Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

FEB 19 2002

10 CFR 50.9

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

In the Matter of) Docket No.50-390
Tennessee Valley Authority)

SUBJECT: WATTS BAR NUCLEAR PLANT - REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING TRITIUM PRODUCTION - HOLTEC ANALYSIS (TAC NO. MB1884)

On November 21, 2001, TVA submitted Holtec International analysis requested in NRC RAI letter dated October 2, 2001. That November letter provided both a proprietary and non-proprietary versions of Holtec, International report HI-2012620.

After submittal of these documents, an email was received from the WBN NRC Project Manager which requested that a clarification of the non-proprietary version of the Holtec International analysis be made to clearly identify the proprietary information that had been removed. The enclosure to this letter provides a new non-proprietary version (Revision 3) which will supercede the previous revision provided in the November 21, 2001, letter. This new version revises the document to identify Appendices A and C to be withheld in accordance with paragraph (b)(4) of 10 CFR 2.790.

Please refer to the November 21, 2001, letter for the original withholding of proprietary information request, the associated Holtec International's affidavit, and Holtec International contact information.

D030

There are no regulatory commitments made by this letter. If you have any questions about this letter, please contact me at (423) 365-1824.

Sincerely,

P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures
cc: See page 3

Subscribed and sworn to before me
on this 19th day of February, 2002

E. Jeannette Long
Notary Public

My Commission expires May 21, 2005

cc (Enclosure):

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. L. Mark Padovan, Senior Project Manager
U.S. Nuclear Regulatory Commission
MS 08G9
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852-2738

U.S. Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

ENCLOSURE

TENNESSEE VALLEY AUTHORITY

WATTS NUCLEAR PLANT (WBN)

UNIT 1

DOCKET NO. 390

HOLTEC, INTERNATIONAL REPORT NUMBER HI-2012620, REVISION 3

NON-PROPRIETARY VERSION

HOLTEC
INTERNATIONAL

Holtec Center 555 Lincoln Drive West Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

QA Record

EVALUATION OF THE EFFECT OF THE USE OF TRITIUM PRODUCING BURNABLE ABSORBER RODS (TPBARS) ON FUEL STORAGE REQUIREMENTS

FOR

WATTS BAR UNIT 1 (TVA)

PROJECT Watts Bar DISCIPLINE N
CONTRACT 99NNO-254781 UNIT 1
DESC. SFP Criticality Analysis - Tritium Production
DWG/DOC NO. HI-2012620 (NP)
SHEET - OF - REV. 03
DATE 01/29/02 ECN/DCN - FILE N2N-056

Holtec Report No: HI-2012620 (NP), R3

Holtec Project No: 90941

Report Class : SAFETY RELATED

APPROVED
This approval does not relieve the Contractor from any part of his responsibility for the correctness of design, details and dimensions.
Letter No. 30M431
Date: January 29, 2002
TENNESSEE VALLEY AUTHORITY
SDEP. (N) BY D.L. Lundy

NON-PROPRIETARY VERSION

RIMS, WTC A-K

Summary of Revisions

Report HI-2012620

Revision 1: The document is revised to incorporate client comments transmitted to Holtec International by TVA via Letter 30M410 dated April 10, 2001. There are no changes to the conclusions of the report.

Revision 2: The document is revised to incorporate client comments transmitted to Holtec International by TVA via Letter 30M414 dated July, 2001. There are no changes to the conclusions of the report.

Revision 3: The document is revised to incorporate client comments transmitted to Holtec International by TVA via Letter 30M430 dated January 11, 2002. There are no changes to the conclusions of the report.

TABLE OF CONTENTS

1.0 INTRODUCTION AND SUMMARY 1

2.0 ANALYSIS CRITERIA AND ASSUMPTIONS 5

3.0 ACCEPTANCE CRITERIA 6

4.0 DESIGN AND INPUT DATA 7

5.0 METHODOLOGY 8

6.0 ANALYSIS RESULTS 9

7.0 ACCIDENT CONDITIONS AND SOLUBLE BORON REQUIREMENTS... 12

8.0 OTHER BURNABLE POISON ROD INSERTS IN THE FUEL ASSEMBLIES.. 13

9.0 CRITICALITY ANALYSES RESULTS AND CONCLUSIONS 14

10.0 REFERENCES 15

TABLES (Total 11)

FIGURES (Total 4)

APPENDIX A List of Holtec's QA Approved Computer Codes List

APPENDIX B Benchmark Calculations

APPENDIX C List of CASMO4 and KENO-Va Input Files

LIST OF TABLES

4.1 Design Basis Fuel Assembly Specifications 16

6.1 Reactivity Effects of Density Tolerance in the Watts Bar Spent Fuel Racks.......... 17

6.2 Reactivity Effects of Temperature and Void in Watts Bar Spent Fuel Racks.. 18

6.3 Reactivity Effects of Fuel Enrichment Tolerance in Watts Bar Spent Fuel Racks... 19

6.4 Reactivity Effects of Abnormal and Accident Conditions in Watts Bar Spent Fuel Racks 20

6.5 Summary of the Criticality Safety Analyses for Checkerboard Storage of 2 Fresh and 2 Spent Fuel Assemblies In Watts Bar Racks (Arrangement 2).......... 21

6.6 Summary of the Criticality Safety Analyses for Face Adjacent Storage of Spent Fuel Assemblies In Watts Bar Racks (Arrangement 1)............................... 22

6.7 Summary of the Criticality Safety Analyses for Checkerboard Storage of 3 Fresh Fuel Assemblies and 1 Water Cell in Watts Bar Racks (Arrangement 3)............. 23

6.8 Summary of the Criticality Safety Analyses for Storage of Fresh Fuel Assemblies, Containing 32 IFBA rods, in Watts Bar Racks (Arrangement 4)...... 24

6.9 Summary of the Analyses of the Postulated Accidents in the Watts Bar Spent Fuel Storage Racks.. 25

6.10 Comparison of the Reactivity of Fuel Assemblies Depleted with Different Burnable Poison Rod Types.. 26

6.11 Comparison of the Reactivity Effects of Depletion with Different Poison Materials 27

LIST OF FIGURES

Figure 1	Minimum Burnup of Spent Fuel in 2x2 Checkerboard Arrangement of Spent and Fresh Fuel of 4.95% Enrichment (Arrangement 2)	28
Figure 2	Minimum Burnup for Unrestricted Storage of Spent Fuel of Various Initial Enrichments (Arrangement 1)	29
Figure 3	Comparison of the Reactivity of Fuel Assemblies Depleted with Different Burnable Poison Rod Types	30
Figure 4	Fuel Storage Cell Cross-Section	31

1.0 INTRODUCTION AND SUMMARY

1.1 Objectives and General Description

The objective of the criticality safety analysis documented in this report is to evaluate the safe storage configuration of fresh and spent fuel assemblies in the Watts Bar Nuclear Plant spent fuel storage racks. This new analysis is performed with fuel assemblies containing tritium producing burnable absorber rods (TPBARs). Previous analysis performed by Holtec International [9] determined the safe storage patterns for spent fuel in the racks for fuel containing no burnable poison rods. In addition to the TPBARs, the presence of other burnable poison rods such as WABAs and IFBA rods in the fuel assemblies has also been addressed in the present analysis. Credit is taken for integral fuel burnable absorber (IFBA) rods and fuel burnup, where appropriate. Soluble boron in pool water is used to protect against a mis-loaded assembly accident, where necessary. The analysis uses the KENO5a Monte Carlo code with the 238-group cross-section library developed by the Oak Ridge National Laboratory as the primary code for the calculations. CASMO4 was used for calculation of fuel depletion effects and manufacturing tolerances. As permitted in the USNRC guidelines, parametric evaluations were performed for each of the manufacturing tolerances and the associated reactivity uncertainties were combined statistically. All calculations were made for an explicit modeling of the fuel and storage cell geometries to define the enrichment-burnup combinations for spent fuel configurations that assure a safe storage of fresh and spent fuel in the pool.

The following configurations of fresh and spent fuel storage in the Watts Bar racks have been analyzed in this report. The fuel was assumed to have initially contained Integral Fuel Burnable Absorber (IFBA) Rods and TPBARs, which are removed at the time the assemblies are placed in storage.

1. Storage of spent fuel with credit for burnup only.
2. Checkerboard of two fresh Fuel (initial enrichment of 4.95±0.05 wt%) and two spent fuel assemblies.

3. Checkerboard storage of three fresh fuel assemblies (initial enrichment of 4.95±0.05 wt%) and one cell containing only water or water and non-fuel materials.

4. Storage of fresh fuel, containing IFBA rods, in the racks with no other restrictions, other than that the assemblies contain at least 32 IFBA rods (1.25x).

Postulated accident conditions, where a fresh fuel assembly without IFBA rods, is inadvertently placed into a cell intended to remain empty or to contain a spent fuel or fresh fuel assemblies with IFBA rods, have also been evaluated.

1.2 Summary of Results

Arrangement 1

Previous analyses performed [Reference 9] showed that the required burnup for the spent fuel (initial enrichment of 4.95±0.05wt%) in this configuration was 6.75 GWD/MTU. The required burnup for the fuel assemblies containing TPBARs remain the same. A summary of the calculations, for fuel with an initial enrichment of 4.95±0.05wt%, is given in Table 6.6. The required burnup for other initial enrichment is shown in Figure 2.

Arrangement 2

Previous analyses performed in [Reference 9] for a 2x2 checkerboard arrangement showed that the required burnup for the spent fuel (initial enrichment of 4.95±0.05 wt %) in this configuration was 20 GWD/MTU. In the present analysis, the required burnup for the fuel assemblies containing TPBARs remain the same. A summary of the calculations, for fuel with an initial enrichment of 4.95±0.05 wt%, is given in Table 6.5. The required burnups for other initial enrichments are shown in Figure 1.

Arrangement 3

In this arrangement, 3 fresh fuel assemblies are checker boarded with 1 water cell in a 2x2 array. This arrangement was found to be acceptable for fresh fuel storage without any additional restriction. A summary of the calculations, for fuel with an initial enrichment of 4.95±0.05wt%, is given in Table 6.7. Analyses were also performed to determine the limiting amount of water that can be displaced in order to checkerboard non-fissile bearing components (such as a boral coupon tree, thimble plug etc.) with fresh fuel. It was conservatively determined that 75% of water can be safely displaced in empty cells by non-fissile bearing components. Cells containing items such as TPBAR consolidation baskets and baskets containing discarded materials may be considered water cells, as long as the material is non-fissile and no more than 75% of the water is displaced. These analyses also confirm that non-fuel bearing assembly components (i.e. thimble plugs, rod cluster control assemblies (RCCAs) etc.) may be stored in the fuel assemblies without affecting the storage requirements for the assemblies.

Arrangement 4

In this arrangement, fresh fuel assemblies containing integral burnable absorber rods (IFBA) are stored face adjacent to each other. The fuel assemblies were assumed to contain 16, 32 and 48 IFBA rods. Calculations show that the fuel assemblies containing a minimum of 32 IFBA rods can be stored in the storage cells without any credit for burnup, with a maximum $k_{eff} \leq 0.95$ including bias and uncertainties. A summary of the calculations, for fuel with an initial enrichment of 4.95±0.05wt% and containing 32 IFBA rods (at 1.25x), is given in Table 6.8. Assemblies with a greater number of IFBA rods would exhibit a lower reactivity.

Interface Requirements

When arrangements 2 and 3 are placed adjacent to each other in the pool, there should be a barrier row of empty cells between the two arrays to prevent fresh fuel assemblies from being adjacent to each other in these arrays.

Accident Condition

Evaluation of postulated accident conditions demonstrate that 55 ppm of soluble boron in the spent fuel pool is sufficient to maintain $k_{eff} \leq 0.95$, including calculational biases and all uncertainties under the most serious postulated fuel handling or mis-loading accident. Recent USNRC guidelines allow partial credit for soluble boron, and this would be more than adequate to protect against the most serious fuel handling accident. Normal soluble boron levels are maintained above 2000 ppm in the spent fuel pool.

2.0 ANALYSIS CRITERIA AND ASSUMPTIONS

To assure the true reactivity will always be less than the calculated reactivity, the following conservative analysis criteria or assumptions were used.

- Criticality safety analyses were based upon an infinite radial array of cells; i.e., no credit was taken for radial neutron leakage, except for evaluating the rack boundaries accident conditions where neutron leakage is inherent.

- Minor structural materials were neglected; i.e., spacer grids were conservatively assumed to be replaced by water.

- The analyses assumed a temperature of 4 °C, which is the temperature of highest water density and highest reactivity in poisoned racks.

- The analyses assumed a Westinghouse V5H 17x17 fuel assembly, which was found to be the most reactive of the fuel assembly types in use at Watts Bar Nuclear Plant, for the burnup appropriate to the analysis.

- The density of the fuel was assumed to be 97% of the nominal theoretical density, with a tolerance of ± 2%.

- Boron-10 was used to simulate the Li-6 in the TPBARs, since CASMO-4 does not include Li-6 in the cross-section library. To accomplish this, the number density of B-10 was adjusted to give the same absorption cross section as the Li-6 by KENO-Va calculations. This is a conservative assumption since the B-10 (Li-6) was not depleted.

- No credit is taken for the presence of the Uranium-236 isotope in the fuel for this analysis.

