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September 25, 2008

AEP:NRC:8431
10 CFR 50.90

·U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Units 1 and 2
            Docket Nos. 50-315 and 50-316
            License Amendment Request to Modify Allowable Storage Patterns in Spent Fuel
            Storage Racks

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for
Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend the Appendix A Technical
Specifications (TS) to Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to modify
two TS figures showing allowable locations for nuclear fuel in the spent fuel pool storage racks.
These figures show two different allowable storage patterns for four of the storage rack modules.
I&M proposes to modify these two figures such that fuel may be located in any of these four
individual modules in accordance with either figure. The proposed amendment will allow continued
placement of new and intermediate burnup fuel in the spent fuel pool as the storage racks approach
a full condition.

Enclosure 1 to this letter provides an affirmation statement pertaining to the information contained
herein. Enclosure 2 provides I&M's evaluation of the proposed change. Enclosure 3 provides a
technical evaluation by Holtec International regarding the proposed change. Attachments 1 and 2
to this letter provide TS pages marked to show changes for Unit 1 and Unit 2, respectively.
Associated TS Bases changes will be made in accordance with the CNP Bases Control Program.

I&M requests approval of the proposed change by August 31, 2009, to support fuel movements for
the fall 2009 Unit 1 refueling outage. I&M will inform the Nuclear Regulatory Commission (NRC)
Licensing Project Manager if changes to the Unit 1 operating schedule affect the date by which
approval is requested. The proposed change will be implemented within 45 days of NRC approval.
Copies of this letter and its attachments are being transmitted to the Michigan Public Service
Commission and Michigan Department of Environmental Quality, in accordance with the
requirements of 10 CFR 50.91.

There are no new regulatory commitments made in this letter. Should you have any questions,
please contact Mr. John A. Zwolinski, Regulatory Affairs Manager, at (269) 466-2478.

Sincerely,

Lawrence J. Weber
Site Vice President

JRW/rdw

Enclosures:

    1.   Affirmation
    2.   Proposed License Amendment to Modify Allowable Storage Patterns in Four Spent
        Fuel Storage Modules
    3.   Holtec Technical Evaluation

Attachments:

    1.   Donald C. Cook Nuclear Plant Unit 1 Technical Specification Pages Marked To Show
        Changes
    2.   Donald C. Cook Nuclear Plant Unit 2 Technical Specification Pages Marked To Show
        Changes

c: T. A. Beltz, NRC Washington, DC
        J. L. Caldwell, NRC Region III
        K. D. Curry, Ft. Wayne AEP, w/o enclosures/attachments
        J. T. King, MPSC
        MDEQ – WHMD/RPS
        NRC Resident Inspector

AFFIRMATION

I, Lawrence J. Weber, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.
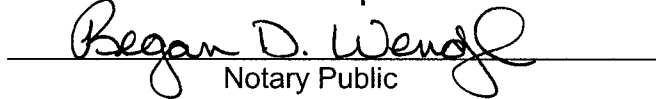
Indiana Michigan Power Company

Lawrence J. Weber

Lawrence J. Weber
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 25th DAY OF September , 2008

Regan D. Wendzel
Notary Public

My Commission Expires _____

REGAN D. WENDZEL
Notary Public, Berrien County, MI
My Commission Expires Jan. 21, 2009

**Proposed License Amendment to Modify Allowable Storage Patterns in
Four Spent Fuel Storage Modules**

Documents referenced in this enclosure are identified in Section 8.0.

1.0   DESCRIPTION

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for
Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend the Appendix A
Technical Specifications (TS) to Facility Operating Licenses DPR-58 and DPR-74. I&M
proposes to modify two TS figures showing allowable locations for nuclear fuel in the spent fuel
pool storage racks. These figures show two different allowable storage patterns for four of the
storage rack modules. I&M proposes to modify these two figures such that fuel may be located
in any of these four individual modules in accordance with either figure. The proposed
amendment will allow continued placement of new and intermediate burnup fuel in the spent
fuel pool as the storage racks approach a full condition.

2.0   PROPOSED CHANGE

I&M proposes to modify Figure 4.3-1 and Figure 4.3-2 in the Unit 1 and Unit 2 TS as follows:

- Add the alpha-numeric designations A1, C1, E1, and A5, to the four fuel modules that
  comprise the vertical row on the left side of the figure.
- Add a note indicating that the storage pattern for any of these individual modules, A1, C1,
  E1, or A5, may be in accordance with Figure 4.3-1 or Figure 4.3-2.
- Delete the value for the total number of cells in each of the three storage regions
  (Region 1, Region 2, and Region 3), and on Figure 4.3-2 only, delete the value for the
  total number of cells that are to be left empty.

Attachments 1 and 2 to this letter provide TS pages marked to show proposed changes for
Unit 1 and Unit 2, respectively. New text on these pages is indicated by an underline, and
deleted text is indicated by a strikethrough. New clean Unit 1 and Unit 2 TS pages with
proposed changes incorporated will be provided to the Nuclear Regulatory Commission (NRC)
Licensing Project Manager when requested. Associated TS Bases changes will be made in
accordance with the CNP Bases Control Program.

3.0   BACKGROUND

3.1   Spent Fuel Storage Rack Design and Operation

The current CNP spent fuel storage racks were installed in the early 1990s. The design and
operation of the storage racks were detailed in the 1991 license amendment request transmitted
by Reference 1. The associated amendment was approved by the NRC in 1993 as
documented by Reference 2. A summary of the storage rack design and operation is provided
below. Additional details are provided in Reference 1.

Irradiated fuel from Unit 1 and Unit 2 is stored in spent fuel storage racks located in a common spent fuel pool. New fuel is also stored in the spent fuel storage racks prior to being placed in the reactor vessel during refueling operations. The storage racks are comprised of twenty-three free-standing, self-supporting storage rack modules that rest on the floor of the spent fuel pool. Each module is divided into cells which hold individual fuel elements. The twenty-three modules contain a total of 3613 storage cells. There are several sizes of modules with varying numbers of cells. Within each module, the cells are grouped in parallel rows with a nominal distance between centers of 8.97 inches in both directions. The modules are constructed primarily of stainless steel. The only other material used in the modules is a boron carbide and aluminum neutron absorber with the product name "Boral." Each cell wall includes a Boral panel. The module baseplates extend outwards such that a two inch nominal gap is established between the adjacent walls of neighboring modules. Consequently, there are two neutron Boral absorber panels and a nominal two inch water flux trap between fuel assemblies located in peripheral cells of adjacent modules.

In establishing allowable fuel loading patterns for the modules, the cells are assigned to one of three regions or assigned as empty. Region 1 cells are designated for new fuel or irradiated fuel of any burnup. Region 2 cells are designated for high burnup fuel. Region 3 cells are designated for intermediate or high burnup fuel. In the configuration shown in TS Figure 4.3-1, the Region 1 cells are located in alternate cells along the outside edges of the outside modules (where neutron leakage reduces reactivity), and alternate cells along the edges between two modules (where the water gap provides a flux-trap which reduces reactivity). High burnup fuel in Region 2 provides a low-reactivity barrier between new fuel assemblies and fuel of intermediate burnup in Region 3. The configuration shown in TS Figure 4.3-1 is designated as the "Normal Storage Pattern."

As shown in TS Figure 4.3-2, another storage pattern, termed the "Interim Storage Pattern," is also permitted. This pattern allows the central cells in four modules to be loaded in a checkerboard pattern of Region 1 cells and empty cells. The modules in which the checkerboard pattern is allowed are identified as modules A1, C1, E1, and A5 in Reference 1 and Figure 9.7-2 of the Updated Final Safety Analysis Report (UFSAR). The checkerboard pattern provides more Region 1 cells than the pattern shown in TS Figure 4.3-1. The checkerboard pattern was intended primarily to facilitate a temporary full core off-load and storage of new fuel when needed for refueling outages prior to the time that the central cells in modules A1, C1, E1, or A5 are beginning to fill with intermediate or high burnup fuel.

