouse OFA at 15 GWD/MTU. In the Type 2 storage cells, the modified Westinghouse Standard assembly is again the more reactive assembly. For the Type 3 storage cell calculations, the fresh fuel cell contained either 5 wt% ^{235}U enriched modified Westinghouse OFA assemblies or Standard assemblies and the spent fuel locations contained 2.22 wt% ^{235}U equivalent burned modified Westinghouse Standard fuel assemblies. In this case the array containing the modified Westinghouse OFA assembly is more reactive than the modified Westinghouse Standard fuel assembly. Based on these results, it is concluded that the modified Westinghouse OFA fuel assembly is the most conservative design basis fresh fuel assembly and the modified Westinghouse Standard fuel assembly is the most conservative design basis spent fuel assembly for evaluation of the spent fuel storage rack.

The damaged rod storage basket was modeled conservatively as an array of 0.6-inch OD SS-304 tubes, with 0.05-inch wall thickness, positioned in a nominal 8 x 8 array with a nominal square pitch of 7/8-inches. In one corner of the array 16 tubes have been replaced by a 3 x 3 array of the same geometry tubes spaced on a nominal square pitch of 1.0-inch. Thus, a total of 57 fuel pins can be stored in the storage basket. The damaged rod storage basket is precluded from further consideration on the basis that the K_{eff} of the basket when filled with 5.0 wt% enriched fuel rods and placed in the Region 2, Type 1 Racks is 0.88357 ± 0.00079. Since this value is much lower than the design basis for the spent fuel pool (less than unity including all biases and uncertainties), the damaged rod storage basket can be placed in either Region 1 or Region 2 with no burnup restrictions.

The fuel rod consolidation canister is employed to store burned fuel rods removed from multiple, typically two or less, fuel assemblies. Table 5.1-4 lists the pertinent dimensions of the canister and Figure 5.1-2 illustrates the geometry of the container. The purpose of these canisters is to increase the storage capacity of the pool consistent with the load bearing capability of the pool structure by removing fuel rods from the burned fuel assembly cage structure and storing the rods in the consolidated fuel rod canister at a reduced water to fuel ratio. The canister was modeled using the dimensions provided in Table 5.1-4. The upper tolerance value of the outer dimension of the canister and the lower tolerance value of the thickness of the steel enclosure were used to maximize the capacity of the canister. The divider plate was not modeled for conservatism.

The KENO calculations were performed with both the Westinghouse Standard and Westinghouse OFA fuel rods. The pitch of the fuel rods inside the canister was varied to obtain the near-optimum pitch for the canister. The fuel rods were enriched to 1.30 w/o ^{235}U, a conservative value for Region 2 Type 1 cells. The KENO results are shown in Table 5.1-5. These results show the most reactive case occurs when 225 fuel rods from the Westinghouse Standard fuel assembly are optimally spaced in the canister. This value

of K_{eff} is, however, lower than for the case of the design basis spent fuel assembly in a Type 1 cell. It can therefore be concluded that results based on loading design basis spent fuel assemblies in Type 1 cells is bounding and the canisters can be excluded from further treatment. This argument is also extended to other type storage cells in the spent fuel pool so as to permit use of the consolidated canisters in those locations.

5.2 Non-SNM Stored in Spent Fuel Pool

Other items may be stored in the RG&E spent fuel pool in addition to fresh or discharged fuel assemblies. These items, in general, fall into the category of Non-Special Nuclear Material(SNM). These items are non-multiplying and, in general, are parasitic to the spent fuel rack local reactivity. Some of the items which fall under this category that can be safely stored in the spent fuel pool are: Dummy Canisters containing Non-SNM, Consolidation Hardware, Dummy Fuel Assemblies, Trash Basket containing full length control rods, etc. The general rule for safely storing these types of items is very simple: any non-multiplying and non-fissile item can be safely stored in any cell location. Note that neutron sources are considered to be non-multiplying and non-fissile.

Table 5.1-1
Summary of Fuel Assembly Characteristics

CHARACTERISTICS	B&W	EXXON SOLID	WESTINGHOUSE MOX	EXXON ANNULAR	EXXON STANDARD	WESTINGHOUSE STANDARD	WESTINGHOUSE OFA
PELLET ID (in)	N/A	N/A	N/A	.1108	N/A	N/A	N/A
PELLET OD (in)	.3615	.3565	.3659	.3505	.3565	.3669	.3444
FUEL ROD CLAD ID (in)	.37	.364	.3734	.358	.364	.3734	.3514
FUEL ROD CLAD OD (in)	.422	.424	.422	.417	.424	.422	.400
FUEL ROD CLAD MATERIAL	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4/ ZIRLO
GUIDE TUBE ID (in)	.5075	.507	.5075	.510	.521	.505	.490
GUIDE TUBE OD (in)	.5375	.538	.5375	.540	.524	.539	.528
GUIDE TUBE MATERIAL	SS-304	SS-304L	SS-304	ZIRC-4	ZIRC-4	SS	ZIRC-4/ ZIRLO
ENRICHMENT w/o U_{235}	3.22	3.29	3.09	3.7	5.00	5.00	5.00
GRAMS U_{235}/cm	34.98	35.74	33.56*	34.00	52.82	55.95	49.29

N/A – Not applicable

* **"Equivalent ^{235}U or Fissile"**

Table 5.1-2
Input Parameters for Fuel Assembly Models

Description	Exxon Std.	W Std.	W OFA
Rods/Assy.	179	179	179
Guide Tubes/ Assy.	16	16	16
Instrument Tubes / Assy.	1**	1**	1**
IHM wt., Kg/Assy.	370 –374.5	383 – 398	349 – 356.5
Rod Pitch, in.	0.556	0.556	0.556
Pellet OD, in.	0.3565 ± 0.0008	0.3669 ± 0.0008	0.3444 ± 0.0008
Pellet Density, % TD	95 ± 2.0	95 ± 2.0	95 ± 2.0
Max. Enrichment, wt%	5.0 ± 0.05	5.0 ± 0.05	5.0 ± 0.05
Pellet Dish Factor, %	1.187 ± 2.0	1.187 ± 2.0	1.1926 ± 2.0
Active Fuel Length, in.	141 –144	141 – 144	141 – 144
Clad OD, in.	0.424 ± 0.0025	0.422 ± 0.0025	0.400 ± 0.0025
Clad Thickness, in.	0.030 ± 0.0025	0.0243 ± 0.0025	0.0243 ± 0.0025
Clad Material	Zirc.-4	Zirc.-4	Zirc.-4/ Zirlo
Guide Tube OD, in.	0.524 ± 0.005	0.539 ± 0.005	0.528 ± 0.005
Guide Tube Thickness, in.	0.015 ± 0.0055	0.017 ± 0.0055	0.019 ± 0.0055
Guide Tube Mat.	Zirc.-4	SS*	Zirc.-4/ Zirlo
Inst. Tube OD, in.	0.424 ± 0.005	0.422 ± 0.005	0.399 ± 0.005
Inst. Tube Thickness, in.	0.039 ± 0.004	0.0240 ± 0.004	0.0235 ± 0.004
Inst. Tube Mat.	Zirc.-4	SS	Zirc.-4/ Zirlo

* Modeled conservatively as Zirc.-4 in KENO Model
** Instrument tube replaced by fuel rod in KENO Model

Table 5.1-3
Summary of Reactivity Evaluations for Different Fuel Assembly Designs in Different Storage Cell Types

Storage Cell Type	Burnup*	Fuel Assembly Type		
		W Std.	W OFA	Exxon Std.
Type 1	0	1.34304 ± 0.00072	1.34231 ± 0.00071	1.33739 ± 0.00075
	15	1.18702 ± 0.00066	1.18322 ± 0.00065	1.18405 ± 0.00067
	45	0.94668 ± 0.00065	0.90955 ± 0.00054	0.93660 ± 0.00059
Type 2	15	1.04031 ± 0.00078	1.03938 ± 0.00077	1.03611 ± 0.00078
	45	0.82970 ± 0.00064	0.79400 ± 0.00065	0.81538 ± 0.00063
Type 3	#	0.94218 ± 0.00094	0.94890 ± 0.00092	

* GWD/MTU

\# 5 wt% ^{235}U Enriched Fuel Assembly in Fresh Fuel Storage Location with 2.22 wt% ^{235}U Equivalent Modified Westinghouse Standard Fuel Assembly in Spent Fuel Storage Location.

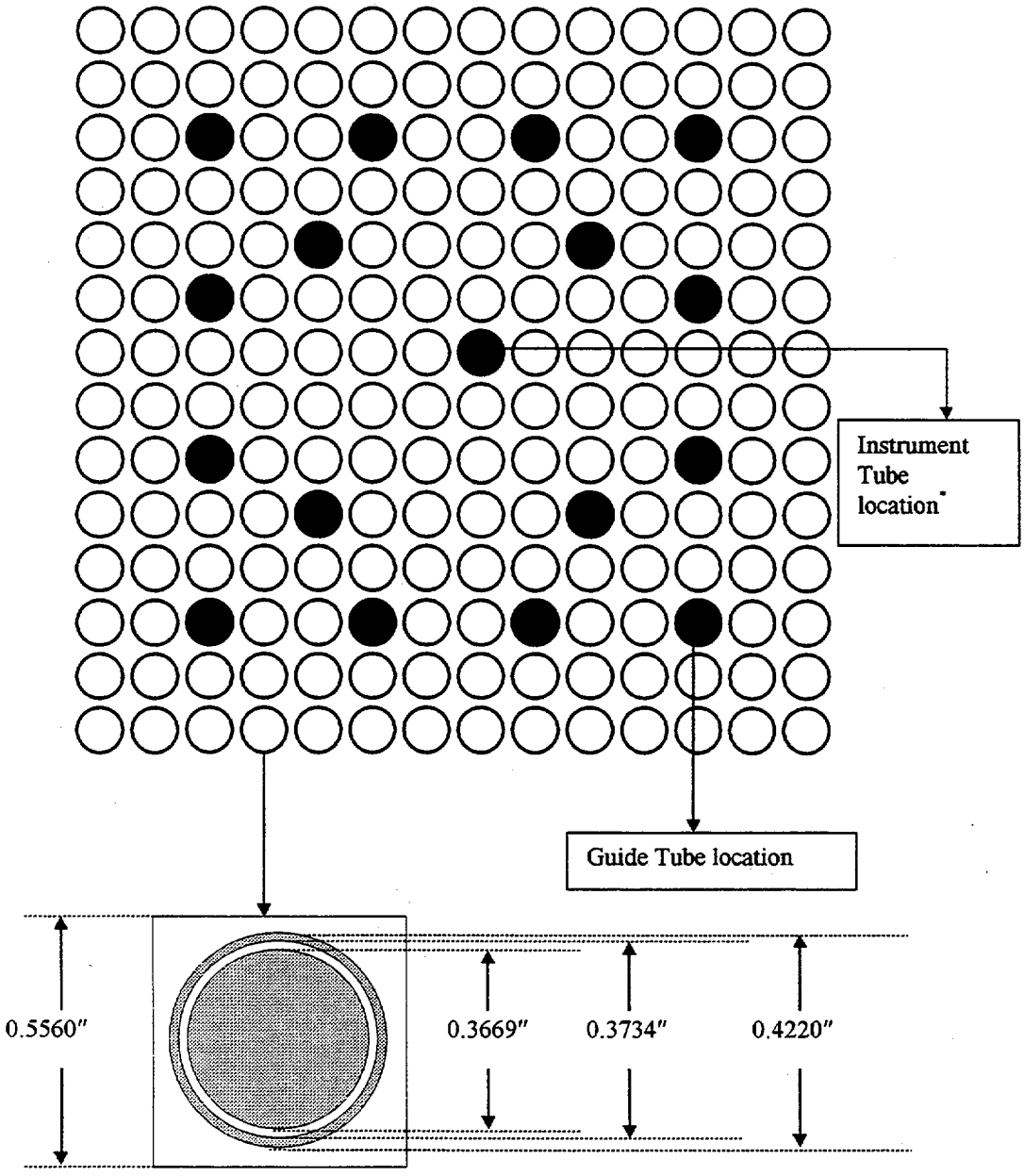
Table 5.1-4
Consolidation Canister Specifications

Description	Values
Outer square dimension, in.	8.00 ± 0.02
Wall thickness, in.	0.093 ± 0.004
Height, in.	
Including Lids at top/bottom	168 ± 0.06
Without Lids at top/bottom	156 1/4 ± 0.06
Canister lid height, in. – top/bottom	5 7/8
Material of construction	
Body	SS-304
Lids	SS-304
Divider Plate	SS-304
Divider Plate	
Thickness, in.	0.093 ± 0.004
Centered within, in.	1/32
Length, in.	153 5/16 ± 0.06
Max. rods/ container	2 x 179

Table 5.1-5
KENO Calculated Multiplication Factors for an infinite array of Consolidated Canisters in Region 2 Type 1 Cells

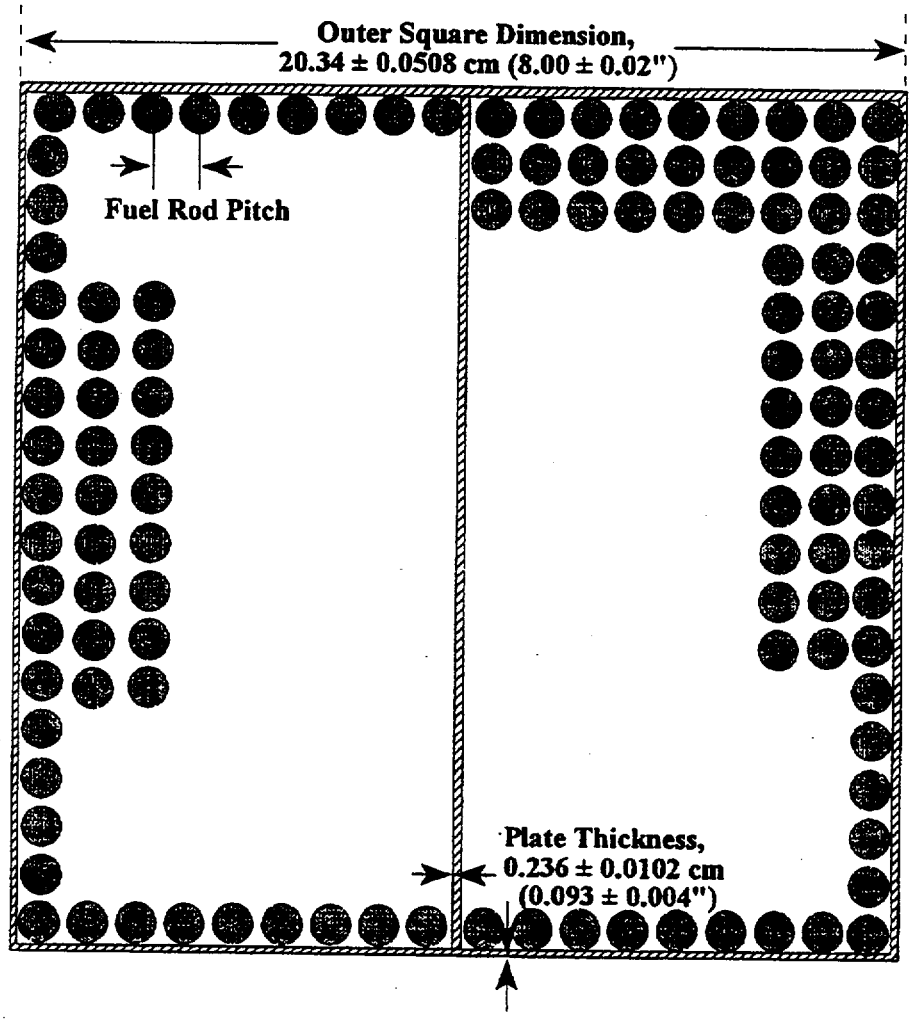
No. of Fuel Rods in Canister	KENO Calculated Multiplication Factor	
	Westinghouse Standard	Westinghouse OFA
169	0.94229 ± 0.00060	0.92094 ± 0.00058
196	0.95884 ± 0.00063	0.94537 ± 0.00064
225	0.95910 ± 0.00065	0.95644 ± 0.00063
256	0.94474 ± 0.00061	0.95444 ± 0.00062

Figure 5.1-1
Geometrical View of Westinghouse 14x14 Standard Fuel Assembly



* modeled with a fuel rod in this location

Figure 5.2-1
Illustration of Fuel Rod Consolidation Canister



6.0 Modeling of Axial Burnup Distributions

A key aspect of the burnup credit methodology employed in this analysis is the inclusion of an axial burnup profile correlated with feed enrichment and discharge burnup of the burned fuel assemblies. This effect is important in the analysis of the spent fuel pool characteristics since the majority of spent fuel assemblies stored in the pool have a discharge burnup well beyond the limit for which the assumption of an uniform axial burnup shape is conservative. Input to this analysis is based on axial burnup profile data provided by RGE, as documented in Reference 1 and provided herein, for typical fuel assemblies having various feed enrichments and discharge burnups. Table 6.0-1 lists pertinent data on the fuel assemblies for which axial profile data were provided. The selected assemblies covered a range of enrichments, axial blanket enrichments, core positions, and different cycles. The majority of the fuel assemblies were the OFA type with axial blankets, which are more typical of later cycle operation in the Ginna reactor, but four assemblies were of the Exxon standard fuel assembly type with no axial blanket. Figures 6.0-1 through 6.0-4 show the normalized axial burnup shapes for the fuel assemblies having average burnups in the four intervals: 10 to 20, 20 to 30, 30 to 40, and 40 to 50 GWD/MTU. In general, the axial burnup profiles are quite similar. However, for the OFA assemblies with natural uranium blankets, the relative burnup is higher in all nodes except the upper and lower two nodes which contain the axial blankets. An apparent inconsistency is seen for the case of assembly E60; however, for this case the axial blankets contain 2.6 wt% ^{235}U hence a lower burnup is seen for the central nodes and a higher burnup for the blanket nodes relative to the other OFA assemblies.

From the standpoint of reactivity contributions to the overall fuel storage rack K_{eff}, the natural uranium blanket regions of the OFA assemblies make a negligible contribution. Assembly E60 with the 2.6 wt% ^{235}U axial blanket does exhibit a similar axial burnup profile to the non-blanket assembly Q16. However, it is the top nodes which provide the most reactivity due to irradiation temperature effects and in this regime assembly Q16 exhibits a more rapid fall-off of burnup. In addition, the reduced enrichment in the blanket of assembly E60 would have some diminishment of the rack reactivity contribution from these blanket nodes.

On these latter bases, the axial profile data from assembly Q16 was selected as being a conservative representation of the typical burnup profile for the Ginna core. Table 6.0-2 lists the relative axial profile obtained from the typical fuel cycle analyses for this assembly as a function of end-of-cycle burnup. Figure 6.0-5 provides a plot of the absolute burnup as a function of end-of-cycle burnup. Figure 6.0-6 provides a plot of the absolute burnup as a function of height for each cycle of irradiation and Figure 6.0-7 shows the relative distribution.

Based on a review of articles on burnup credit, e.g., Reference 14 , an eight node axial model was implemented for the storage rack calculations. The principal objective of this model versus that of Reference 1 is added definition of the burnup distribution at the top of the fuel assembly. In effect, the top three nodes of the model of Reference 1 each spanned 6.15 inches whereas for the eight node model the top four nodes were taken to be 6.15, 3.075, 6.15, and 3.075 inches in height, respectively. The KENO model assumes an active fuel height of 144 inches whereas most of the past and current fuel had a height of about 141 inches. To accommodate the added height, the extra length of fuel is assumed to be in the middle of the fuel column.

DIT was used to generate the isotopic concentrations for each segment of the axial profile. Table 6.0-3 lists the fuel and moderator temperatures employed in the spectral calculations for each node in the eight zone axial model. These values are the same as were employed in Reference 1 and are based on mid-cycle temperature profiles for the Ginna core. The axial burnup profile was taken to be that for the 40 to 50 GWD/MTU burnup interval discussed in the previous section since the relative axial shapes for non-blanketed fuel shown in Figure 6.0-6 are very similar. These node dependent moderator and fuel temperature data and power profile data were employed in DIT to deplete the fuel to the desired burnup for each initial enrichment and each axial zone. The values of assembly average burnups versus feed enrichment for which burned fuel assemblies were simulated are tabulated as follows:

3 wt%	**4 wt%**	**5 wt%**
MWD/MTU	MWD/MTU	MWD/MTU
0	0	0
15,000	15,000	15,000
25,000	15,000	25,000
35,000	35,000	35,000
	45,000	45,000
		55,000

The DIT computed isotopic concentrations were transferred into the KENO models of the storage cells using the same formalism as in Reference 1. That is, the ^{235}U, ^{238}U, ^{236}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{16}O, and equilibrium ^{149}Sm at shutdown are represented explicitly in the KENO models. All other fuel isotopics are represented by an equivalent ^{10}B concentration; the magnitude of this concentration is determined by matching the DIT K_{eff} value with KENO to a one sigma tolerance level.

All KENO models of storage cells containing burned fuel assemblies employ the explicit eight axial zone representation of the burned fuel assemblies as described above. For simulating fuel assemblies with burnups greater than 45 GWD/MTU, the eight zone axial model was slightly modified to accommodate the increase in burnup. It is assumed that the axial burnup distribution varies in the same rate from 45 GWD/MTU and beyond as that from 35 GWD/MTU to 45 GWD/MTU. Using the data from the columns 6 and 7 of Table 6.0-2, the eight zone axial model for burnups greater than 45 GWD/MTU is determined. This modified model is applied only for those assemblies that have an initial enrichment of 5.0 w/o ^{235}U. Table 6.0-4 shows the details of this model. Figure 6.0-8 provides a pictorial view of the axial zones employed in the eight zone axial model. Appendix A contains a listing of the isotopic number densities employed in the KENO calculations. There are a total of 29 tables. The first nine tables give the assembly averaged isotopics for a 3 wt% ^{235}U W-Std fuel assembly plus the isotopics by axial zone for each of the eight zones. The second nine tables give similar data for a 4 wt% ^{235}U W-Std fuel assembly and the third nine tables give similar data for a 5 wt% ^{235}U W-Std fuel assembly. The last two tables provide the assembly average number densities for the W-OFA and EXXON Std. fuel assemblies, respectively, at 4 wt% ^{235}U.

Table 6.0-1
Ginna Fuel Assemblies Used for Axial Shape Evaluation

Fuel Assembly ID	Assembly Type	Cycle	Burnup (GWD/MTU)
A62	OFA, Blanket	21	15.19
A62	OFA, Blanket	22	27.21
A62	OFA, Blanket	23	33.97
A62	OFA, Blanket	26	48.48
D77	OFA, Blanket	24	12.96
D77	OFA, Blanket	25	27.87
D77	OFA, Blanket	26	44.11
C63	OFA, Blanket	23	12.88
C63	OFA, Blanket	24	27.08
C63	OFA, Blanket	25	39.67
C63	OFA, Blanket	26	45.03
C56	OFA, Blanket	23	14.22
C56	OFA, Blanket	24	27.62
C56	OFA, Blanket	25	32.30
C56	OFA, Blanket	26	37.51
E60	OFA, Blanket	25	15.65
E60	OFA, Blanket	26	34.42
Q16	Exxon Std, No Blkt.	14	10.72
Q16	Exxon Std, No Blkt.	15	23.01
Q16	Exxon Std, No Blkt.	16	33.53
Q16	Exxon Std, No Blkt.	17	44.84

Table 6.0-2
Relative Axial Shapes for Typical Non-Axial Blanket Standard Fuel Assemblies

Assy Burnup, GWD/MTU =			10.715	23.011	33.526	44.844
Assy ID & Cycle of Irradiation =			Q16 Cy 14	Q16 Cy 15	Q16 Cy 16	Q16 Cy 17
Node	Ht.(in)	Midpt.(in)	Relative Burnup			
1	6.15	3.075	0.490901	0.485724	0.472708	0.48818
2	12.30	9.225	0.806533	0.813611	0.806777	0.813375
3	18.45	15.375	0.987681	1.000956	0.997793	1.00252
4	24.60	21.525	1.074382	1.084308	1.082622	1.081126
5	30.75	27.675	1.112366	1.117726	1.116626	1.106949
6	36.90	33.825	1.12018	1.128938	1.12808	1.113705
7	43.05	39.975	1.130751	1.130677	1.129869	1.113839
8	49.20	46.125	1.130098	1.128373	1.127602	1.111498
9	55.35	52.275	1.127298	1.12468	1.124023	1.108554
10	61.50	58.425	1.123472	1.120377	1.119967	1.105544
11	67.65	64.575	1.119832	1.116031	1.11591	1.102667
12	73.80	70.725	1.115912	1.111642	1.111853	1.099835
13	79.95	76.875	1.112086	1.107383	1.108036	1.097293
14	86.10	83.025	1.107699	1.102864	1.104098	1.094773
15	92.25	89.175	1.10294	1.09817	1.099922	1.092164
16	98.40	95.325	1.097154	1.092564	1.095031	1.089399
17	104.55	101.475	1.088661	1.084612	1.087872	1.08514
18	110.70	107.625	1.074382	1.071879	1.07606	1.077848
19	116.85	113.775	1.04797	1.04776	1.052914	1.062015
20	123.00	119.925	0.997107	1.000217	1.006144	1.025176
21	129.15	126.075	0.898553	0.903872	0.909622	0.936067
22	135.30	132.225	0.713299	0.714658	0.717413	0.745652
23	141.45	138.375	0.415119	0.41315	0.409473	0.446793

Table 6.0-3
Relative Power, and Fuel and Moderator Temperatures for the Eight Zone Model

Zone No.	Height (in.)	Relative Power	Fuel Temperature (°F)	Moderator Temperature (°F)
1	6.15	0.488	991.022	544.190
2	6.15	0.813	1101.020	545.018
3	6.15	1.003	1211.018	545.360
4	107.1	1.092	1218.956	574.034
5	6.15	0.936	1138.010	603.860
6	3.075	0.841	1085.522	604.526
7	6.15	0.624	980.528	605.741
8	3.075	0.297	875.516	606.488

Table 6.0-4
Relative Power, and Fuel and Moderator Temperatures for the Eight Zone Model (used only for burnups > 45 GWD/MTU, 5.00wt%)

Zone No.	Height (in.)	Relative Power	Fuel Temperature (°F)	Moderator Temperature (°F)
1	6.15	0.488	991.022	544.190
2	6.15	0.813	1101.020	545.018
3	6.15	1.003	1211.018	545.360
4	107.1	1.084	1218.956	574.034
5	6.15	0.975	1138.010	603.860
6	3.075	0.882	1085.522	604.526
7	6.15	0.672	980.528	605.741
8	3.075	0.361	875.516	606.488

Figure 6.0-1
Typical Ginna Axial Burnup Shapes for Burnups Between 10 and 20 GWD/MTU