- No axial blankets were assumed to be present in the fuel rods. The entire active fuel length was assumed to have the same enrichment.

- WABAs or TPBARs and IFBA rods were assumed to be present during the operating life of the fuel assemblies. This penalty is bounding for the fuel assemblies, which operate without poison rods.

3.0 ACCEPTANCE CRITERIA

The primary acceptance criterion is that the effective multiplication factor (k_{eff}) of the racks shall remain less than or equal to 0.95, under normal conditions. The maximum k_{eff} includes calculation uncertainties and reactivity effects of mechanical tolerances, under the postulated accident of the loss of all soluble boron. Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage.
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December, 1981.
- ANSI-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum L. Kopp to Timothy Collins, August 19, 1998.
- Code of Federal Regulation 10CFR50.68, "Criticality Accident Requirements"

4.0 DESIGN AND INPUT DATA

4.1 Fuel Assembly and Component Design Specifications

Two different fuel assembly designs were considered in the analyses; the Westinghouse 17x17 V5H and Robust designs. Table 4.1 provides the pertinent design details for the fuel assembly types. Calculations were performed for fuel operating with both the TPBARs and the WABA components. Design specifications for the TPBARs are obtained from Reference 7. The compositions of the fuel assemblies containing either IFBA or WABA rods were obtained from Reference 8.

4.2 Storage Racks

The storage rack design is described in detail in Reference 9. A schematic of the fuel storage cell model, used in this analysis, is shown in Figure 4. The tolerances in the dimensions are also presented in Reference 9, and have been used in the present analysis.

4.3 Operating Parameters

The core operating parameters for performing the depletion calculations were obtained from Reference 8. The principal core operating parameters, used in this study, are summarized in the table below.

Core Operating Parameters	Value
Fuel Temperature (°F)	1370
Moderator Temperature (°F)	592
Average Soluble Boron in Moderator (ppm)	700

5.0 METHODOLOGY

The criticality analyses were performed principally with the three-dimensional NITAWL-KENO5a Monte Carlo code package [1]. NITAWL was used with the 238-group SCALE-4.3 cross-section library and the Nordheim integral treatment for resonance shielding effects. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0030 ± 0.0012 (95%/95%) [2]. CASMO4, a two-dimensional deterministic code [4] using transmission probabilities, was used for depletion (burnup) calculations and to evaluate the small (differential) reactivity effects of manufacturing tolerances. Validity of the CASMO4 code was established by comparison with KENO5a calculations for comparable rack cases.

In the geometric model used in the calculations, each fuel rod, and associated cladding and each fuel assembly were explicitly described. Reflecting boundary conditions effectively defined an infinite radial array of storage cells. In the axial direction, a 30-cm water reflector was used to conservatively describe axial neutron leakage. Each stainless steel box and the water gaps [8, 9] were described in the calculational model. The fuel cladding material was assumed to be zirconium.

Monte Carlo (KENO5a) calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the KENO5a calculated reactivities, a minimum of 1 million neutron histories were accumulated in each calculation, generally resulting in a statistical uncertainty of about ±0.0003 Δk (1σ). Three-dimensional KENO5a calculations were necessary to describe the geometry of the checkerboard cases. However, KENO5a cannot perform depletion calculations. Depletion calculations were performed with CASMO4 with explicit description of the fission product nuclide concentration. To compensate for those fission product nuclides, which cannot be described in KENO5a, an equivalent boron-10 in the fuel was determined which produced the same reactivity in KENO5a as the CASMO4 result. This methodology incorporates approximately 40 of the most important fission products, accounting for all but about 1% in k. The remaining 1 % in k is included by the equivalent B-10 concentration in the fuel.

6.0 ANALYSIS RESULTS

6.1 Bounding Fuel Assembly

Calculations were done, using CASMO4, to evaluate the reactivity of the fuel assemblies currently in use or anticipated for storage in Watts Bar spent fuel racks. Calculations show that the Westinghouse 17x17 V5H fuel assembly exhibits the highest reactivity at the burnups of interest in this analysis (from 0 to 35 GWD/MTU) and were used in all the subsequent calculations. Beyond 35 GWD/MTU burnup, the Westinghouse Robust fuel design becomes slightly more reactive, but this does not affect the present analyses.

Burnup, GWD/MTU	W 17x17 V5H	W 17x17 ROBUST
0	0.9792	0.9776
10	0.9132	0.9109
20	0.8629	0.8608
30	0.8193	0.8174
35	0.7990	0.7973

6.2 Evaluation of Manufacturing Tolerance Uncertainties

CASMO4 calculations were made to determine the uncertainties in reactivity associated with density and enrichment tolerances. The uncertainties associated with the other mechanical tolerances have been assumed to be the same as that reported in the earlier analysis [9]. The reactivity effects of each independent tolerance were combined statistically. All fuel and rack dimensions and their dimensional tolerances are obtained from References 8 and 9. The reactivity effects of the tolerances are listed in Tables 6.5-6.8.

For estimating the reactivity uncertainties associated with tolerances in fuel enrichment and density, conservative tolerances of ± 0.05% in enrichment and ±2% in UO_2 density were assumed. The reactivity uncertainty associated with the fuel density tolerance is listed in Table 6.1. The reactivity uncertainties for the tolerance in fuel enrichment are listed in Table 6.3.

6.3 Uncertainty in Depletion Calculations

The uncertainty in depletion calculations is part of the methodology uncertainty and was taken as 5% of the reactivity decrement from beginning-of-life to the burnup of concern for the spent fuel [5]. This methodology uncertainty is included in the calculations of the final k_{eff} in Table 6.5 and 6.6.

6.4 Uncertainty in TPBAR Loading

Since CASMO4 does not include Li-6 (as used in the TPBARs), an equivalent boron was used to stimulate the absorption in Li-6. Since this approximation could introduce some uncertainty, a sensitivity analysis was made by doubling the boron concentration in the simulated TPBAR's*. Results of this analysis showed that the effect on the residual reactivity was virtually negligible. For the most sensitive storage configuration (checkerboard of 2 fresh assemblies with 2 assemblies burned to 20 GWD/MTU), doubling the TPBAR absorption resulted in only a 0.0005 Δk increase in reactivity, and would not affect the other configurations.

6.5 Eccentric Location of Fuel Assemblies

The fuel assemblies are nominally stored in the center of the storage cells. Eccentric positioning of fuel assemblies in the cells normally results in a reduction in reactivity for poisoned racks. Calculations have been made confirming negative reactivity effect of the eccentric positioning fuel assemblies at the position of closest approach. These calculations confirm that the normal centered position is the most reactive.

* The TPBARs are removed when the assembly is placed in storage. Therefore, the TPBAR composition only affects the residual reactivity after TPBAR removal

6.6 Temperature and Void Effects

Temperature effects were also evaluated, using CASMO4, in the temperature range from 4°C to 120°C and the results are listed in Table 6.2. These results show that the temperature coefficient of reactivity is negative. The void coefficient of reactivity (boiling conditions) was also found to be negative for the Watts Bar racks.

6.7 Reactivity Effect of the Axial Burnup Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower ends. At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of high neutron leakage. Consequently, it would be expected that over most of the burnup history, distributed burnup fuel assemblies would exhibit a slightly lower reactivity than that calculated for the average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup.

The effect of the axial burnup distribution on reactivity was studied, on a generic basis, in detail by Turner [6]. Reference 6 indicates that below 30 GWD/MTU, the axial burnup penalty is negative. Since all the required burnups in this analysis are substantially lower than about 30 GWD/MTU, an axial burnup penalty was not required in any of the four different storage patterns investigated.

7.0 ACCIDENT CONDITIONS AND SOLUBLE BORON REQUIREMENTS

Soluble boron is required to protect against the accident of a mis-loaded fuel assembly. The accident analyses corresponding to all the storage configurations investigated in this analysis are summarized below:

- Fresh fuel assembly misloaded into a cell intended to store a spent fuel assembly in Arrangement 1
- Fresh fuel assembly misloaded into a cell intended to store a spent fuel assembly in Arrangement 2
- Fresh fuel assembly misloaded into a location intended to be a water cell in the Arrangement 3
- Fresh fuel assembly, without any IFBA, misloaded into a cell intended to store a fresh fuel assembly with 32 IFBA rods, in Arrangement 4

Table 6.9 summarizes the k_{eff} for each of these accident analyses. The results show that the most serious postulated accident condition with the misplacement of a fresh fuel assembly occurs in arrangement 3. In this case, a fresh fuel assembly is misplaced in the location of a water cell. Calculations show that 55 ppm of soluble boron would be required to maintain the k_{eff} in the rack below the regulatory requirement of 0.95, including bias and uncertainties. Misplacement of a fuel assembly outside the periphery of a storage module, or a dropped assembly lying on top of the rack would have a smaller reactivity effect.

8.0 OTHER BURNABLE POISON ROD INSERTS IN THE FUEL ASSEMBLIES

The fuel assemblies used at the Watts Bar may contain poison rods other than the TPBARs. Analyses show that the fuel assemblies containing TPBARs are more reactive than those containing BPRAs and are essentially the same as those with WABAs, at the burnups of interest. At higher burnups, the fuel assemblies with TPBARs exhibit higher reactivity. The results are summarized in Table 6.10, and illustrated in Figure 3.

With IFBA rods present, similar calculations show that the WABA case yields a slightly higher reactivity than the TPBAR case. These results are tabulated in Table 6.11. The difference does not affect the results given in this report. The spent fuel calculated without IFBA present bounds all other cases.

9.0 CRITICALITY ANALYSES RESULTS AND CONCLUSIONS

Four different storage configurations of fresh and spent fuel assemblies in the Watts Bar spent fuel pool have been evaluated in this analysis. The results indicate that these storage patterns of fresh fuel assemblies (4.95±0.05 wt% enrichment) and spent fuel assemblies meets the regulatory requirements. The results for the different arrangements are summarized in Tables 6.5 to 6.8. Results show that the burnup requirements for the storage arrangements 1 and 2 remain the same as those reported in Reference 9. A summary of the conditions evaluated and the conclusions are given below:

- Spent fuel assemblies may be stored in unrestricted locations provided that they satisfy the burnup-enrichment combinations identified in Figure 2 (minimum of 6.75 MWD/Kg-U burnup for fuel of 4.95±0.05 wt% initial enrichment). Fuel of 3.8 wt% or less U^{235} may be also stored without restrictions.
- Storage of two fresh fuel assemblies (4.95±0.05 wt% initial enrichment) in a 2x2 checkerboard array with two spent fuel assemblies, whose burnup-enrichment combination is shown in Figure 1 (minimum of 20 MWD/Kg-U burnup for fuel of 4.95±0.05 wt% initial enrichment), satisfy the regulatory requirements.
- Checkerboard arrangement of 3 fresh fuel assemblies and 1 empty cell satisfy the regulatory requirements for fuel storage in the racks.
- Fresh fuel assemblies, of 4.95±0.05 wt% initial enrichment, containing a minimum of 32 (1.25x) IFBA rods may be stored face adjacent to each other in the spent fuel storage racks. These may also be stored face adjacent to spent fuel assemblies satisfying burnup-enrichment combinations in Figure 2 (minimum of 6.75 MWD/Kg-U burnup for fuel of 4.95±0.05 wt% initial enrichment).
- A water cell will always be less reactive than an irradiated fuel assembly. Conservatively, 75% of the water may be safely displaced from a cell by non-fissile materials and the cell may still be considered a water cell.
- Accident analysis show that only 55 ppm of soluble boron is required to mitigate the effects of the most serious postulated fuel misplacement and maintain the k_{eff} below 0.95, including all uncertainties and biases.

10.0 REFERENCES

1. R.M. Westfall, et. al., "NITAWL-S: Scale System Module for Performing Resonance Shielding and Working Library Production" in SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation., NUREG/CR- 0200, 1979.

   L.M. Petrie and N.F. Landers, "KENO Va. An improved Monte Carlo Criticality Program with Subgrouping" in SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation., NUREG/V-0200, 1979.

2. M.G. Natrella, Experimental Statistics, National Bureau of Standards, Handbook 91, August 1963.

3. J.F. Briesmeister, Ed., "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A", Los Alamos National Laboratory, LA-12625-M (1993).

4. A. Ahlin, M. Edenius, H. Haggblom, "CASMO- A Fuel Assembly Burnup Program," AE-RF-76-4158, Studsvik report (proprietary).

   A. Ahlin and M. Edenius, "CASMO- A Fast Transport Theory Depletion Code for LWR Analysis," ANS Transactions, Vol. 26, p. 604, 1977.