3.2   Current Licensing Basis Criticality Analyses

The amendment request transmitted by Reference 1 addressed the applicable design considerations for the spent fuel storage modules. The principal design consideration of interest with respect to the amendment proposed by this letter is criticality control. A summary of the criticality analyses detailed in Reference 1 is provided below.

Conditions Analyzed

Criticality analyses of two conditions were performed. Analyses were performed for normal operating conditions in which fuel is located as shown in TS Figure 4.3-1 or Figure 4.3-2.

Analyses were also performed for abnormal or accident conditions in which a fuel assembly is placed in a location not in accordance with TS Figure 4.3-1.

Analysis Assumptions

Except as noted, the following conservative assumptions were applied to both the normal and the abnormal or accident condition analyses to assure that the actual reactivity remains lower than calculated reactivity:

- A room temperature (20 degrees centigrade) moderator was assumed because it yields the highest reactivity within the operating temperature range.
- No poison (soluble boron) in the spent fuel pool water was assumed for the normal criticality analysis.
- Neutron leakage was only assumed for module boundary storage cells.
- Neutron absorption in minor structural members was disregarded.

Additional margins of conservatism resulted from the time that irradiated fuel is in the spent fuel pool which leads to a decrease in reactivity, and the negative moderator coefficient of reactivity inherent in the pool water. These margins insure the actual reactivity of the TS Figure 4.3-1 and Figure 4.3-2 configurations will remain less than the maximum calculated reactivity (k_{eff}) of 0.94, which in turn satisfies the TS requirement that k_{eff} be maintained ≤ 0.95.

The analyses were performed by modeling the standard Westinghouse 15 x 15 fuel assemblies containing UO_2 with a maximum enrichment of 4.95 ± 0.05 weight percent U-235 used in Unit 1. The fuel assemblies were assumed to be uniform, having the high reactivity characteristic to the central enriched zone. This model yields higher reactivity values than those for standard and optimized 17 x 17 Westinghouse and Advanced Nuclear Fuel (ANF) designs used in Unit 2.

Methodology

The KENO-5a Monte Carlo computer package was used in conjunction with the 27-group SCALE cross-section library and the NITAWL subroutine for U-238 resonance shielding effects. A total of 1,250,000 histories were run to guarantee calculation convergence, thus, minimizing statistical variances. The two-dimensional transport theory code CASMO-3 was used to determine equivalent enrichments, as burnup capability was not available.

Normal Operating Condition Analyses

The criticality calculations for TS Figure 4.3-1 or Figure 4.3-2 configurations yielded a maximum reactivity of 0.940. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations including mechanical tolerances. All uncertainties are statistically combined, such that the final k_{eff} will be equal to or less than 0.95 with a 95% probability at a 95% confidence level. This satisfies the TS 4.3.1.1.b requirement that k_{eff} be maintained ≤ 0.95.

Abnormal or Accident Condition Analyses

The assumed abnormal or accident condition for the TS Figure 4.3-1 storage pattern consisted of placement of a new fuel assembly in a Region 2 position where the remainder of the rack is filled with fuel of the highest permissible reactivity. This condition could result in a reactivity increase of +0.065 δk. The calculation determined that a spent fuel pool boron concentration of 450 parts per million (ppm) would be sufficient to provide a k_{eff} of 0.95. A boron concentration well in excess of this value is assured by TS 3.7.15 which requires that the spent fuel pool boron concentration be $\geq$2400 ppm when fuel assemblies are stored in the pool and a storage pool verification has not been performed since the last movement of fuel assemblies in the pool.

3.3   Reason for Requesting Amendment

The cells in the center of modules A1, C1, E1, and A5 are normally kept empty so they can be loaded in the interim checkerboard pattern shown in TS Figure 4.3-2 to support refueling outages. However, the cells in modules other than A1, C1, E1, and A5 are approaching a full condition. It will then be necessary to load irradiated fuel into modules A1, C1, E1, and A5 in accordance with the normal pattern shown in TS Figure 4.3-1. Once the normal pattern is used in one of the four modules (A1, C1, E1, and A5), the current TS Figure 4.3-1 and Figure 4.3-2 would preclude use of the interim checkerboard pattern in the other three modules.

Without the allowance to use the interim pattern in the other three modules, it will be necessary to rearrange fuel in modules other than A1, C1, E1, and A5 to accomplish upcoming refueling outages while maintaining compliance with requirements identified in Section B.5.b of NRC Order EA-02-026. This would significantly increase the number and distance of fuel moves needed to support the outages. Eventually, a condition will be reached in which there will not be enough Region 1 cells available to accommodate the new fuel and low burnup fuel for a core reload. Therefore, the allowance to use the interim checkerboard pattern in some of the four modules and the normal pattern in others will minimize the number and distance of fuel moves for upcoming outages, and will ultimately allow continued plant operation until fuel is relocated to dry cask storage facilities (expected to begin in mid-2011).

Accordingly, I&M is requesting a TS change that would allow fuel to be located in any of these four individual modules (A1, C1, E1, or A5) in accordance with either the normal pattern of Figure 4.3-1 or the interim checkerboard pattern of TS Figure 4.3-2.

4.0   TECHNICAL ANALYSIS

4.1   Design Considerations

To support the TS change proposed by this amendment request, I&M contracted Holtec International to evaluate the effect of the proposed changes on the design considerations addressed in Reference 1. The Holtec technical evaluation is provided as Enclosure 3 to this letter. The Holtec technical evaluation determined that compliance with the design criteria addressed in Reference 1 will not be impacted by the proposed change. I&M has verified that there have been no changes to the CNP design and licensing basis subsequent to NRC approval of Reference 1 that would alter the conclusions documented in the technical

evaluation. Note that the Holtec technical evaluation uses the term "fresh fuel," while the TS use the term "new fuel." Both terms refer to fuel that has not been subjected to a neutron flux in an operating reactor.

In addition to evaluating the effect of the proposed change on the design considerations addressed in Reference 1, Holtec performed an analysis of an abnormal or accident case in which a fuel assembly is placed in a location not in accordance with the interim checkerboard pattern shown in TS Figure 4.3-2. As described in Section 4.1 of Enclosure 3, the Holtec analysis used similar assumptions to the analysis documented in Reference 1, but used a newer computer code, MCNP4a, rather than the KENO-5a code. Use of the MCNP4a code has been previously approved by the NRC as described in Section 7.0 of this enclosure. Section 4.1 of Enclosure 3 documents that the maximum k_{eff} resulting from placement of a fuel assembly in a location not in accordance with the interim checkerboard pattern shown in TS Figure 4.3-2 would be significantly less than 0.95 with the boron concentration required during fuel movement by the TS.

4.2   Specific TS Changes

The reasons for the specific changes to TS Figure 4.3-1 and TS Figure 4.3-2 are described below.

Addition of alpha-numeric designations A1, C1, E1, and A5

The alpha-numeric designations were added to differentiate the modules involved in the proposed change (those on the left side of the figure) from the remainder of the modules. The alpha-numeric designation for each of the four modules corresponds to that shown in Figure 2.1.1 of Reference 1 and Figure 9.7-2 of the UFSAR.

Addition of notes indicating that the storage pattern for any of the individual modules (A1, C1, E1, or A5) may be in accordance with Figure 4.3-1 or Figure 4.3-2.