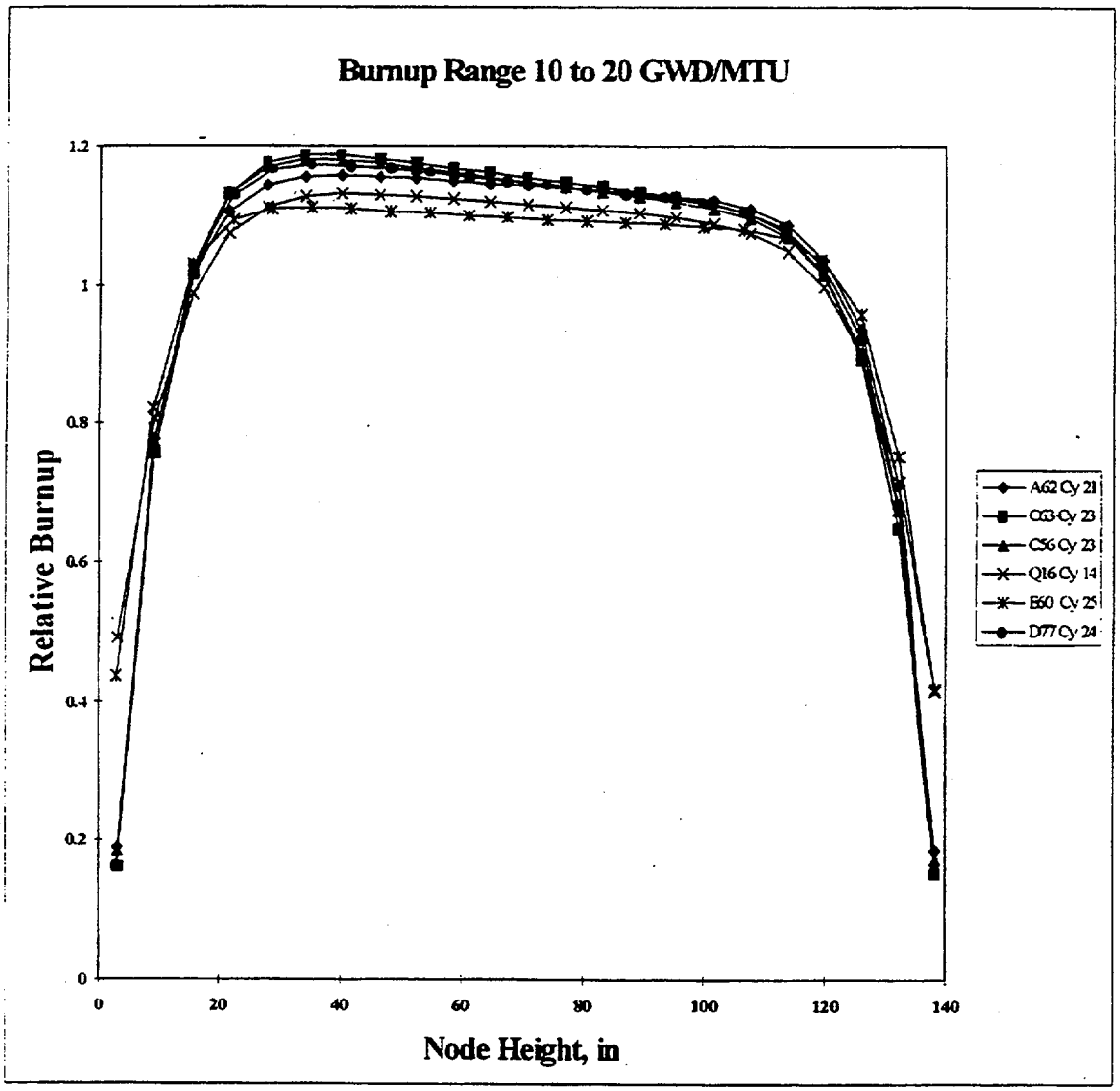


Figure 6.0-2
Typical Ginna Axial Burnup Shape for Burnups Between 20 and 30 GWD/MTU

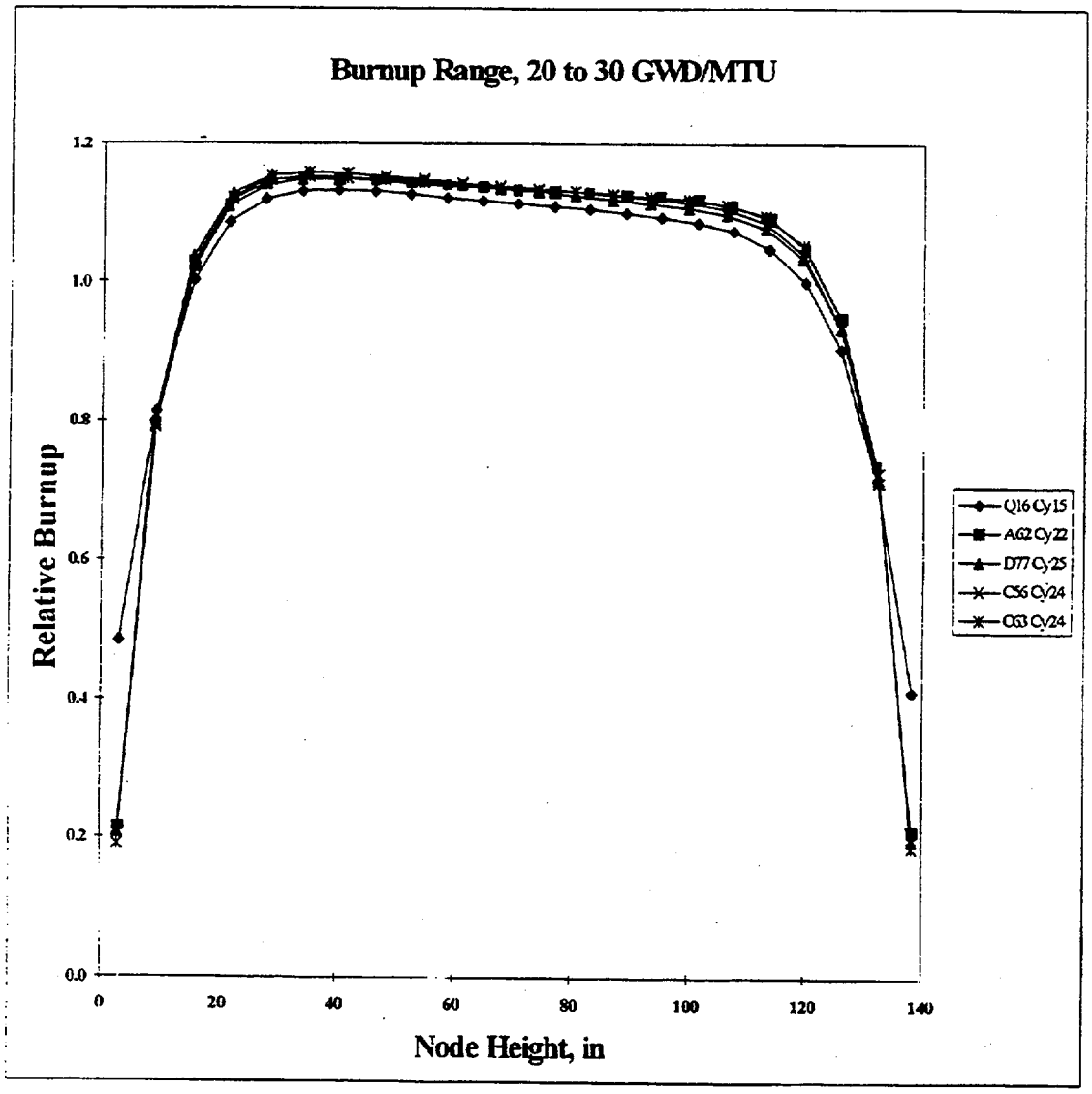


Figure 6.0-3
Typical Ginna Axial Burnup Shapes for Burnups Between 30 and 40 GWD/MTU

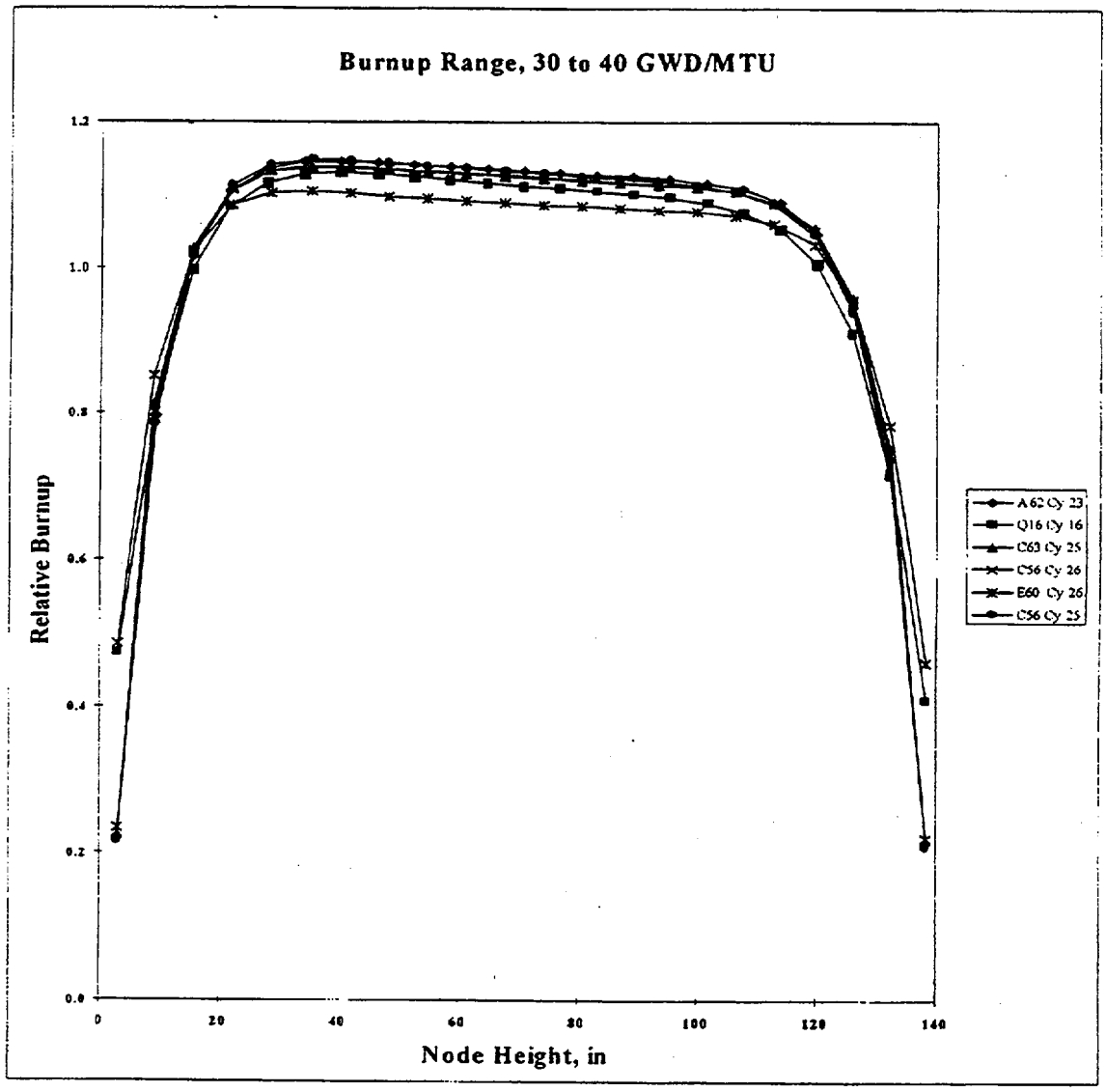


Figure 6.0-4
Typical Ginna Axial Burnup Shapes for Burnups Between 40 and 50 GWD/MTU

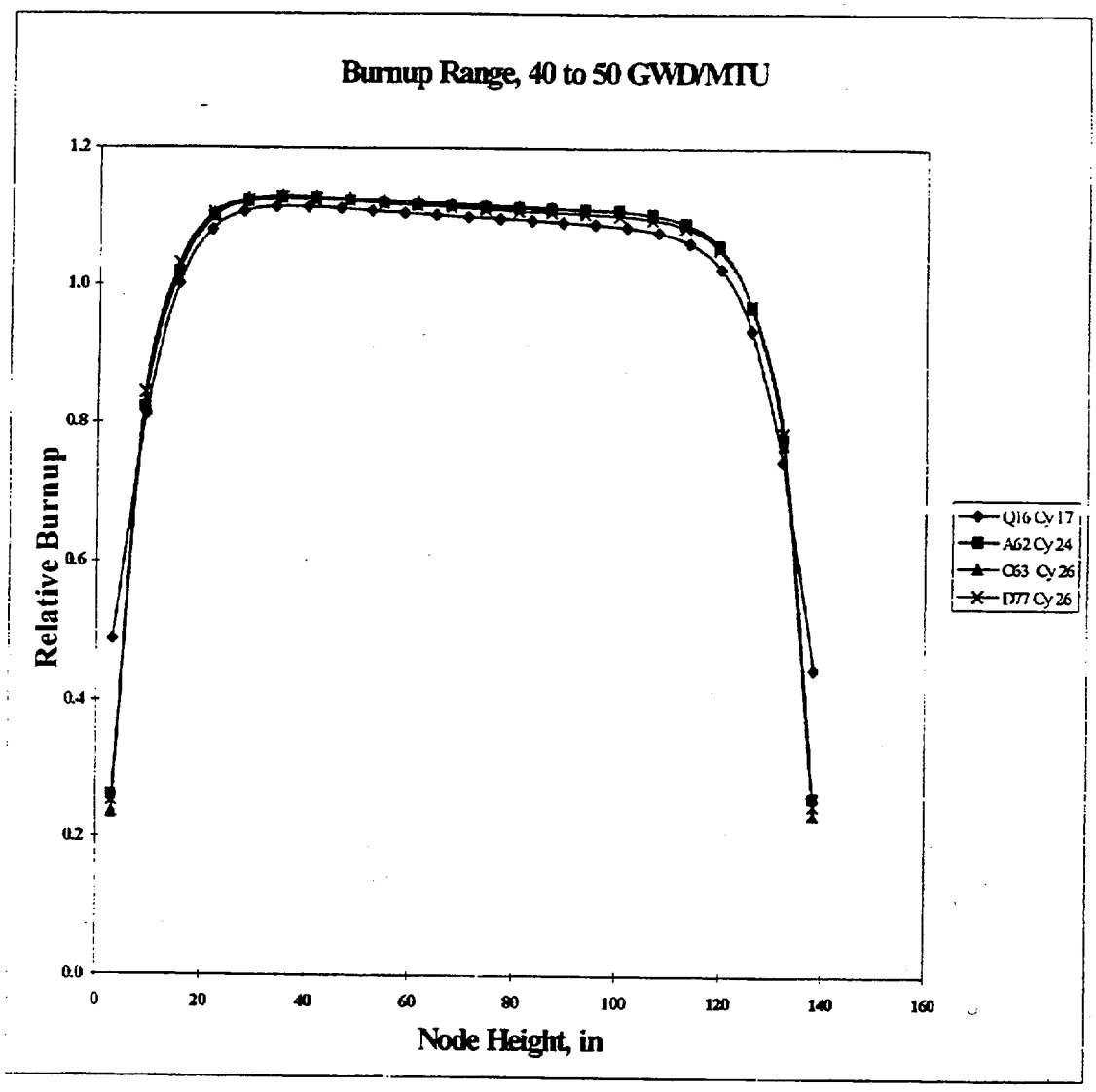


Figure 6.0-5
Non-Axial Blanket Shapes Used for Analysis

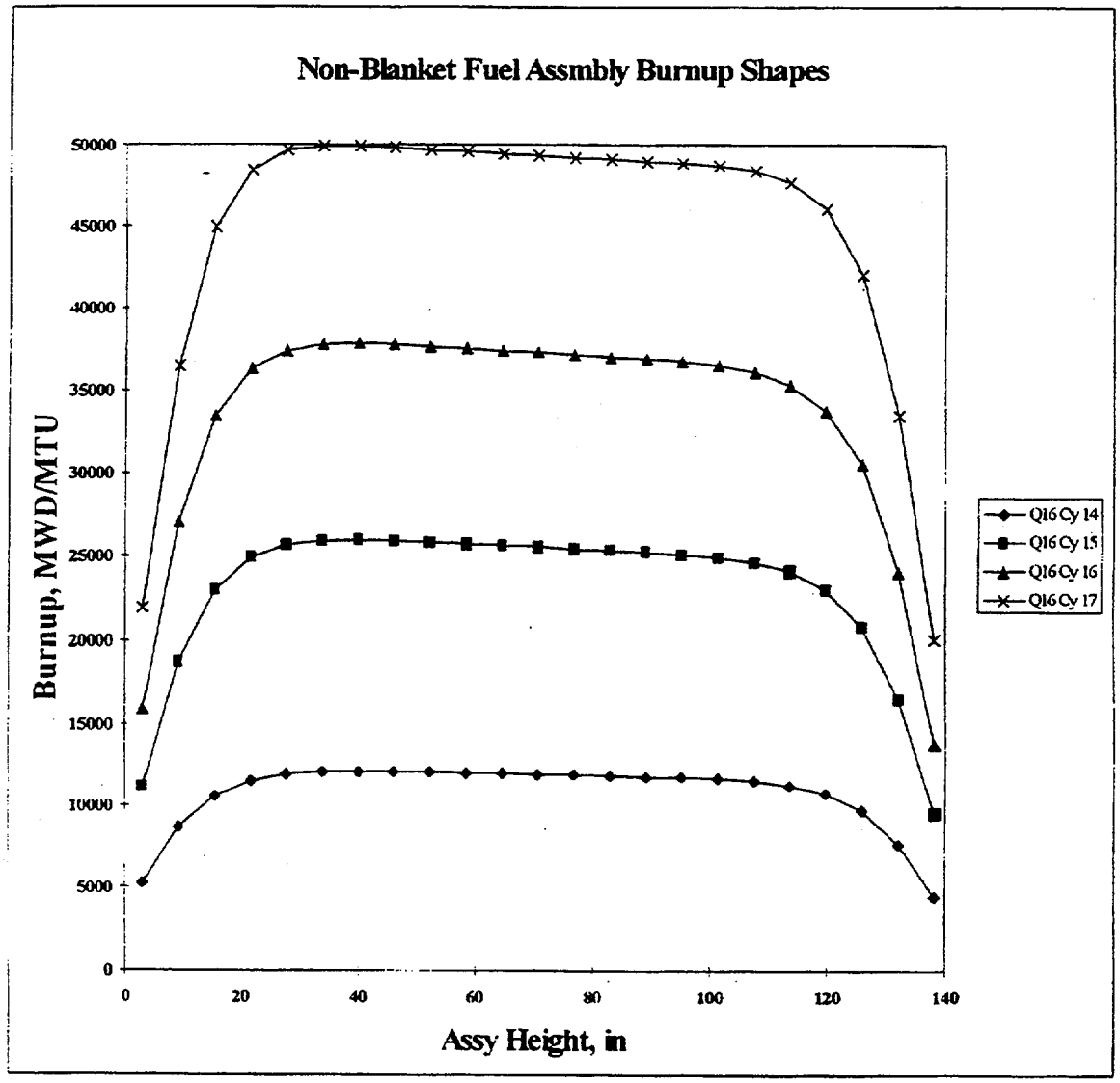


Figure 6.0-6
Relative Non-Blanket Axial Shapes Used in Analysis

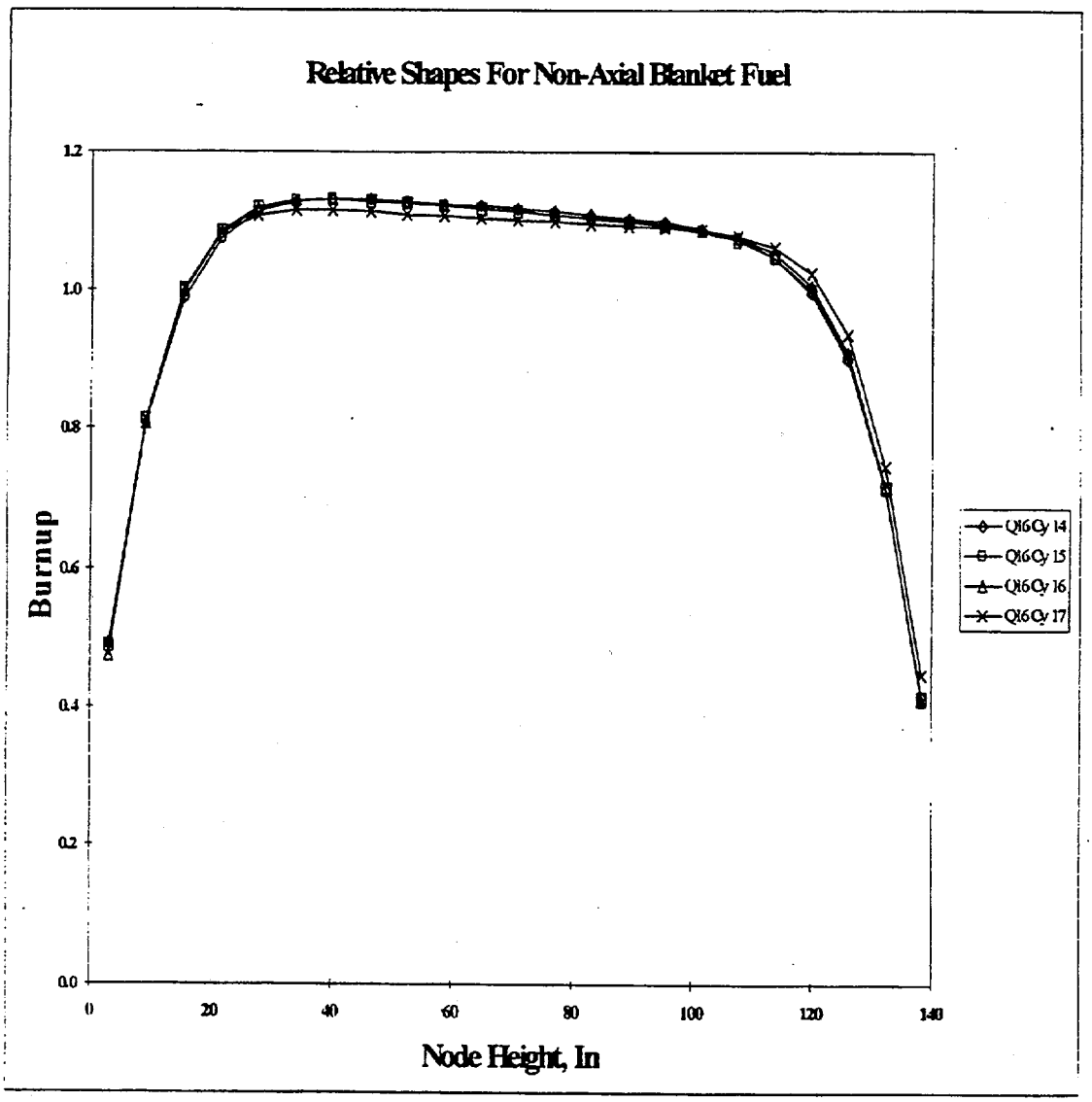


Figure 6.0-7
Illustration of the Eight Zone Axial Model for the Reference Axial Shape

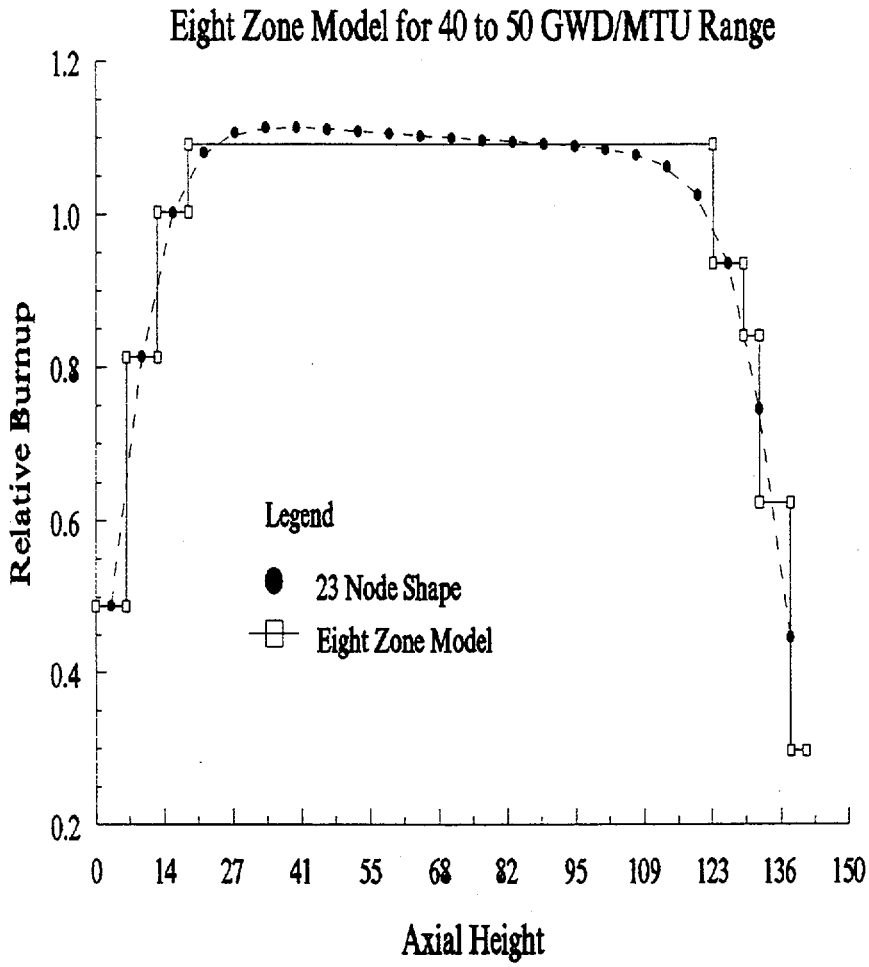
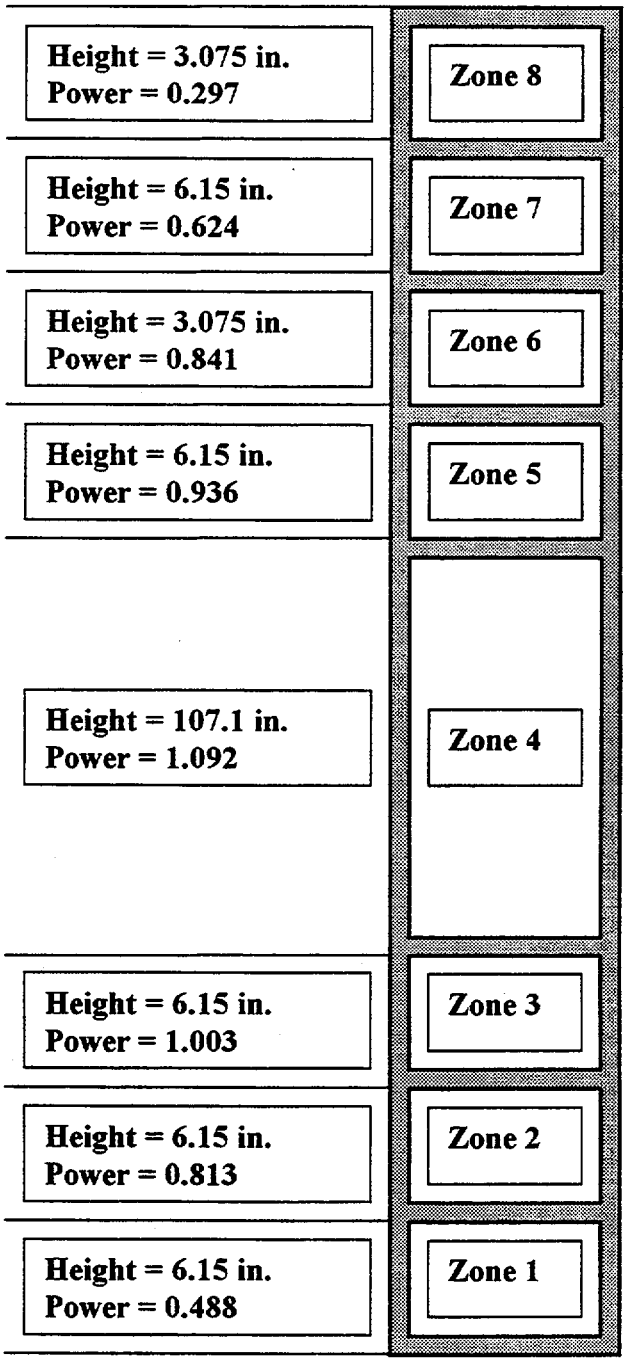


Figure 6.0-8
Sketch of Axial Zones Employed in Fuel Assembly



7.0 Tolerance/Uncertainty Evaluation for Individual Cell Types

Previous sections described the various types of fuel assembly storage racks within the spent fuel storage pool and the KENO models employed to represent infinite arrays of individual cell types and arrays of storage rack types. In addition, the method of modeling the axial profiles of fuel assembly burnup, moderator temperature, and fuel temperature were discussed in so far as their use in reactivity equivalencing fuel assemblies of different burnup histories are concerned.