   D. Knott, "CASMO4 Benchmark Against Critical Experiments", Studsvik Report SOA-94/13 (Proprietary).

   M. Edenius et al., "CASMO4, A Fuel Burnup Program, Users Manual" Studsvik Report SOA/95/1.

5. L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum, L. Kopp to Timothy Collins, August 19,1998.

6. S. E. Turner, "Uncertainty Analysis - Burnup Distributions," Presented at the 1988 DOE/SANDIA Technical Meeting on Fuel Burnup Credit

7. Attachments to the TVA Letter 30M394 from D.L.Lundy (TVA) to K.K. Niyogi (Holtec), dated August 21, 2000.

8. TVA Letter 30M400 to Holtec International, dated January 8, 2001

9. Holtec Report HI-961513, Revision 2, Holtec Project Number 10371

Table 4.1 Design Basis Fuel Assembly Specifications [8]

Fuel Assembly	W 17x17 V5H	W 17x17 ROBUST
Clad O.D., in	0.374	0.374
Clad I.D., in	0.329	0.329
Clad Material	Zircaloy-4	Zirlo
Pellet Diameter, in	0.3225	0.3225
Density, g/cc	10.631	10.631
Maximum Enrichment %	4.95±0.05 w/o	4.95±0.05 w/o
Active Fuel Length, in	144	144
Number Fuel Rods	264	264
Fuel Rod Pitch	0.496	0.496
Number of Thimbles	25	25
Thimble O.D.	0.474	0.482
Thimble I.D.	0.442	0.442

Table 6. 1. Reactivity Effects of Density Tolerance in the Watts Bar Spent Fuel Racks.

BURNUP, GWD/MTU	REFERENCE	FUEL DENSITY	
	k_{inf}	k_{inf}	Δk
0	0.9776	0.9795	0.0019
10	0.9107	0.9122	0.0015
20	0.8606	0.8621	0.0015
30	0.8173	0.8189	0.0016
35	0.7971	0.7989	0.0018

Table 6.2 Reactivity Effects of Temperature and Void in Watts Bar Spent Fuel Racks.

BURNUP, GWD/MT	T = 4 ^{0}C	T = 20 ^{0}C		T = 120 ^{0}C		T = 120 ^{0}C + VOID	
	k_{inf}	k_{inf}	Δk^+	k_{inf}	Δk^*	k_{inf}	Δk^{**}
0	0.9792	0.9776	-0.0016	0.9548	-0.0228	0.9262	-0.0286
10	0.9132	0.9107	-0.0025	0.8895	-0.0212	0.8619	-0.0276
20	0.8629	0.8606	-0.0023	0.8404	-0.0202	0.8134	-0.0270
30	0.8193	0.8173	-0.0020	0.7984	-0.0189	0.7720	-0.0264
35	0.7990	0.7971	-0.0019	0.7790	-0.0181	0.7529	-0.0261

\+ difference with results @ 4 °C
* difference with results @ 20 °C
** difference with results at 120 °C and no void

Table 6.3 Reactivity Effects of Fuel Enrichment Tolerance in Watts Bar Spent Fuel Racks.

BURNUP, GWD/MTU	**REFERENCE**	**ENRICHMENT TOLERANCE**	
	k_{inf}	k_{inf}	Δk
0	0.9776	0.9793	0.0017
10	0.9107	0.9124	0.0017
20	0.8606	0.8624	0.0018
30	0.8173	0.8191	0.0018
35	0.7971	0.7990	0.0019

Table 6.4 Reactivity Effects of Abnormal and Accident Conditions in Watts Bar Spent Fuel Racks.

ACCIDENT/ABNORMAL CONDITIONS	REACTIVITY EFFECT
Temperature increase (See Table 6.2)	Negative
Void (Boiling) (See Table 6.2)	Negative
Misplacement of a fresh fuel assembly	Positive: most serious misplacement accident requires 55 ppm soluble boron
Eccentric Positioning of Fuel Assemblies	Negative

Table 6.5 Summary of the Criticality Safety Analyses for Checkerboard Storage of 2 Fresh and 2 Spent Fuel Assemblies In Watts Bar Racks (Arrangement 2).

Reference k_{eff}	0.9233
Required Burnup of the Spent Fuel Assemblies	20 GWD/MTU
Keno5a Bias	0.0030
Temperature Correction to 4 °C	0.0023
Axial Burnup Distribution Penalty	Not Applicable
KENO5a Bias Uncertainty	0.0012
KENO Statistics (95/95) Uncertainty (1.7 *σ)	0.0009
Mechanical Tolerance Uncertainty	0.0059
Density Tolerance Uncertainty	0.0019
Enrichment Tolerance Uncertainty	0.0018
Depletion Uncertainty	0.0059
Fuel Eccentric Positioning Uncertainty	Negative
Statistical Combination of Uncertainties	0.0089
Maximum k_{eff}	0.9375
Regulatory Limiting k_{eff}	0.9500

Table 6.6 Summary of the Criticality Safety Analyses for Face Adjacent Storage of Spent Fuel Assemblies In Watts Bar Racks (Arrangement 1).

Reference k_{eff}	0.9271
Required Burnup of the Spent Fuel Assemblies for 4.95+/-0.05 wt% initial enrichment	6.75 GWD/MTU
Keno5a Bias	0.0030
Temperature Correction to 4 °C	0.0022
Axial Burnup Distribution Penalty	Not Applicable*
KENO5a Bias Uncertainty	0.0012
KENO Statistics (95/95) Uncertainty (1.7 *σ)	0.0009
Mechanical Tolerance Uncertainty	0.0059
Density Tolerance Uncertainty	0.0016
Enrichment Tolerance Uncertainty	0.0017
Depletion Uncertainty	0.0023
Fuel Eccentricity Uncertainty	Negative
Statistical Combination of Uncertainties	0.0069
Maximum k_{eff}	0.9392
Regulatory Limiting k_{eff}	0.9500

* Reactivity dominated by once-burned assemblies, which suppresses the axial burnup penalty.

Table 6.7 Summary of the Criticality Safety Analyses for Checkerboard Storage of 3 Fresh Fuel Assemblies and 1 Water Cell in Watts Bar Racks (Arrangement 3).

Reference k_{eff}	0.9131
Keno5a Bias	0.0030
Temperature Correction to 4 °C	0.0016
Axial Burnup Distribution Penalty	Not Applicable
KENO5a Bias Uncertainty	0.0012
KENO Statistics (95/95) Uncertainty (1.7 * σ)	0.0010
Mechanical Tolerance Uncertainty	0.0059
Density Tolerance Uncertainty	0.0019
Enrichment Tolerance Uncertainty	0.0017
Depletion Uncertainty	Not Applicable
Fuel Eccentricity Uncertainty	Negative
Statistical Combination of Uncertainties	0.0066
Maximum k_{eff}	0.9243
Regulatory Limiting k_{eff}	0.9500

Table 6.8 Summary of the Criticality Safety Analyses for Storage of Fresh Fuel Assemblies, containing 32 IFBA rods, in Watts Bar Racks (Arrangement 4).

Reference k_{eff}	0.9365
Keno5a Bias	0.0030
Temperature Correction to 4 °C	0.0016
Axial Burnup Distribution Penalty	Not Applicable
KENO5a Bias Uncertainty	0.0012
KENO Statistics (95/95) Uncertainty (1.7 *σ)	0.0010
Mechanical Tolerance Uncertainty	0.0059
Density Tolerance Uncertainty	0.0019
Enrichment Tolerance Uncertainty	0.0017
Depletion Uncertainty	Not Applicable
Fuel Eccentricity Uncertainty	Negative
Statistical Combination of Uncertainties	0.0066
Maximum k_{eff}	0.9477
Regulatory Limiting k_{eff}	0.9500

Table 6.9 Summary of the Analyses of the Postulated Accidents in the Watts Bar Spent Fuel Storage Racks.

Description of Accident	K_{eff}	Calculation Biases, Penalty and Uncertainties	Total k_{eff}
Fresh fuel assembly misloaded in the location of a spent fuel assembly in Arrangement 1 (face adjacent storage)	0.9292	0.0121	0.9413
Fresh fuel assembly misloaded in the location of a spent fuel assembly in Arrangement 2 (checkerboard loading)	0.9292	0.0140	0.9432
Fresh fuel assembly misloaded in the location of a water cell in Arrangement 3	0.9435	0.0112	0.9547
Fresh fuel assembly, without IFBA rods, misloaded in the location of a fresh fuel assembly, with 32 IFBA rods, in Arrangement 4	0.9375	0.0112	0.9487

Table 6.10 Comparison of the Reactivity of Fuel Assemblies Depleted with Different Burnable Poison Rod Types.

BURNUP, GWD/MTU	W-V5H with TPBAR	W-V5H with BPRA	W-V5H WITH WABA
	k_{inf}	k_{inf}	k_{inf}
0	0.9792	0.9792	0.9792
10	0.9132	0.9120	0.9137
20	0.8629	0.8581	0.8634
30	0.8193	0.8087	0.8177
35	0.7990	0.7833	0.7926

Table 6.11 Comparison of the Reactivity Effects of Depletion with Different Poison Materials

	k_{eff} in rack				
Burnup, MWD/KgU	IFBA	No	No	Yes	Yes
	TPBAR	No	Yes	Yes	No
	WABA	Yes	No	No	Yes
0		0.9792	0.9792	0.9402	0.9402
10		0.9137	0.9132	0.9016	0.9018
15		0.8876	0.8869	0.8808	0.8813
20		0.8634	0.8629	0.8600	0.8604
25		0.8403	0.8405	0.8392	0.8390
30		0.8177	0.8193	0.8188	0.8173
35		0.7926	0.7990	0.7990	0.7951

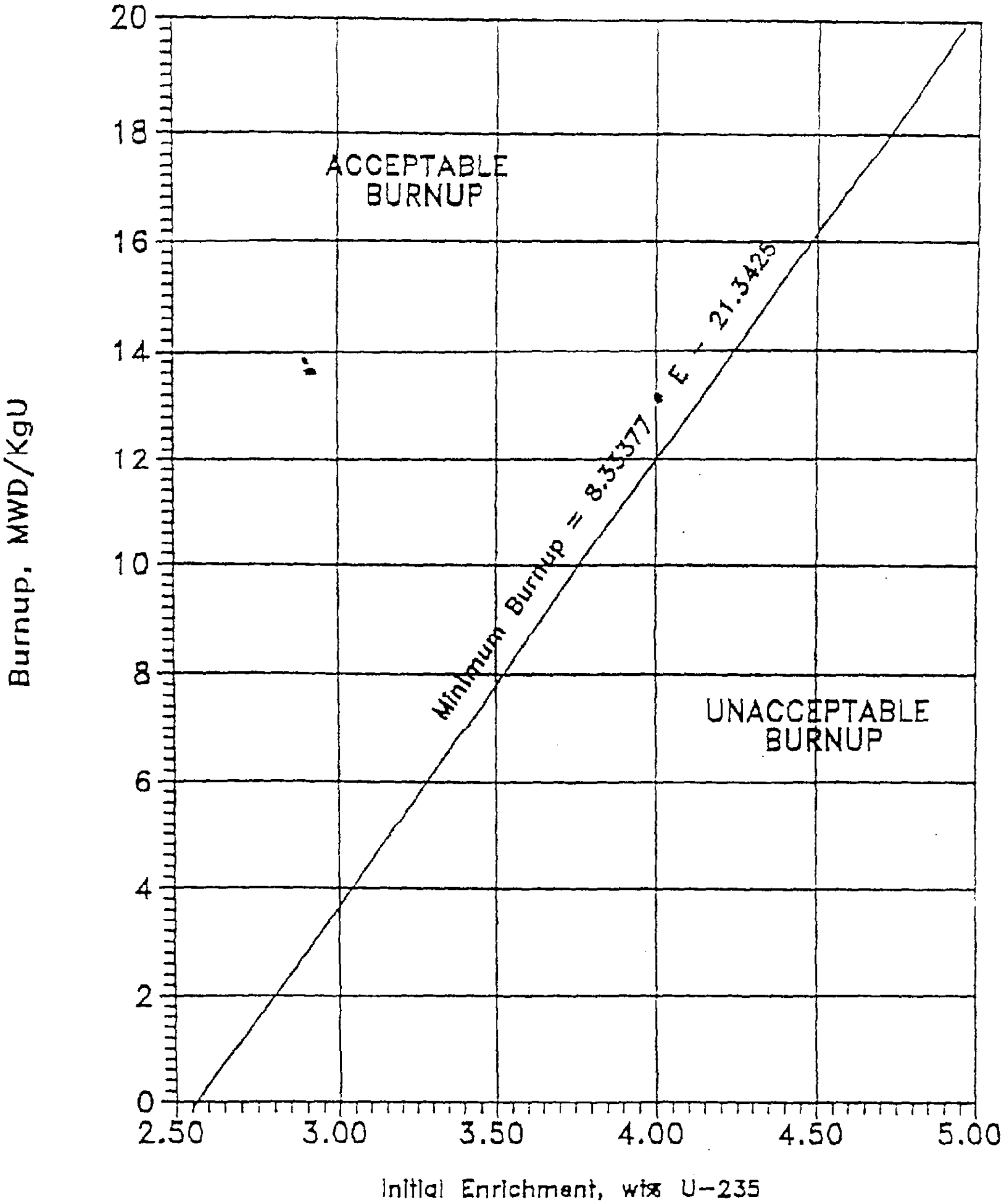

Figure 1 Minimum Burnup of Spent Fuel in 2x2 Checkerboard Arramgement of Spent and Fresh Fuel of 4.95% Enrichment (Arrangement 2)