The notes were added to accomplish the fundamental objective of the proposed amendment, i.e., to allow any of the specified modules to be loaded in accordance with either the normal pattern (Figure 4.3-1) or the interim checkerboard pattern (Figure 4.3-2).

Deletion of the value for the total number of cells in Region 1, Region 2, Region 3, and for Figure 4.3-2 only, deletion of the value for the total number of empty cells.

The values for the total number of cells in each category were deleted because the values can change depending on the loading pattern used for each of the individual modules A1, C1, E1, and A5. The analyses discussed in the Holtec technical evaluation provided as Enclosure 3 to this letter demonstrate that compliance with the storage patterns specified in TS Figure 4.3-1 and Figure 4.3-2 in accordance with this proposed amendment will assure safety. Therefore, the specific values for the total number of cells in each category are not important to safety, and the inclusion of these values in TS is not required by 10 CFR 50.36(d)(4).

5.0   REGULATORY SAFETY ANALYSIS

5.1   No Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend the Appendix A Technical Specifications (TS) to Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to modify two TS figures showing allowable locations for nuclear fuel in the spent fuel pool storage racks. These figures show two different allowable storage patterns for four of the storage rack modules. I&M proposes to modify these two figures such that fuel may be located in any of the four individual modules in accordance with either figure. I&M has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

    Response: No

    The accidents and events of concern involving fuel located in the spent fuel pool storage racks are a criticality accident, a fuel handling accident, and inadequate decay heat removal. The proposed change will not increase the probability of a criticality accident because analyses demonstrate that sub-criticality will be maintained for the fuel storage configurations allowed by the change. The proposed change will not increase the probability of a fuel handling accident because it does not affect the manner in which fuel is moved or handled. The proposed change will decrease the number of fuel moves needed for upcoming refueling outages. The proposed change will not increase the probability of inadequate decay heat removal because thermal-hydraulic analyses demonstrating adequate heat removal will remain valid for the storage configurations allowed by the change. Therefore, the probability of occurrence of a previously evaluated accident will not be significantly increased.

    The proposed change does not adversely affect the ability to perform the intended safety functions of any system, structure, or component (SSC) credited for mitigating a criticality accident, a fuel handling accident, or inadequate decay heat removal. Therefore, the consequences of a previously evaluated accident will not be significantly increased.

    Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

    Response: No

The proposed change does not alter the design function or operation of any SSC. The proposed change does not affect the capability of the SSCs involved with the storage of fuel in the spent fuel pool to perform their function. As a result, no new failure mechanisms, malfunctions, or accident initiators are created. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The margins of safety involved with the storage of fuel in the spent fuel pool are the margins associated with the prevention of criticality, mitigation of a fuel handling accident, and assurance of adequate decay heat removal. The proposed amendment involves no change in the capability of any SSC that maintains these margins. Therefore, there is no significant reduction in a margin of safety as a result of the proposed amendment.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, I&M concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2   Applicable Regulatory Requirements/Criteria

As described below, compliance with the applicable regulatory requirements and criteria is not affected by the proposed amendment.

Code of Federal Regulations

10 CFR 50.36 requires that the TS for a facility include a description of design features, such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety, and are not covered by Safety Limits, Limiting Safety System Settings, Limiting Control Settings, Limiting Conditions for Operation, or Surveillance Requirements. TS Figures 4.3-1 and Figure 4.3-2 will continue to satisfy this criterion in that the fuel storage patterns allowed by the figures will assure safety with respect to the licensing basis design considerations.

10 CFR 70.24 requires that licensees authorized to possess specified quantities and forms of special nuclear material maintain provisions for monitoring criticality and mitigating the consequences of criticality accidents. By Reference 3, the Nuclear Regulatory Commission (NRC) granted I&M an exemption from these requirements for CNP. The basis for the exemption was that inadvertent or accidental criticality will be precluded through compliance with the TS, the geometric spacing of fuel assemblies in the new fuel storage facility and spent fuel storage pool, and administrative controls imposed on fuel handling procedures. The basis for this exemption remains valid in that geometric spacing of fuel assemblies will continue to maintain k_{eff} less than the TS limit of 0.95. Additionally, appropriate administrative controls on fuel handling will be maintained.

Plant Specific Design Criteria (PSDC)

The CNP PSDC are described in Section 1.4 of the UFSAR. These criteria differ from the criteria stated in Appendix A of 10 CFR 50 which were published in 1971 after the CNP construction permits were issued. CNP PSDC Criterion 66, "Prevention of Fuel Storage Criticality," requires that criticality in the spent fuel pool be prevented by physical systems or processes. Compliance with this criterion will be unaffected by the proposed change in that, as described in Enclosure 3, criticality will be prevented by the allowed storage configurations.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health or safety of the public.

6.0   ENVIRONMENTAL CONSIDERATIONS

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0   PRECEDENT

The proposed change is specific to the configuration of the CNP spent fuel pool, the CNP storage rack arrangement, and the CNP allowable fuel storage patterns. Consequently, there are no apparent precedent license amendments that allow a change identical to that proposed for CNP. However, a license amendment for Crystal River Nuclear Plant was approved by the NRC (Reference 4) based on use of the MCNP4a three-dimensional Monte Carlo code. The MCNP4a code was used for the analysis of a postulated misloading accident in the interim storage configuration at CNP as described in Enclosure 3.

8.0   REFERENCES

1. Letter from E. E. Fitzpatrick, I&M, to NRC Document Control Desk, "Spent Fuel Pool Reracking Technical Specification Changes," dated July 26, 1991.

2. Letter from W. M. Dean, NRC, to E. E. Fitzpatrick, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Amendment Nos. 169 And 152 to Facility Operating License Nos. DPR-58 and DPR-74 (TAC Nos. M80615 and M80616)," dated January 14, 1993 (ML021060153).

3. Letter from J. B. Hickman, NRC, to E. E. Fitzpatrick, I&M, "Exemption from 10 CFR 70.24 Criticality Monitoring Requirements – Donald C. Cook Nuclear Plant, Units 1 and Unit 2 (TAC Nos. M95197 and M95198)," dated October 28, 1996 (ML021070041).

4. Letter from S. N. Bailey, NRC, to D. E. Young, Crystal River Nuclear Plant, "Crystal River, Unit 3 - Issuance of Amendment Regarding Fuel Storage Patterns in the Spent Fuel Pool (TAC No. MD3308)," dated October 25, 2007 (ML072910317).

HOLTEC TECHNICAL EVALUATION

1.0   Introduction

This position paper documents the acceptability of a proposed configuration change for the D.C. Cook spent fuel pool. Each of the major technical disciplines as documented in Holtec Report HI-90488, "Licensing Report for Storage of Densification of D.C. Cook Spent Fuel Pool" [1] are addressed in the following subsections.

2.0   Background

The D.C. Cook spent fuel pool was re-racked in the early 1990's with high density spent fuel storage racks to achieve the maximum spent fuel capacity possible as shown in Figure 1. All racks installed were non-flux-trap style racks, with a single neutron absorber panel (Boral) between each fuel assembly. The racks were approved by the USNRC to accommodate a mixed zone three region (MZTR) storage scheme where fresh fuel is interspersed with high burnup fuel on the edge of the rack, with moderate burnup fuel stored in the center of each rack. Figure 2 (Figure 4-1 from [1]) shows the normal storage pattern for the spent fuel pool with the following fuel stored in each of the 3 regions:

- Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 ± 0.05 wt% ^{235}U, or spent fuel regardless of the discharge fuel burnup.
- Region 2 is designed to accommodate fuel of 4.95 wt% initial enrichment burned to at least 50,000 MWD/MTU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity.
- Region 3 is designed to accommodate fuel of 4.95 wt% initial enrichment burned to at least 38,000 MWD/MTU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity.