Using the above input, analytic models were developed to perform the quantitative evaluations necessary to demonstrate the effective multiplication factor for the spent fuel pool is less than unity with zero boron present in the pool. Applicable biases to be factored into this evaluation are: (1) the methods bias deduced from the validation analyses of pertinent critical experiments, and (2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range of 50 to 212 °F.

A second allowance is based on a 95/95 confidence level assessment of tolerances and uncertainties; included in the summation of variances are the following:

a) the 95/95 confidence level methods variance,

b) the 95/95 confidence level calculational uncertainty,

c) fuel rod manufacturing tolerance,

d) storage rack fabrication tolerances,

e) tolerance due to positioning the fuel assembly in the storage cell.

Items a) and b) are based on the calculational methods validation analyses. For Item c), the fuel rod manufacturing tolerance for the reference design fuel assembly is assumed to consist of four components: an increase in fuel enrichment from 5.0 to 5.05 wt% ^{235}U, an increase in pellet density from 95 to 97 %TD, an increase in pellet OD from 0.3669 to 0.3677 inches, and a decrease in clad OD from 0.422 to 0.4195 inches; the individual contributions of each change are combined by taking the square root of the sum of the squares of each component.

For Item d), the following uncertainty components are evaluated. In the Type 1 racks, the L-insert thickness is decreased from 0.062 to 0.059 inches and the cell tube wall thickness is decreased from 0.090 to 0.086 inches. In the Type 2 racks, the SS wall thickness is decreased from 0.20 to 0.182 cm. and the gap between adjacent storage cells is decreased from 0.232 to 0.15 cm.; since the BSS wall thickness is taken at the minimum tolerance level in all analyses, no uncertainty component is calculated for this material. In the Type 3 racks, the SS wall thickness is decreased from 0.20 to 0.182 cm. and the cell pitch is decreased from 23.45 to 23.25 cm.; again, the BSS wall thickness is taken at the lower tolerance level for all calculations.

In the case of the tolerance due to positioning of the fuel assembly in the storage cells, all nominal calculations are carried out with fuel assemblies centered in the storage cells. Table 7.0-1 provides a summary of the KENO cases used in the calculation of biases and uncertainties for the zero soluble boron condition in the infinite array models for each rack type. Table 7.0-2 provides a summary of the biases and uncertainties calculated for the zero soluble boron condition in the infinite array models for each rack type.

Table 7.0-1
Keno Calculated K_{eff} values for the Various Physical Tolerance Cases

Case	Description	K_{eff} from KENO
Region 2 Type 1 Cells		
1	Nominal Case at 5.0 w/o U-235, Fresh	1.39469 ± 0.00075
2	Increase in U-235 Enrichment	1.39638 ± 0.00075
3	Increase in Stack Density	1.39543 ± 0.00072
4	Increase in Pellet Diameter	1.39392 ± 0.00075
5	Decrease in Clad Diameter	1.39487 ± 0.00070
Statistical Sum of F/A Manufacturing Tolerances		0.00427
6	Decrease in L-Insert Thickness	1.39495 ± 0.00075
7	Decrease in SS Wall Thickness	1.39514 ± 0.00070
Statistical Sum of Rack Fabrication Tolerances		0.00259
Region 2 Type 2 Cells		
8	Nominal Case at 5.0 w/o U-235, Fresh	1.22776 ± 0.00084
9	Increase in U-235 Enrichment	1.23061 ± 0.00089
10	Increase in Stack Density	1.22934 ± 0.00084
11	Increase in Pellet Diameter	1.22757 ± 0.00086
12	Decrease in Clad Diameter	1.22929 ± 0.00082
Statistical Sum of F/A Manufacturing Tolerances		0.00664
13	Decrease in Wall Thickness	1.22893 ± 0.00090
14	Decrease in Minimum Gap	1.23131 ± 0.00085
Statistical Sum of Rack Fabrication Tolerances		0.00599
Region 1 Type 3 Cells		
15	Nominal Case at 5.0 w/o U-235, Fresh	0.94890 ± 0.00094
16	Increase in U-235 Enrichment	0.95138 ± 0.00094
17	Increase in Stack Density	0.95098 ± 0.00092
18	Increase in Pellet Diameter	0.94896 ± 0.00091
19	Decrease in Clad Diameter	0.95181 ± 0.00094
Statistical Sum of F/A Manufacturing Tolerances		0.00782
20	Decrease in SS Wall Thickness	0.94969 ± 0.00092
21	Decrease in Cell Pitch	0.96094 ± 0.00092
Statistical Sum of Rack Fabrication Tolerances		0.01415

Table 7.0-2
Summary of Biases and Uncertainties for the Zero Soluble Boron Condition

Description	Region 1	Region 2		
	Type 3	Type 1	Type 2	Type 4(*)
Methodology Bias and Calculational Penalties				
KENO Bias (ΔK_{bias})	0.00259	0.00259	0.00259	0.00259
Penalties				
Pool Temperature Penalty (50 °F – 212 °F)	0.00133	0.00218	0.00207	0.00207
Assembly Off-Center Placement Penalty	0.00000	0.00000	0.00000	0.00000
Sum of Penalties (ΔK_{pen})	0.00133	0.00218	0.00207	0.00207
Total = $\Delta K_{bias} + \Delta K_{pen}$	**0.00392**	**0.00477**	**0.00466**	**0.00466**
Tolerance and Statistical Uncertainties				
Fuel Assembly Manufacturing Tolerance	0.00782	0.00427	0.00664	0.00664
Rack Fabrication Tolerance	0.01415	0.00259	0.00599	0.00599
KENO Methodology Uncertainty	0.00684	0.00684	0.00684	0.00684
Total (Statistically Combined)	**0.01755**	**0.00847**	**0.01126**	**0.01126**
Total Adjustment to K_{eff}	**0.02147**	**0.01324**	**0.01592**	**0.01592**

(*) – The biases and uncertainties for Region 2, Type 2 cells are conservatively applied to the Region 2, Type 4 cells.

8.0 No Soluble Boron 95/95 K_{eff} Calculational Results

8.1 Region 1 Type 3 Cells

One objective of this analysis was to demonstrate the existing spent fuel assembly burnup versus initial enrichment storage curve for Region 1 Type 3 storage cells remains valid. The Region 1 Type 3 cells employ a checkerboard storage pattern to store fresh and burned fuel assemblies in alternate locations. Figure 10.1-1 of Section 10 displays the existing spent fuel assembly burnup versus initial enrichment storage curve for Region 1 Type 3 cells; the latter data were obtained from Figure 4.1-1 of Reference 1. Note that the maximum fresh fuel enrichment assumed for the generation of the latter figure was 4.0 wt% ^{235}U. For enrichments greater than 4.0 wt% ^{235}U, Reference 1 supplied the necessary IFBA requirements to meet the criticality safety design basis.

One of the calculations contained in Table 4.1-3 of Reference 1 was reproduced for the purposes of this report. Specifically, fresh fuel assemblies with an enrichment of 4.0 wt% ^{235}U were checkerboarded with burned fuel assemblies with an equivalent fresh fuel enrichment equal to 2.22 wt% ^{235}U. The quoted KENO K_{eff} value from Table 4.1-3 of Reference 1 is 0.91977 without any biases or uncertainties. The KENO calculated K_{eff} value obtained with the methodology of this report is 0.92317, again without any biases or uncertainties. This comparison indicates good agreement for the stated storage configuration.

The enrichment of the fresh fuel assemblies in the checkerboard of Region 1 Type 3 cells was then increased to 5.0 w/o U-235. Again, the KENO model contained no IFBAs. The results of the KENO calculations are displayed in Table 8.1-1. The KENO calculated K_{eff} value, assuming no IFBAs, is 0.94890 without biases and uncertainties. The biases and uncertainties for Region 1 Type 3 cells reported in Table 7.0-2 as 0.02147 delta K_{eff} units. Therefore, the maximum K_{eff} value for Region 1 Type 3 cells assuming a checkerboard of fresh fuel assemblies with an enrichment equal to 5.0 w/o U235 (and no IFBAs) and burned fuel assemblies with a fresh equivalent enrichment equal to 2.22 w/o U-235 is less than 0.975 at zero ppm.

Therefore, Figure 10.1-1 of Section 10 can be employed to determine the burnup versus initial enrichment data for burned fuel assemblies checkerboarded with fresh fuel assemblies in Region 1 Type 3 cells. The fresh fuel assemblies in this configuration can have a maximum enrichment equal to 5.0 wt% ^{235}U (nominal) with no IFBAs.

8.2 Region 2 Type 1 Cells

As described in Section 6.0, the spent fuel storage rack analysis model employs an eight axial zone representation of spent/burned fuel assemblies in the storage racks in an evaluation of burnup credit for a given spent fuel assembly. In the case of the Type 1 storage cells, K_{eff} was evaluated for an infinite array of storage cells over a range of feed enrichment values up to 5 wt% ^{235}U and fuel assembly average burnups up to 55 GWD/T. An added dimension to this data is included by incorporating the effect of Pu-241 decay over 5-year intervals up to 20 years. These data are then employed to deduce the Base burnup limits versus initial feed enrichment and fuel assembly burnup for a given target K_{eff} value at zero soluble boron. The target value of K_{eff} is selected to be less than unity by an amount sufficient to cover the expected magnitude of analytical biases and uncertainties in these analyses, i.e., approximately 0.02 ΔK_{eff}.

More efficient utilization of the Type 1 storage racks can be attained by a further refinement of the Base burnup limits. This refinement involves the definition of Upper and Lower burnup limits which differ from the Base burnup limits by approximately ± 5000 MWD/MTU. Using this approach, burned assemblies which do not meet the Base burnup limit, but could meet the Lower burnup limit, are placed in a checkerboard arrangement with assemblies meeting the Upper burnup limit. The resulting K_{eff} for the checkerboard should meet the K_{eff} limit for the Type 1 storage racks.

Table 8.2-1 lists the KENO K_{eff} values computed with the eight axial zone model for the Type 1 storage cells versus ^{241}Pu decay time, feed enrichment, and fuel assembly average burnup. The K_{eff} values versus burnup, for each different Pu-241 decay time, were fitted to polynomial expressions as shown in Table 8.2-2 at initial enrichments equal to 3.0 w/o, 4.0 w/o, and 5.0 w/o U235. Figures 8.2-1 through 8.2-3 illustrate the variation of K_{eff} with burnup and decay time for initial enrichments equal to 3.0 w/o, 4.0 w/o, and 5.0 w/o U-235 respectively.

The Base burnup limits are evaluated using approximately 0.98 for the target K_{eff}, thus allowing about 0.02 for biases and uncertainties. The Upper and Lower burnup limits are deduced by an iterative process. Using the Base burnup limits, initial values of the Upper and Lower burnup limits are selected by using a delta of approximately 5000 MWD/MTU. Next, the Base, Upper, and Lower burnup values are converted to equivalent K_{eff} values using the appropriate polynomial fits to K_{eff} versus burnup and decay times. The resulting Upper and Lower burnup K_{eff} values are numerically averaged and compared to the Base burnup K_{eff} value. Depending on this comparison, adjustments to the positive and/or negative decrements in burnup may be made and the process is repeated until an acceptable level of convergence is obtained. Table 8.2-3 provides the converged set of Base, Upper and Lower burnup limits derived by this process. Table 8.2-4 contains the corresponding K_{eff} values obtained from the appropriate polynomial. The right-hand column in Table 8.2-4 lists the average K_{eff} value for the Upper and Lower columns.

To test the veracity of this iterative process using the numerical addition of the K_{eff} values for the checkerboarded cells, five test problems were run to compare the results of averaging the K_{eff} values deduced for the checkerboarded array of cells having fuel assemblies at the upper and lower burnup limits by: 1) averaging the K_{eff} values inferred from the polynomial fits, and 2) explicit KENO checkerboard results. Table 8.2-5 summarizes the results of these comparative calculations.

The entries in the K_{eff} (Average) column of Table 8.2-5 are based on a numerical averaging of the K_{eff} values deduced from the polynomial expressions for the two cells whereas the KENO values are for explicit eight axial zone checkerboard calculations. In general, the results of the explicit KENO calculations demonstrate that the averaging process produces good estimates for K_{eff}. The largest differences in K_{eff} are seen in the cases where there are large differences between the upper and lower assembly burnups, as in cases 3 and 5. In these cases, the numerically averaged value for K_{eff} is conservative relative to the explicit KENO calculated result. This latter effect can be attributed to the difference in the axial reactivity distribution between the two storage cells which is explicitly evaluated in the eight axial zone KENO calculation but neglected in the numerical averaging of K_{eff} values for adjacent cells.

To demonstrate that any "Upper" assembly (for any decay period) may be checkerboarded with any "Lower" assembly (for any decay period), a very simple and convincing argument was constructed. If the highest reactivity "Upper" assembly is checkerboarded with the highest reactivity "Lower" assembly and the resulting K_{eff}, with biases and uncertainties, is less than unity at zero soluble boron, then all checkerboard combinations would produce acceptable reactivity results. A survey of the K_{eff} values given in Table 8.2-4 indicates that the highest K_{eff} e value for any "Lower" assembly is 1.06568 (which corresponds to an assembly with an equivalent fresh fuel enrichment equal to 1.60 w/o) and that the highest K_{eff} value for any "Upper" assembly is 0.94384 (4.0 w/o, 43,155 MWD/T, and 20 years decay). Case 5 in Table 8.2-5 represents a close and conservative representation of this worst combination in that the "Upper" assembly was conservatively modeled with a K_{eff} value equal to 0.94443. The explicit KENO calculated result for this limiting combination is 0.97735, without biases and uncertainties. From Table 7.0-2, the biases and uncertainties applicable to the Region 2 Type 1 cells is 0.01324 delta K_{eff} units. Therefore, at zero soluble boron, the maximum expected K_{eff} value (including biases and uncertainties) for any checkerboard combination of "Upper" and "Lower" assemblies is 0.99059.

Table 8.2-6 contains the polynomial expressions for the Base, Upper, and Lower burnup curves for the 0, 5, 10, 15, and 20 years of ^{241}Pu decay. Note that these expressions incorporate an additional 5% allowance for burnup determination uncertainty.

8.3 Region 2 Type 2 Cells

The determination of the Base burnup limit for the Type 2 storage cells employs the same approach as in the previous section for the Type 1 cells. Since the Type 2 racks employ a checkerboard of SS and BSS storage cells, the K_{eff} value for a checkerboard array of fuel assemblies having the same initial feed enrichments but dissimilar burnups will depend upon which assembly is placed in the BSS storage cells. The more reactive condition occurs with the lower burnup assembly in the SS storage cell, consequently the more conservative design approach is to place fuel assemblies meeting the Upper burnup limit in the BSS storage cells and fuel assemblies meeting the Lower burnup limit in the SS storage cell. The reactivity effect of reversing the placement of assemblies varies between 0.005 to 0.025 units of ΔK.

Table 8.3-1 lists the KENO K_{eff} values computed with the eight axial zone model for the infinite array of checkerboarded Type 2 storage cells versus ^{241}Pu decay time, feed enrichment, and fuel assembly average burnup. As in the case of the Type 1 cells, the K_{eff} values versus burnup for each different ^{241}Pu decay time are fitted by polynomial expressions. The polynomial expressions are displayed in Table 8.3-2 for initial enrichments equal to 3.0 w/o, 4.0 w/o, and 5.0 w/o U-235. Figures 8.3-1 through 8.3-3 illustrate the variation of K_{eff} with burnup and decay time for initial enrichments equal to 3.0 w/o, 4.0 w/o, and 5.0 w/o U-235 respectively.

The Base burnup limits are again evaluated using approximately 0.98 for the target K_{eff} value and the Upper and Lower burnup limits are again evaluated by the iterative process employed for the Type 1 cells. Table 8.3-3 provides the converged set of Base, Upper and Lower burnup limits derived by this process. Table 8.3-4 contains the corresponding K_{eff} values obtained from the appropriate polynomial. The right-hand column in Table 8.3-4 lists the average K_{eff} value for the Upper and Lower columns.

To test the iterative process employed in determining the Upper and Lower burnup limits, the cases of Table 8.3-5 were run. The K_{eff} (KENO) column lists the KENO derived K_{eff} values obtained from the eight axial zone checkerboard calculation and the right-hand column lists the result obtained by the numerical averaging approach. The first case provides a comparison of the K_{eff} values derived from an explicit KENO calculation versus that from the numerical averaging of the Upper and Lower cell K_{eff} values; the difference in magnitude is small and the averaging method is conservative. The second case illustrates the effect of a checkerboard loading of a burned fuel assembly with an equivalent enrichment fresh assembly. In this case too, the averaging process overestimates the magnitude of K_{eff} since it does not include the effect of the decreased reactivity of the upper nodes for the fresh assembly. In the third case, fresh fuel equivalent fuel assemblies are employed in both type cells and the averaging process, once again conservatively agrees with the KENO calculation.

To demonstrate that any "Upper" assembly (for any decay period) may be checkerboarded with any "Lower" assembly (for any decay period), the same simple and convincing argument that was applied to the Type 1 cells was also applied to the Type 2 cells. If the highest reactivity "Upper" assembly is checkerboarded with the highest reactivity "Lower" assembly and the resulting K_{eff}, with biases and uncertainties, is less than unity at zero soluble boron, then all checkerboard combinations would produce acceptable reactivity results. A survey of the K_{eff} values given in Table 8.3-4 indicates that the highest K_{eff} value for any "Lower" assembly is 1.01217 (3.0 w/o U-235, 7550 MWD/T, 20 years decay) and that the highest K_{eff} value for any "Upper" assembly is 0.93536 (5.0 w/o U-235, 42,500 MWD/T, and 0 years decay). The numerical average of this checkerboard arrangement is 0.97760. (Note that this numerical average was obtained by weighting the K_{eff} of the "Lower" assembly by 55 % and the K_{eff} for the "Upper" assembly by 45 %. The relative weighting factors, 55 % and 45 %, were deduced to be conservative from explicit KENO calculations.) From Table 7.0-2, the biases and uncertainties applicable to the Region 2 Type 2 cells is 0.01592 delta K_{eff} units. Therefore, at zero soluble boron, the maximum expected K_{eff} value (including biases and uncertainties) for any checkerboard combination of "Upper" and "Lower" Type 2 assemblies is 0.99353.

Table 8.3-6 lists the polynomial expressions for the Base, Upper, and Lower burnup curves for the 0, 5, 10, 15, and 20 years of ^{241}Pu decay. Note that these expressions incorporate a 5% allowance for burnup determination uncertainty.

8.4 Region 2 Type 4 Cells

The full core KENO model was employed to address the applicability of the Region 2 Type 2 burnup versus enrichment curves for the storage of fuel assemblies in Region 2 Type 4 cells. The Type 4 cells were modeled as a single row of cells above and below the Type 1 cells. A water gap of 3 cm between the Type 4 and the Type 1 cells was utilized for this purpose. It is expected that the applicability of the Region 2 Type 2 curves will be conservative for the Region 2 Type 4 cells because they are located at the periphery of the spent fuel pool. The inherent leakage for Region 2 Type 4 cells will be higher than the leakage for interior cells. Two cases were analyzed for this model and the results are tabulated in Table 8.4-1. In one case, the Region 2 Type 4 cells were loaded with fresh fuel assemblies with the same enrichment as that of the Region 2 Type 2 cells and in the other case, the enrichment in the Region 2 Type 4 cells was reduced to 0.1 w/o ^{235}U. The KENO calculated results for these two cases indicate that the Region 2 Type 4 cells do not affect the multiplication factor for the rest of the spent fuel pool.

8.5 Entire Spent Fuel Pool

A KENO model for the entire spent fuel pool at the zero boron condition was constructed as discussed in Section 4.5 of this report. The following equivalent enrichments were employed for each of the cell types in this model which is intended to represent the most reactive representation of the full core model since it places burned fuel representations in the most reactive regions of the spent fuel pool.

Cell Type	Description
Region 1 Type 3	5.00 w/o Fresh in BSS cells and 2.22 w/o Fresh, zero ^{241}Pu decay in SS/BSS cells
Region 2 Type 1	3.0 w/o, 25 GWD/MTU, 15 Years ^{241}PU Decay F/A in all cells
Region 2 Type 2	5.0 w/o, 35 GWD/MTU, zero ^{241}Pu decay F/A in all cells
Region 2 Type 4	1.96 w/o Fresh F/A, zero ^{241}Pu decay in all cells

The calculated KENO multiplication factor for the spent fuel pool using this model is 0.98051 ± 0.00067, without biases and uncertainties, with no soluble boron. This multiplication factor for the spent fuel pool is driven by Region 2. Region 1 does not significantly contribute to the reactivity of the entire spent fuel pool because the infinite array multiplication factor for the Region 1 cells is less than 0.950; see Table 8.1-1. Therefore, the applicable biases and uncertainties term for the entire spent fuel pool can be based upon the maximum calculated value for Region 2. The maximum biases and uncertainty term for Region 2 cells is 0.01592. Thus, the final K_{eff} value, including biases and uncertainties, for the entire spent fuel pool model is 0.98051 + 0.01592 or 0.99643. Note that this value meets the criticality safety criterion for the spent fuel pool with no soluble boron.

Table 8.1-1
KENO Results for Infinite Array of Region 1 Type 3 Cells
(No Soluble Boron)

Case	Description	K_{eff} from KENO
1	4.00 w/o Fresh, OFA F/A in BSS Cell, 2.220 w/o Fresh F/A in SS/BSS Cell	0.92317 ± 0.00090
2	5.00 w/o Fresh, OFA F/A in BSS Cell, 2.220 w/o Fresh F/A in SS/BSS Cell	0.94890 ± 0.00094

Table 8.2-1
KENO K_{eff} Values versus Decay Time, Feed Enrichment, and Assembly Average Burnup for the Region 2, Type 1 Storage Cells with No Soluble Boron

Decay Time	w/o ^{235}U	KENO V.a K_{eff} Values						
		0 GWD/MTU	15 GWD/MTU	25 GWD/MTU	35 GWD/MTU	45 GWD/MTU	55 GWD/MTU	65 GWD/MTU
0.0	0.940	0.86511±0.00058						
	1.250	0.97420±0.00061						
	1.600	1.06568±0.00063						
	3.000	1.26696±0.00072	1.09337±0.00066	1.01357±0.00067	0.94652±0.00070			
	4.000	1.34250±0.00073		1.09849±0.00064	1.03296±0.00074	0.97916±0.00071	0.93200±0.00072	
	5.000	1.39469±0.00075			1.10334±0.00069	1.04455±0.00070	0.98642±0.00072	0.93141±0.00071
5 yr.	3.000	1.26696±0.00072	1.08397±0.00066	0.99827±0.00066	0.93115±0.00067			
	4.000	1.34250±0.00073		1.08787±0.00066	1.02209±0.00070	0.96130±0.00077	0.91246±0.00070	
	5.000	1.39469±0.00075			1.08896±0.00069	1.03386±0.00071	0.96953±0.00072	0.91061±0.00071
10 yr.	3.000	1.26696±0.00072	1.08010±0.00063	0.98945±0.00063	0.91894±0.00068			
	4.000	1.34250±0.00073		1.08035±0.00071	1.00957±0.00068	0.95106±0.00078	0.89831±0.00072	
	5.000	1.39469±0.00075			1.08093±0.00065	1.02122±0.00074	0.95536±0.00075	0.89300±0.00069
15 yr.	3.000	1.26696±0.00072	1.07414±0.00065	0.98034±0.00065	0.90961±0.00066			
	4.000	1.34250±0.00073		1.07349±0.00067	1.00317±0.00071	0.93984±0.00074	0.88648±0.00073	
	5.000	1.39469±0.00075			1.07390±0.00072	1.01562±0.00074	0.94443±0.00068	0.88327±0.00074
20 yr.	3.000	1.26696±0.00072	1.06911±0.00066	0.97568±0.00065	0.90030±0.00070			
	4.000	1.34250±0.00073		1.06930±0.00067	0.99653±0.00069	0.93271±0.00073	0.87716±0.00074	
	5.000	1.39469±0.00075			1.07041±0.00072	1.01083±0.00075	0.93746±0.00071	0.86936±0.00074

Table 8.2-2
Polynomial Expressions for KENO Derived K_{eff} Values versus Feed Enrichment and ^{241}Pu Decay Time for Region 2, Type 1 Cells

Feed Enrich. (w/o ^{235}U)	^{241}Pu Decay Time (yr.)	Polynomial for K_{eff} versus Burnup (GWD/MTU)
3.0	0	K_{eff} = 6.37500E-05 $(BU)^2$ – 1.05300E-02 (BU) + 1.23698E+00
	5	K_{eff} = 9.29000E-05 $(BU)^2$ – 1.22860E-02 (BU) + 1.24736E+00
	10	K_{eff} = 9.67000E-05 $(BU)^2$ – 1.28930E-02 (BU) + 1.25174E+00
	15	K_{eff} = 1.15350E-04 $(BU)^2$ – 1.39940E-02 (BU) + 1.25810E+00
	20	K_{eff} = 9.02500E-05 $(BU)^2$ – 1.29530E-02 (BU) + 1.24310E+00
4.0	0	K_{eff} = - 8.48333E-07 $(BU)^3$ + 1.47725E-04 $(BU)^2$ – 1.31048E-02 (BU) + 1.34704E+00
	5	K_{eff} = 1.16000E-06 $(BU)^3$ – 9.68500E-05 $(BU)^2$ – 3.92800E-03 (BU) + 1.22848E+00
	10	K_{eff} = - 1.08500E-06 $(BU)^3$ + 1.75275E-04 $(BU)^2$ – 1.46379E-02 (BU) + 1.35370E+00
	15	K_{eff} = 4.96667E-07 $(BU)^3$ - 1.72000E-05 $(BU)^2$ – 7.35342E-03 (BU) + 1.26032E+00
	20	K_{eff} = - 1.13333E-07 $(BU)^3$ + 5.66500E-05 $(BU)^2$ – 1.03672E-02 (BU) + 1.29484E+00
5.0	0	K_{eff} = 4.15000E-07 $(BU)^3$ – 5.28750E-05 $(BU)^2$ – 3.64837E-03 (BU) + 1.27798E+00
	5	K_{eff} = 2.44000E-06 $(BU)^3$ – 3.75550E-04 $(BU)^2$ + 1.27610E-02 (BU) + 9.97759E-01
	10	K_{eff} = 1.60833E-06 $(BU)^3$ – 2.47875E-04 $(BU)^2$ + 6.09879E-03 (BU) + 1.10216E+00
	15	K_{eff} = 3.82333E-06 $(BU)^3$ – 5.80700E-04 $(BU)^2$ + 2.21804E-02 (BU) + 8.45018E-01
	20	K_{eff} = 3.17667E-06 $(BU)^3$ – 4.97800E-04 $(BU)^2$ + 1.85386E-02 (BU) + 8.95165E-01

Table 8.2-3
Base, Upper, and Lower Burnups (MWD/MTU), without 5% Burnup Uncertainty, versus Initial Feed Enrichment and Pu-241 Decay Times for Infinite Array of Region 2, Type 1 Cells

Decay Time	w/o ^{235}U	Burnup (MWD/MTU)		
		Base	Upper	Lower
All	1.25	0		
	0.94		0	
	1.60			0
0.0	3.000	29690	36690	25190
	4.000	44825	52575	39825
	5.000	56167	63917	50667
5 yr.	3.000	27445	33445	23445
	4.000	41715	48465	37715
	5.000	53372	60122	49372
10 yr.	3.000	26235	31485	22235
	4.000	39765	46515	35765
	5.000	51259	58009	47259
15 yr.	3.000	25045	30045	21045
	4.000	38435	44435	34435
	5.000	50000	56000	46000
20 yr.	3.000	24530	29280	20330
	4.000	37405	43155	33205
	5.000	49202	54952	45002

Table 8.2-4
K_{eff} Values Evaluated from Polynomials for the Base, Upper, and Lower Burnups versus Enrichment and Pu-241 Decay Times for Region 2 Type 1 Cells with No Soluble Boron .