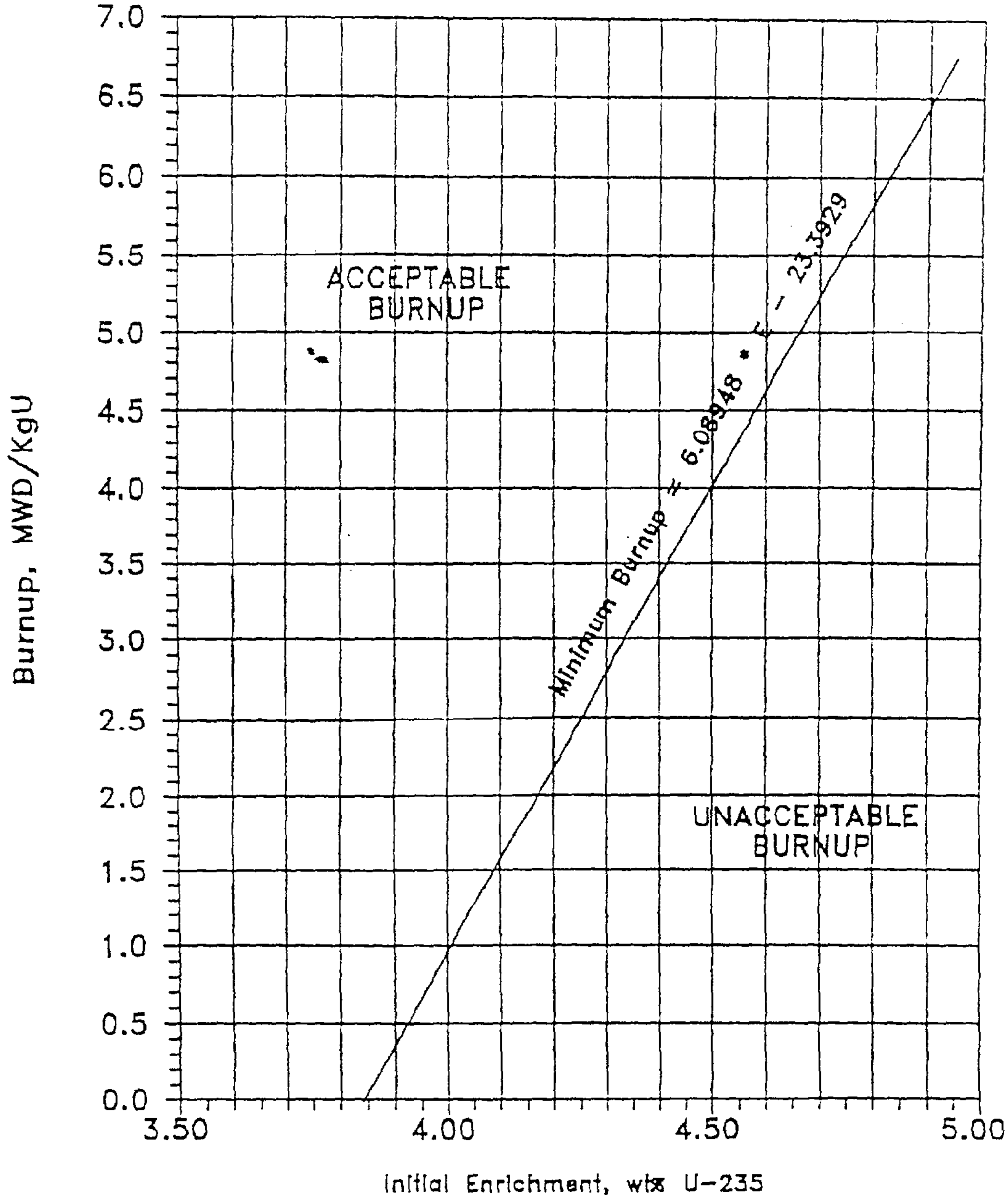

Figure 2 Minimum Burnup for Unrestricted Storage of Spent Fuel of Various Initial Enrichments (Arrangement 1)

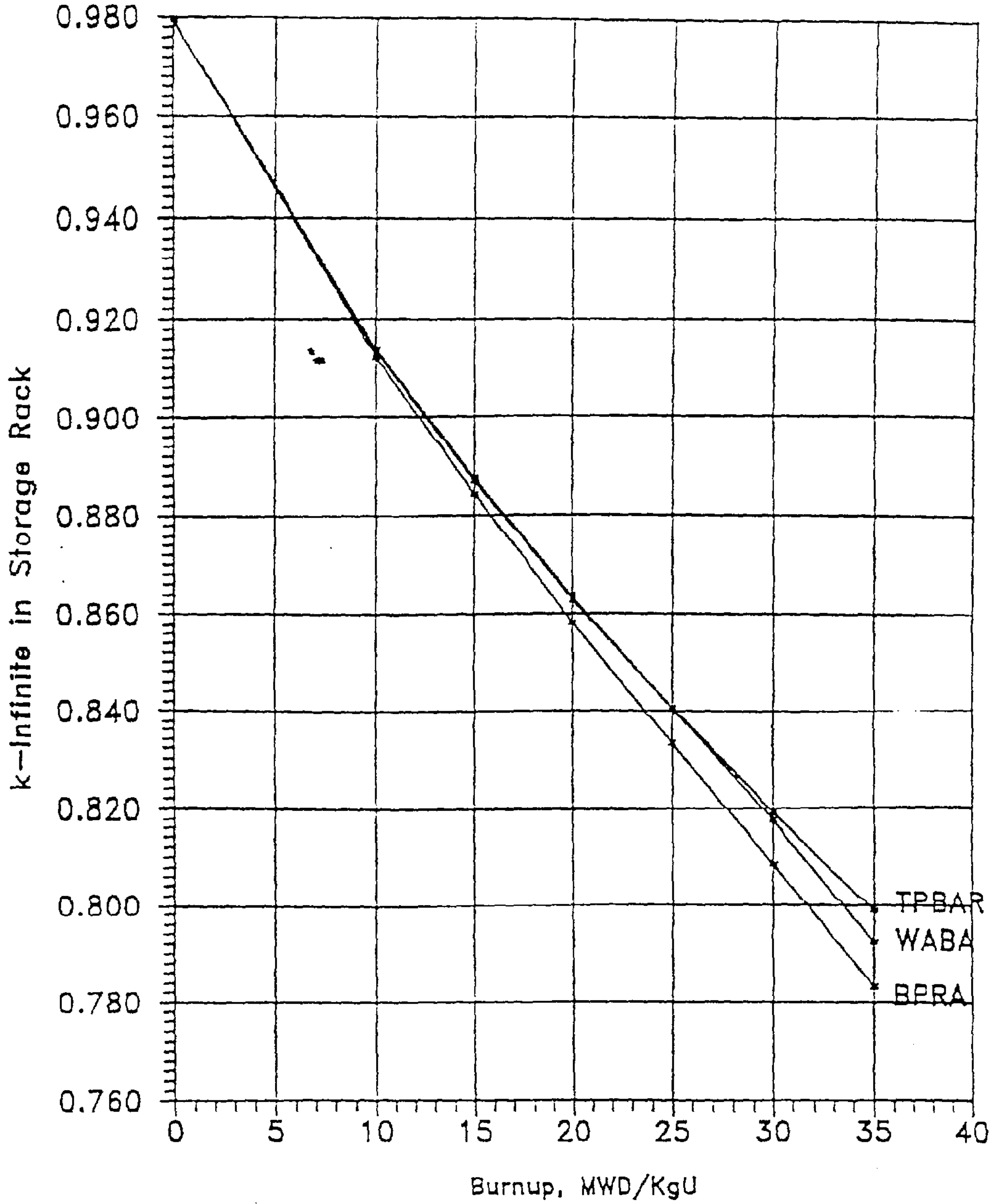

Fig. 3 Comparison of the Reactivity of Fuel Assemblies Depleted with Different Burnable Poison Rod Types

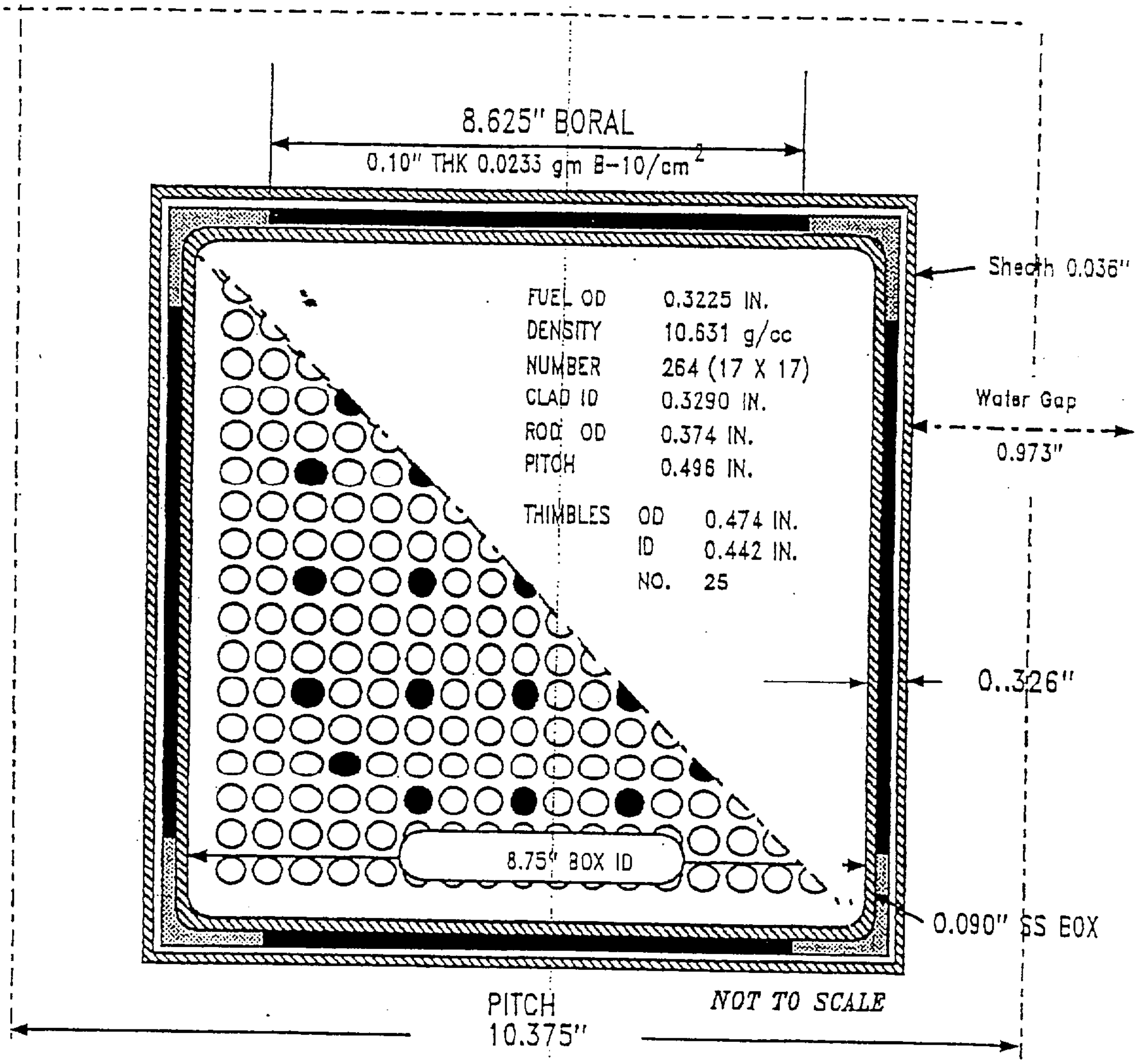

FIGURE 4 FUEL STORAGE CELL CROSS SECTION

APPENDIX A List of Holtec's QA Approved Computer Codes List

The list of Holtec's QA approved computer codes consists of all the codes that have been developed or verified by Holtec International for its use in nuclear safety-related applications. This information, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, analysis and licensing of a similar product. This list is, therefore, deemed proprietary and is not presented in this non-proprietary version of HI-2012620.

APPENDIX B: BENCHMARK CALCULATIONS

(Total of 26 Pages Including This Page)

Note: This appendix was taken from a different report. Hence, the next page is labeled "Appendix 4A, Page 1".

APPENDIX 4A: BENCHMARK CALCULATIONS

4A.1 INTRODUCTION AND SUMMARY

Benchmark calculations have been made on selected critical experiments, chosen, in so far as possible, to bound the range of variables in the rack designs. Two independent methods of analysis were used, differing in cross section libraries and in the treatment of the cross sections. MCNP4a [4A.1] is a continuous energy Monte Carlo code and KENO5a [4A.2] uses group-dependent cross sections. For the KENO5a analyses reported here, the 238-group library was chosen, processed through the NITAWL-II [4A.2] program to create a working library and to account for resonance self-shielding in uranium-238 (Nordheim integral treatment). The 238 group library was chosen to avoid or minimize the errors† (trends) that have been reported (e.g., [4A.3 through 4A.5]) for calculations with collapsed cross section sets.

In rack designs, the three most significant parameters affecting criticality are (1) the fuel enrichment, (2) the ^{10}B loading in the neutron absorber, and (3) the lattice spacing (or water-gap thickness if a flux-trap design is used). Other parameters, within the normal range of rack and fuel designs, have a smaller effect, but are also included in the analyses.

Table 4A.1 summarizes results of the benchmark calculations for all cases selected and analyzed, as referenced in the table. The effect of the major variables are discussed in subsequent sections below. It is important to note that there is obviously considerable overlap in parameters since it is not possible to vary a single parameter and maintain criticality; some other parameter or parameters must be concurrently varied to maintain criticality.