Additionally, an interim storage pattern was licensed to accommodate a full-core offload of low-burned fuel where the center region of four racks (A1, C1, E1 and A5) in the pool are replaced with a checkerboard of fresh fuel and empty storage cells as shown in Figure 3 (Figure 4-2 from [1]).

The following subsections provide technical justification for allowing any combination of the four rack modules to contain either fuel assemblies meeting the Region 3 criteria or a checkerboard of fresh fuel and empty storage cells in the central region of these modules. The numbering of the following subsections matches the section numbering in [1].

3.0   Construction of Rack Module

No changes are proposed to the construction of the rack modules, only changes to the acceptable loading configurations in the spent fuel pool. Therefore, the discussion in [1] on construction of the rack modules is not affected.

4.0   Criticality Safety Analysis

The criticality analysis presented in [1] addresses both the normal storage configuration and the interim storage configuration separately. The MZTR concept is based on neutronically isolating fresh fuel assemblies by surrounding them with high burned fuel, while placing moderately burned fuel in the center of the rack. The storage cells in each rack module are of the non-flux trap type, with a single sheet of neutron absorber between adjacent assemblies. Each rack module is separated from adjacent rack modules by a nominal spacing of 2 inches with neutron absorber on the outside surfaces of the rack module, effectively creating a "flux-trap[1]" between rack modules. Additionally, fuel assemblies in the center region (Region 3) of adjacent racks are separated by four rows of fuel and the flux-trap between rack modules, ensuring that the central regions of adjacent racks are neutronically decoupled.

KENO5a models were created which represent a single rack containing the MZTR loading pattern. These models represent an 11x11 storage rack, as shown in Figure 4. Reflecting boundary conditions were used on all sides of the model, effectively creating an infinite array of 11x11 storage racks. Additional models for the interim storage configuration were created with the same geometry, except the Region 3 fuel assemblies were replaced with a checkerboard of Region 1 fuel assemblies (fresh fuel) and empty storage locations. The results show that the maximum k_{eff} of the interim and normal storage configurations are statistically equivalent (within 2σ) and therefore either storage configuration is acceptable. Given that the analysis demonstrates that either of the two configurations is acceptable for a single rack module, any combination of the four storage rack modules with the center region containing either Region 3 fuel assemblies or a checkerboard of fresh fuel and empty storage cells is acceptable.

The previous analysis made use of reactivity equivalencing, which was the subject of an NRC issue summary 2001-12 [2]. The reactivity equivalencing method used in the analysis was reviewed and determined that the equivalent enrichments were calculated in *the storage cell configuration* and therefore are acceptable. For accident and abnormal conditions, 450 ppm of soluble boron in the spent fuel pool was credited to ensure that the maximum k_{eff} would remain less than or equal to 0.95. Given that the spent fuel pool has a Technical Specification 3.7.15

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[1] A flux-trap construction implies that there is a water gap between adjacent storage cells such that the neutrons emanating from a fuel assembly are thermalized before reaching an adjacent assembly (i.e., in the water gap between assemblies).

requirement that the fuel storage pool concentration be $\geq$ 2400ppm during fuel movement, the reactivity in the event of an accident or abnormal condition will be much lower than 0.95.

Based on the discussion above and the fact that the previous analysis was sufficient to bound the proposed configuration, no new analysis was performed to justify the proposed configuration. However, a single calculation was performed to address an accident condition not analyzed when the racks were first licensed. The analysis for this accident conditions is presented in the following subsections.

4.1   Criticality Analysis – Misloading Accident in Interim Storage Configuration

During the review of the licensing report [1] to justify the proposed configuration, the existing analysis analyzed the misloading accident in the normal storage configuration, and showed that a soluble boron amount of 450 ppm was necessary to ensure that the maximum k_{eff} remained below 0.95. However, the misloading accident in the interim storage configuration where a fresh, unburned assembly is placed into a storage location intended to remain empty, was not performed. The criticality safety analysis is performed to determine the necessary soluble boron amount to ensure that the maximum k_{eff} remained below 0.95 for the misloaded assembly in the interim storage configuration, where the misloaded assembly is face adjacent to four fresh fuel assemblies.

4.1.1   Calculational Approach

The basic approach for the additional calculation was to model a misplaced assembly in the center of the storage rack in the interim storage configuration, where 4 fresh fuel assemblies are face adjacent to the misloaded assembly. This location was chosen to maximize k_{eff} as the misplaced assembly is adjacent to high reactivity, fresh fuel. The calculational model, as shown in Figure 5, is a 5x5 array of storage cells, with a checkerboard of fresh fuel assemblies and empty storage cells and the center storage cell of the model including the misloaded fresh fuel assembly (in a location intended to be empty). This model captures the reactivity effect of the misloaded fresh fuel assembly in the interim storage configuration without the use of reactivity equivalencing that was used for the spent fuel in the original analysis [1]. Additionally, reflecting boundary conditions are used in the radial direction, effectively creating an infinite model in the radial direction, where a misloaded assembly is present in every 5x5 array of cells. Finally, the uncertainties and bias from the previous analysis [1] were included in the calculation of the maximum k_{eff} in a conservative manner where applicable, as described in the relevant subsections.

4.1.2   Methodology

The principal method for the criticality analysis is the three-dimensional Monte Carlo code MCNP4a [4]. MCNP4a is a continuous energy three-dimensional Monte Carlo code developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been used previously

and verified for criticality analyses and has all of the necessary features for this analysis. Additionally, this code has been used in all recent spent fuel storage rack licensing efforts by Holtec International that have been reviewed and approved by the USNRC. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V and ENDF/B-VI supplied with the code.

Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009 with an uncertainty of ± 0.0011 for MCNP4a, evaluated with a 95% probability at the 95% confidence level [3]. The calculations for this analysis utilize the same computer platform and cross-section libraries used for the benchmark calculations discussed in Appendix A.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information has been used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in storage rack criticality calculations. Based on these studies, a minimum of 10,000 histories were simulated per cycle, a minimum of 20 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). Further, the output was reviewed to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between calculational precision and computational time.

The maximum k_{eff} is determined from the MCNP4a calculated k_{eff}, the calculational bias and the applicable uncertainties and tolerances (bias uncertainty, calculational uncertainty, rack tolerances, fuel tolerances) using the following formula:

$$\text{Max } k_{eff} = \text{Calculated } k_{eff} + \text{biases} + [\Sigma_i \, (\text{Uncertainty}_i)^2]^{1/2}$$

In the geometric models used for the calculations, each fuel rod and its cladding were described explicitly and reflecting boundary conditions were used in the radial direction which has the effect of creating an infinite radial array of storage cells.

4.1.3 Input data

All rack and fuel input data provided in [1] is sufficient to perform the criticality analysis for the misloaded assembly in the interim storage configuration. Fuel data for the design basis assembly was provided in Table 4.4 of [1] and rack data was provided in Figure 4-4 of [1][2]. The design

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[2] Table 4.4 and Figure 4-4 of [1] have different dimensions for the guide tube (thimble) i.d. and o.d., but the same thickness. The dimensions in Table 4.4 are used in the analysis. The discrepancy between the dimensions for the guide tube has a negligible effect on reactivity.

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basis assembly is the Westinghouse 15x15 assembly, with an enrichment of 5.0 wt%, which was found in [1] to have the highest reactivity of all assembly types present in the D.C. Cook spent fuel pool.

4.1.4 Assumptions/Conservatisms

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were employed:

1) Neutron absorption in minor structural members is neglected, i.e., spacer grids are analytically replaced by water.