Decay Time	w/o ^{235}U	K_{eff} Values Deduced from Polynomials			
		Base	Upper	Lower	Average
All	1.25	0.97420			
	0.94		0.86511		
	1.60			1.06568	
0.0	3.000	0.98054	0.93645	1.01218	0.97432
	4.000	0.98003	0.94310	1.00585	0.97448
	5.000	0.97979	0.93714	1.01137	0.97425
5 yr.	3.000	0.98015	0.94037	1.01038	0.97537
	4.000	0.98030	0.94267	1.00480	0.97374
	5.000	0.98002	0.93775	1.00601	0.97188
10 yr.	3.000	0.98005	0.94166	1.01287	0.97727
	4.000	0.98056	0.94285	1.00474	0.97380
	5.000	0.98010	0.93578	1.00653	0.97116
15 yr.	3.000	0.97997	0.94178	1.01468	0.97823
	4.000	0.98048	0.94319	1.00699	0.97509
	5.000	0.98020	0.93748	1.00870	0.97309
20 yr.	3.000	0.97967	0.94121	1.01707	0.97914
	4.000	0.98038	0.94384	1.00891	0.97637
	5.000	0.98058	0.93782	1.01082	0.97432

Table 8.2-5
Verification of Checkerboarding Calculations for
Region 2 Type 1 Storage Cells with No Soluble Boron

Calc. No.	"Upper" Assembly Attributes	"Lower" Assembly Attributes	K_{eff} (KENO)	K_{eff} (Average)
1	Enrichment = 5.00 wt% Burnup = 55 GWD/MTU 20 Years Pu-241 Decay	Enrichment = 5.00 wt% Burnup = 45 GWD/MTU 20 Years Pu-241 Decay	0.97538	0.97415
2	Enrichment = 4.00 wt% Burnup = 45 GWD/MTU 15 Years Pu-241 Decay	Enrichment = 4.00 wt% Burnup = 45 GWD/MTU 10 Years Pu-241 Decay	0.97510	0.97471
3	Enrichment = 5.00 wt% Burnup = 55 GWD/MTU 20 Years Pu-241 Decay	Enrichment = 3.00 wt% Burnup = 25 GWD/MTU 0 Years Pu-241 Decay	0.97265	0.97552
4	Enrichment = 0.94 wt% Fresh Fuel	Enrichment = 1.60 wt% Fresh Fuel	0.97240	0.96540
5	Enrichment = 5.00 wt% Burnup = 55 GWD/MTU 15 Years Pu-241 Decay	Enrichment = 1.60 wt% Fresh Fuel	0.97735	1.00506

Table 8.2-6
Required Fuel Assembly Burnup (MWD/MTU) versus Initial Enrichment (w/o) for Base, Upper, and Lower Region 2 Type 1 Curves expressed as third-order polynomials (With 5% Burnup Uncertainty)

No Decay

Curve Type	Polynomial Describing the Curve
Base	$BU = [- 328.21 (w/o)^3 + 2042.01 (w/o)^2 + 12984.65 (w/o) - 18780.43] * 1.05$
Upper	$BU = [- 404.48 (w/o)^3 + 2582.27 (w/o)^2 + 12774.89 (w/o) - 13954.14] * 1.05$
Lower	$BU = [- 146.29 (w/o)^3 - 141.00 (w/o)^2 + 21034.78 (w/o) - 32695.48] * 1.05$

5 Years Decay

Curve Type	Polynomial Describing the Curve
Base	$BU = [- 211.40 (w/o)^3 + 1230.25 (w/o)^2 + 13479.90 (w/o) - 18359.26] * 1.05$
Upper	$BU = [- 316.33 (w/o)^3 + 2114.46 (w/o)^2 + 11923.01 (w/o) - 12813.22] * 1.05$
Lower	$BU = [- 80.78 (w/o)^3 - 337.13 (w/o)^2 + 19618.80 (w/o) - 30196.15] * 1.05$

10 Years Decay

Curve Type	Polynomial Describing the Curve
Base	$BU = [- 129.75 (w/o)^3 + 539.03 (w/o)^2 + 14557.64 (w/o) - 18785.86] * 1.05$
Upper	$BU = [- 415.02 (w/o)^3 + 3212.30 (w/o)^2 + 7899.84 (w/o) - 9919.52] * 1.05$
Lower	$BU = [- 11.16 (w/o)^3 - 884.09 (w/o)^2 + 20131.53 (w/o) - 29901.46] * 1.05$

15 Years Decay

Curve Type	Polynomial Describing the Curve
Base	$BU = [- 153.98 (w/o)^3 + 935.29 (w/o)^2 + 12540.31 (w/o) - 16836.04] * 1.05$
Upper	$BU = [- 332.21 (w/o)^3 + 2574.07 (w/o)^2 + 8663.42 (w/o) - 10142.14] * 1.05$
Lower	$BU = [- 67.14 (w/o)^3 - 106.83 (w/o)^2 + 16621.95 (w/o) - 26046.64] * 1.05$

20 Years Decay

Curve Type	Polynomial Describing the Curve
Base	$BU = [- 32.98 (w/o)^3 - 143.24 (w/o)^2 + 15097.94 (w/o) - 18584.19] * 1.05$
Upper	$BU = [- 228.66 (w/o)^3 + 1704.89 (w/o)^2 + 10401.10 (w/o) - 11093.56] * 1.05$
Lower	$BU = [43.24 (w/o)^3 - 1057.87 (w/o)^2 + 18680.22 (w/o) - 27357.33] * 1.05$

Table 8.3-1
KENO K_{eff} Values versus Decay Time, Feed Enrichment, and Assembly Average Burnup for an Infinite Checkerboard Array of Region 2 Type 2 Storage Cells with No Soluble Boron

Decay Time	w/o ^{235}U	KENO K_{eff} Values					
		0 GWD/MTU	5 GWD/MTU	15 GWD/MTU	25 GWD/MTU	35 GWD/MTU	45 GWD/MTU
0.0	2.200	1.01190±0.00077					
	1.960	0.97587±0.00076					
	1.700	0.92971±0.00074					
	3.000	1.10176±0.00080	1.04354±0.00079	0.95529±0.00076	0.88459±0.00070		
	4.000	1.17438±0.00085		1.03414±0.00077	0.96556±0.00074	0.91011±0.00074	
	5.000	1.22776±0.00084			1.02705±0.00075	0.97060±0.00072	0.92497±0.00075
5 yr.	3.000	1.10176±0.00080	1.04230±0.00079	0.94535±0.00078	0.87300±0.00072		
	4.000	1.17438±0.00085		1.02789±0.00077	0.95369±0.00074	0.89495±0.00077	
	5.000	1.22776±0.00084			1.01845±0.00075	0.96222±0.00078	0.91250±0.00079
10 yr.	3.000	1.10176±0.00080	1.04402±0.00076	0.94166±0.00075	0.86539±0.00070		
	4.000	1.17438±0.00085		1.02546±0.00075	0.94764±0.00073	0.88930±0.00073	
	5.000	1.22776±0.00084			1.01282±0.00080	0.95545±0.00076	0.90407±0.00079
15 yr.	3.000	1.10176±0.00080	1.04066±0.00079	0.93360±0.00073	0.85911±0.00072		
	4.000	1.17438±0.00085		1.02382±0.00075	0.94383±0.00077	0.87991±0.00079	
	5.000	1.22776±0.00084			1.00914±0.00079	0.94761±0.00077	0.89712±0.00076
20 yr.	3.000	1.10176±0.00080	1.04224±0.00077	0.93448±0.00076	0.85403±0.00069		
	4.000	1.17438±0.00085		1.02019±0.00075	0.93948±0.00075	0.87855±0.00078	
	5.000	1.22776±0.00084			1.00665±0.00076	0.94645±0.00076	0.89295±0.00078

Table 8.3-2
Polynomial Expressions for KENO Derived K_{eff} Values versus Feed Enrichment and ^{241}Pu Decay Time for Region 2, Type 2 Cells

Feed Enrich. (w/o ^{235}U)	^{241}Pu Decay Time (yr.)	Polynomial for K_{eff} versus Burnup (GWD/MTU)
3.0	0	K_{eff} = 8.77500E-05 (BU) ^{2} - 1.05800E-02 (BU) + 1.09425E+00
	5	K_{eff} = 1.23000E-04 (BU) ^{2} - 1.21550E-02 (BU) + 1.10000E+00
	10	K_{eff} = 1.30450E-04 (BU) ^{2} - 1.28450E-02 (BU) + 1.10498E+00
	15	K_{eff} = 1.62850E-04 (BU) ^{2} - 1.39630E-02 (BU) + 1.10640E+00
	20	K_{eff} = 1.36550E-04 (BU) ^{2} - 1.35070E-02 (BU) + 1.10636E+00
4.0	0	K_{eff} = 6.56500E-05 (BU) ^{2} - 9.48400E-03 (BU) + 1.16163E+00
	5	K_{eff} = 7.73000E-05 (BU) ^{2} - 1.05120E-02 (BU) + 1.16818E+00
	10	K_{eff} = 9.74000E-05 (BU) ^{2} - 1.16780E-02 (BU) + 1.17872E+00
	15	K_{eff} = 8.03500E-05 (BU) ^{2} - 1.12130E-02 (BU) + 1.17394E+00
	20	K_{eff} = 9.89000E-05 (BU) ^{2} - 1.20270E-02 (BU) + 1.17834E+00
5.0	0	K_{eff} = 5.41000E-05 (BU) ^{2} - 8.89100E-03 (BU) + 1.21551E+00
	5	K_{eff} = 3.25500E-05 (BU) ^{2} - 7.57600E-03 (BU) + 1.18751E+00
	10	K_{eff} = 2.99500E-05 (BU) ^{2} - 7.53400E-03 (BU) + 1.18245E+00
	15	K_{eff} = 5.52000E-05 (BU) ^{2} - 9.46500E-03 (BU) + 1.21127E+00
	20	K_{eff} = 3.35000E-05 (BU) ^{2} - 8.03000E-03 (BU) + 1.18646E+00

Table 8.3-3
Base, Upper, and Lower Burnups (MWD/MTU), without 5% Burnup Uncertainty, versus Initial Feed Enrichment and Pu-241 Decay Times for Infinite Array of Region 2, Type 2 Cells.

Decay Time	w/o ^{235}U	Burnup (MWD/MTU)		
		Base	Upper	Lower
All	1.96	0		
	1.55		0	
	2.20			0
0.0	3.000	12500	18500	9250
	4.000	23500	30500	19500
	5.000	34500	42500	29500
5 yr.	3.000	11500	17500	8250
	4.000	21750	28750	17750
	5.000	32500	40500	27500
10 yr.	3.000	11250	17250	8000
	4.000	21000	28000	17000
	5.000	31500	39500	26500
15 yr.	3.000	10950	15950	7700
	4.000	20650	26650	16650
	5.000	30500	37500	25500
20 yr.	3.000	10800	15800	7550
	4.000	20250	26250	16250
	5.000	30000	37000	25000

Table 8.3-4
K_{eff} Values Evaluated from Polynomials for the Base, Upper, and Lower Burnups versus Enrichment and Pu-241 Decay Times for Region 2, Type 2 Cells with No Soluble Boron

Decay Time	w/o ^{235}U	K_{eff} Values Deduced from Polynomials			
		Base	Upper	Lower	Average
All	1.96	0.97587			
	1.70		1.01190		
	2.20			0.92971	
0.0	3.000	0.97571	0.92855	1.00389	0.96999
	4.000	0.97501	0.93344	1.00166	0.97096
	5.000	0.97316	0.93536	1.00031	0.97108
5 yr.	3.000	0.97648	0.92496	1.00809	0.97068
	4.000	0.97611	0.92985	1.00595	0.97170
	5.000	0.97567	0.93407	1.00379	0.97241
10 yr.	3.000	0.97698	0.92222	1.01057	0.97081
	4.000	0.97644	0.92810	1.00834	0.97223
	5.000	0.97485	0.93159	1.00383	0.97132
15 yr.	3.000	0.97303	0.92512	1.00854	0.97100
	4.000	0.97665	0.93218	1.00952	0.97472
	5.000	0.97394	0.93396	1.00581	0.97347
20 yr.	3.000	0.97641	0.92704	1.01217	0.97386
	4.000	0.97535	0.93078	1.00902	0.97381
	5.000	0.97571	0.93521	1.00665	0.97450

Table 8.3-5
Verification of Checkerboarding Calculations for Region 2 Type 2 Storage Cells with No Soluble Boron

Calc. No.	"Upper" Assembly Attributes	"Lower" Assembly Attributes	K_{eff} KENO	K_{eff} (Average)
1	Enrichment = 4.00 w/o Burnup = 25 GWD/MTU 20 Years Pu-241 Decay	Enrichment = 5.00 w/o Burnup = 25 GWD/MTU 15 Years Pu-241 Decay	0.97452	0.97779
2	Enrichment = 4.00 w/o Burnup = 25 GWD/MTU 20 Years Pu-241 Decay	Enrichment = 2.20 w/o Fresh Fuel	0.96902	0.97931
3	Enrichment =1.70 w/o Fresh Fuel	Enrichment = 2.20 w/o Fresh Fuel	0.97322	0.97491

Table 8.3-6
Required Fuel Assembly Burnup (MWD/MTU) versus Initial Enrichment (w/o) for Base, Upper, and Lower Region 2 Type 2 Curves expressed as third-order polynomials (With 5% Burnup Uncertainty)

No Decay

Curve Type	Polynomial Describing the Curve
Base	BU = [164.35 (W/O) ^{3} - 1972.20 (W/O) ^{2} + 18724.43 (W/O) - 30360.98] * 1.05
Upper	BU = [89.75 (W/O) ^{3} - 1077.01 (W/O) ^{2} + 16218.30 (W/O) - 22885.06] * 1.05
Lower	BU = [215.77 (W/O) ^{3} - 2714.29 (W/O) ^{2} + 21266.37 (W/O) - 35946.43] * 1.05

5 Years Decay

Curve Type	Polynomial Describing the Curve
Base	BU = [212.48 (W/O) ^{3} - 2299.71 (W/O) ^{2} + 18486.38 (W/O) - 28998.57] * 1.05
Upper	BU = [169.35 (W/O) ^{3} - 1782.25 (W/O) ^{2} + 17459.64 (W/O) - 23411.25] * 1.05
Lower	BU = [205.85 (W/O) ^{3} - 2345.24 (W/O) ^{2} + 18300.10 (W/O) - 31101.19] * 1.05

10 Years Decay

Curve Type	Polynomial Describing the Curve
Base	BU = [295.46 (W/O) ^{3} - 3170.49 (W/O) ^{2} + 21011.49 (W/O) - 31227.44] * 1.05
Upper	BU = [244.34 (W/O) ^{3} - 2557.11 (W/O) ^{2} + 19609.08 (W/O) - 25160.53] * 1.05
Lower	BU = [287.70 (W/O) ^{3} - 3202.38 (W/O) ^{2} + 20771.83 (W/O) - 33261.90] * 1.05

15 Years Decay

Curve Type	Polynomial Describing the Curve
Base	BU = [158.32 (W/O) ^{3} - 1824.86 (W/O) ^{2} + 16616.11 (W/O) - 26749.29] * 1.05
Upper	BU = [57.23 (W/O) ^{3} - 611.78 (W/O) ^{2} + 12864.88 (W/O) - 18683.90] * 1.05
Lower	BU = [116.07 (W/O) ^{3} - 1442.86 (W/O) ^{2} + 14755.36 (W/O) - 26714.29] * 1.05

20 Years Decay

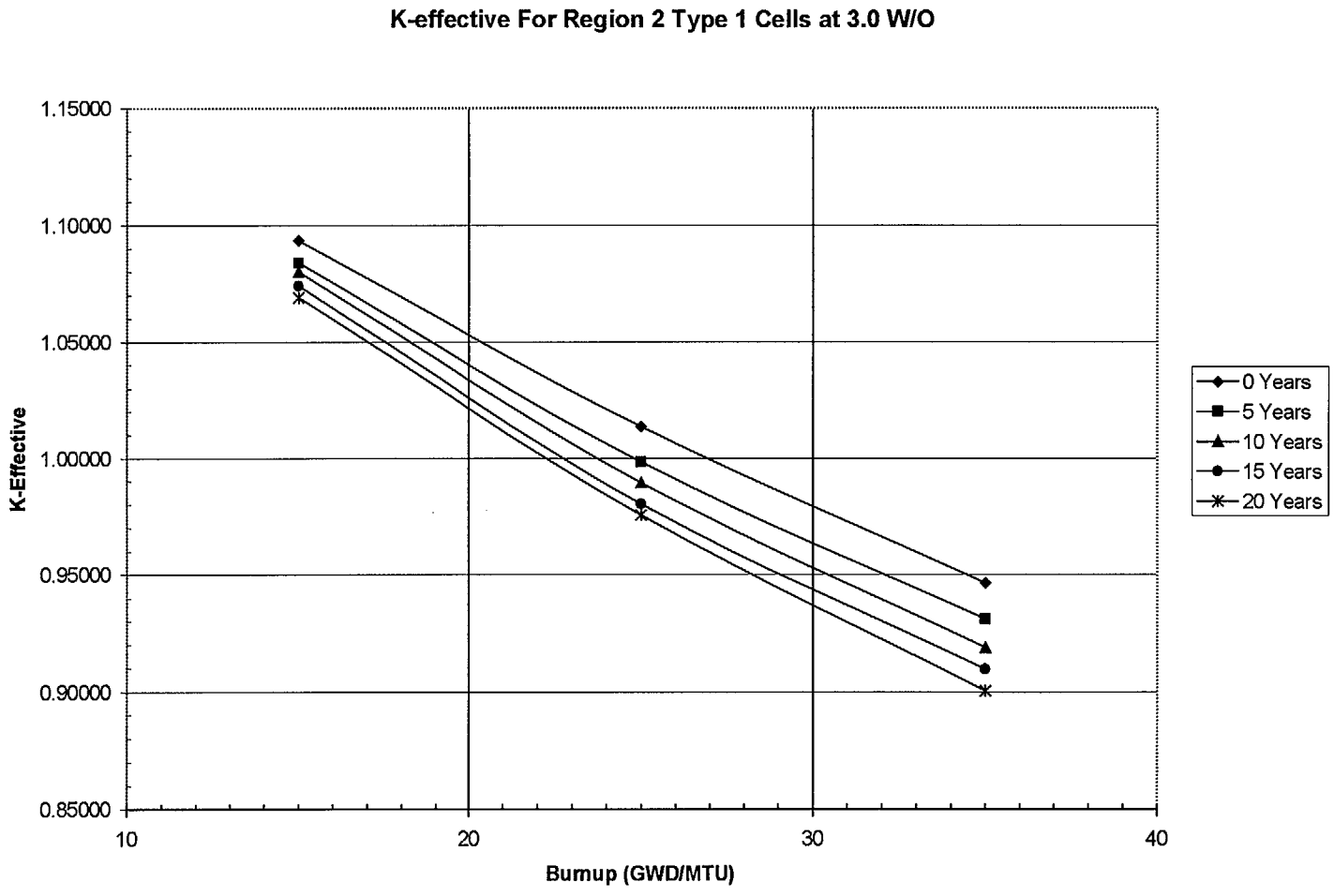
Curve Type	Polynomial Describing the Curve
Base	BU = [200.05 (W/O) ^{3} - 2250.57 (W/O) ^{2} + 17802.24 (W/O) - 27752.86] * 1.05
Upper	BU = [96.31 (W/O) ^{3} - 1005.71 (W/O) ^{2} + 13926.52 (W/O) - 19528.54] * 1.05
Lower	BU = [155.26 (W/O) ^{3} - 1838.10 (W/O) ^{2} + 15822.12 (W/O) - 27565.48] * 1.05

Table 8.4-1
KENO Results for Entire Fuel Pool (No Soluble Boron)

Case	Description	K_{eff} from KENO
1	F/A with Equivalent Fresh w/o in all Cells*, 1.960 w/o Fresh F/A in Type 4 Cell	0.97228 ± 0.00063
2	F/A with Equivalent Fresh w/o in all Cells*, 0.100 w/o Fresh F/A in Type 4 Cell	0.97273 ± 0.00062

* 5 wt% Fresh F/A in Region 1 Type 3 cells with 2.22 wt% Fresh in the SS/BSS cells; 1.25 wt% Fresh F/A in Region 2 Type 1 cells; 1.96 wt% fresh F/A in Region 2 Type 2 cells. All fuel representations with zero ^{241}Pu decay.