One possible way of representing the data is through a spectrum index that incorporates all of the variations in parameters. KENO5a computes and prints the "energy of the average lethargy causing fission" (EALF). In MCNP4a, by utilizing the tally option with the identical 238-group energy structure as in KENO5a, the number of fissions in each group may be collected and the EALF determined (post-processing).

---

† Small but observable trends (errors) have been reported for calculations with the 27-group and 44-group collapsed libraries. These errors are probably due to the use of a single collapsing spectrum when the spectrum should be different for the various cases analyzed, as evidenced by the spectrum indices.

Figures 4A.1 and 4A.2 show the calculated k_{eff} for the benchmark critical experiments as a function of the EALF for MCNP4a and KENO5a, respectively (UO_2 fuel only). The scatter in the data (even for comparatively minor variation in critical parameters) represents experimental error[†] in performing the critical experiments within each laboratory, as well as between the various testing laboratories. The B&W critical experiments show a larger experimental error than the PNL criticals. This would be expected since the B&W criticals encompass a greater range of critical parameters than the PNL criticals.

Linear regression analysis of the data in Figures 4A.1 and 4A.2 show that there are no trends, as evidenced by very low values of the correlation coefficient (0.13 for MCNP4a and 0.21 for KENO5a). The total bias (systematic error, or mean of the deviation from a k_{eff} of exactly 1.000) for the two methods of analysis are shown in the table below.

Calculational Bias of MCNP4a and KENO5a	
MCNP4a	0.0009±0.0011
KENO5a	0.0030±0.0012

The bias and standard error of the bias were derived directly from the calculated k_{eff} values in Table 4A.1 using the following equations[††], with the standard error multiplied by the one-sided K-factor for 95% probability at the 95% confidence level from NBS Handbook 91 [4A.18] (for the number of cases analyzed, the K-factor is ~2.05 or slightly more than 2).

$$\bar{k} = \frac{1}{n} \sum_{i}^{n} k_i \qquad (4A.1)$$

† A classical example of experimental error is the corrected enrichment in the PNL experiments, first as an addendum to the initial report and, secondly, by revised values in subsequent reports for the same fuel rods.

†† These equations may be found in any standard text on statistics, for example, reference [4A.6] (or the MCNP4a manual) and is the same methodology used in MCNP4a and in KENO5a.

$$\sigma^2_{\bar{k}} = \frac{\sum_{i=1}^{n} k_i^2 - (\sum_{i=1}^{n} k_i)^2 / n}{n\ (n-1)} \qquad (4A.2)$$

$$Bias = (1 - \bar{k}\ ) \pm K\ \sigma_{\bar{k}} \qquad (4A.3)$$

where k_i are the calculated reactivities of n critical experiments; $\sigma_{\bar{k}}$ is the unbiased estimator of the standard deviation of the mean (also called the standard error of the bias (mean)); K is the one-sided multiplier for 95% probability at the 95% confidence level (NBS Handbook 91 [4A.18]).*

Formula 4.A.3 is based on the methodology of the National Bureau of Standards (now NIST) and is used to calculate the values presented on page 4.A-2. The first portion of the equation, (1- $\bar{k}$), is the actual bias which is added to the MCNP4a and KENO5a results. The second term, $K\sigma_{\bar{k}}$, is the uncertainty or standard error associated with the bias. The K values used were obtained from the National Bureau of Standards Handbook 91 and are for one-sided statistical tolerance limits for 95% probability at the 95% confidence level. The actual K values for the 56 critical experiments evaluated with MCNP4a and the 53 critical experiments evaluated with KENO5a are 2.04 and 2.05, respectively.

The bias values are used to evaluate the maximum k_{eff} values for the rack designs. KENO5a has a slightly larger systematic error than MCNP4a, but both result in greater precision than published data [4A.3 through 4A.5] would indicate for collapsed cross section sets in KENO5a (SCALE) calculations.

4A.2 Effect of Enrichment

The benchmark critical experiments include those with enrichments ranging from 2.46 w/o to 5.74 w/o and therefore span the enrichment range for rack designs. Figures 4A.3 and 4A.4 show the calculated k_{eff} values (Table 4A.1) as a function of the fuel enrichment reported for the critical experiments. Linear regression analyses for these data confirms that there are no trends, as indicated by low values of the correlation coefficients (0.03 for MCNP4a and 0.38 for KENO5a). Thus, there are no corrections to the bias for the various enrichments.

As further confirmation of the absence of any trends with enrichment, a typical configuration was calculated with both MCNP4a and KENO5a for various enrichments. The cross-comparison of calculations with codes of comparable sophistication is suggested in Reg. Guide 3.41. Results of this comparison, shown in Table 4A.2 and Figure 4A.5, confirm no significant difference in the calculated values of k_{eff} for the two independent codes as evidenced by the 45° slope of the curve. Since it is very unlikely that two independent methods of analysis would be subject to the same error, this comparison is considered confirmation of the absence of an enrichment effect (trend) in the bias.

4A.3 Effect of ^{10}B Loading

Several laboratories have performed critical experiments with a variety of thin absorber panels similar to the Boral panels in the rack designs. Of these critical experiments, those performed by B&W are the most representative of the rack designs. PNL has also made some measurements with absorber plates, but, with one exception (a flux-trap experiment), the reactivity worth of the absorbers in the PNL tests is very low and any significant errors that might exist in the treatment of strong thin absorbers could not be revealed.

Table 4A.3 lists the subset of experiments using thin neutron absorbers (from Table 4A.1) and shows the reactivity worth (Δk) of the absorber.[†]

No trends with reactivity worth of the absorber are evident, although based on the calculations shown in Table 4A.3, some of the B&W critical experiments seem to have unusually large experimental errors. B&W made an effort to report some of their experimental errors. Other laboratories did not evaluate their experimental errors.

To further confirm the absence of a significant trend with ^{10}B concentration in the absorber, a cross-comparison was made with MCNP4a and KENO5a (as suggested in Reg. Guide 3.41). Results are shown in Figure 4A.6 and Table 4A.4 for a typical geometry. These data substantiate the absence of any error (trend) in either of the two codes for the conditions analyzed (data points fall on a 45° line, within an expected 95% probability limit).

---

† The reactivity worth of the absorber panels was determined by repeating the calculation with the absorber analytically removed and calculating the incremental (Δk) change in reactivity due to the absorber.

4A.4 Miscellaneous and Minor Parameters

4A.4.1 Reflector Material and Spacings

PNL has performed a number of critical experiments with thick steel and lead reflectors.[†] Analysis of these critical experiments are listed in Table 4A.5 (subset of data in Table 4A.1). There appears to be a small tendency toward overprediction of k_{eff} at the lower spacing, although there are an insufficient number of data points in each series to allow a quantitative determination of any trends. The tendency toward overprediction at close spacing means that the rack calculations may be slightly more conservative than otherwise.

4A.4.2 Fuel Pellet Diameter and Lattice Pitch

The critical experiments selected for analysis cover a range of fuel pellet diameters from 0.311 to 0.444 inches, and lattice spacings from 0.476 to 1.00 inches. In the rack designs, the fuel pellet diameters range from 0.303 to 0.3805 inches O.D. (0.496 to 0.580 inch lattice spacing) for PWR fuel and from 0.3224 to 0.494 inches O.D. (0.488 to 0.740 inch lattice spacing) for BWR fuel. Thus, the critical experiments analyzed provide a reasonable representation of power reactor fuel. Based on the data in Table 4A.1, there does not appear to be any observable trend with either fuel pellet diameter or lattice pitch, at least over the range of the critical experiments applicable to rack designs.

4A.4.3 Soluble Boron Concentration Effects

Various soluble boron concentrations were used in the B&W series of critical experiments and in one PNL experiment, with boron concentrations ranging up to 2550 ppm. Results of MCNP4a (and one KENO5a) calculations are shown in Table 4A.6. Analyses of the very high boron concentration experiments (> 1300 ppm) show a tendency to slightly overpredict reactivity for the three experiments exceeding 1300 ppm. In turn, this would suggest that the evaluation of the racks with higher soluble boron concentrations could be slightly conservative.

---

† Parallel experiments with a depleted uranium reflector were also performed but not included in the present analysis since they are not pertinent to the Holtec rack design.

4A.5 MOX Fuel

The number of critical experiments with PuO_2 bearing fuel (MOX) is more limited than for UO_2 fuel. However, a number of MOX critical experiments have been analyzed and the results are shown in Table 4A.7. Results of these analyses are generally above a k_{eff} of 1.00, indicating that when Pu is present, both MCNP4a and KENO5a overpredict the reactivity. This may indicate that calculation for MOX fuel will be expected to be conservative, especially with MCNP4a. It may be noted that for the larger lattice spacings, the KENO5a calculated reactivities are below 1.00, suggesting that a small trend may exist with KENO5a. It is also possible that the overprediction in k_{eff} for both codes may be due to a small inadequacy in the determination of the Pu-241 decay and Am-241 growth. This possibility is supported by the consistency in calculated k_{eff} over a wide range of the spectral index (energy of the average lethargy causing fission).

4A.6 References

[4A.1] J.F. Briesmeister, Ed., "MCNP4a - A General Monte Carlo N-Particle Transport Code, Version 4A; Los Alamos National Laboratory, LA-12625-M (1993).

[4A.2] SCALE 4.3, "A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", NUREG-0200 (ORNL-NUREG-CSD-2/U2/R5, Revision 5, Oak Ridge National Laboratory, September 1995.

[4A.3] M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-G Group ENDF/B-Y Cross-Section Library for Use in Criticality Safety Analyses", NUREG/CR-6102 (ORNL/TM-12460) Oak Ridge National Laboratory, September 1994.

[4A.4] W.C. Jordan et al., "Validation of KENOV.a", CSD/TM-238, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, December 1986.

[4A.5] O.W. Hermann et al., "Validation of the Scale System for PWR Spent Fuel Isotopic Composition Analysis", ORNL-TM-12667, Oak Ridge National Laboratory, undated.

[4A.6] R.J. Larsen and M.L. Marx, An Introduction to Mathematical Statistics and its Applications, Prentice-Hall, 1986.

[4A.7] M.N. Baldwin et al., Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel, BAW-1484-7, Babcock and Wilcox Company, July 1979.

[4A.8] G.S. Hoovier et al., Critical Experiments Supporting Underwater Storage of Tightly Packed Configurations of Spent Fuel Pins, BAW-1645-4, Babcock & Wilcox Company, November 1991.

[4A.9] L.W. Newman et al., Urania Gadolinia: Nuclear Model Development and Critical Experiment Benchmark, BAW-1810, Babcock and Wilcox Company, April 1984.

[4A.10] J.C. Manaranche et al., "Dissolution and Storage Experimental Program with 4.75 w/o Enriched Uranium-Oxide Rods," Trans. Am. Nucl. Soc. 33: 362-364 (1979).

[4A.11] S.R. Bierman and E.D. Clayton, Criticality Experiments with Subcritical Clusters of 2.35 w/o and 4.31 w/o ^{235}U Enriched UO_2 Rods in Water with Steel Reflecting Walls, PNL-3602, Battelle Pacific Northwest Laboratory, April 1981.

[4A.12] S.R. Bierman et al., Criticality Experiments with Subcritical Clusters of 2.35 w/o and 4.31 w/o ^{235}U Enriched UO_2 Rods in Water with Uranium or Lead Reflecting Walls, PNL-3926, Battelle Pacific Northwest Laboratory, December, 1981.

[4A.13] S.R. Bierman et al., Critical Separation Between Subcritical Clusters of 4.31 w/o ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons, PNL-2615, Battelle Pacific Northwest Laboratory, October 1977.

[4A.14] S.R. Bierman, Criticality Experiments with Neutron Flux Traps Containing Voids, PNL-7167, Battelle Pacific Northwest Laboratory, April 1990.

[4A.15] B.M. Durst et al., Critical Experiments with 4.31 wt % ^{235}U Enriched UO_2 Rods in Highly Borated Water Lattices, PNL-4267, Battelle Pacific Northwest Laboratory, August 1982.

[4A.16] S.R. Bierman, Criticality Experiments with Fast Test Reactor Fuel Pins in Organic Moderator, PNL-5803, Battelle Pacific Northwest Laboratory, December 1981.

[4A.17] E.G. Taylor et al., Saxton Plutonium Program Critical Experiments for the Saxton Partial Plutonium Core, WCAP-3385-54, Westinghouse Electric Corp., Atomic Power Division, December 1965.

[4A.18] M.G. Natrella, Experimental Statistics, National Bureau of Standards, Handbook 91, August 1963.