2) The effective multiplication factor of an infinite radial array of fuel assemblies was used in the analysis.

3) An enrichment of 5.0 wt% is conservatively used for all fresh fuel assemblies in the analysis. The analysis in [1] used a maximum enrichment of 4.95 wt% with a 0.05 wt% enrichment tolerance. The reactivity effect of the enrichment tolerance is conservatively included in the calculation of the maximum k_{eff} as shown in Table 1.

4) The boron loading in the neutron absorber panels is assumed to be at the minimum of 0.0300 g ^{10}B/cm^2 (nominal of 0.0345 g ^{10}B/cm^2). The reactivity effect for the neutron absorber loading tolerance from [1] is conservatively included in the rack manufacturing tolerances as shown in Table 1.

5) The presence of burnable absorbers (B_4C, Gadolinium, Erbium, IFBA) in fresh fuel is neglected. This is conservative as burnable absorbers would reduce the reactivity of the fresh fuel assembly.

6) The calculation consists of a 5x5 array with a misloaded fuel assembly in the center storage cell. Reflecting boundary conditions conservatively create an infinite array of a single misloaded assembly in every 5x5 array.

7) The biases and uncertainties are applied in a conservative manner to maximize the maximimum k_{eff}. The axial burnup effect and depletion uncertainty from [1] is applied although this model contains only fresh fuel, to account for the high burned fuel in the interim storage configuration.

4.1.5 Computer Codes

The following computer codes were used during this analysis.

- MCNP4a [4] is a three-dimensional continuous energy Monte Carlo code developed at Los Alamos National Laboratory. This code offers the capability of performing full three-dimensional calculations for the loaded storage racks. MCNP4a was run on the PCs at Holtec.

4.1.6 Temperature Effects

Pool water temperature effects on reactivity in the spent fuel storage racks were previously calculated and the results were presented in Table 4.6 of [1]. The results show that the spent fuel pool temperature coefficient of reactivity is negative, i.e. a lower temperature results in a higher reactivity.

In MCNP4a, the Doppler treatment and cross-sections are valid only at 300K (27 °C). Therefore, a Δk is determined from 27 °C to 4 °C from Table 4.6 of [1], and is included in the final k_{eff} calculation as a bias. The bias shown in Table 1 for 27 °C (80.33 °F) to 4 °C (39.2 °F).

4.1.7 Uncertainties Due to Manufacturing Tolerances

In the calculation of the final k_{eff}, the effect of manufacturing tolerances on reactivity must be included. CASMO-3 was used to perform these calculations in the previous analysis [1]. As prescribed in [5], the methodology employed to calculate the tolerance effects combine both the worst-case bounding value and sensitivity study approaches. The evaluations include tolerances of the rack dimensions and tolerances of the fuel dimensions. As for the bounding assembly, calculations are performed for a nominal initial enrichment of 4.95 wt% ^{235}U. The reference condition is the condition with nominal dimensions and properties. To determine the Δk associated with a specific manufacturing tolerance, the k_{inf} calculated for the reference condition is compared to the k_{inf} from a calculation with the tolerance included. Note that for the individual parameters associated with a tolerance, no statistical approach is utilized. Instead, the full tolerance value was utilized to determine the maximum reactivity effect. All of the Δk values from the various tolerances are statistically combined (square root of the sum of the squares) to determine the final reactivity allowance for manufacturing tolerances. Only the Δk values in the positive direction (increasing reactivity) were used in the statistical combination. The fuel and rack tolerances included in this analysis are shown in Table 1.

4.1.8 Criticality Analysis

The calculational models for the misloaded assembly situation consist of 5x5 checkerboard of assemblies with reflective boundary conditions on all four sides. Figure 5 shows the MCNP4a calculational model of the spent fuel storage cells, as drawn by the two-dimensional plotter in MCNP4a for the analyzed accident condition. This model is a checkerboard of fresh fuel

assemblies and empty storage locations, with the center location filled with a fresh fuel assembly rather than remaining empty. Reactivity effects of manufacturing tolerances and uncertainties are taken from [1] and applied to the accident condition analyzed here. A summary of the calculation of the maximum k_{eff} with a maximum enrichment of 5.0 wt% ^{235}U is shown in Table 1. The result of the analysis shows that the maximum k_{eff} under this accident scenario is significantly less than 0.95 with 800 ppm of soluble boron credited. The 800 ppm is much less than the 2400 ppm required during fuel movement by the Technical Specifications.

5.0   Thermal-Hydraulic Considerations

The thermal hydraulic evaluation provided in [1] considers several core-offload scenarios such as normal discharge and back-to-back core offloads. The discharge scenarios are not dependant upon the loading configurations illustrated in Figure 2 and Figure 3. Therefore, the thermal analysis bounds any loading combination allowed by the proposed configuration change.

6.0   Rack Structural Considerations

The rack structural/seismic analysis provided in [1] considers all racks to be fully loaded with spent fuel. This analysis is bounding for both the normal storage configuration and the interim storage configuration, and therefore is also bounding for the proposed configuration.

7.0   Accident Analysis and Miscellaneous Structural Evaluations

The accident analysis and miscellaneous structural evaluations presented in [1] considers such events as a dropped fuel assembly, local cell wall buckling, analysis of welded joints in an isolated hot cell and crane uplift loads. In all cases these events are not affected by the proposed configuration.

8.0   Static and Dynamic Analyses of Fuel Pool Structure

The spent fuel pool structural analysis provided in [1] assumes all racks to be fully loaded with spent fuel. This analysis is bounding for both the normal storage configuration and the interim storage configuration, and therefore is also bounding for the proposed configuration.

9.0   Radiological Evaluation

The radiological analysis of a fuel handling accident presented in [1] is based upon a single assembly at the maximum burnup and shortest cooling time. The normal operational dose rates around the spent fuel pool due to the proposed configuration would be no higher than the dose
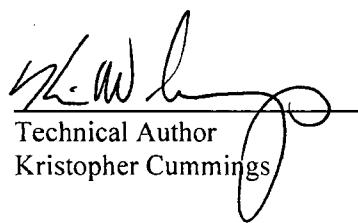
rates with the configuration shown in Figure 2. Therefore, the accident and the normal operational radiological analyses in [1] are not affected by the proposed configuration.
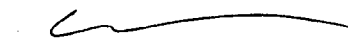
10.0   In-Service Surveillance Program

The in-service surveillance of the Boral neutron absorber coupons is not affected by the proposed change as no hardware changes are being implemented.