Figure 8.2-1



KENO K_{eff} Values for Region 2 Type 1 Cells, for 3.0 W/O

Figure 8.2-2
KENO K_{eff} Values for Region 2 Type 1 Cells, 4.0 W/O

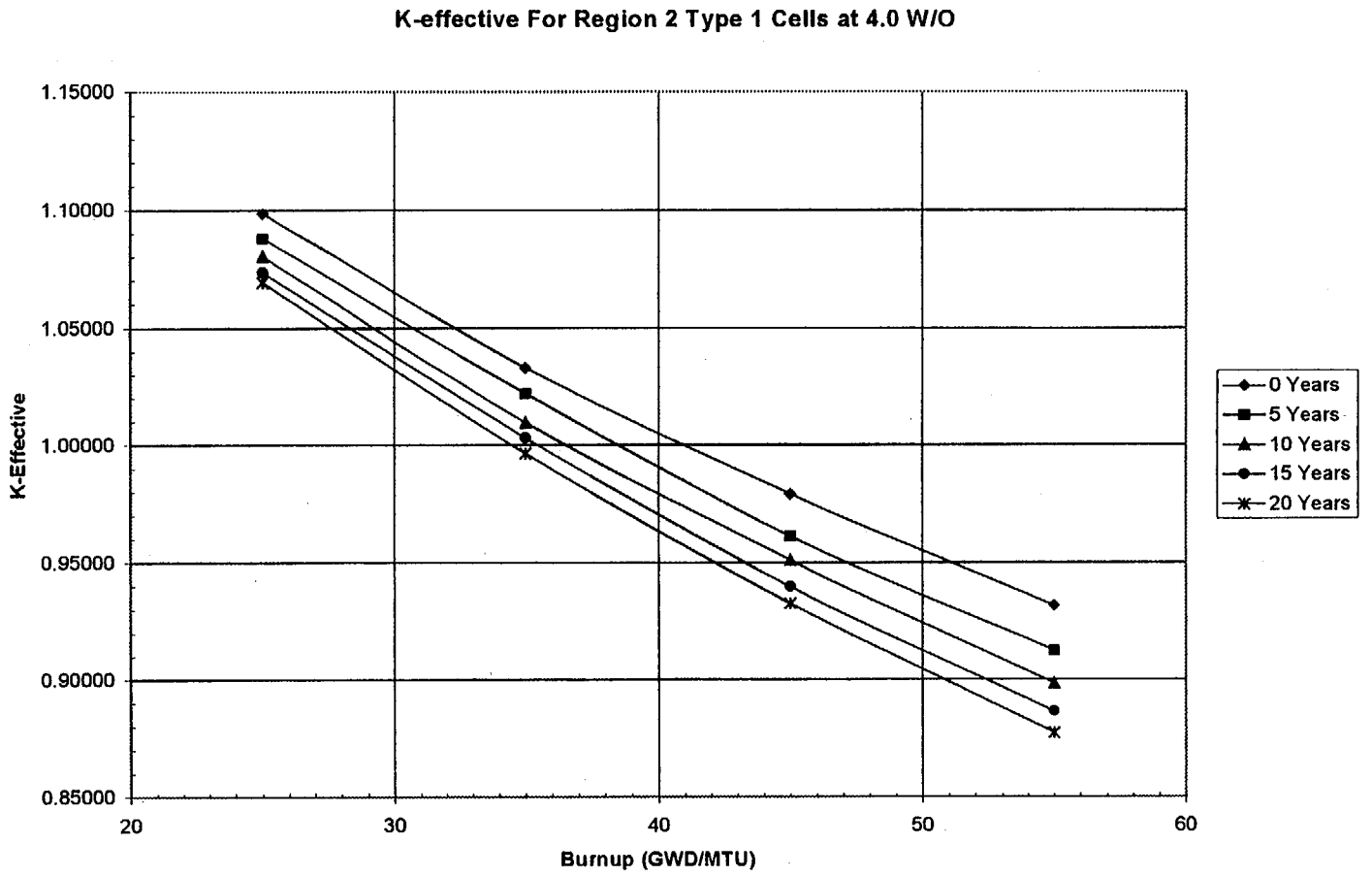


Figure 8.2-3
KENO K_{eff} Values for Region 2 Type 1 Cells, 5.0 W/O

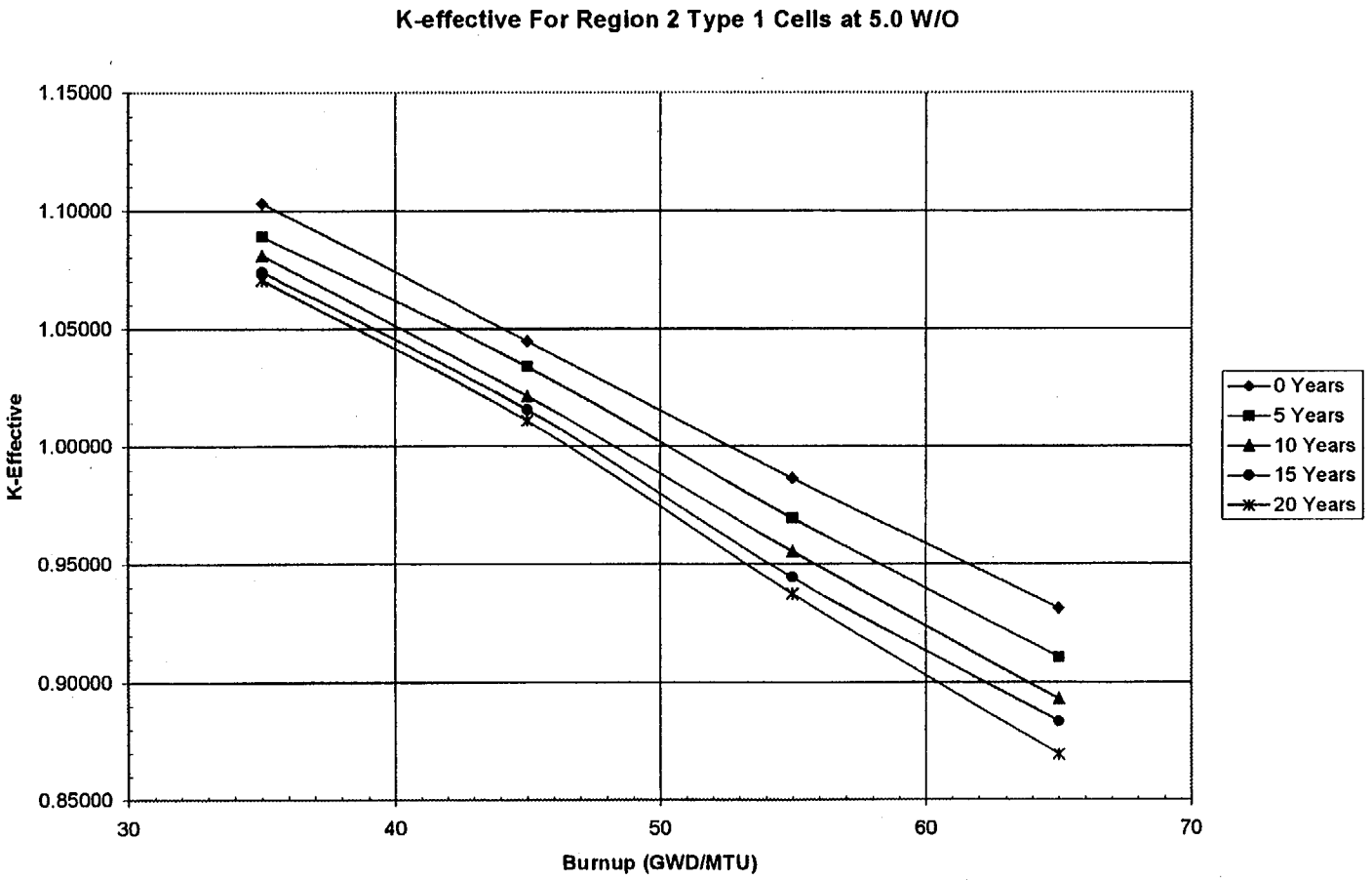


Figure 8.3-1
KENO K_{eff} Values for Region 2 Type 2 Cells, 3.0 W/O

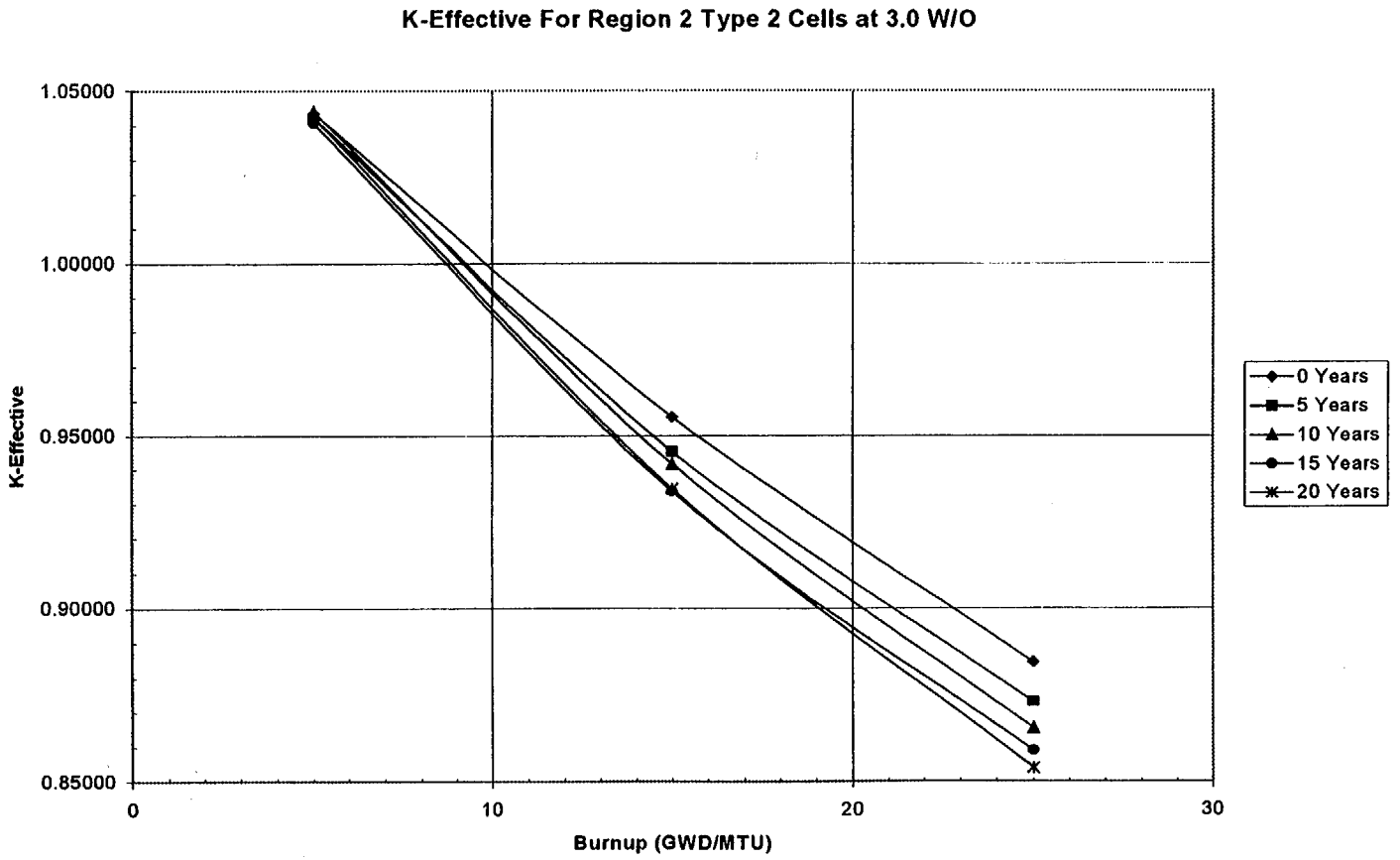


Figure 8.3-2
KENO K_{eff} Values for Region 2 Type 2 Cells, 4.0 W/O

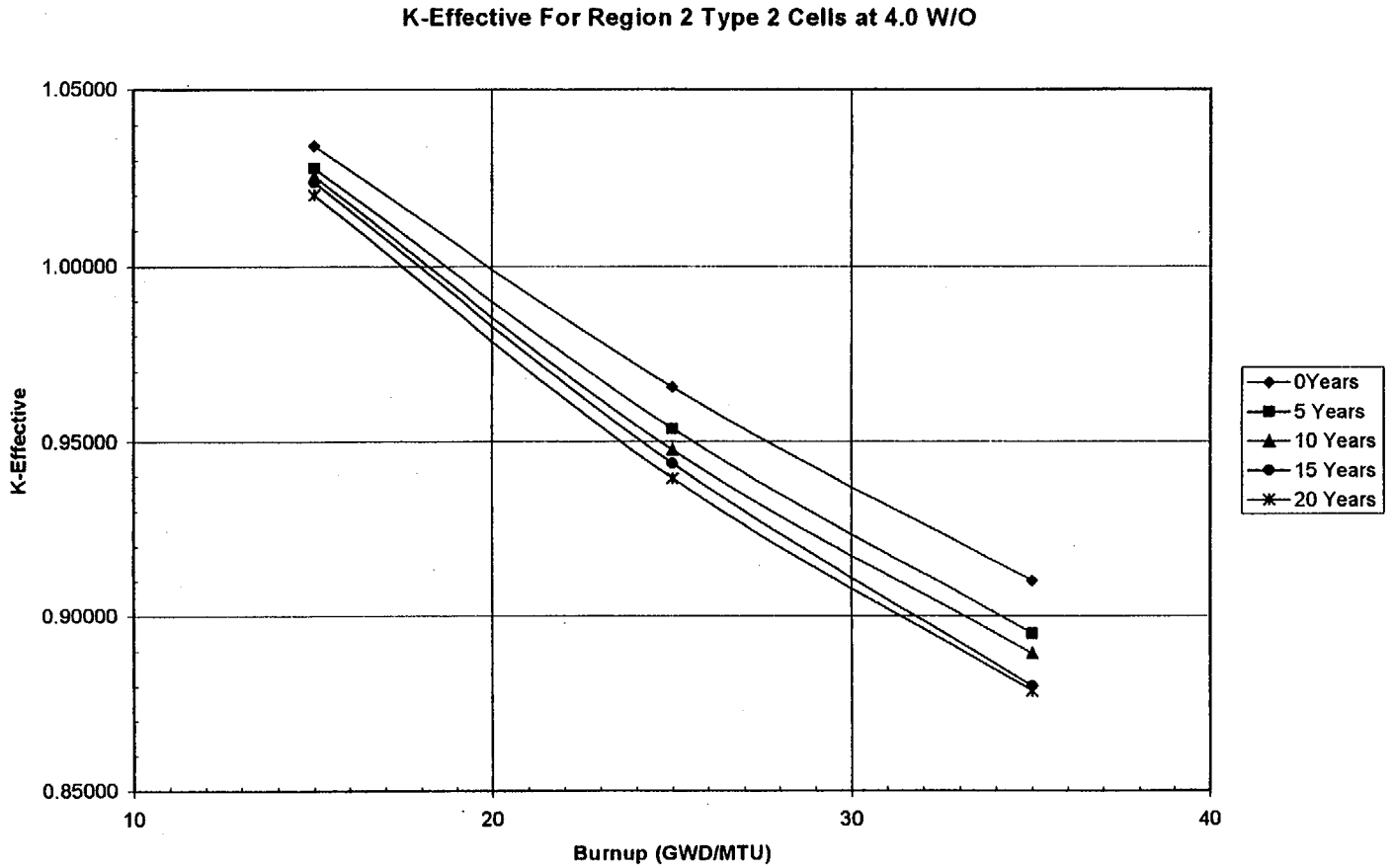
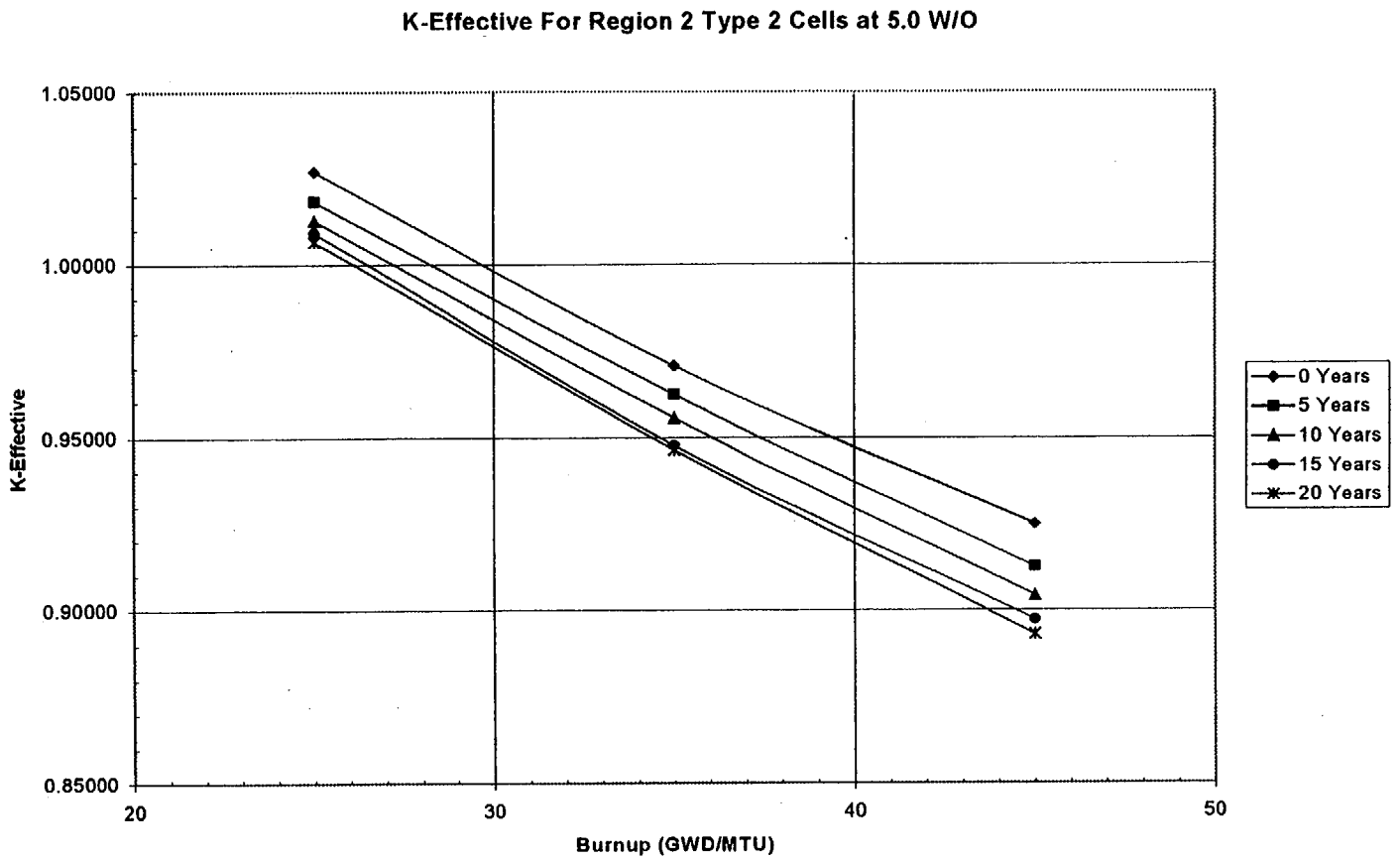


Figure 8.3-3
KENO V.a K_{eff} Values for Region 2 Type 2 Cells, 5.0 W/O



9.0 Soluble Boron K_{eff} Calculational Results

The NRC Safety Evaluation Report (SER) for WCAP-14416-P is given in Reference 3; Page 9 of the enclosure to Reference 3 defines the soluble boron requirement as follows. The total soluble boron credit requirement is defined as the sum of three quantities:

$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm),

$SBC_{95/95}$ = soluble boron requirement for 95/95 $K_{eff} \leq 0.95$ (ppm),

SBC_{RE} = soluble boron required for reactivity equivalencing methodologies (ppm),

SBC_{PA} = soluble boron required for $K_{eff} \leq 0.95$ under accident conditions (ppm).

Each of these terms will be discussed in the following subsections.

9.1 Soluble Boron Determination to Maintain K_{eff} Less Than 0.95

Tables 9.1-1 and 9.1-2 contain KENO calculated multiplication factors for the entire Ginna spent fuel pool at 0, 200 and 400 ppm of soluble boron. Table 9.1-1 contains this data assuming the fresh fuel equivalent enrichments for all the cells. The values utilized in the KENO calculations are also listed in Table 9.1-1. Table 9.1-2 contains similar data except that Region 2, Type1 and Region 2, Type 2 cells were simulated with "burned fuel descriptions (based on the eight zone axial model)." The burned fuel utilized in this calculation, whose description is listed in Table 9.1-2, is based on highly burnt fuel assemblies in the Type 1 and Type 2 cells. The two sets of data were generated to demonstrate that the soluble boron worth is different between fresh equivalent fuel assemblies and realistically described burned fuel assemblies.

The last column in Table 9.1-1 and Table 9.1-2 is labeled "Delta K_{eff}." Delta K_{eff} is equal to the K_{eff} (plus the one sigma value) of the case with soluble boron minus K_{eff} (less the one sigma value) of the unborated spent fuel pool. The reference K_{eff} value of the unborated spent fuel pool (and the associated one sigma value) is given in each table at zero ppm.

Since the soluble boron worth in Table 9.1-2 is less than the soluble boron worth in Table 9.1-1, the soluble boron worth data from Table 9.1-2 will be employed to deduce the soluble boron concentration necessary to maintain K_{eff} less than 0.95 (including biases and uncertainties). and to compensate for the reactivity equivalencing methodologies which could increase the multiplication factor of the spent fuel pool.

The amount of soluble boron required to maintain K_{eff} less than 0.95 including biases and uncertainties is determined based on the results from Table 9.1-2. The soluble boron concentration (ppm) required to reduce the K_{eff} of the entire spent fuel pool by 0.050 delta K_{eff} units is conservatively determined by employing the minimum soluble boron worth data contained in Table 9.1-2. By linear interpolation of the data in Table 9.1-2 (between 200 and 400 ppm) the amount of soluble boron required to reduce the K_{eff} of the entire spent fuel pool by 0.050 delta K_{eff} units is equal to 377.0 ppm.

9.2 Soluble Boron Determination for Reactivity Equivalencing Methods

The soluble boron credit (ppm) required for reactivity equivalencing methodologies was determined by converting the uncertainty in fuel assembly reactivity and the uncertainty in absolute fuel assembly burnup values to a soluble boron concentration (ppm) necessary to compensate for these two uncertainties. The first term, uncertainty in fuel assembly reactivity, is calculated by employing a depletion reactivity uncertainty equal to 0.005 delta K_{eff} units per 30,000 MWD/T of assembly burnup (obtained from Reference 3) and multiplying by the maximum amount of assembly burnup credited in a Region analysis. The highest assembly burnup credited is 63,917 MWD/T; this value is employed for Region 2 Type 1 cells at an initial fuel assembly enrichment equal to 5.0 w/o ^{235}U. Therefore, the uncertainty in fuel assembly reactivity is equal to 0.01065 delta K_{eff} units.

The uncertainty in absolute fuel assembly burnup values is conservatively calculated as 5% of the maximum fuel assembly burnup credited in a Region analysis. The maximum fuel assembly burnup credited in this analysis is 63,917 MWD/T. Such a fuel assembly is used in Region 2 Type 1 cells at an initial fuel enrichment of 5.0 wt% ^{235}U. The uncertainty in the burnup value is determined to be 3,200 MWD/T. The reactivity associated with a delta-burnup of 3,200 MWD/T at 60,000 MWD/T for a Region 2 Type 1 cell is calculated to be 0.01740 delta K_{eff} units.

The total of these two reactivity effects is equal to 0.02805 (0.01065+0.01740) delta K_{eff} units. By linear interpolation of the data in Table 9.1-2 the soluble boron concentration (ppm) necessary to compensate for this reactivity is equal to 207 ppm.

9.3 Soluble Boron Determination to Mitigate Accidents

The soluble boron concentration (ppm) required to maintain K_{eff} less than or equal to 0.95 under accident conditions is determined by first surveying all possible events which increase the K_{eff} value of the spent fuel pool. The accident event which produced the largest increase in spent fuel pool K_{eff} value is employed to determine the required soluble boron concentration necessary to mitigate this and all less severe accident events.

Several fuel mishandling events and one seismic event were simulated with KENO to assess the possible increase in the K_{eff} value of the Ginna spent fuel pool. The fuel mishandling events all assumed that a fresh 5.0 w/o ^{235}U assembly (*Accident* F/A) with no IFBAs was mislocated into any cell of the spent fuel pool intended for less reactive fuel assemblies. The fuel mishandling events assumed for this analysis did not consider the possibility of locating a fresh fuel assembly in one of the Region 2 Type 4 locations because the effect is conservative relative to a mishandling accident in the Region 2 Type 2 cells. The seismic event results in the reduction of the gap between all the modules to 0.1 cm. The inter-module gap was originally simulated based on the minimum fabricated values.

A survey of the various fuel mishandling events considered in an earlier analysis (Section 4.3.6, Reference 1) indicated that the most disruptive accident would be the misplacement of a fresh fuel assembly with 5.0 w/o ^{235}U in a Type 1 location. Therefore, all the other mishandling events were simulated with equivalent fresh fuel assemblies in all the cells, this event was simulated with burnt fuel assemblies occupying the Region 2 Type 1 and Type 2 locations. Table 9.3-1 shows the various mishandling accidents (including the location) simulated and the associated KENO calculated K_{eff} value. The description of the fuel assemblies occupying the cells is the same as that shown in Table 9.1-1. Table 9.3-2 shows the mishandling accident in the Type 1 cells and the burned fuel description is provided in Table 9.3-2. In order to determine the amount of soluble boron required to mitigate this accident, the same scenario is simulated, this time at a soluble boron concentration of 400 ppm.

The last column in Table 9.3-1 is labeled "Delta K_{eff}". Delta K_{eff} is equal to the K_{eff} (plus the one sigma value) of the fuel mishandling scenario minus K_{eff} (less the one sigma value) of the nominal spent fuel pool. The reference K_{eff} value of the spent fuel pool (and the associated one sigma value) provided in Table 9.3-1 was obtained from Table 9.1-1. The reference K_{eff} value for the spent fuel pool for the scenario described in Table 9.3-2 is based on a KENO simulation and is provided in the Table.

The data in Table 9.3-1 and Table 9.3-2 indicate that the highest worth of a fuel mishandling event is based on a misloading in the Type 1 racks with a worth of 5.953 % delta K_{eff}. Simulation of the same accident at a soluble boron concentration of 400 ppm indicates that it is more than sufficient to mitigate the accident.

The soluble boron concentration (ppm) necessary to compensate for this reactivity insertion is conservatively calculated, based on linear interpolation, from the results given on Table 9.3-2, to be equal to 381 ppm.

As noted above, the misplaced fuel assembly was cited in Reference 1 as being the most adverse postulated mishandling event. Other dropped assembly events such as, for example, the postulated Rack Type 2 T-bone fuel assembly accident configuration and the postulated deep drop type accidents, have little effect on the local K_{eff} of the spent fuel storage rack as indicated by the analyses cited in Table 4.3-14 of Reference 1. In the case of the T-bone accident, the ΔK_{eff} resulting from a fuel assembly lying on top of the storage rack is quite minimal due to the relatively large separation distance between the top of the fuel columns for the assemblies standing vertical in the storage rack and the fuel assembly lying across the top of the rack. In the case of the postulated deep drop accident, the deflection of the base plate is limited by the height of the pedestals supporting the rack above the concrete floor and the structural design of the rack. A significant fraction of the base plate deflection distance would be taken up by the fuel assembly structure below the active fuel columns. Thus, one would again conclude the misplaced fuel assembly accident overshadows the reactivity insertion resulting from the postulated deep drop event.

9.4 Summary of Soluble Boron Requirements

Soluble boron in the spent fuel pool coolant is used in this criticality safety analysis to offset the reactivity allowances for calculational uncertainties in modeling, storage rack fabrication tolerances, fuel assembly design tolerances, and postulated accidents. The total soluble boron requirement, SBC_{TOTAL}, is defined by the following equation.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm),

$SBC_{95/95}$ = soluble boron requirement for 95/95 $K_{eff} \leq 0.95$ (ppm),

SBC_{RE} = soluble boron required for reactivity equivalencing methodologies (ppm),

SBC_{PA} = soluble boron required for $K_{eff} \leq 0.95$ under accident conditions (ppm).

The magnitude of the above components is:

$SBC_{95/95}$ = 377 ppm

SBC_{RE} = 207 ppm

SBC_{PA} = 381 ppm

SBC_{TOTAL} = 965 ppm

Therefore, a total of 965 ppm of soluble boron is required to maintain K_{eff} less than 0.95 (including all biases and uncertainties) assuming the most limiting single fuel mishandling accident. Note that this soluble boron concentration assumes an atomic fraction for ^{10}B equal to 0.199. For a ^{10}B isotopic fraction equal to 0.197 (as recently measured at RGE), the total soluble boron concentration, required to maintain the same concentration of B-10 atoms, would be equal to 975 ppm.

Table 9.1-1
K_{eff} as a Function of Soluble Boron Level
(All Fresh Equivalent)

Cell Type	Description	
Region 1 Type 3	5.00 w/o Fresh in BSS cells and 2.22 w/o Fresh in SS/BSS cells	
Region 2 Type 1	1.25 w/o Fresh F/A in all cells	
Region 2 Type 2	1.96 w/o Fresh F/A in all cells	
Region 2 Type 4	1.96 w/o Fresh F/A in all cells	
Soluble Boron Concentration, ppm	**K_{eff}**	**Delta K_{eff}**
0	0.97228 ± 0.00063	0.0
200	0.92163 ± 0.00058	-0.04944
400	0.88112 ± 0.00067	-0.08986

Table 9.1-2
K_{eff} as a Function of Soluble Boron Level
(Burned Fuel Assembly Descriptions)

Cell Type	Description	
Region 1 Type 3	5.00 w/o Fresh in BSS cells and 2.22 w/o Fresh in SS/BSS cells	
Region 2 Type 1	5.0 w/o, 55 GWD/MTU, F/A in all cells, zero ^{241}Pu decay	
Region 2 Type 2	5.0 w/o, 35 GWD/MTU, F/A in all cells, zero ^{241}Pu decay	
Region 2 Type 4	1.96 w/o Fresh F/A in all cells	
Soluble Boron Concentration, ppm	**K_{eff}**	**Delta K_{eff}**
0	0.98642 ± 0.00072	0.0
200	0.95784 ± 0.00070	-0.02716
400	0.93260 ± 0.00068	-0.05242

Table 9.3-1
K_{eff} for Assumed Accident Events

(All Fresh Fuel Equivalent Descriptions, reference K_{eff} =0.97228 ± 0.00063)

Description of Accident	K_{eff}	Delta K_{eff}
Seismic event (min gap = 0.1 cm)	0.97478 ± 0.00065	0.00378
Accident F/A in Type 2, SS Cell	0.98483 ± 0.00085	0.01403
Accident F/A in Type 2 / Type 3 Interface (SW Corner in Module 3E)	0.97315 ± 0.00063	0.00213
Accident F/A in Type 3, BSS/SS Cell	0.97133 ± 0.00064	0.00032
Accident F/A in Type 3 near the Fuel Elevator Area	0.97174 ± 0.00067	0.00076

Table 9.3-2
K_{eff} for the Assumed Fuel Mishandling Event in Type 1 Cells

Cell Type	Description
Region 1 Type 3	5.00 w/o Fresh in BSS cells and 2.22 w/o Fresh in SS/BSS cells
Region 2 Type 1	3.0 w/o, 25 GWD/MTU, 15 Years ^{241}PU Decay F/A in all cells
Region 2 Type 2	5.0 w/o, 35 GWD/MTU, F/A in all cells, zero ^{241}Pu decay
Region 2 Type 4	1.96 w/o Fresh F/A in all cells

Description	K_{eff}
Reference Value of K_{eff}	0.98051 ± 0.00067
Accident F/A placed adjacent to a Type 1 "Lower" F/A (1.60 w/o fresh) with the remaining cells occupied by burnt Type 1 "Base" F/As.	1.03872 ± 0.00065
Simulation of the same accident at a soluble boron concentration of 400 ppm	0.97596 ± 0.00070

10.0 Summary of Results

The following sections contain the results in the form of plots for all the storage cells in the Ginna Spent fuel pool.

10.1 Burnup versus Enrichment Storage Curve for Region 1 Type 3 Cells

Table 10.1-1 shows that Base burnup as a function of enrichment for the Region 1 Type 3 Cells (with the 5% burnup uncertainty). The storage curves for the same are shown in Figure 10.1-1.

10.2 Burnup versus Enrichment Storage Curves for Region 2 Type 1 Cells

The following is a description of the storage curves for Region 2 Type 1 Cells:

- Table 10.2-1 shows the base, upper and lower burnups (with 5% uncertainty) for Region 2 Type 1 cells for various initial enrichments for 0, 5, 10, 15 and 20 years of ^{241}Pu decay.
- Figure 10.2-1 shows the burnup versus enrichment storage curves for no ^{241}Pu decay.
- Figure 10.2-2 shows the burnup versus enrichment storage curves for 5 years ^{241}Pu decay.
- Figure 10.2-3 shows the burnup versus enrichment storage curves for 10 years ^{241}Pu decay.
- Figure 10.2-4 shows the burnup versus enrichment storage curves for 15 years ^{241}Pu decay.
- Figure 10.2-5 shows the burnup versus enrichment storage curves for 20 years ^{241}Pu decay.

10.3 Burnup versus Enrichment Storage Curves for Region 2 Type 2 Cells

The following is a description of the storage curves for Region 2 Type 2 Cells

- Table 10.3-1 shows the base, upper and lower burnups (with 5% uncertainty) for Region 2 Type 2 cells for various initial enrichments for 0, 5, 10, 15 and 20 years of ^{241}Pu decay.
- Figure 10.3-1 shows the burnup versus enrichment storage curves for no ^{241}Pu decay.
- Figure 10.3-2 shows the burnup versus enrichment storage curves for 5 years ^{241}Pu decay.
- Figure 10.3-3 shows the burnup versus enrichment storage curves for 10 years ^{241}Pu decay.
- Figure 10.3-4 shows the burnup versus enrichment storage curves for 15 years ^{241}Pu decay.
- Figure 10.3-5 shows the burnup versus enrichment storage curves for 20 years ^{241}Pu decay.

10.4 Burnup versus Enrichment Storage Curves for Region 2 Type 4 Cells

Since the Type 2 Base storage curves are found to be conservative for Type 4 cells, the same curves (base curves only) may be used for Type 4 cells

10.5 Total Soluble Boron Requirement

The total soluble boron (sum of all the three components) required to make the K_{eff} (including all biases and uncertainties) less than or equal to 0.95 is determined to be 965 ppm. The soluble boron concentration with the adjustment for the ^{10}B atomic fraction of 0.197 is determined to be 975 ppm.

10.6 Allowable Storage Configurations

For the purpose of storage, the fuel assemblies in the Ginna Spent Fuel pool can be classified into three types – A, B and C. The "A" fuel assemblies are those that have met the base burnup criterion. These fuel assemblies can be stored directly in the spent fuel pool adjacent to any other "A" fuel assembly. The Type "B" fuel assemblies are those that have satisfied the criterion for the "Lower" curves. These fuel assemblies can be stored only by checker-boarding them with those that have met the criterion for the "Upper" curves. To enable checker-boarding, the type "A" fuel assemblies are further classified into "A1" and "A2" types. The "A1" assemblies are those that meet the "Upper" curve criterion and can be used in a checker-boarding configuration to store the type "B" assemblies. The "A2" fuel assemblies are those that cannot be used for checker-boarding with the "B" fuel assemblies. The "C" fuel assemblies are those that cannot be checker-boarded and hence can only be stored if they are surrounded on all sides by empty locations.