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF [1] (eV) MCNP4a	EALF [1] (eV) KENO5a
1	B&W-1484 (4A.7)	Core I	2.46	0.9964 ± 0.0010	0.9898 ± 0.0006	0.1759	0.1753
2	B&W-1484 (4A.7)	Core II	2.46	1.0008 ± 0.0011	1.0015 ± 0.0005	0.2553	0.2446
3	B&W-1484 (4A.7)	Core III	2.46	1.0010 ± 0.0012	1.0005 ± 0.0005	0.1999	0.1939
4	B&W-1484 (4A.7)	Core IX	2.46	0.9956 ± 0.0012	0.9901 ± 0.0006	0.1422	0.1426
5	B&W-1484 (4A.7)	Core X	2.46	0.9980 ± 0.0014	0.9922 ± 0.0006	0.1513	0.1499
6	B&W-1484 (4A.7)	Core XI	2.46	0.9978 ± 0.0012	1.0005 ± 0.0005	0.2031	0.1947
7	B&W-1484 (4A.7)	Core XII	2.46	0.9988 ± 0.0011	0.9978 ± 0.0006	0.1718	0.1662
8	B&W-1484 (4A.7)	Core XIII	2.46	1.0020 ± 0.0010	0.9952 ± 0.0006	0.1988	0.1965
9	B&W-1484 (4A.7)	Core XIV	2.46	0.9953 ± 0.0011	0.9928 ± 0.0006	0.2022	0.1986
10	B&W-1484 (4A.7)	Core XV ''	2.46	0.9910 ± 0.0011	0.9909 ± 0.0006	0.2092	0.2014
11	B&W-1484 (4A.7)	Core XVI ''	2.46	0.9935 ± 0.0010	0.9889 ± 0.0006	0.1757	0.1713
12	B&W-1484 (4A.7)	Core XVII	2.46	0.9962 ± 0.0012	0.9942 ± 0.0005	0.2083	0.2021
13	B&W-1484 (4A.7)	Core XVIII	2.46	1.0036 ± 0.0012	0.9931 ± 0.0006	0.1705	0.1708

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff}		EALF[1] (eV)	
				MCNP4a	KENO5a	MCNP4a	KENO5a
14	B&W-1484 (4A.7)	Core XIX	2.46	0.9961 ± 0.0012	0.9971 ± 0.0005	0.2103	0.2011
15	B&W-1484 (4A.7)	Core XX	2.46	1.0008 ± 0.0011	0.9932 ± 0.0006	0.1724	0.1701
16	B&W-1484 (4A.7)	Core XXI	2.46	0.9994 ± 0.0010	0.9918 ± 0.0006	0.1544	0.1536
17	B&W-1645 (4A.8)	S-type Fuel, w/886 ppm B	2.46	0.9970 ± 0.0010	0.9924 ± 0.0006	1.4475	1.4680
18	B&W-1645 (4A.8)	S-type Fuel, w/746 ppm B	2.46	0.9990 ± 0.0010	0.9913 ± 0.0006	1.5463	1.5660
19	B&W-1645 (4A.8)	SO-type Fuel, w/1156 ppm B	2.46	0.9972 ± 0.0009	0.9949 ± 0.0005	0.4241	0.4331
20	B&W-1810 (4A.9)	Case 1 1337 ppm B	2.46	1.0023 ± 0.0010	NC	0.1531	NC
21	B&W-1810 (4A.9)	Case 12 1899 ppm B	2.46/4.02	1.0060 ± 0.0009	NC	0.4493	NC
22	French (4A.10)	Water Moderator 0 gap	4.75	0.9966 ± 0.0013	NC	0.2172	NC
23	French (4A.10)	Water Moderator 2.5 cm gap	4.75	0.9952 ± 0.0012	NC	0.1778	NC
24	French (4A.10)	Water Moderator 5 cm gap	4.75	0.9943 ± 0.0010	NC	0.1677	NC
25	French (4A.10)	Water Moderator 10 cm gap	4.75	0.9979 ± 0.0010	NC	0.1736	NC
26	PNL-3602 (4A.11)	Steel Reflector, 0 separation	2.35	NC	1.0004 ± 0.0006	NC	0.1018

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF[1] (eV) MCNP4a	EALF[1] (eV) KENO5a
27	PNL-3602 (4A.11)	Steel Reflector, 1.321 cm sepn.	2.35	0.9980 ± 0.0009	0.9992 ± 0.0006	0.1000	0.0909
28	PNL-3602 (4A.11)	Steel Reflector, 2.616 cm sepn	2.35	0.9968 ± 0.0009	0.9964 ± 0.0006	0.0981	0.0975
29	PNL-3602 (4A.11)	Steel Reflector, 3.912 cm sepn.	2.35	0.9974 ± 0.0010	0.9980 ± 0.0006	0.0976	0.0970
30	PNL-3602 (4A.11)	Steel Reflector, infinite sepn.	2.35	0.9962 ± 0.0008	0.9939 ± 0.0006	0.0973	0.0968
31	PNL-3602 (4A.11)	Steel Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3282
32	PNL-3602 (4A.11)	Steel Reflector, 1.321 cm sepn.	4.306	0.9997 ± 0.0010	1.0012 ± 0.0007	0.3016	0.3039
33	PNL-3602 (4A.11)	Steel Reflector, 2.616 cm sepn.	4.306	0.9994 ± 0.0012	0.9974 ± 0.0007	0.2911	0.2927
34	PNL-3602 (4A.11)	Steel Reflector, 5.405 cm sepn.	4.306	0.9969 ± 0.0011	0.9951 ± 0.0007	0.2828	0.2860
35	PNL-3602 (4A.11)	Steel Reflector, Infinite sepn. "	4.306	0.9910 ± 0.0020	0.9947 ± 0.0007	0.2851	0.2864
36	PNL-3602 (4A.11)	Steel Reflector, with Boral Sheets	4.306	0.9941 ± 0.0011	0.9970 ± 0.0007	0.3135	0.3150
37	PNL-3926 (4A.12)	Lead Reflector, 0 cm sepn.	4.306	NC	1.0003 ± 0.0007	NC	0.3159
38	PNL-3926 (4A.12)	Lead Reflector, 0.55 cm sepn.	4.306	1.0025 ± 0.0011	0.9997 ± 0.0007	0.3030	0.3044
39	PNL-3926 (4A.12)	Lead Reflector, 1.956 cm sepn.	4.306	1.0000 ± 0.0012	0.9985 ± 0.0007	0.2883	0.2930

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF[1] (eV) MCNP4a	EALF[1] (eV) KENO5a
40	PNL-3926 (4A.12)	Lead Reflector, 5.405 cm sepn.	4.306	0.9971 ± 0.0012	0.9946 ± 0.0007	0.2831	0.2854
41	PNL-2615 (4A.13)	Experiment 004/032 - no absorber	4.306	0.9925 ± 0.0012	0.9950 ± 0.0007	0.1155	0.1159
42	PNL-2615 (4A.13)	Experiment 030 - Zr plates	4.306	NC	0.9971 ± 0.0007	NC	0.1154
43	PNL-2615 (4A.13)	Experiment 013 - Steel plates	4.306	NC	0.9965 ± 0.0007	NC	0.1164
44	PNL-2615 (4A.13)	Experiment 014 - Steel plates	4.306	NC	0.9972 ± 0.0007	NC	0.1164
45	PNL-2615 (4A.13)	Exp. 009 1.05% Boron-Steel plates	4.306	0.9982 ± 0.0010	0.9981 ± 0.0007	0.1172	0.1162
46	PNL-2615 (4A.13)	Exp. 012 1.62% Boron-Steel plates	4.306	0.9996 ± 0.0012	0.9982 ± 0.0007	0.1161	0.1173
47	PNL-2615 (4A.13)	Exp. 031 - Boral plates	4.306	0.9994 ± 0.0012	0.9969 ± 0.0007	0.1165	0.1171
48	PNL-7167 (4A.14)	Experiment 214R - with flux trap	4.306	0.9991 ± 0.0011	0.9956 ± 0.0007	0.3722	0.3812
49	PNL-7167 (4A.14)	Experiment 214V3 - with flux trap	4.306	0.9969 ± 0.0011	0.9963 ± 0.0007	0.3742	0.3826
50	PNL-4267 (4A.15)	Case 173 - 0 ppm B	4.306	0.9974 ± 0.0012	NC	0.2893	NC
51	PNL-4267 (4A.15)	Case 177 - 2550 ppm B	4.306	1.0057 ± 0.0010	NC	0.5509	NC
52	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 21	20% Pu	1.0041 ± 0.0011	1.0046 ± 0.0006	0.9171	0.8868

Table 4A.1

Summary of Criticality Benchmark Calculations

	Reference	Identification	Enrich.	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a	EALF[†] (eV) MCNP4a	EALF[†] (eV) KENO5a
53	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 43	20% Pu	1.0058 ± 0.0012	1.0036 ± 0.0006	0.2968	0.2944
54	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 13	20% Pu	1.0083 ± 0.0011	0.9989 ± 0.0006	0.1665	0.1706
55	PNL-5803 (4A.16)	MOX Fuel - Type 3.2 Exp. 32	20% Pu	1.0079 ± 0.0011	0.9966 ± 0.0006	0.1139	0.1165
56	WCAP-3385 (4A.17)	Saxton Case 52 PuO2 0.52" pitch	6.6% Pu	0.9996 ± 0.0011	1.0005 ± 0.0006	0.8665	0.8417
57	WCAP-3385 (4A.17)	Saxton Case 52 U 0.52" pitch	5.74	1.0000 ± 0.0010	0.9956 ± 0.0007	0.4476	0.4580
58	WCAP-3385 (4A.17)	Saxton Case 56 PuO2 0.56" pitch	6.6% Pu	1.0036 ± 0.0011	1.0047 ± 0.0006	0.5289	0.5197
59	WCAP-3385 (4A.17)	Saxton Case 56 borated PuO2	6.6% Pu	1.0008 ± 0.0010	NC	0.6389	NC
60	WCAP-3385 (4A.17)	Saxton Case 56 U 0.56" pitch	5.74	0.9994 ± 0.0011	0.9967 ± 0.0007	0.2923	0.2954
61	WCAP-3385 (4A.17)	Saxton Case 79 PuO2 0.79" pitch	6.6% Pu	1.0063 ± 0.0011	1.0133 ± 0.0006	0.1520	0.1555
62	WCAP-3385 (4A.17)	Saxton Case 79 U 0.79" pitch	5.74	1.0039 ± 0.0011	1.0008 ± 0.0006	0.1036	0.1047

Notes: NC stands for not calculated.

† EALF is the energy of the average lethargy causing fission.

†† These experimental results appear to be statistical outliers (>3σ) suggesting the possibility of unusually large experimental error. Although they could justifiably be excluded, for conservatism, they were retained in determining the calculational basis.

Table 4A.2

COMPARISON OF MCNP4a AND KENO5a CALCULATED REACTIVITIES[†]
FOR VARIOUS ENRICHMENTS

Enrichment	Calculated k_{eff} ± 1σ	
	MCNP4a	KENO5a
3.0	0.8465 ± 0.0011	0.8478 ± 0.0004
3.5	0.8820 ± 0.0011	0.8841 ± 0.0004
3.75	0.9019 ± 0.0011	0.8987 ± 0.0004
4.0	0.9132 ± 0.0010	0.9140 ± 0.0004
4.2	0.9276 ± 0.0011	0.9237 ± 0.0004
4.5	0.9400 ± 0.0011	0.9388 ± 0.0004

† Based on the GE 8x8R fuel assembly.

Table 4A.3

MCNP4a CALCULATED REACTIVITIES FOR CRITICAL EXPERIMENTS WITH NEUTRON ABSORBERS

Ref.	Experiment		Δk Worth of Absorber	MCNP4a Calculated k_{eff}	EALF* (eV)
4A.13	PNL-2615	Boral Sheet	0.0139	0.9994±0.0012	0.1165
4A.7	B&W-1484	Core XX	0.0165	1.0008±0.0011	0.1724
4A.13	PNL-2615	1.62% Boron-steel	0.0165	0.9996±0.0012	0.1161
4A.7	B&W-1484	Core XIX	0.0202	0.9961±0.0012	0.2103
4A.7	B&W-1484	Core XXI	0.0243	0.9994±0.0010	0.1544
4A.7	B&W-1484	Core XVII	0.0519	0.9962±0.0012	0.2083
4A.11	PNL-3602	Boral Sheet	0.0708	0.9941±0.0011	0.3135
4A.7	B&W-1484	Core XV	0.0786	0.9910±0.0011	0.2092
4A.7	B&W-1484	Core XVI	0.0845	0.9935±0.0010	0.1757
4A.7	B&W-1484	Core XIV	0.1575	0.9953±0.0011	0.2022
4A.7	B&W-1484	Core XIII	0.1738	1.0020±0.0011	0.1988
4A.14	PNL-7167	Expt 214R flux trap	0.1931	0.9991±0.0011	0.3722

*EALF is the energy of the average lethargy causing fission.

Table 4A.4

COMPARISON OF MCNP4a AND KENO5a
CALCULATED REACTIVITIES† FOR VARIOUS ^{10}B LOADINGS

^{10}B, g/cm^2	Calculated k_{eff} ± 1σ	
	MCNP4a	KENO5a
0.005	1.0381 ± 0.0012	1.0340 ± 0.0004
0.010	0.9960 ± 0.0010	0.9941 ± 0.0004
0.015	0.9727 ± 0.0009	0.9713 ± 0.0004
0.020	0.9541 ± 0.0012	0.9560 ± 0.0004
0.025	0.9433 ± 0.0011	0.9428 ± 0.0004
0.03	0.9325 ± 0.0011	0.9338 ± 0.0004
0.035	0.9234 ± 0.0011	0.9251 ± 0.0004
0.04	0.9173 ± 0.0011	0.9179 ± 0.0004

† Based on a 4.5% enriched GE 8x8R fuel assembly.