11.0   References

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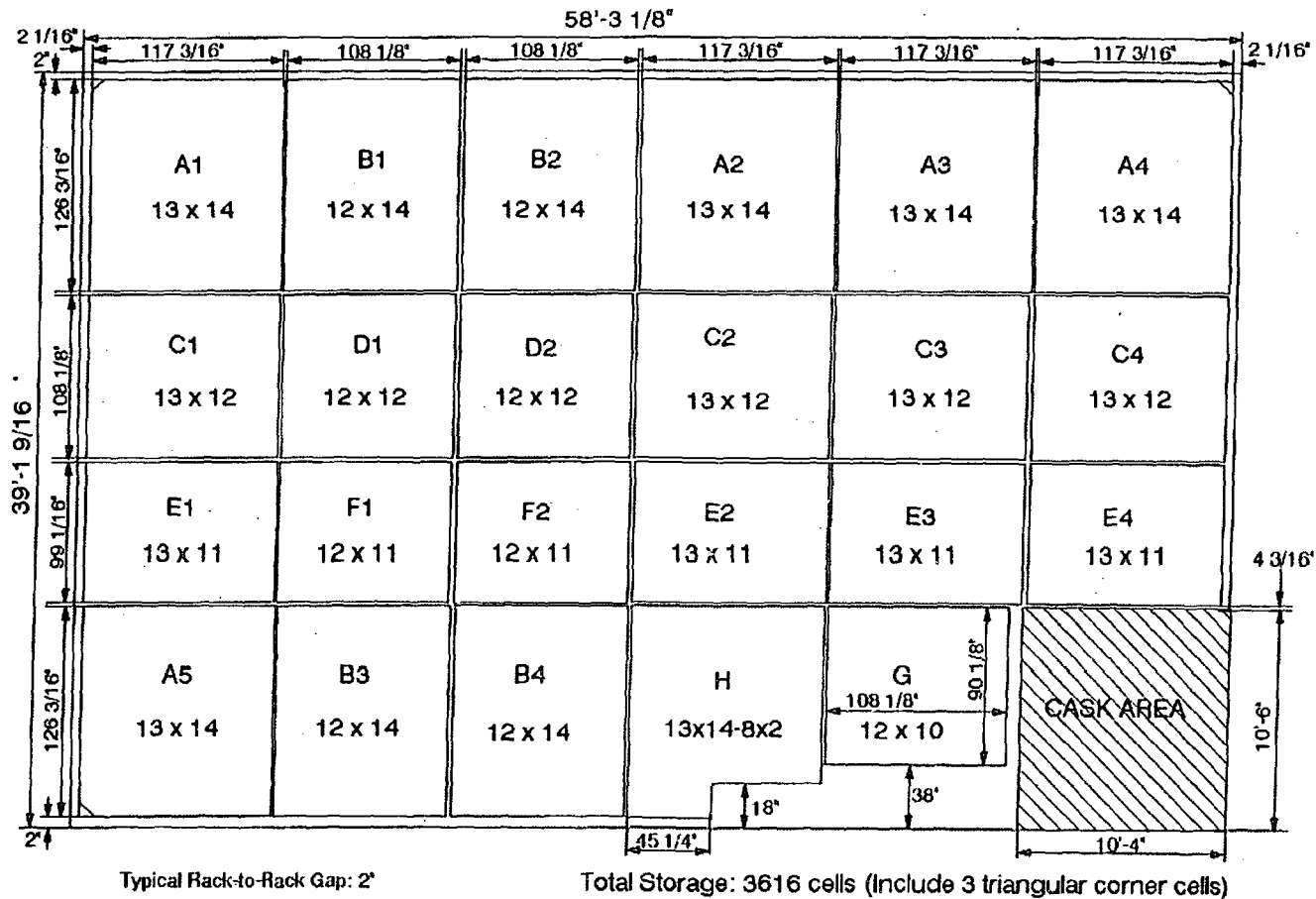
Figure 1: D.C. Cook Spent Fuel Pool Layout

Typical Rack-to-Rack Gap: 2"   Total Storage: 3616 cells (Include 3 triangular corner cells)

■ 504 Region 1 Cells   □ 1439 Region 2 Cells   ▓ 1670 Region 3 Cells
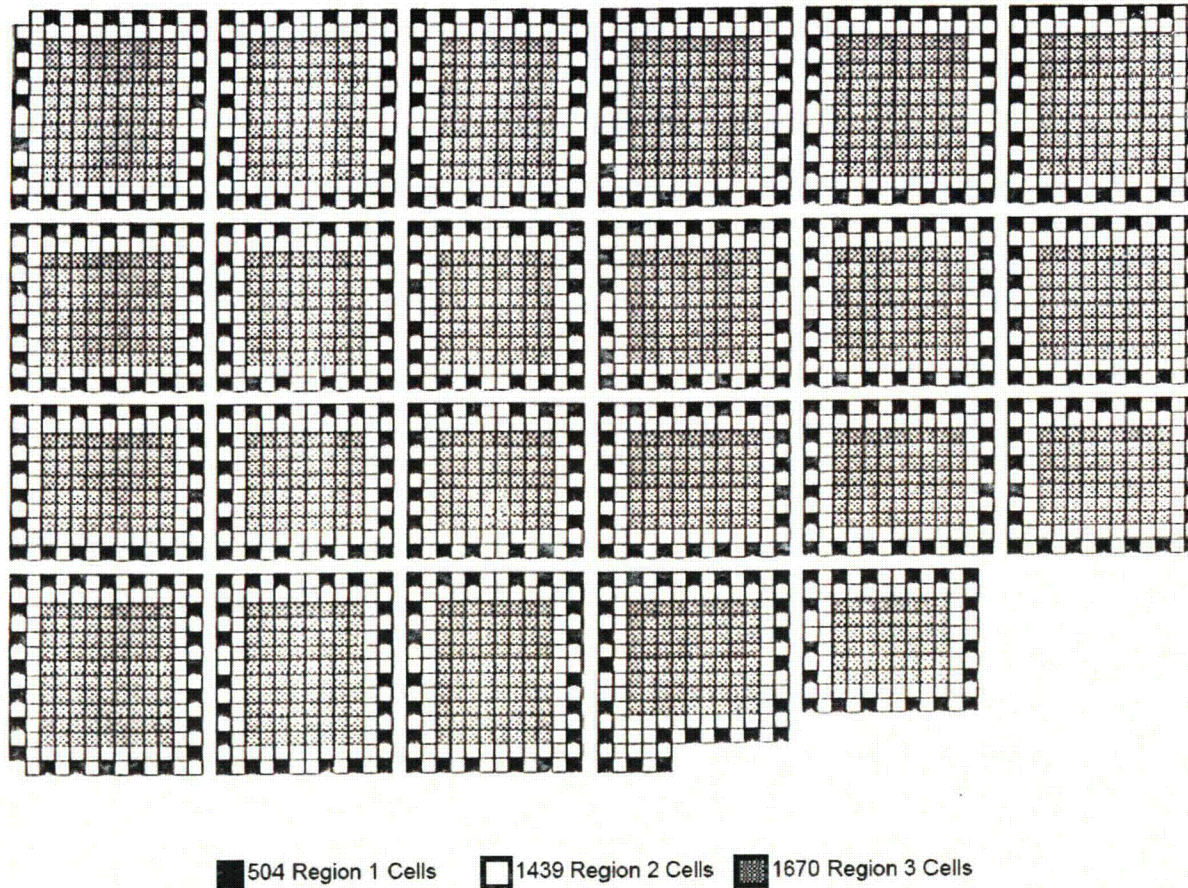
Figure 2: Normal Storage Pattern