The allowable Storage configurations for Region 1 cells are illustrated in Figure 10.6-1; Figures 10.6-2 and 10.6-3 illustrate the allowed storage configurations for Region 2 Type 1 and Type 2 storage cells, respectively. Each of the latter three figures also address the storage restrictions on the fuel rod consolidation canisters, the damaged fuel rod baskets, and non-SNM materials. It is noted that these burned fuel storage requirements were evaluated using a conservative representation of the Westinghouse Standard fuel assembly which envelopes the burned assembly characteristics for all types of assemblies which have passed through the Ginna reactor to date.

Table 10.1-1
Burnup Vs Initial Enrichment Results for Infinite Array of Region 1 Type 3 Cells (with 5% Burnup Uncertainty)

Enrichment (w/o ^{235}U)	Base Burnup (MWD/MTU)
2.220	0
3.000	9450
4.000	18900
5.000	29400

Note that these burnup values correspond to that required for the burnt fuel assemblies that are loaded in Region 1 Type 3 Cells. The limit for the fresh fuel for this region is 5.00 w/o ^{235}U(nominal).

Table 10.2-1
Base, Upper, and Lower Burnups (MWD/MTU), with 5% Burnup Uncertainty, versus Initial Feed Enrichment and Pu-241 Decay Times for Infinite Array of Region 2, Type 1 Cells

Decay Time	w/o ^{235}U	Burnup (MWD/MTU)		
		Base	Upper	Lower
All	1.25	0		
	0.94		0	
	1.60			0
0.0	3.000	31175	38525	26450
	4.000	47066	55204	41816
	5.000	58975	67113	53200
5 yr.	3.000	28818	35117	24617
	4.000	43801	50888	39601
	5.000	56041	63128	51841
10 yr.	3.000	27547	33059	23347
	4.000	41753	48841	37553
	5.000	53822	60909	49622
15 yr.	3.000	26297	31547	22097
	4.000	40357	46657	36157
	5.000	52500	58800	48300
20 yr.	3.000	25757	30744	21347
	4.000	39275	45313	34865
	5.000	51662	57700	47252

Table 10.3-1
Base, Upper, and Lower Burnups (MWD/MTU), with 5% Burnup Uncertainty, versus Initial Feed Enrichment and Pu-241 Decay Times for Infinite Array of Region 2, Type 2 Cells.

Decay Time	w/o ^{235}U	Burnup (MWD/MTU)		
		Base	Upper	Lower
All	1.96	0		
	1.55		0	
	2.20			0
0.0	3.000	13125	19425	9713
	4.000	24675	32025	20475
	5.000	36225	44625	30975
5 yr.	3.000	12075	18375	8663
	4.000	22838	30188	18638
	5.000	34125	42525	28875
10 yr.	3.000	11813	18113	8400
	4.000	22050	29400	17850
	5.000	33075	41475	27825
15 yr.	3.000	11498	16748	8085
	4.000	21683	27983	17483
	5.000	32025	39375	26775
20 yr.	3.000	11340	16590	7928
	4.000	21263	27563	17063
	5.000	31500	38850	26250

Figure 10.1-1 Region 1 Type 3 Cells

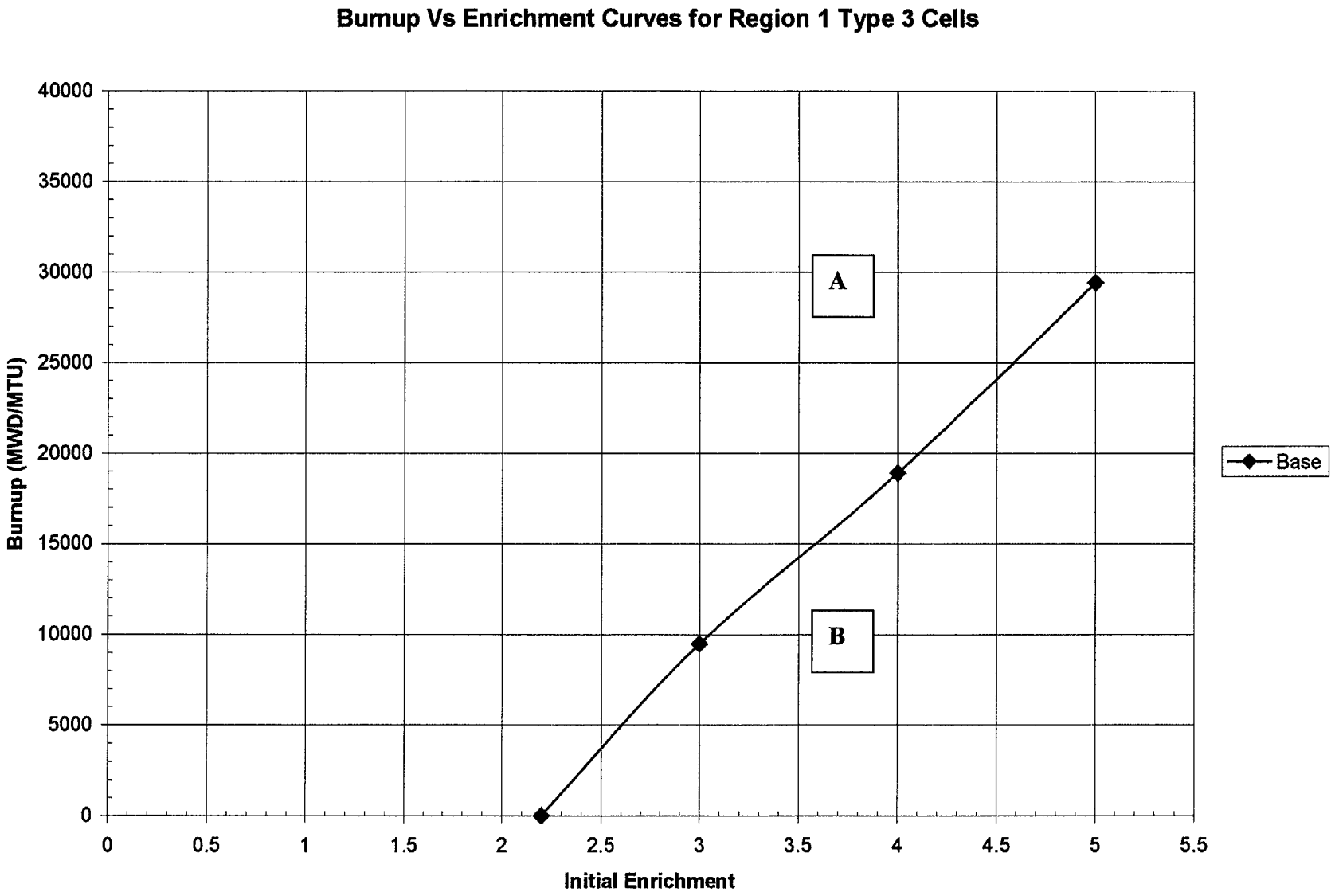


Figure 10.2-1 Region 2 Type 1 Cells (No Pu-241 Decay)

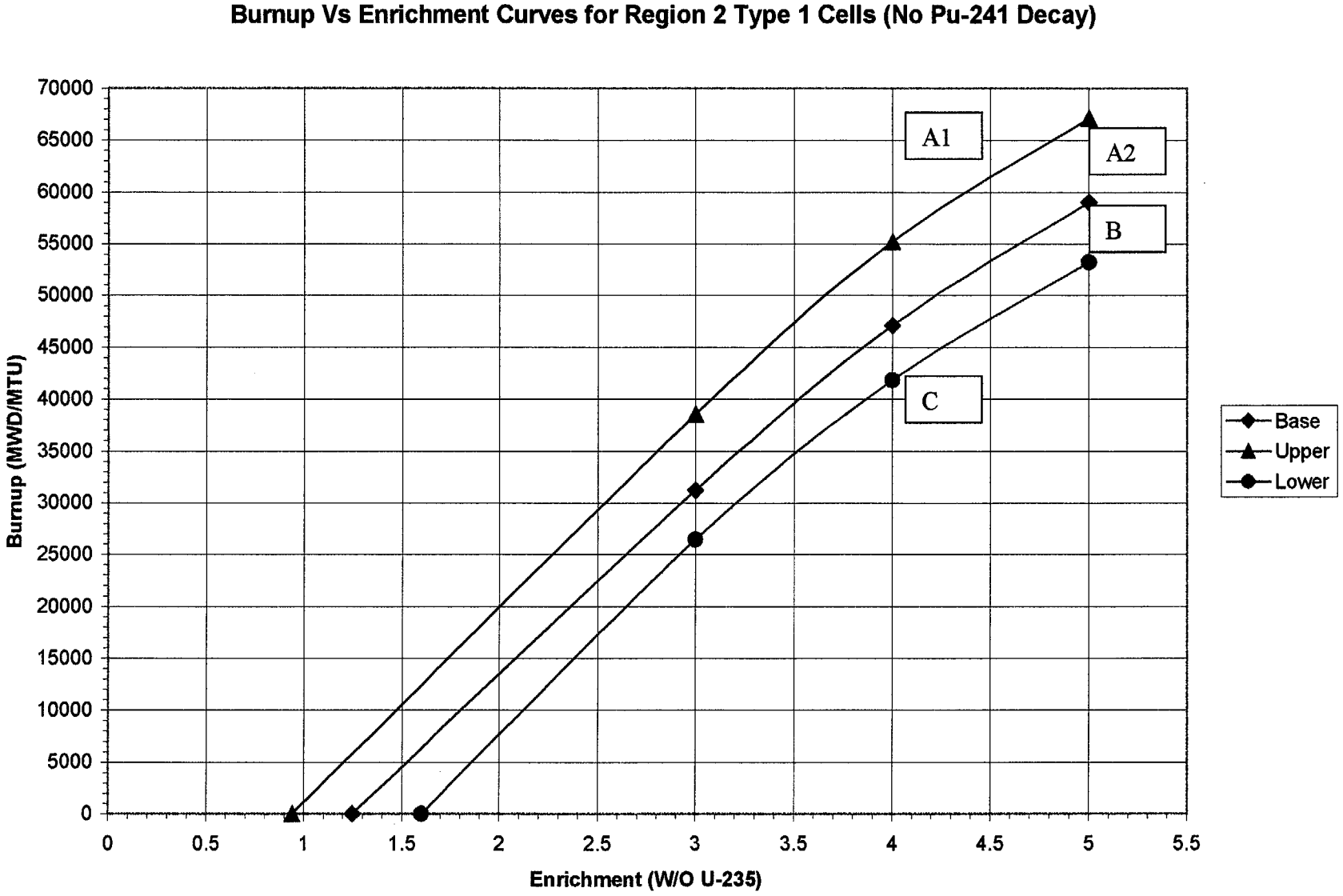


Figure 10.2-2 Region 2 Type 1 Cells (5 Years Pu-241 Decay)

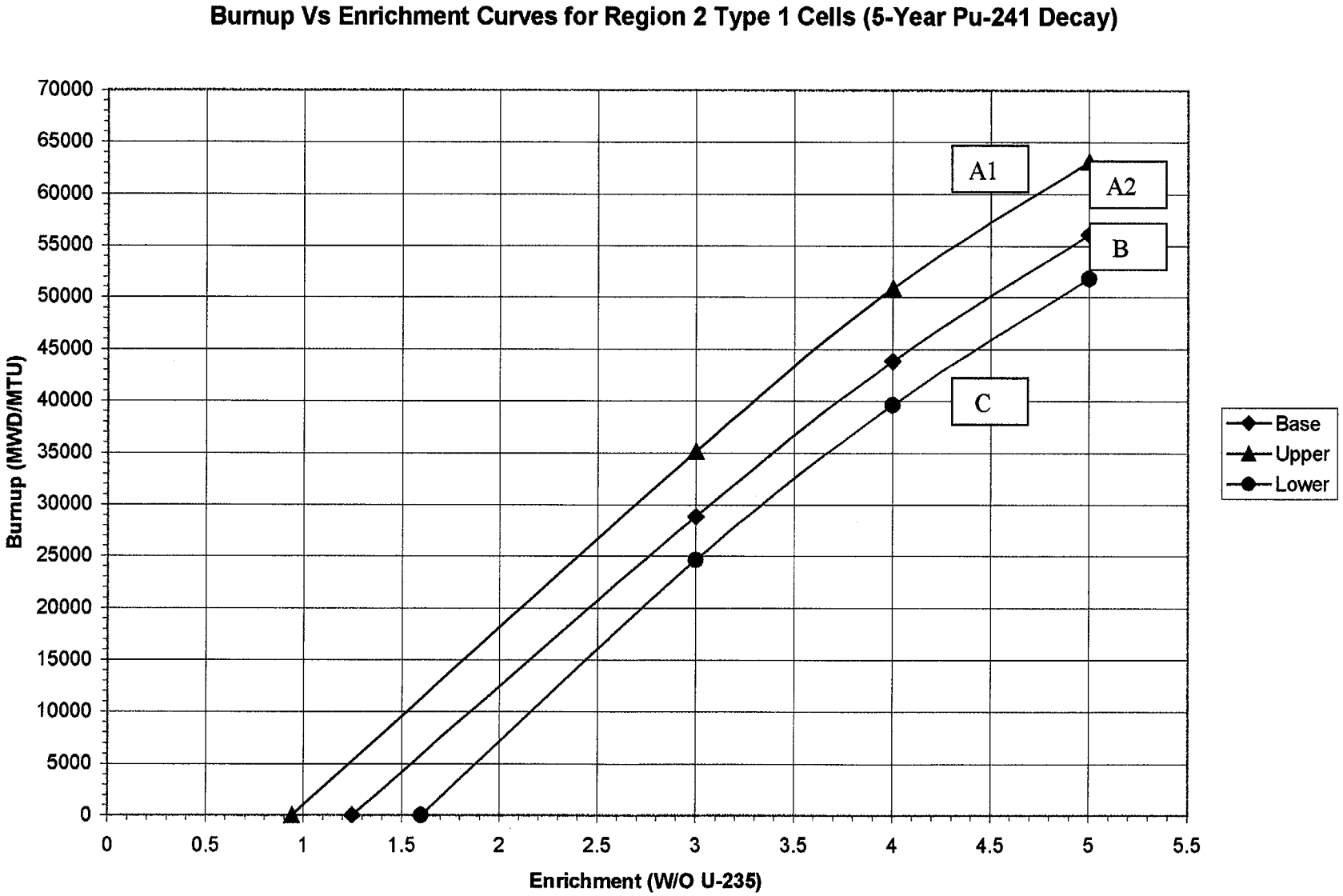


Figure 10.2-3 Region 2 Type 1 Cells (10 Years Pu-241 Decay)

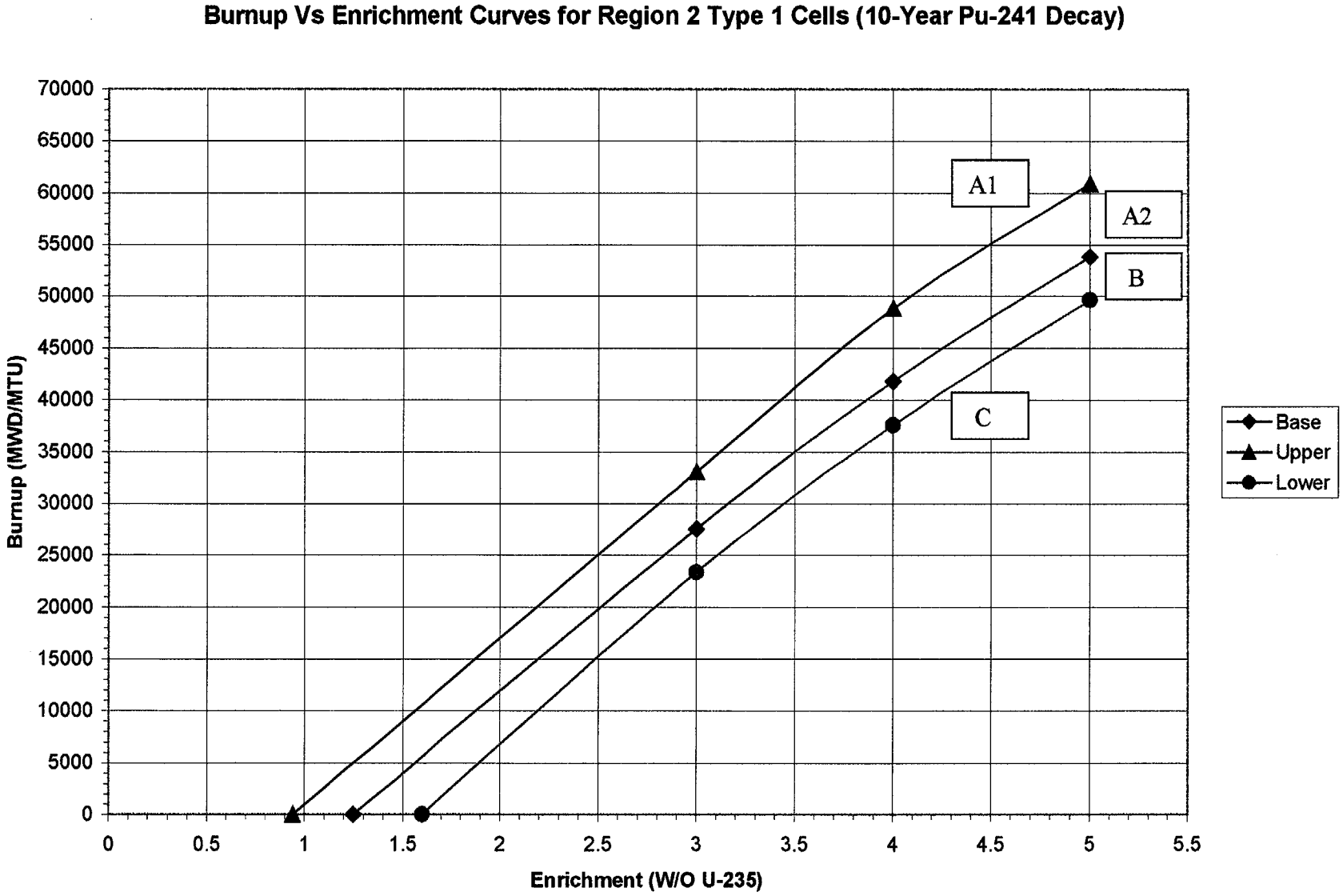


Figure 10.2-4 Region 2 Type 1 Cells (15 Years Pu-241 Decay)

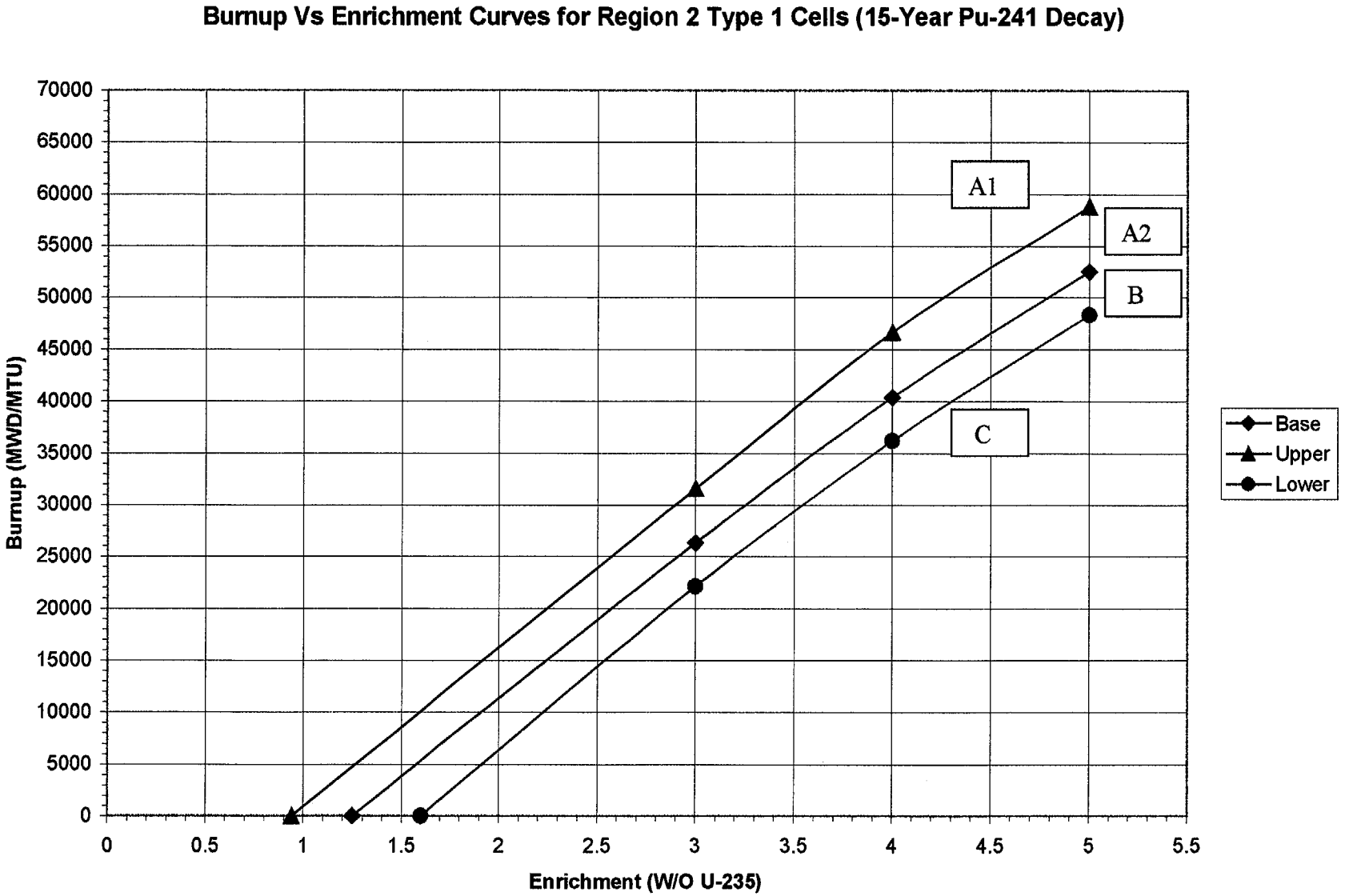


Figure 10.2-5 Region 2 Type 1 Cells (20 Years Pu-241 Decay)

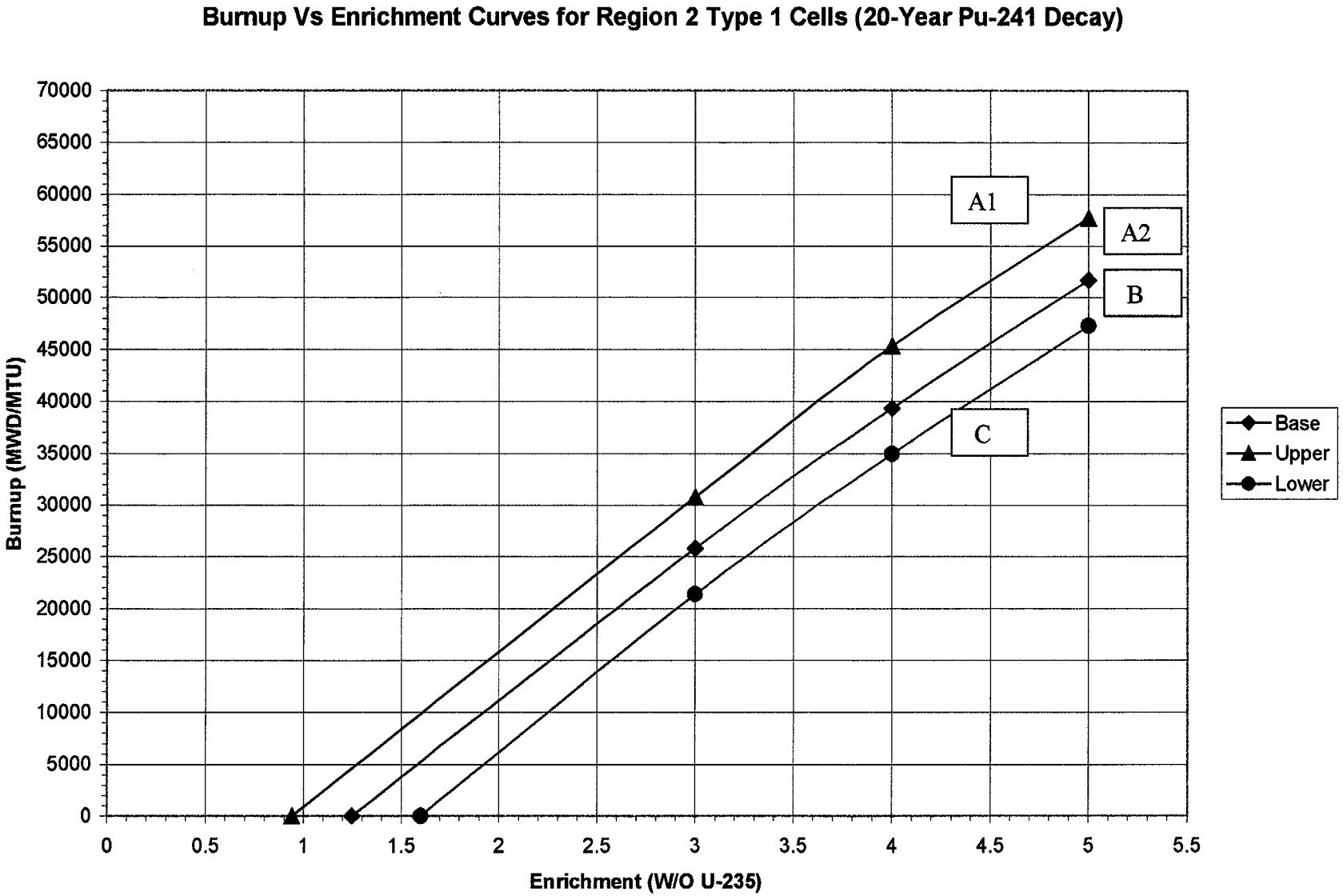


Figure 10.3-1 Region 2 Type 2 Cells (No Pu-241 Decay)