Table 4A.5

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH THICK LEAD AND STEEL REFLECTORS[†]

Ref.	Case	E, wt%	Separation, cm	MCNP4a k_{eff}	KENO5a k_{eff}
4A.11	Steel Reflector	2.35	1.321	0.9980±0.0009	0.9992±0.0006
		2.35	2.616	0.9968±0.0009	0.9964±0.0006
		2.35	3.912	0.9974±0.0010	0.9980±0.0006
		2.35	∞	0.9962±0.0008	0.9939±0.0006
4A.11	Steel Reflector	4.306	1.321	0.9997±0.0010	1.0012±0.0007
		4.306	2.616	0.9994±0.0012	0.9974±0.0007
		4.306	3.405	0.9969±0.0011	0.9951±0.0007
		4.306	∞	0.9910±0.0020	0.9947±0.0007
4A.12	Lead Reflector	4.306	0.55	1.0025±0.0011	0.9997±0.0007
		4.306	1.956	1.0000±0.0012	0.9985±0.0007
		4.306	5.405	0.9971±0.0012	0.9946±0.0007

† Arranged in order of increasing reflector-fuel spacing.

Table 4A.6

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH VARIOUS SOLUBLE BORON CONCENTRATIONS

Reference	Experiment	Boron Concentration, ppm	Calculated k_{eff} MCNP4a	Calculated k_{eff} KENO5a
4A.15	PNL-4267	0	0.9974 ± 0.0012	-
4A.8	B&W-1645	886	0.9970 ± 0.0010	0.9924 ± 0.0006
4A.9	B&W-1810	1337	1.0023 ± 0.0010	-
4A.9	B&W-1810	1899	1.0060 ± 0.0009	-
4A.15	PNL-4267	2550	1.0057 ± 0.0010	-

Table 4A.7

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH MOX FUEL

Reference	Case*	MCNP4a		KENO5a	
		k_{eff}	EALF**	k_{eff}	EALF**
PNL-5803 [4A.16]	MOX Fuel - Exp. No. 21	1.0041 ± 0.0011	0.9171	1.0046 ± 0.0006	0.8868
	MOX Fuel - Exp. No. 43	1.0058 ± 0.0012	0.2968	1.0036 ± 0.0006	0.2944
	MOX Fuel - Exp. No. 13	1.0083 ± 0.0011	0.1665	0.9989 ± 0.0006	0.1706
	MOX Fuel - Exp. No. 32	1.0079 ± 0.0011	0.1139	0.9966 ± 0.0006	0.1165
WCAP-3385-54 [4A.17]	Saxton @ 0.52" pitch	0.9996 ± 0.0011	0.8665	1.0005 ± 0.0006	0.8417
	Saxton @ 0.56" pitch	1.0036 ± 0.0011	0.5239	1.0047 ± 0.0006	0.5197
	Saxton @ 0.56" pitch borated	1.0008 ± 0.0010	0.6389	NC	NC
	Saxton @ 0.79" pitch	1.0063 ± 0.0011	0.1520	1.0133 ± 0.0006	0.1555

Note: NC stands for not calculated

* Arranged in order of increasing lattice spacing.