⊠ 158 Empty Locations   ■ 661 Region 1 Cells   ☐ 1439 Region 2 Cells   ▦ 1355 Region 3 Cells
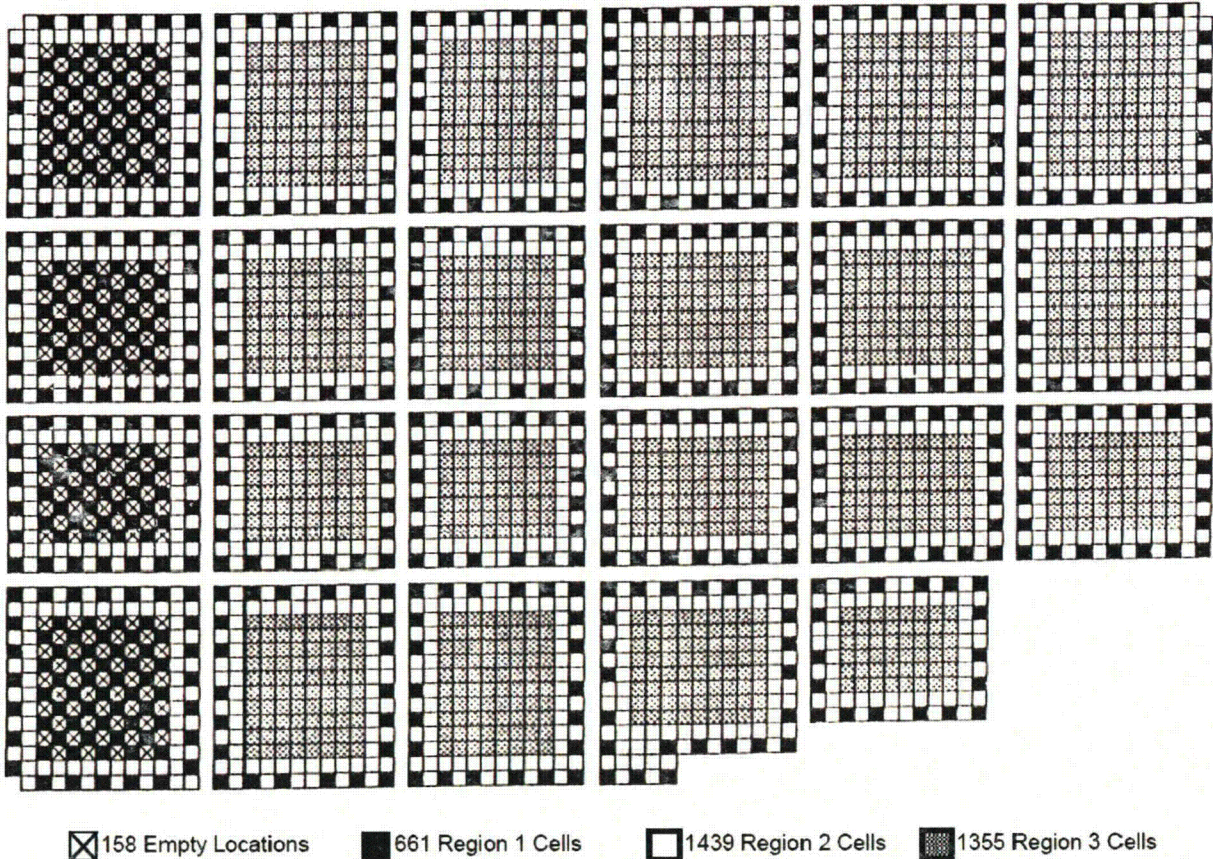
Figure 3: Interim Storage Pattern
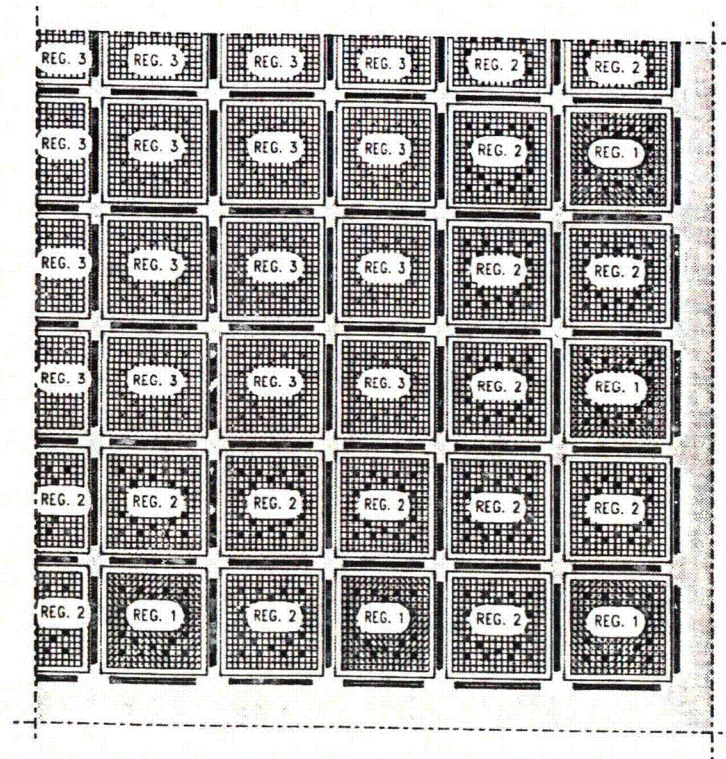
Figure 4: Criticality Analysis KENO model for the Normal Storage Configuration at D.C. Cook[3]

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[3] The KENO model for the interim storage configuration is identical except that the Region 3 fuel assemblies are replaced by a checkerboard of Region 1 fuel assemblies (fresh fuel) and empty storage locations.

Figure 5: A Two-Dimensional Representation of the Misloaded Assembly Calculation Model used for the Interim Storage Configuration. This Figure was Drawn (To Scale) with the Two-Dimensional Plotter in MCNP4a.

Table 1
Summary of the Criticality Safety Analyses for a Misloaded Fresh Fuel Assembly in the Interim
Storage Configuration with Soluble Boron

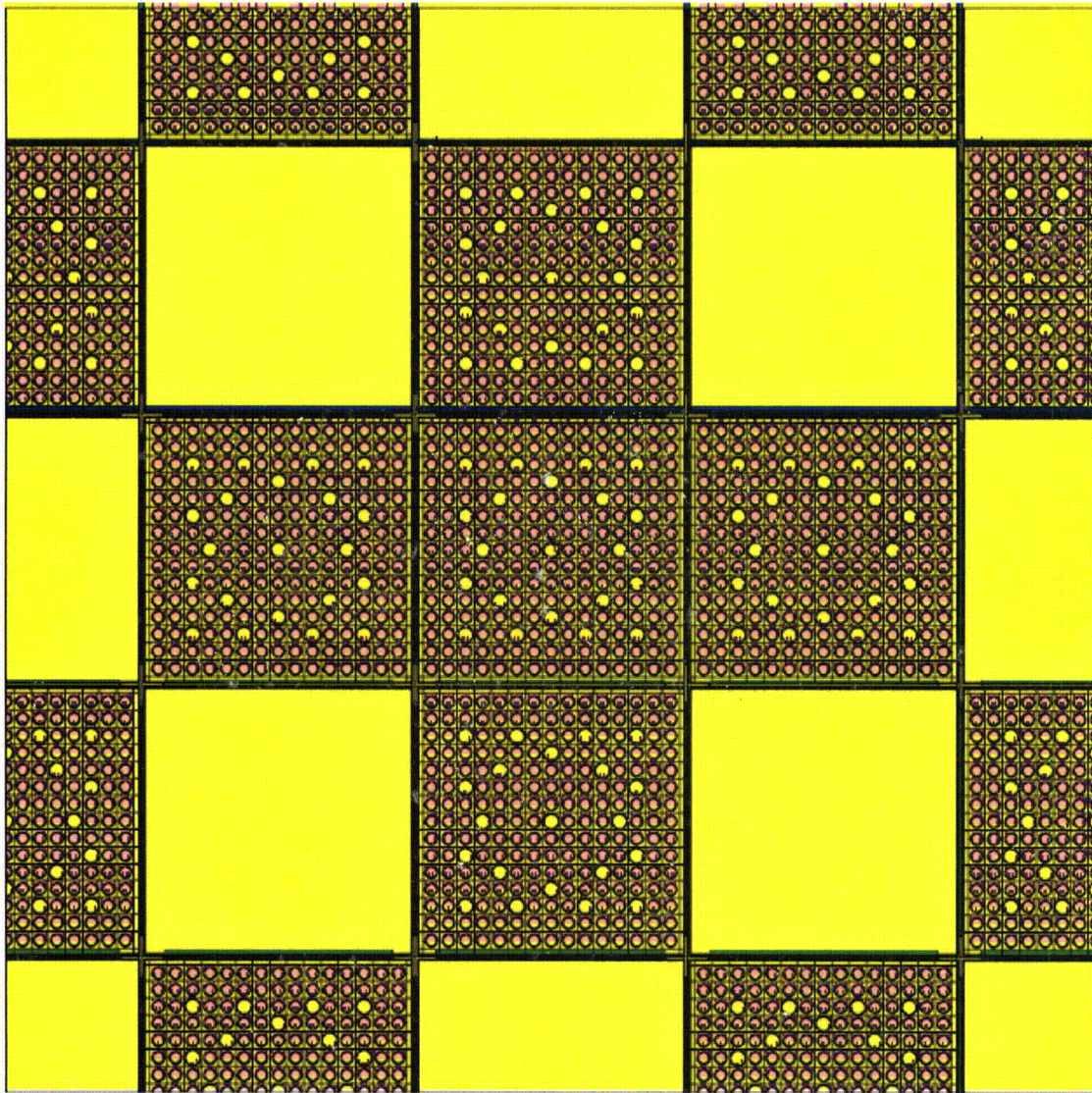| Design Basis Burnup at 5.0 wt% ^{235}U	0 GWD/MTU
Soluble Boron	800 ppm
MCNP Filename	d5r2b[4]
Uncertainties	
Bias Uncertainty (95%/95%)	± 0.0011
Calculational Statistics (95%/95%, 2.0×σ)	± 0.0014
Fuel Eccentricity	± 0.0019
Rack Tolerances[5]	± 0.0078
Fuel Tolerances[6]	± 0.0049
Depletion Uncertainty[7]	± 0.0047
Statistical Combination of Uncertainties	± 0.0107
Reference k_{eff} (MCNP4a)	0.9108
Total Uncertainty (above)	0.0107
Axial Burnup Effect[8]	0.0037
Temperature Bias[9]	0.0034
Calculational Bias (see Appendix A)	0.0009
Maximum k_{eff}	**0.9295**
Regulatory Limiting k_{eff}	**0.9500**