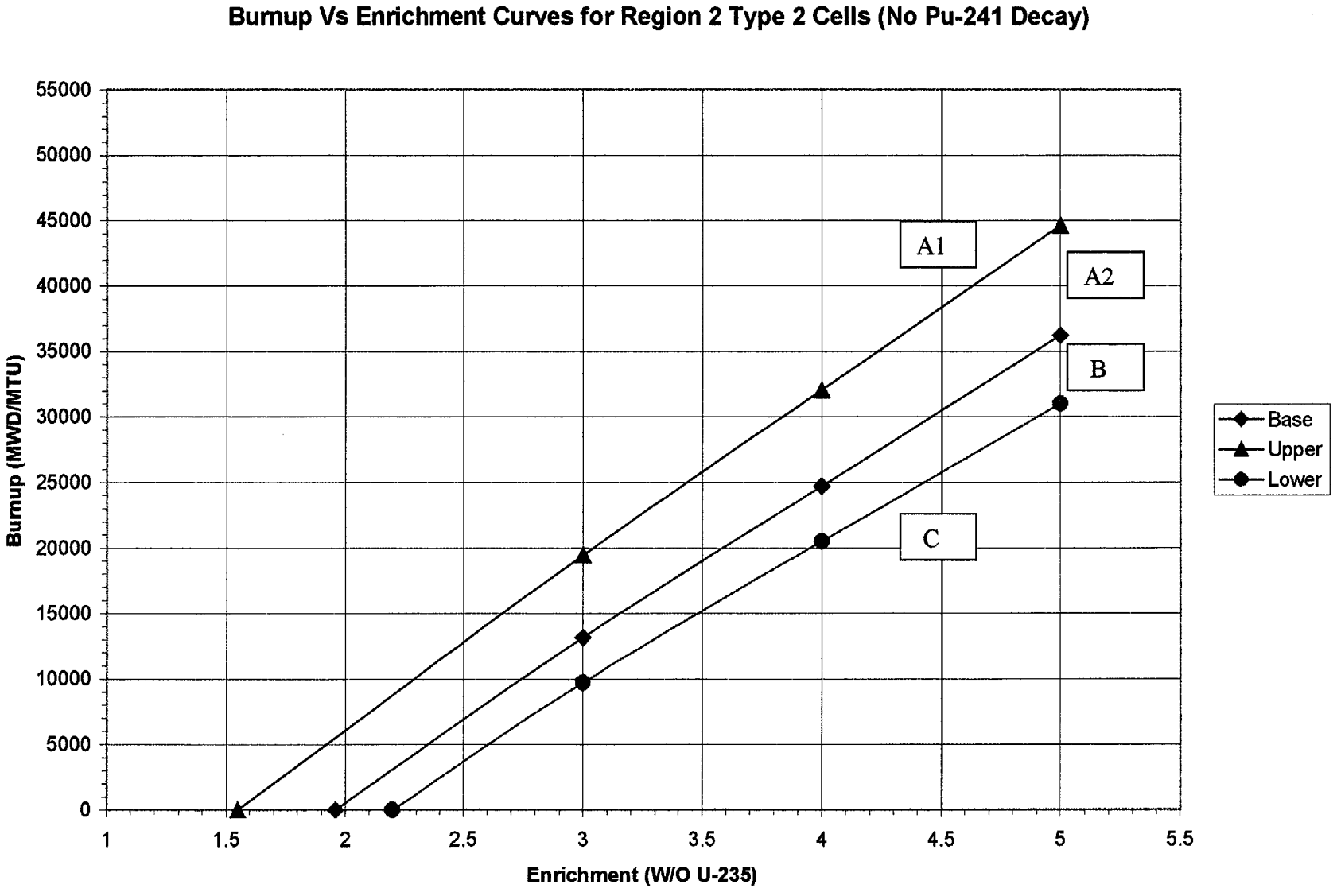


Figure 10.3-2 Region 2 Type 2 Cells (5 Years Pu-241 Decay)

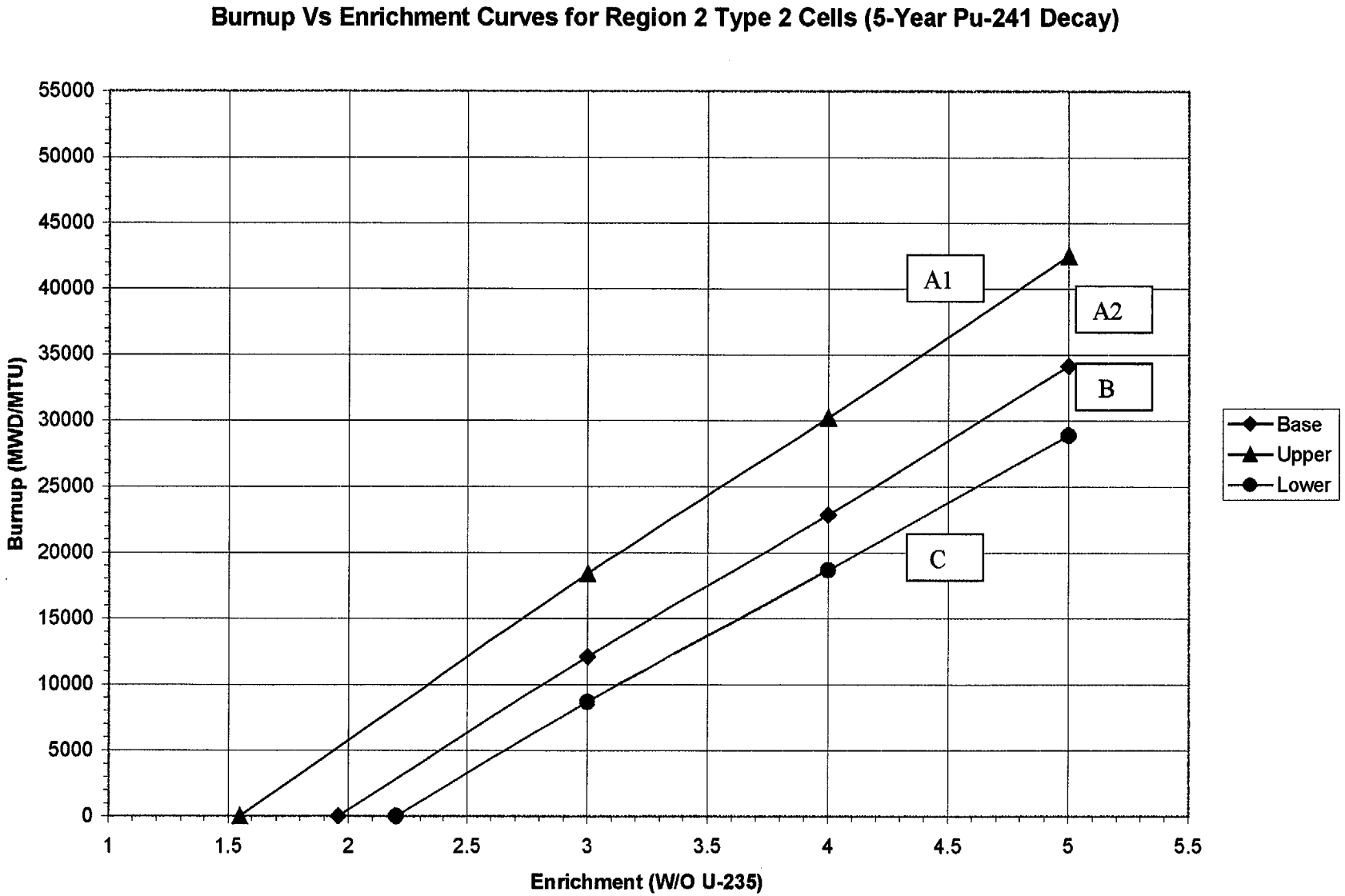


Figure 10.3-3 Region 2 Type 2 Cells (10 Years Pu-241 Decay)

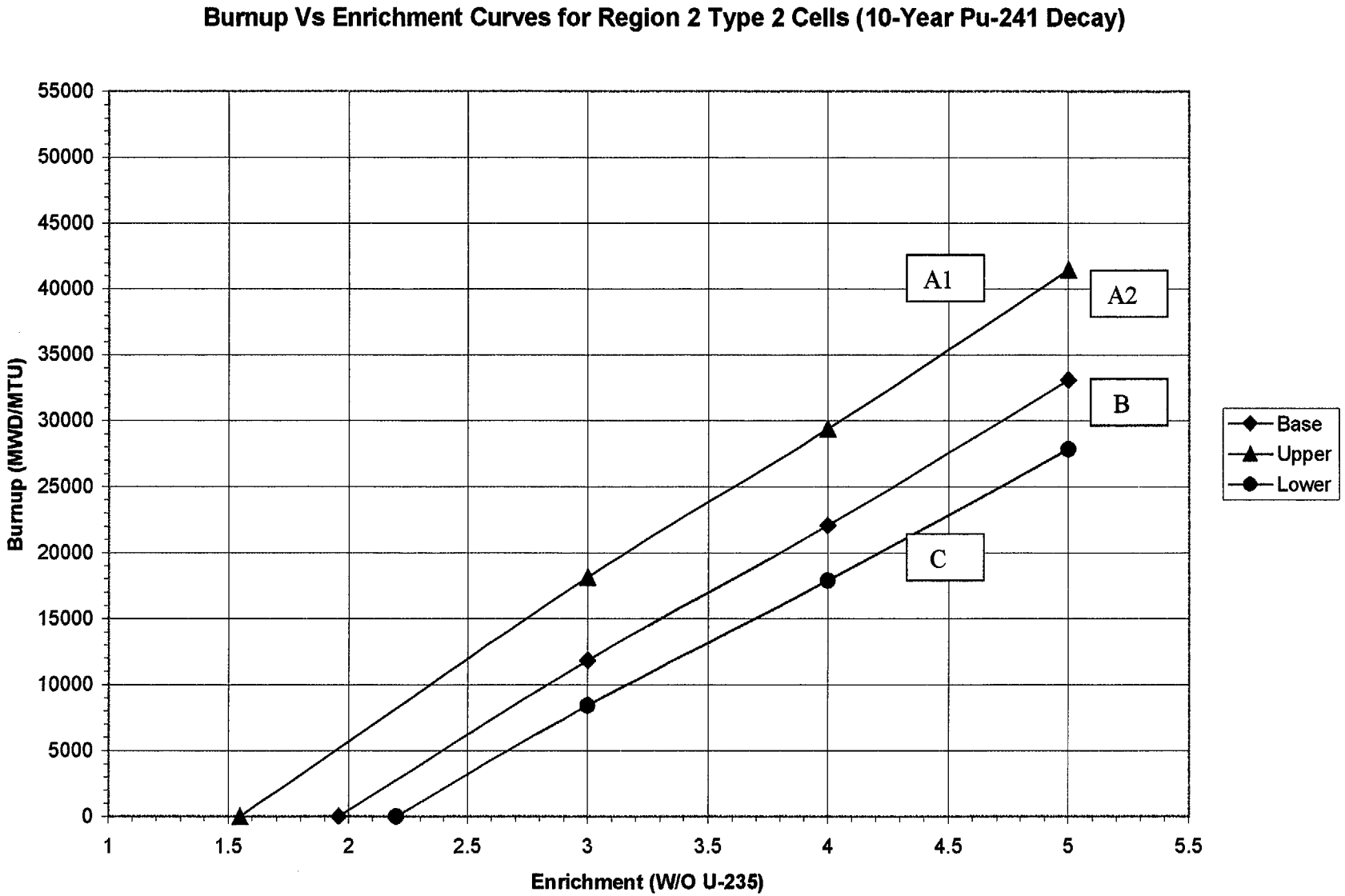


Figure 10.3-4 Region 2 Type 2 Cells (15 Years Pu-241 Decay)

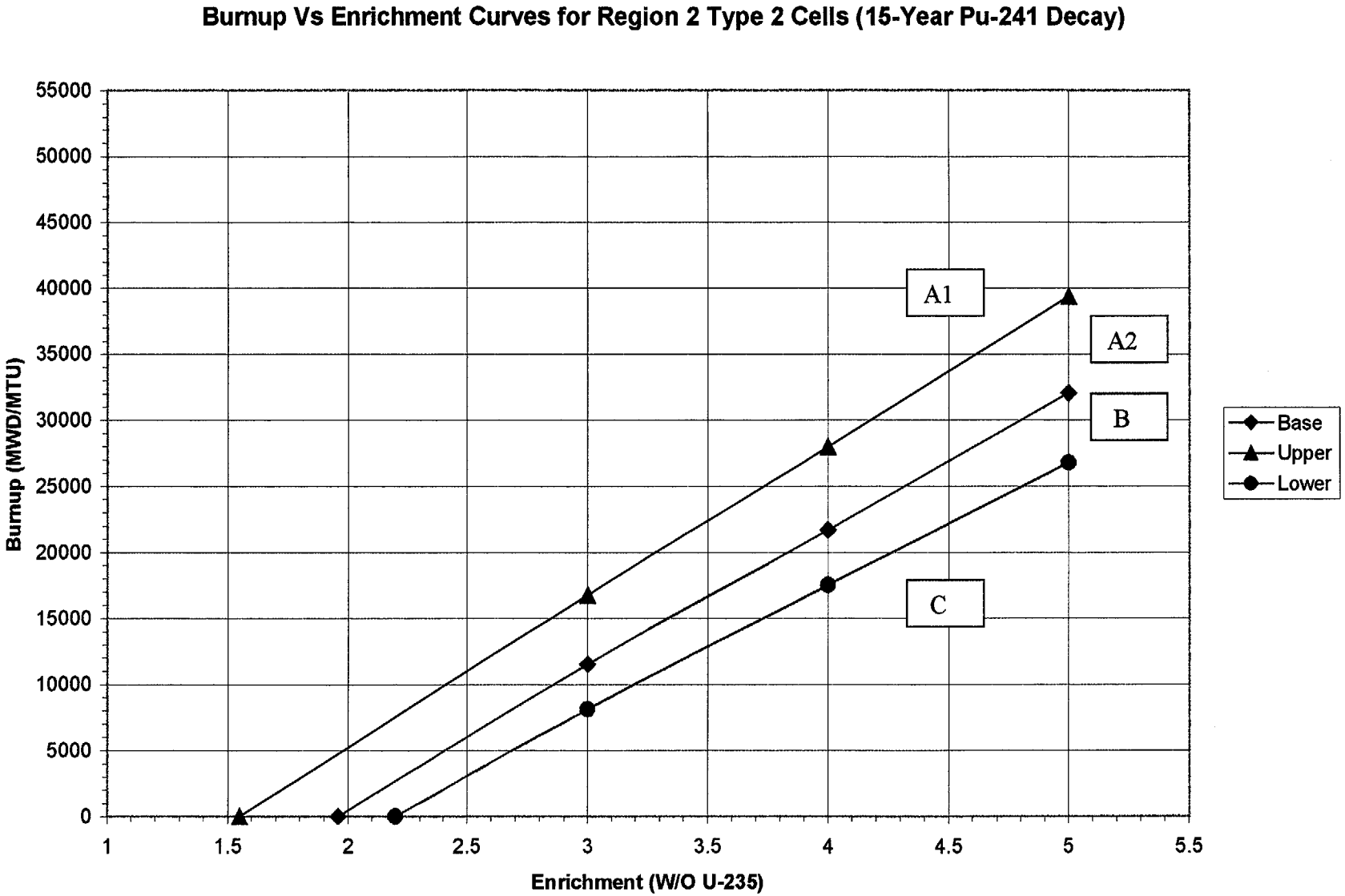


Figure 10.3-5 Region 2 Type 2 Cells (20 Years Pu-241 Decay)

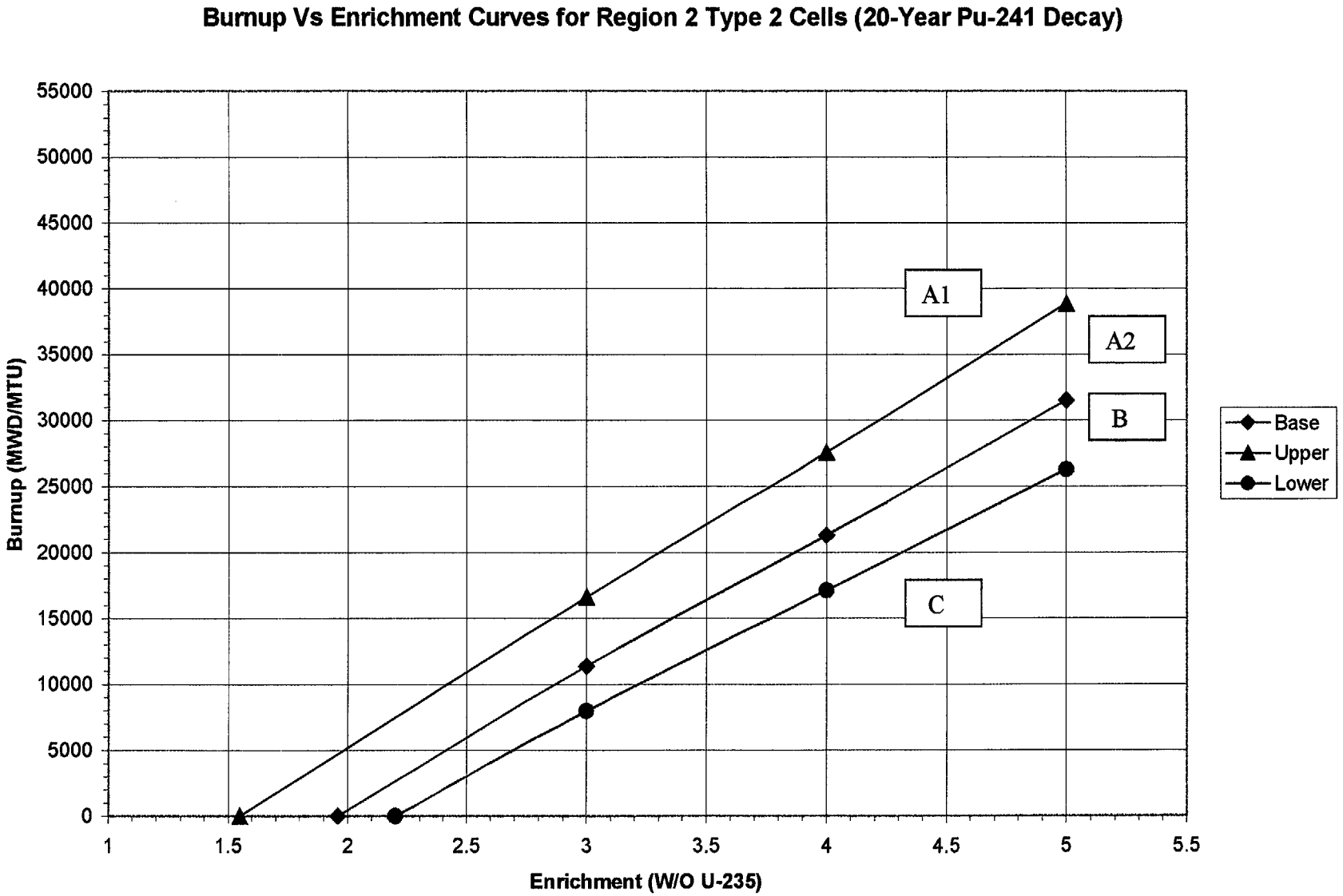
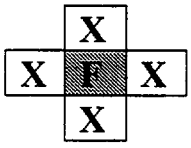
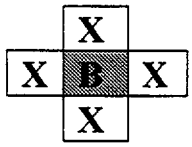


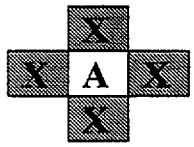
Figure 10.6-1
Sketch of Allowable Loading Configurations for Region 1



X = A or Empty or non-SNM



X = A or Empty or non-SNM



X = A, B, F, or Empty or non-SNM

Cell With Integral Lead-in Funnel

Cell Without Integral Lead-in Funnel

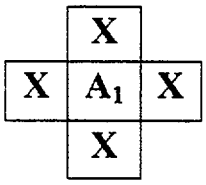
F = Fresh Fuel Assembly

A = Either a Fuel Assembly or a Fuel Rod Consolidation Canister containing Rods with Burnup and Enrichment in Area A of Figure 10.1-1.

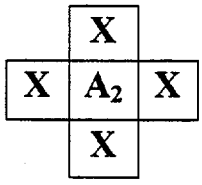
B = Either a Fuel Assembly or a Fuel Rod Consolidation Canister containing Rods with Burnup and Enrichment in Area B of Figure 10.1-1.

NOTE: Full or Partially Loaded Damaged Rod Storage Basket May Be Stored in Any Location.

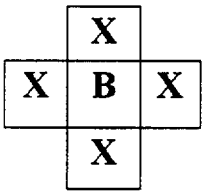
Figure 10.6-2
Sketch of Allowable Loading Configurations for Region 2 Type 1 Cells



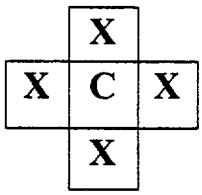
X = A_1, A_2 B_2, Empty, or non-SNM



X = A_1, A_2, Empty, or non-SNM



X = A_1, Empty, or non-SNM



X = Empty or non-SNM

A_1 = Either a Fuel Assembly or a Fuel Rod Consolidation Canister Containing Rods with Burnup and Enrichment in Area A_1 of Figures 10.2-1 through 10.2-5.

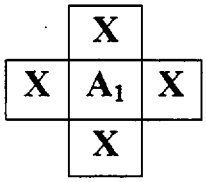
A_2 = Either a Fuel Assembly or a Fuel Rod Consolidation Canister Containing Rods with Burnup and Enrichment in Area A_2 of Figures 10.2-1 through 10.2-5.

B = Either a Fuel Assembly or a Fuel Rod Consolidation Canister Containing Rods with Burnup and Enrichment in Area B of Figures 10.2-1 through 10.2-5.

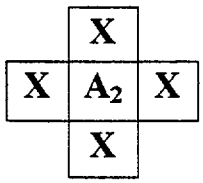
C = Either a Fuel Assembly or a Fuel Rod Consolidation Canister Containing Rods with Burnup and Enrichment in Area C of Figures 10.2-1 through 10.2-5.

NOTE: Full or Partially Loaded Damaged Rod Storage Basket May Be Stored in Any Location.

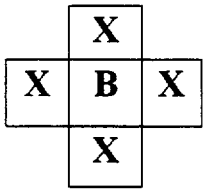
Figure 10.6-3
Sketch of Allowable Loading Configurations for Region 2 Type 2 Cells



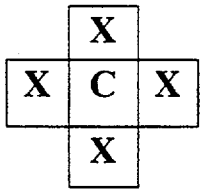
X = A_1, A_2 B_2, Empty, or non-SNM



X = A_1, A_2, Empty, or non-SNM



X = A_1, Empty, or non-SNM



X = Empty or non-SNM

A_1 = Either a Fuel Assembly or a Fuel Rod Consolidation Canister Containing Rods with Burnup and Enrichment in Area A_1 of Figures 10.3-1 through 10.3-5.

A_2 = Either a Fuel Assembly or a Fuel Rod Consolidation Canister Containing Rods with Burnup and Enrichment in Area A_2 of Figures 10.3-1 through 10.3-5.

B = Either a Fuel Assembly or a Fuel Rod Consolidation Canister Containing Rods with Burnup and Enrichment in Area B of Figures 10.3-1 through 10.3-5.

C = Either a Fuel Assembly or a Fuel Rod Consolidation Canister Containing Rods with Burnup and Enrichment in Area C of Figures 10.3-1 through 10.3-5.

NOTE: Full or Partially Loaded Damaged Rod Storage Basket May Be Stored in Any Location.

11.0 References

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APPENDIX A. Isotopic Number Densities employed in KENO calculations

Table 1A. W-Std Design Basis Fuel Assembly

Enrichment = 3.0 W/O U-235, Assembly Average Case

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	8.9177E-08	5.7338E-04	2.2190E-05	2.2178E-02	5.0711E-05	4.8218E-06	1.1995E-06	1.32307
15.00	1.0507E-07	3.8581E-04	5.4671E-05	2.2016E-02	9.9108E-05	2.2621E-05	1.1342E-05	1.20920
25.00	1.0995E-07	2.5128E-04	7.5960E-05	2.1838E-02	1.1683E-04	3.9658E-05	2.2848E-05	1.11115
35.00	1.1006E-07	1.5635E-04	8.8709E-05	2.1646E-02	1.2191E-04	5.3058E-05	3.1353E-05	1.02372
45.00	1.0854E-07	9.2569E-05	9.4790E-05	2.1439E-02	1.2202E-04	6.2425E-05	3.6566E-05	0.94973
55.00	1.0707E-07	5.2366E-05	9.5945E-05	2.1219E-02	1.2072E-04	6.8309E-05	3.9382E-05	0.89265
65.00	1.0637E-07	2.8609E-05	9.3819E-05	2.0990E-02	1.1949E-04	7.1656E-05	4.0793E-05	0.85227

Table 2A. W-Std Design Basis Fuel Assembly

Enrichment = 3.0 W/O U-235, Zone 1 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	7.0824E-08	6.3340E-04	1.1376E-05	2.2219E-02	2.7397E-05	1.3807E-06	1.7203E-07	1.35692
15.00	7.7630E-08	5.2328E-04	3.0802E-05	2.2145E-02	6.4545E-05	8.3880E-06	2.6787E-06	1.29585
25.00	8.0672E-08	4.3017E-04	4.6801E-05	2.2068E-02	8.7148E-05	1.7108E-05	7.4076E-06	1.23935
35.00	8.1488E-08	3.5061E-04	5.9969E-05	2.1988E-02	1.0092E-04	2.5991E-05	1.2880E-05	1.18652
45.00	8.0959E-08	2.8265E-04	7.0674E-05	2.1904E-02	1.0901E-04	3.4404E-05	1.8189E-05	1.13603
55.00	7.9645E-08	2.2496E-04	7.9186E-05	2.1816E-02	1.1337E-04	4.2028E-05	2.2885E-05	1.08758
65.00	7.7958E-08	1.7653E-04	8.5719E-05	2.1724E-02	1.1531E-04	4.8687E-05	2.6796E-05	1.04140

Table 3A. W-Std Design Basis Fuel Assembly

Enrichment = 3.0 W/O U-235, Zone 2 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	8.1359E-08	5.9438E-04	1.8317E-05	2.2194E-02	4.2263E-05	3.3363E-06	6.7356E-07	1.33548
15.00	9.3087E-08	4.3055E-04	4.6774E-05	2.2067E-02	8.7946E-05	1.7109E-05	7.5945E-06	1.23847
25.00	9.7039E-08	3.0520E-04	6.7247E-05	2.1930E-02	1.0803E-04	3.1611E-05	1.6956E-05	1.15277
35.00	9.7081E-08	2.0960E-04	8.1384E-05	2.1783E-02	1.1587E-04	4.4231E-05	2.5058E-05	1.07377
45.00	9.5461E-08	1.3868E-04	9.0257E-05	2.1626E-02	1.1775E-04	5.4179E-05	3.0919E-05	1.00176
55.00	9.3470E-08	8.8257E-05	9.4804E-05	2.1458E-02	1.1708E-04	6.1386E-05	3.4680E-05	0.93963
65.00	9.1827E-08	5.4196E-05	9.5981E-05	2.1281E-02	1.1565E-04	6.6200E-05	3.6871E-05	0.88979

Table 4A. W-Std Design Basis Fuel Assembly
Enrichment = 3.0 W/O U-235, Zone 3 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	8.7878E-08	5.7267E-04	2.2158E-05	2.2179E-02	5.0036E-05	4.7126E-06	1.1483E-06	1.32291
15.00	1.0257E-07	3.8317E-04	5.4721E-05	2.2019E-02	9.7700E-05	2.2324E-05	1.1023E-05	1.20733
25.00	1.0674E-07	2.4701E-04	7.6095E-05	2.1844E-02	1.1468E-04	3.9256E-05	2.2269E-05	1.10661
35.00	1.0639E-07	1.5126E-04	8.8844E-05	2.1653E-02	1.1907E-04	5.2533E-05	3.0494E-05	1.01616
45.00	1.0460E-07	8.7547E-05	9.4807E-05	2.1447E-02	1.1864E-04	6.1691E-05	3.5416E-05	0.93964
55.00	1.0297E-07	4.8127E-05	9.5767E-05	2.1226E-02	1.1700E-04	6.7306E-05	3.7971E-05	0.88120
65.00	1.0218E-07	2.5443E-05	9.3437E-05	2.0996E-02	1.1560E-04	7.0377E-05	3.9183E-05	0.84062

Table 5A. W-Std Design Basis Fuel Assembly

Enrichment = 3.0 W/O U-235, Zone 4 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	9.2476E-08	5.6320E-04	2.4000E-05	2.2171E-02	5.4249E-05	5.5411E-06	1.4867E-06	1.31709
15.00	1.0982E-07	3.6460E-04	5.8186E-05	2.1991E-02	1.0304E-04	2.5118E-05	1.3083E-05	1.19490
25.00	1.1478E-07	2.2657E-04	7.9533E-05	2.1795E-02	1.1915E-04	4.3064E-05	2.5243E-05	1.09028
35.00	1.1472E-07	1.3322E-04	9.1258E-05	2.1580E-02	1.2279E-04	5.6467E-05	3.3539E-05	0.99875
45.00	1.1321E-07	7.3933E-05	9.5725E-05	2.1349E-02	1.2215E-04	6.5223E-05	3.8185E-05	0.92486
55.00	1.1206E-07	3.9071E-05	9.5168E-05	2.1103E-02	1.2072E-04	7.0289E-05	4.0476E-05	0.87128
65.00	1.1187E-07	1.9970E-05	9.1572E-05	2.0849E-02	1.1967E-04	7.2915E-05	4.1550E-05	0.83591