** EALF is the energy of the average lethargy causing fission.

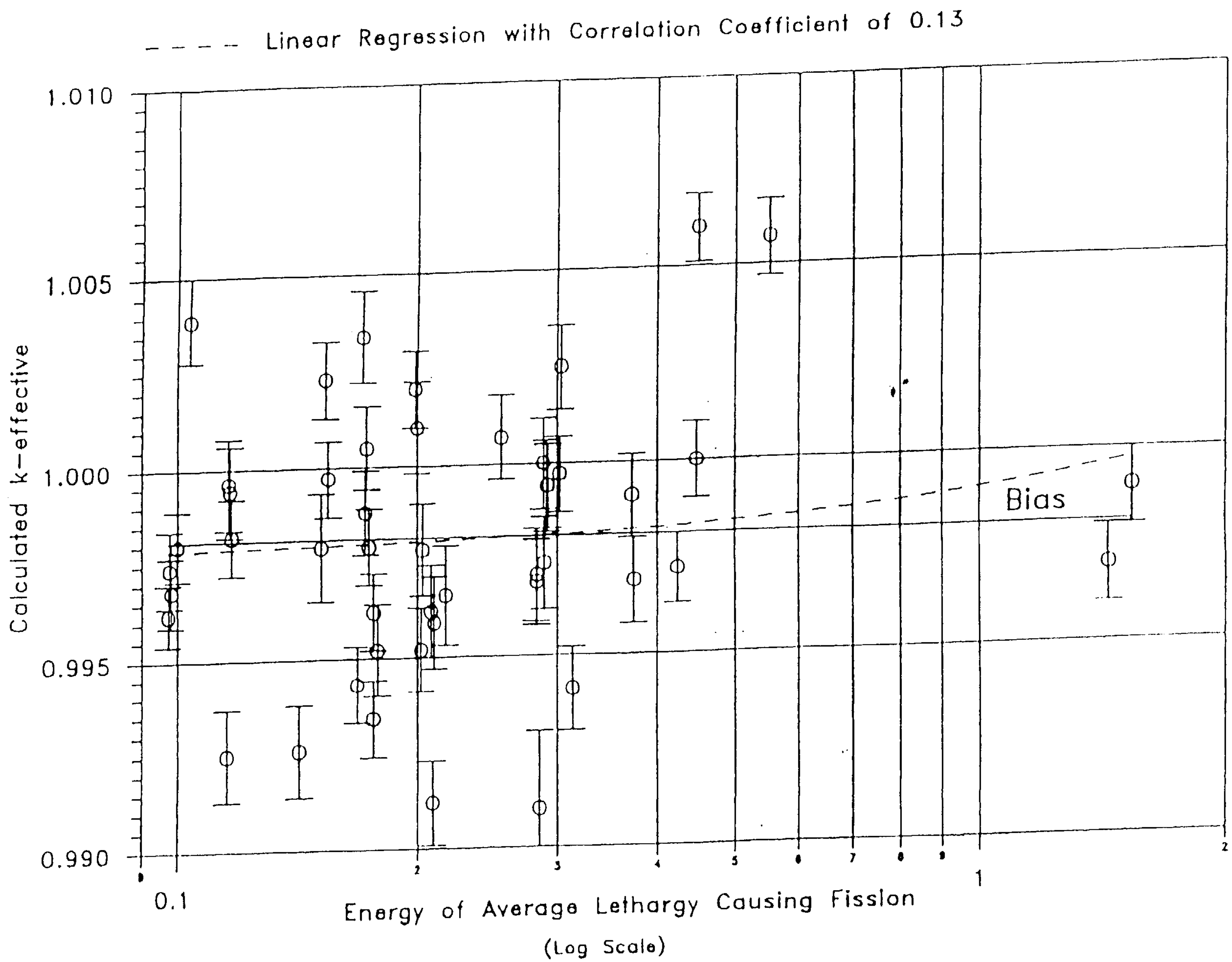

FIGURE 4A.1 MCNP CALCULATED k-eff VALUES for VARIOUS VALUES OF THE SPECTRAL INDEX

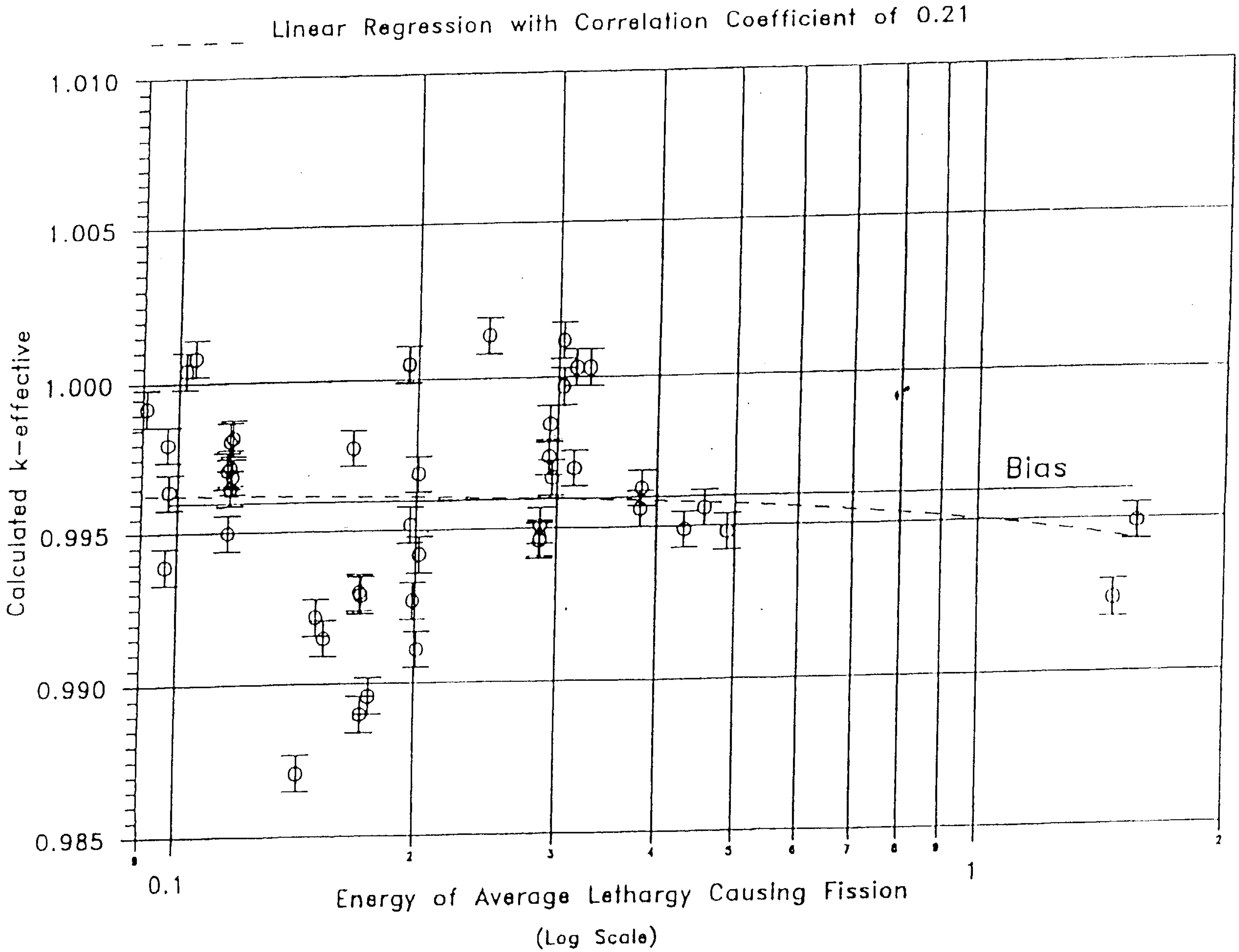

FIGURE 4A.2 KENO5a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

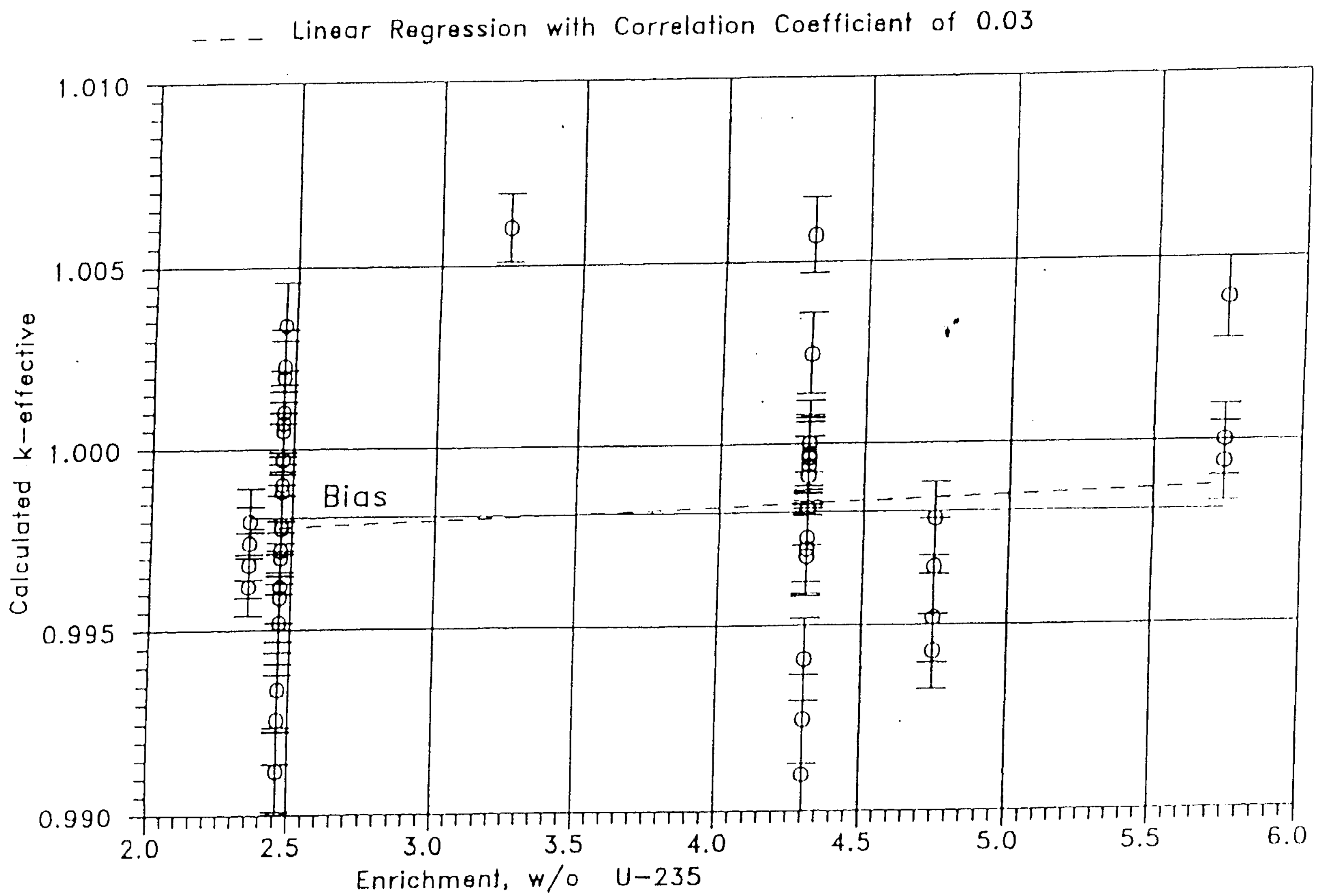

FIGURE 4A.3 MCNP CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

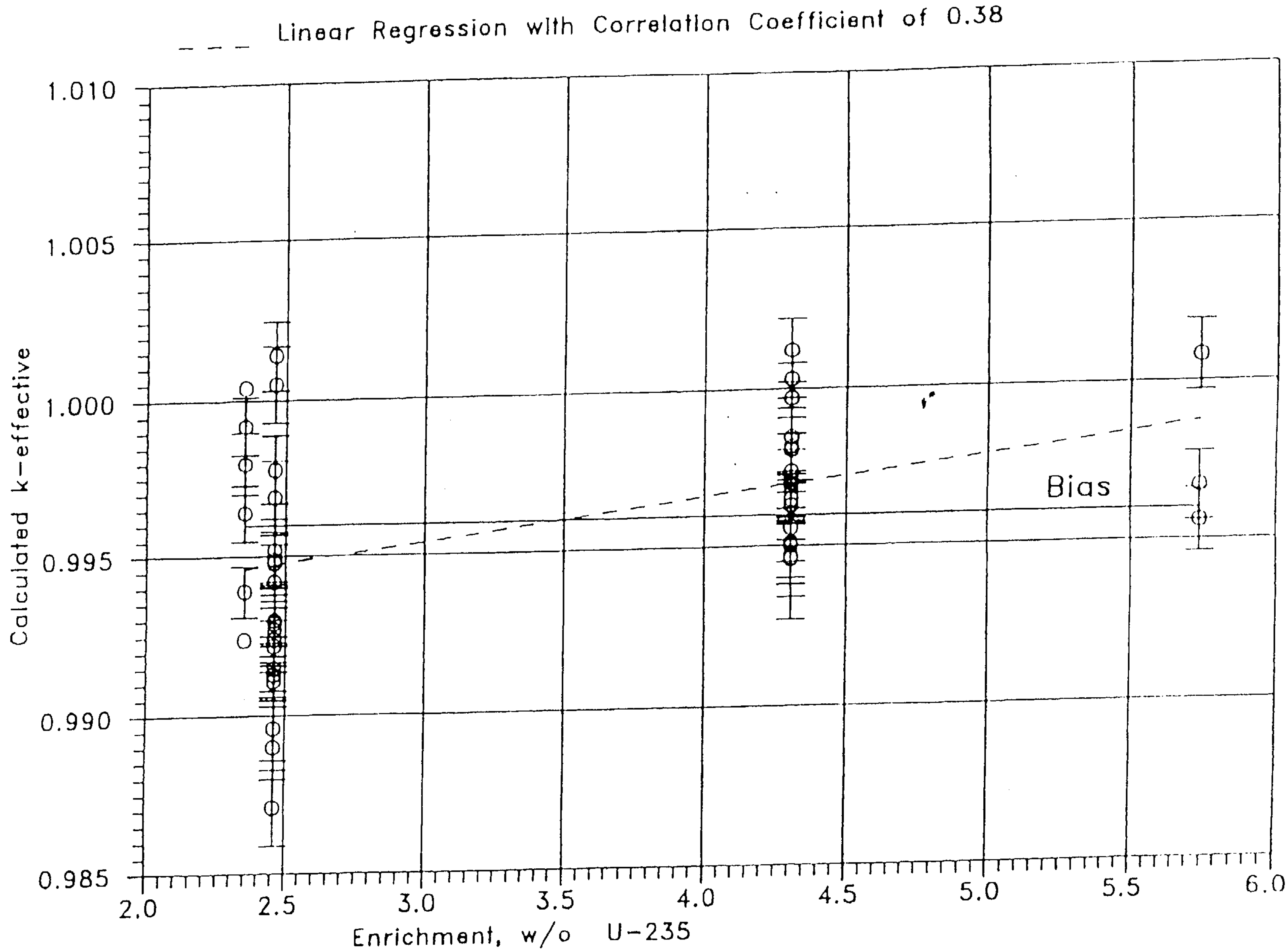

FIGURE 4A.4 KENO CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

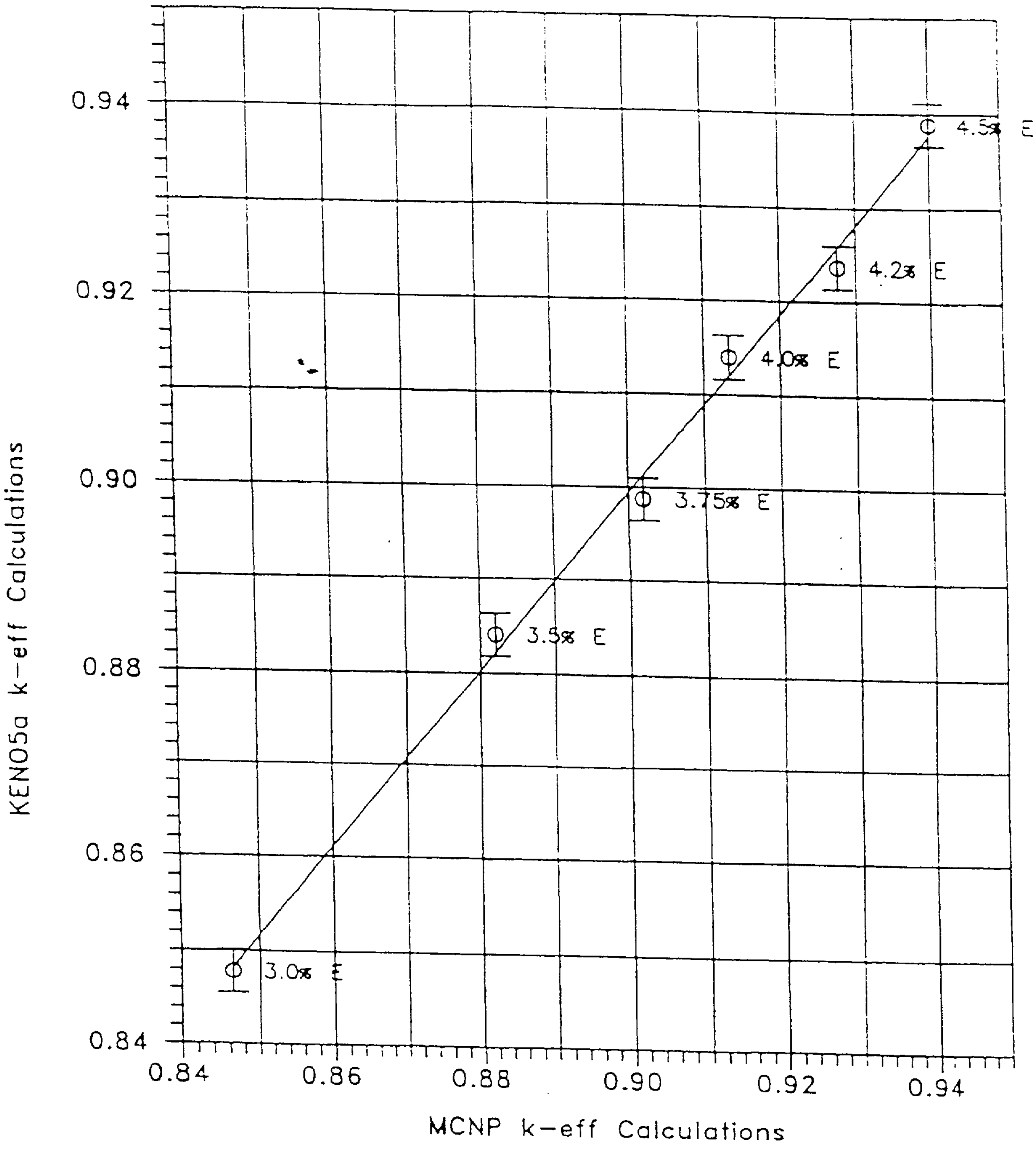

FIGURE 4A.5 COMPARISON OF MCNP AND KENO5A CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

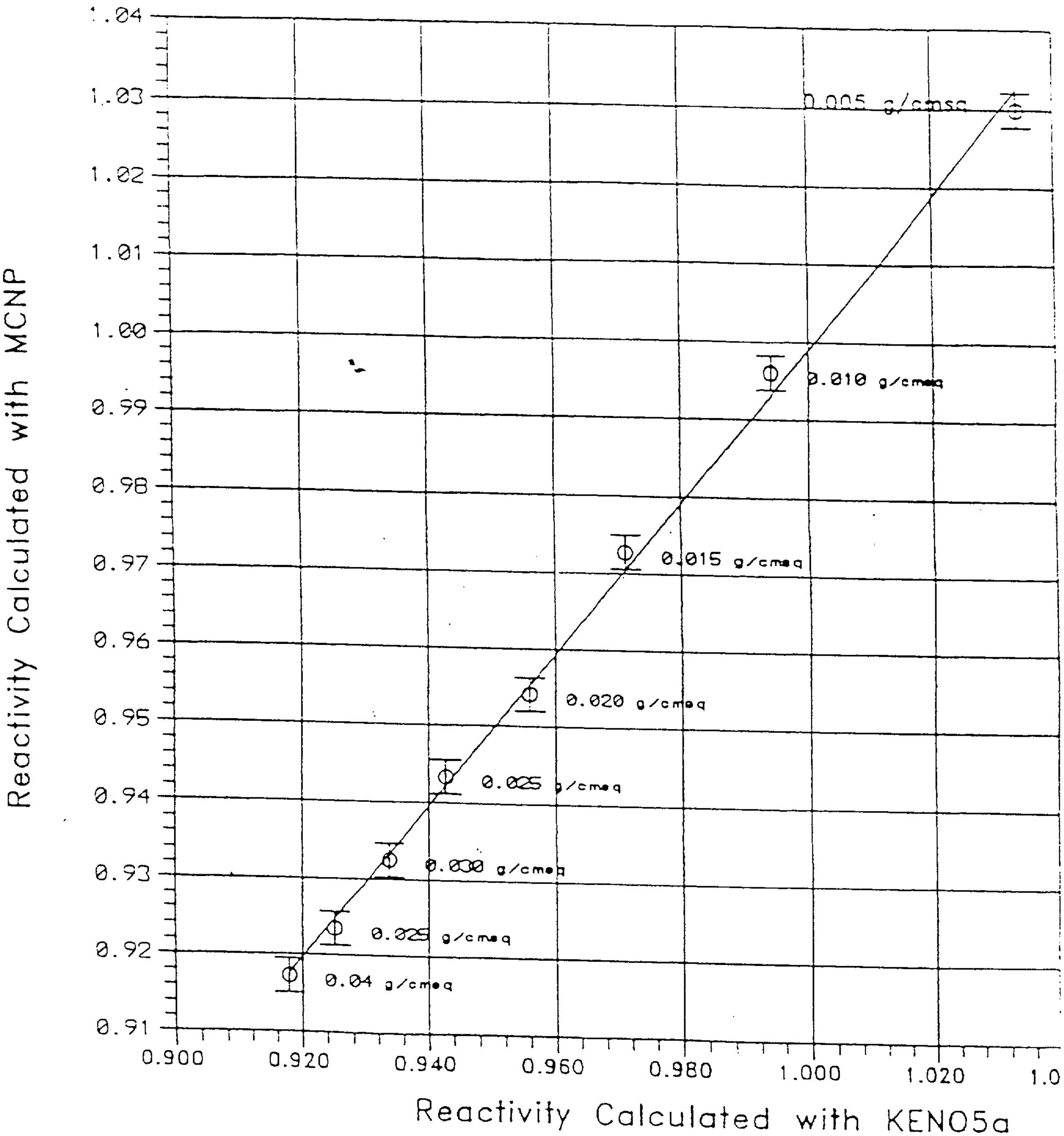

FIGURE 4A.6 COMPARISON OF MCNP AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES

APPENDIX C List of CASMO4 and KENO-Va Input Files

The list of computer files consists of those computer code input files that were used in the analysis and a brief description of each of the input files. This information provides details on the method of analysis and if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, analysis and licensing of a similar product. This list is, therefore, deemed proprietary and is not presented in this in this non-proprietary version of HI-2012620.