Table 6A. W-Std Design Basis Fuel Assembly

Enrichment = 3.0 W/O U-235, Zone 5 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	8.8747E-08	5.8098E-04	2.1025E-05	2.2181E-02	4.9253E-05	4.4788E-06	1.0835E-06	1.32735
15.00	1.0521E-07	4.0338E-04	5.2207E-05	2.2026E-02	9.8407E-05	2.1277E-05	1.0582E-05	1.22134
25.00	1.1104E-07	2.7378E-04	7.3217E-05	2.1859E-02	1.1829E-04	3.7730E-05	2.1900E-05	1.13111
35.00	1.1198E-07	1.7969E-04	8.6485E-05	2.1678E-02	1.2546E-04	5.1163E-05	3.0819E-05	1.05072
45.00	1.1099E-07	1.1362E-04	9.3646E-05	2.1486E-02	1.2708E-04	6.1081E-05	3.6777E-05	0.98106
55.00	1.0970E-07	6.9339E-05	9.6139E-05	2.1283E-02	1.2661E-04	6.7822E-05	4.0359E-05	0.92489
65.00	1.0893E-07	4.1098E-05	9.5318E-05	2.1070E-02	1.2570E-04	7.2067E-05	4.2374E-05	0.88261

Table 7A. W-Std Design Basis Fuel Assembly

Enrichment = 3.0 W/O U-235, Zone 6 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	8.5337E-08	5.9177E-04	1.9087E-05	2.2189E-02	4.5228E-05	3.7663E-06	8.2880E-07	1.33358
15.00	1.0010E-07	4.2688E-04	4.8206E-05	2.2051E-02	9.3200E-05	1.8618E-05	8.7767E-06	1.23635
25.00	1.0575E-07	3.0274E-04	6.8729E-05	2.1903E-02	1.1446E-04	3.3820E-05	1.9063E-05	1.15297
35.00	1.0687E-07	2.0920E-04	8.2610E-05	2.1745E-02	1.2329E-04	4.6836E-05	2.7815E-05	1.07780
45.00	1.0602E-07	1.4025E-04	9.1108E-05	2.1576E-02	1.2611E-04	5.7024E-05	3.4157E-05	1.01065
55.00	1.0461E-07	9.1202E-05	9.5306E-05	2.1398E-02	1.2626E-04	6.4446E-05	3.8317E-05	0.95357
65.00	1.0345E-07	5.7705E-05	9.6215E-05	2.1212E-02	1.2549E-04	6.9506E-05	4.0858E-05	0.90786

Table 8A. W-Std Design Basis Fuel Assembly

Enrichment = 3.0 W/O U-235, Zone 7 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	7.7725E-08	6.1718E-04	1.4504E-05	2.2206E-02	3.5334E-05	2.2850E-06	3.8229E-07	1.34791
15.00	8.8597E-08	4.8502E-04	3.8103E-05	2.2106E-02	7.8632E-05	1.2539E-05	4.9728E-06	1.27242
25.00	9.3535E-08	3.7860E-04	5.6387E-05	2.2001E-02	1.0208E-04	2.4194E-05	1.2261E-05	1.20602
35.00	9.5082E-08	2.9190E-04	7.0427E-05	2.1890E-02	1.1478E-04	3.5258E-05	1.9663E-05	1.14516
45.00	9.4788E-08	2.2161E-04	8.0891E-05	2.1773E-02	1.2122E-04	4.5023E-05	2.6081E-05	1.08845
55.00	9.3612E-08	1.6536E-04	8.8283E-05	2.1650E-02	1.2402E-04	5.3219E-05	3.1186E-05	1.03595
65.00	9.2156E-08	1.2120E-04	9.3041E-05	2.1522E-02	1.2479E-04	5.9811E-05	3.5005E-05	0.98847

Table 9A. W-Std Design Basis Fuel Assembly

Enrichment = 3.0 W/O U-235, Zone 8 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	6.6915E-08	6.5766E-04	7.1606E-06	2.2232E-02	1.8268E-05	6.0609E-07	4.9585E-08	1.36897
15.00	7.2019E-08	5.8637E-04	2.0044E-05	2.2186E-02	4.6672E-05	4.1259E-06	9.3332E-07	1.33217
25.00	7.4919E-08	5.2292E-04	3.1388E-05	2.2139E-02	6.7370E-05	9.0249E-06	3.0004E-06	1.29609
35.00	7.6469E-08	4.6572E-04	4.1450E-05	2.2090E-02	8.2754E-05	1.4455E-05	5.8761E-06	1.26224
45.00	7.7107E-08	4.1389E-04	5.0380E-05	2.2041E-02	9.4259E-05	2.0045E-05	9.1680E-06	1.23010
55.00	7.7112E-08	3.6683E-04	5.8288E-05	2.1990E-02	1.0285E-04	2.5592E-05	1.2593E-05	1.19920
65.00	7.6673E-08	3.2409E-04	6.5267E-05	2.1937E-02	1.0921E-04	3.0972E-05	1.5965E-05	1.16923

Table 10A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Assembly Average Case

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.1113E-07	8.0040E-04	2.4009E-05	2.1957E-02	4.7344E-05	3.5882E-06	8.1008E-07	1.39248
15.00	1.2669E-07	5.8901E-04	6.2205E-05	2.1813E-02	9.9929E-05	1.8301E-05	9.0777E-06	1.28966
25.00	1.3068E-07	4.2263E-04	9.0343E-05	2.1658E-02	1.2351E-04	3.3911E-05	2.0243E-05	1.19981
35.00	1.2907E-07	2.9252E-04	1.1012E-04	2.1489E-02	1.3243E-04	4.7637E-05	2.9863E-05	1.11517
45.00	1.2504E-07	1.9377E-04	1.2265E-04	2.1307E-02	1.3383E-04	5.8502E-05	3.6669E-05	1.03517
55.00	1.2058E-07	1.2226E-04	1.2899E-04	2.1111E-02	1.3179E-04	6.6303E-05	4.0767E-05	0.96358
65.00	1.1684E-07	7.3513E-05	1.3033E-04	2.0903E-02	1.2874E-04	7.1322E-05	4.2841E-05	0.90412

Table 11A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Zone 1 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	9.2423E-08	8.6400E-04	1.2104E-05	2.1993E-02	2.4781E-05	9.8788E-07	1.0898E-07	1.42247
15.00	9.8727E-08	7.4596E-04	3.3694E-05	2.1928E-02	6.1389E-05	6.3818E-06	1.8967E-06	1.36857
25.00	1.0129E-07	6.4135E-04	5.2427E-05	2.1860E-02	8.6089E-05	1.3524E-05	5.6848E-06	1.31819
35.00	1.0155E-07	5.4792E-04	6.8685E-05	2.1790E-02	1.0277E-04	2.1178E-05	1.0508E-05	1.27101
45.00	1.0029E-07	4.6434E-04	8.2715E-05	2.1717E-02	1.1377E-04	2.8813E-05	1.5594E-05	1.22561
55.00	9.8069E-08	3.8983E-04	9.4687E-05	2.1640E-02	1.2066E-04	3.6120E-05	2.0464E-05	1.18114
65.00	9.5310E-08	3.2374E-04	1.0472E-04	2.1561E-02	1.2454E-04	4.2899E-05	2.4848E-05	1.13725

Table 12A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Zone 2 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.0288E-07	8.2293E-04	1.9681E-05	2.1971E-02	3.8968E-05	2.4478E-06	4.4448E-07	1.40367
15.00	1.1414E-07	6.4179E-04	5.2413E-05	2.1859E-02	8.6902E-05	1.3545E-05	5.8338E-06	1.31703
25.00	1.1736E-07	4.9236E-04	7.8180E-05	2.1739E-02	1.1188E-04	2.6275E-05	1.4344E-05	1.23993
35.00	1.1620E-07	3.6892E-04	9.8049E-05	2.1611E-02	1.2391E-04	3.8370E-05	2.2764E-05	1.16695
45.00	1.1282E-07	2.6840E-04	1.1265E-04	2.1474E-02	1.2834E-04	4.8904E-05	2.9693E-05	1.09610
55.00	1.0861E-07	1.8870E-04	1.2252E-04	2.1328E-02	1.2845E-04	5.7446E-05	3.4727E-05	1.02820
65.00	1.0445E-07	1.2778E-04	1.2817E-04	2.1171E-02	1.2640E-04	6.3885E-05	3.7971E-05	0.96592

Table 13A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Zone 3 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.0938E-07	7.9979E-04	2.3940E-05	2.1958E-02	4.6616E-05	3.5020E-06	7.7423E-07	1.39258
15.00	1.2362E-07	5.8655E-04	6.2146E-05	2.1817E-02	9.8351E-05	1.8019E-05	8.7999E-06	1.28884
25.00	1.2680E-07	4.1826E-04	9.0357E-05	2.1664E-02	1.2119E-04	3.3495E-05	1.9705E-05	1.19719
35.00	1.2465E-07	2.8666E-04	1.1019E-04	2.1497E-02	1.2938E-04	4.7100E-05	2.9057E-05	1.11005
45.00	1.2028E-07	1.8707E-04	1.2270E-04	2.1317E-02	1.3011E-04	5.7810E-05	3.5566E-05	1.02714
55.00	1.1561E-07	1.1559E-04	1.2892E-04	2.1122E-02	1.2757E-04	6.5387E-05	3.9359E-05	0.95286
65.00	1.1179E-07	6.7637E-05	1.3002E-04	2.0913E-02	1.2421E-04	7.0128E-05	4.1160E-05	0.89165

Table 14A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Zone 4 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.1442E-07	7.8943E-04	2.6035E-05	2.1950E-02	5.0884E-05	4.1477E-06	1.0143E-06	1.38710
15.00	1.3144E-07	5.6376E-04	6.6619E-05	2.1792E-02	1.0473E-04	2.0498E-05	1.0649E-05	1.27649
25.00	1.3537E-07	3.9004E-04	9.5541E-05	2.1619E-02	1.2695E-04	3.7264E-05	2.2798E-05	1.17991
35.00	1.3326E-07	2.5806E-04	1.1482E-04	2.1431E-02	1.3401E-04	5.1436E-05	3.2568E-05	1.08902
45.00	1.2884E-07	1.6178E-04	1.2589E-04	2.1227E-02	1.3390E-04	6.2070E-05	3.8910E-05	1.00490
55.00	1.2437E-07	9.5746E-05	1.3016E-04	2.1006E-02	1.3102E-04	6.9163E-05	4.2307E-05	0.93277
65.00	1.2103E-07	5.3703E-05	1.2925E-04	2.0772E-02	1.2779E-04	7.3284E-05	4.3759E-05	0.87654

Table 15A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Zone 5 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.1132E-07	8.0841E-04	2.2753E-05	2.1959E-02	4.5944E-05	3.3260E-06	7.2806E-07	1.39597
15.00	1.2758E-07	6.0893E-04	5.9292E-05	2.1822E-02	9.8823E-05	1.7161E-05	8.3956E-06	1.29957
25.00	1.3272E-07	4.5020E-04	8.6696E-05	2.1674E-02	1.2435E-04	3.2092E-05	1.9169E-05	1.21638
35.00	1.3223E-07	3.2403E-04	1.0655E-04	2.1516E-02	1.3560E-04	4.5560E-05	2.8930E-05	1.13900
45.00	1.2910E-07	2.2568E-04	1.1987E-04	2.1346E-02	1.3909E-04	5.6643E-05	3.6340E-05	1.06593
55.00	1.2514E-07	1.5156E-04	1.2757E-04	2.1164E-02	1.3861E-04	6.5085E-05	4.1289E-05	0.99923
65.00	1.2150E-07	9.8078E-05	1.3060E-04	2.0972E-02	1.3644E-04	7.1011E-05	4.4216E-05	0.94164

Table 16A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Zone 6 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.0791E-07	8.1991E-04	2.0599E-05	2.1966E-02	4.1969E-05	2.7782E-06	5.5073E-07	1.40153
15.00	1.2243E-07	6.3630E-04	5.4381E-05	2.1844E-02	9.2754E-05	1.4870E-05	6.8309E-06	1.31335
25.00	1.2748E-07	4.8696E-04	8.0542E-05	2.1713E-02	1.1917E-04	2.8397E-05	1.6324E-05	1.23689
35.00	1.2745E-07	3.6504E-04	1.0036E-04	2.1574E-02	1.3221E-04	4.1051E-05	2.5515E-05	1.16583
45.00	1.2485E-07	2.6677E-04	1.1463E-04	2.1426E-02	1.3751E-04	5.1949E-05	3.3022E-05	1.09817
55.00	1.2121E-07	1.8942E-04	1.2401E-04	2.1268E-02	1.3844E-04	6.0741E-05	3.8502E-05	1.03461
65.00	1.1752E-07	1.3044E-04	1.2917E-04	2.1101E-02	1.3711E-04	6.7394E-05	4.2128E-05	0.97736

Table 17A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Zone 7 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.0035E-07	8.4687E-04	1.5550E-05	2.1982E-02	3.2377E-05	1.6575E-06	2.4730E-07	1.41418
15.00	1.1086E-07	7.0272E-04	4.2307E-05	2.1893E-02	7.6455E-05	9.7682E-06	3.6815E-06	1.34636
25.00	1.1535E-07	5.7954E-04	6.4487E-05	2.1800E-02	1.0345E-04	1.9678E-05	9.9096E-06	1.28595
35.00	1.1618E-07	4.7338E-04	8.2803E-05	2.1702E-02	1.1994E-04	2.9710E-05	1.6946E-05	1.23023
45.00	1.1488E-07	3.8200E-04	9.7691E-05	2.1599E-02	1.2955E-04	3.9177E-05	2.3653E-05	1.17695
55.00	1.1240E-07	3.0391E-04	1.0945E-04	2.1491E-02	1.3456E-04	4.7713E-05	2.9490E-05	1.12534
65.00	1.0936E-07	2.3798E-04	1.1837E-04	2.1378E-02	1.3653E-04	5.5126E-05	3.4251E-05	1.07555

Table 18A. W-Std Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Zone 8 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	8.9033E-08	8.8911E-04	7.5948E-06	2.2004E-02	1.6387E-05	4.2644E-07	3.1121E-08	1.43286
15.00	9.4345E-08	8.1401E-04	2.1676E-05	2.1964E-02	4.3407E-05	3.0526E-06	6.2586E-07	1.40014
25.00	9.6895E-08	7.4490E-04	3.4507E-05	2.1922E-02	6.4463E-05	6.9034E-06	2.1454E-06	1.36787
35.00	9.8112E-08	6.8080E-04	4.6249E-05	2.1879E-02	8.1082E-05	1.1337E-05	4.4234E-06	1.33737
45.00	9.8385E-08	6.2119E-04	5.7006E-05	2.1835E-02	9.4253E-05	1.6051E-05	7.1982E-06	1.30835
55.00	9.7983E-08	5.6561E-04	6.6851E-05	2.1791E-02	1.0468E-04	2.0875E-05	1.0245E-05	1.28044
65.00	9.7089E-08	5.1371E-04	7.5848E-05	2.1744E-02	1.1289E-04	2.5699E-05	1.3394E-05	1.25333

Table 19A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Assembly Average Case

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.3522E-07	1.0292E-03	2.5456E-05	2.1733E-02	4.4753E-05	2.8318E-06	5.8740E-07	1.43861
15.00	1.5065E-07	8.0181E-04	6.7989E-05	2.1602E-02	9.9736E-05	1.5296E-05	7.4135E-06	1.34597
25.00	1.5413E-07	6.1275E-04	1.0153E-04	2.1461E-02	1.2813E-04	2.9417E-05	1.7852E-05	1.26481
35.00	1.5141E-07	4.5580E-04	1.2725E-04	2.1310E-02	1.4132E-04	4.2729E-05	2.7920E-05	1.18774
45.00	1.4560E-07	3.2776E-04	1.4588E-04	2.1147E-02	1.4546E-04	5.4193E-05	3.5947E-05	1.11210
55.00	1.3870E-07	2.2634E-04	1.5808E-04	2.0973E-02	1.4441E-04	6.3299E-05	4.1482E-05	1.03877
65.00	1.3203E-07	1.4937E-04	1.6447E-04	2.0785E-02	1.4078E-04	6.9917E-05	4.4712E-05	0.97087

Table 20A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Zone 1 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.1573E-07	1.0951E-03	1.2703E-05	2.1766E-02	2.2949E-05	7.5983E-07	7.6056E-08	1.46532
15.00	1.2184E-07	9.7195E-04	3.5944E-05	2.1707E-02	5.8764E-05	5.1145E-06	1.4212E-06	1.41752
25.00	1.2400E-07	8.5972E-04	5.6728E-05	2.1645E-02	8.4625E-05	1.1127E-05	4.5076E-06	1.37245
35.00	1.2380E-07	7.5684E-04	7.5335E-05	2.1582E-02	1.0334E-04	1.7783E-05	8.7142E-06	1.33028
45.00	1.2201E-07	6.6241E-04	9.1947E-05	2.1516E-02	1.1669E-04	2.4632E-05	1.3422E-05	1.28982
55.00	1.1916E-07	5.7578E-04	1.0668E-04	2.1447E-02	1.2591E-04	3.1409E-05	1.8190E-05	1.25015
65.00	1.1564E-07	4.9655E-04	1.1960E-04	2.1376E-02	1.3193E-04	3.7933E-05	2.2730E-05	1.21081

Table 21A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Zone 2 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.2634E-07	1.0527E-03	2.0776E-05	2.1746E-02	3.6537E-05	1.9142E-06	3.1781E-07	1.44873
15.00	1.3731E-07	8.6016E-04	5.6734E-05	2.1644E-02	8.5444E-05	1.1157E-05	4.6273E-06	1.37128
25.00	1.4005E-07	6.9412E-04	8.6564E-05	2.1536E-02	1.1401E-04	2.2360E-05	1.2216E-05	1.30211
35.00	1.3812E-07	5.5056E-04	1.1103E-04	2.1421E-02	1.2992E-04	3.3588E-05	2.0477E-05	1.23667
45.00	1.3355E-07	4.2729E-04	1.3053E-04	2.1299E-02	1.3754E-04	4.3999E-05	2.7958E-05	1.17236
55.00	1.2772E-07	3.2306E-04	1.4542E-04	2.1168E-02	1.3974E-04	5.3096E-05	3.3996E-05	1.10838
65.00	1.2152E-07	2.3694E-04	1.5599E-04	2.1030E-02	1.3855E-04	6.0597E-05	3.8393E-05	1.04539

Table 22A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Zone 3 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.3291E-07	1.0286E-03	2.5356E-05	2.1734E-02	4.4008E-05	2.7624E-06	5.6135E-07	1.43884
15.00	1.4694E-07	7.9958E-04	6.7831E-05	2.1606E-02	9.8043E-05	1.5040E-05	7.1774E-06	1.34574
25.00	1.4953E-07	6.0861E-04	1.0140E-04	2.1468E-02	1.2566E-04	2.9009E-05	1.7364E-05	1.26340
35.00	1.4623E-07	4.4988E-04	1.2718E-04	2.1319E-02	1.3811E-04	4.2189E-05	2.7174E-05	1.18454
45.00	1.4001E-07	3.2040E-04	1.4584E-04	2.1158E-02	1.4154E-04	5.3513E-05	3.4920E-05	1.10653
55.00	1.3283E-07	2.1816E-04	1.5799E-04	2.0985E-02	1.3987E-04	6.2439E-05	4.0145E-05	1.03042
65.00	1.2603E-07	1.4118E-04	1.6424E-04	2.0798E-02	1.3577E-04	6.8817E-05	4.3064E-05	0.95978

Table 23A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Zone 4 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.3853E-07	1.0177E-03	2.7646E-05	2.1727E-02	4.8256E-05	3.2864E-06	7.4044E-07	1.43384
15.00	1.5547E-07	7.7374E-04	7.3111E-05	2.1583E-02	1.0516E-04	1.7234E-05	8.8060E-06	1.33408
25.00	1.5882E-07	5.7429E-04	1.0807E-04	2.1427E-02	1.3266E-04	3.2588E-05	2.0408E-05	1.24668
35.00	1.5538E-07	4.1213E-04	1.3392E-04	2.1258E-02	1.4398E-04	4.6628E-05	3.0963E-05	1.16318
45.00	1.4880E-07	2.8345E-04	1.5158E-04	2.1076E-02	1.4617E-04	5.8233E-05	3.8801E-05	1.08163
55.00	1.3987E-07	1.8785E-04	1.6169E-04	2.0884E-02	1.4364E-04	6.6743E-05	4.3542E-05	1.00652
65.00	1.3322E-07	1.1822E-04	1.6571E-04	2.0679E-02	1.3924E-04	7.2513E-05	4.5968E-05	0.93907

Table 24A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Zone 5 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.3613E-07	1.0374E-03	2.4130E-05	2.1735E-02	4.3404E-05	2.6193E-06	5.2553E-07	1.44150
15.00	1.5237E-07	8.2322E-04	6.4751E-05	2.1610E-02	9.8333E-05	1.4307E-05	6.8066E-06	1.35429
25.00	1.5708E-07	6.4378E-04	9.7204E-05	2.1476E-02	1.2836E-04	2.7739E-05	1.6741E-05	1.27867
35.00	1.5568E-07	4.9318E-04	1.2259E-04	2.1333E-02	1.4384E-04	4.0643E-05	2.6716E-05	1.20780
45.00	1.5105E-07	3.6824E-04	1.4161E-04	2.1180E-02	1.5034E-04	5.2069E-05	3.5117E-05	1.13903
55.00	1.5280E-07	2.5214E-04	1.5651E-04	2.0989E-02	1.5139E-04	6.2832E-05	4.2448E-05	1.06114
65.00	1.4652E-07	1.6491E-04	1.6445E-04	2.0784E-02	1.4864E-04	7.0750E-05	4.6851E-05	0.98990

Table 25A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Zone 6 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.3270E-07	1.0494E-03	2.1808E-05	2.1742E-02	3.9511E-05	2.1778E-06	3.9464E-07	1.44649
15.00	1.4713E-07	8.5329E-04	5.9139E-05	2.1630E-02	9.1671E-05	1.2313E-05	5.4607E-06	1.36690
25.00	1.5179E-07	6.8617E-04	8.9700E-05	2.1511E-02	1.2200E-04	2.4328E-05	1.4019E-05	1.29746
35.00	1.5102E-07	5.4316E-04	1.1440E-04	2.1385E-02	1.3905E-04	3.6208E-05	2.3124E-05	1.23267
45.00	1.4721E-07	4.2163E-04	1.3376E-04	2.1252E-02	1.4756E-04	4.7103E-05	3.1274E-05	1.16996
55.00	1.5008E-07	3.0224E-04	1.5053E-04	2.1082E-02	1.5083E-04	5.8118E-05	3.9159E-05	1.09647
65.00	1.4416E-07	2.0868E-04	1.6100E-04	2.0900E-02	1.4963E-04	6.6754E-05	4.4445E-05	1.02730

Table 26A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Zone 7 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.2511E-07	1.0774E-03	1.6398E-05	2.1756E-02	3.0232E-05	1.2847E-06	1.7395E-07	1.45774
15.00	1.3541E-07	9.2531E-04	4.5560E-05	2.1675E-02	7.4302E-05	7.9497E-06	2.8377E-06	1.39691
25.00	1.3955E-07	7.9072E-04	7.0720E-05	2.1590E-02	1.0364E-04	1.6496E-05	8.1466E-06	1.34219
35.00	1.3986E-07	6.7071E-04	9.2405E-05	2.1501E-02	1.2317E-04	2.5506E-05	1.4626E-05	1.29172
45.00	1.3786E-07	5.6373E-04	1.1090E-04	2.1408E-02	1.3578E-04	3.4375E-05	2.1244E-05	1.24331
55.00	1.4403E-07	4.4190E-04	1.3063E-04	2.1280E-02	1.4511E-04	4.5111E-05	2.9461E-05	1.18094
65.00	1.3972E-07	3.3902E-04	1.4566E-04	2.1145E-02	1.4884E-04	5.4566E-05	3.6176E-05	1.12075

Table 27A. W-Std Design Basis Fuel Assembly

Enrichment = 5.0 W/O U-235, Zone 8 Case
(Burnup Corresponds to Assembly Average)

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.1176E-07	1.1208E-03	7.9583E-06	2.1776E-02	1.5097E-05	3.2427E-07	2.1362E-08	1.47454
15.00	1.1889E-07	1.0432E-03	2.2972E-05	2.1739E-02	4.0928E-05	2.3974E-06	4.5102E-07	1.44523
25.00	1.2115E-07	9.7037E-04	3.6930E-05	2.1701E-02	6.1955E-05	5.5475E-06	1.6149E-06	1.41640
35.00	1.2206E-07	9.0172E-04	4.9948E-05	2.1662E-02	7.9233E-05	9.2686E-06	3.4533E-06	1.38896
45.00	1.2202E-07	8.3678E-04	6.2096E-05	2.1622E-02	9.3481E-05	1.3310E-05	5.7954E-06	1.36279
55.00	1.3349E-07	7.2498E-04	8.2589E-05	2.1545E-02	1.1381E-04	2.1177E-05	1.1240E-05	1.31550
65.00	1.3292E-07	6.2392E-04	1.0049E-04	2.1465E-02	1.2785E-04	2.9144E-05	1.7078E-05	1.27138

Table 28A. W-OFA Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Assembly Average Case

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.0400E-07	7.9950E-04	2.3452E-05	2.1963E-02	4.3298E-05	3.3386E-06	6.7701E-07	1.40472
15.00	1.1480E-07	5.8346E-04	6.1272E-05	2.1832E-02	9.0194E-05	1.7524E-05	7.9710E-06	1.29986
25.00	1.1591E-07	4.1104E-04	8.9565E-05	2.1689E-02	1.0930E-04	3.2706E-05	1.8006E-05	1.20151
35.00	1.1241E-07	2.7527E-04	1.0969E-04	2.1532E-02	1.1459E-04	4.5921E-05	2.6432E-05	1.10370
45.00	1.0717E-07	1.7275E-04	1.2243E-04	2.1359E-02	1.1319E-04	5.6062E-05	3.1951E-05	1.00747
55.00	1.0204E-07	1.0051E-04	1.2861E-04	2.1169E-02	1.0929E-04	6.2864E-05	3.4775E-05	0.91992
65.00	9.8055E-08	5.4151E-05	1.2938E-04	2.0962E-02	1.0532E-04	6.6703E-05	3.5763E-05	0.84934

Table 29A. EXXON Design Basis Fuel Assembly

Enrichment = 4.0 W/O U-235, Assembly Average Case

Burnup [GWD/MTU]	Isotopic Number Densities used in KENO (Atoms / [Barn-Cm])							K_{eff} from DIT
	Sm-149	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	
5.00	1.0893E-07	8.0009E-04	2.3861E-05	2.1959E-02	4.6241E-05	3.5241E-06	7.7300E-07	1.39379
15.00	1.2303E-07	5.8731E-04	6.1953E-05	2.1819E-02	9.7195E-05	1.8110E-05	8.7777E-06	1.29062
25.00	1.2611E-07	4.1919E-04	9.0129E-05	2.1667E-02	1.1942E-04	3.3610E-05	1.9634E-05	1.19863
35.00	1.2386E-07	2.8751E-04	1.0999E-04	2.1501E-02	1.2719E-04	4.7188E-05	2.8914E-05	1.11063
45.00	1.1940E-07	1.8769E-04	1.2257E-04	2.1322E-02	1.2765E-04	5.7840E-05	3.5335E-05	1.02649
55.00	1.1464E-07	1.1590E-04	1.2888E-04	2.1128E-02	1.2494E-04	6.5334E-05	3.9035E-05	0.95071
65.00	1.1072E-07	6.7702E-05	1.3006E-04	2.0920E-02	1.2147E-04	6.9989E-05	4.0744E-05	0.88810