[4] All input files for the calculations are stored in the directory \Projects\1784\Technical Evalautions\1784001 and its subdirectories on the Holtec server.

[5] This includes manufacturing tolerances of the ^{10}B loading in the Boral panel (0.0061), Boral panel width (0.0009), storage cell ID (0.0015), storage cell wall thickness (0.0009) and water gap between adjacent racks (0.0045).

[6] This includes manufacturing tolerances of the fuel density (0.0035) and fuel enrichment (0.0034)

[7] Depletion uncertainty from [1] for 4.95 wt% fuel burned to 50 GWD/MTU.

[8] From Table 4.2 of [1]. Conservatively included to account for burned fuel in the periphery of the rack. This is the axial burnup effect for the normal storage configuration where the rack is filled with spent and fresh fuel as shown in Figure 4.

[9] Calculated as the difference in reactivity for a temperature of 80.33 °F (300K) to 39.2°F from Table 4.6 of [1] for fresh fuel. The temperature bias is calculated by linear interpolation between the points provided in Table 4.6 of [1] for Region 1.

Appendix A
Benchmark Calculations

(total number of pages: 26 including this page)

Note: because this appendix was taken from a previously NRC approved report, 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).

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

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

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

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

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[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}		EALF [†] (eV)	
				MCNP4a	KENO5a	MCNP4a	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 [†] (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}		EALF [†] (eV)	
				MCNP4a	KENO5a	MCNP4a	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}		EALF [†] (eV)	
				MCNP4a	KENO5a	MCNP4a	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}		EALF [†] (eV)	
				MCNP4a	KENO5a	MCNP4a	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 $k_{eff} \pm 1\sigma$	
	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 $k_{eff} \pm 1\sigma$	
	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	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	0.5289	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.
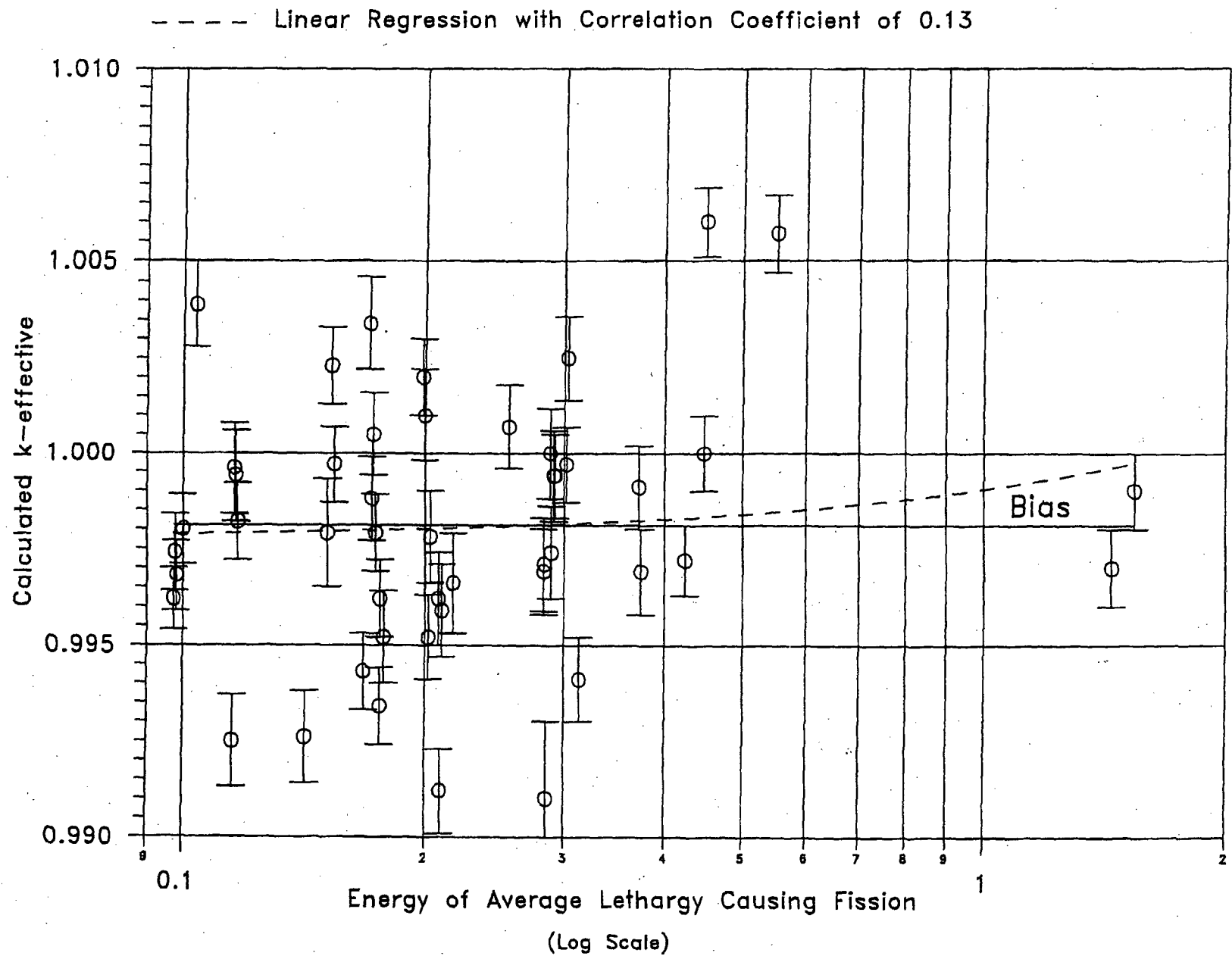
FIGURE 4A.1   MCNP CALCULATED k—eff VALUES for
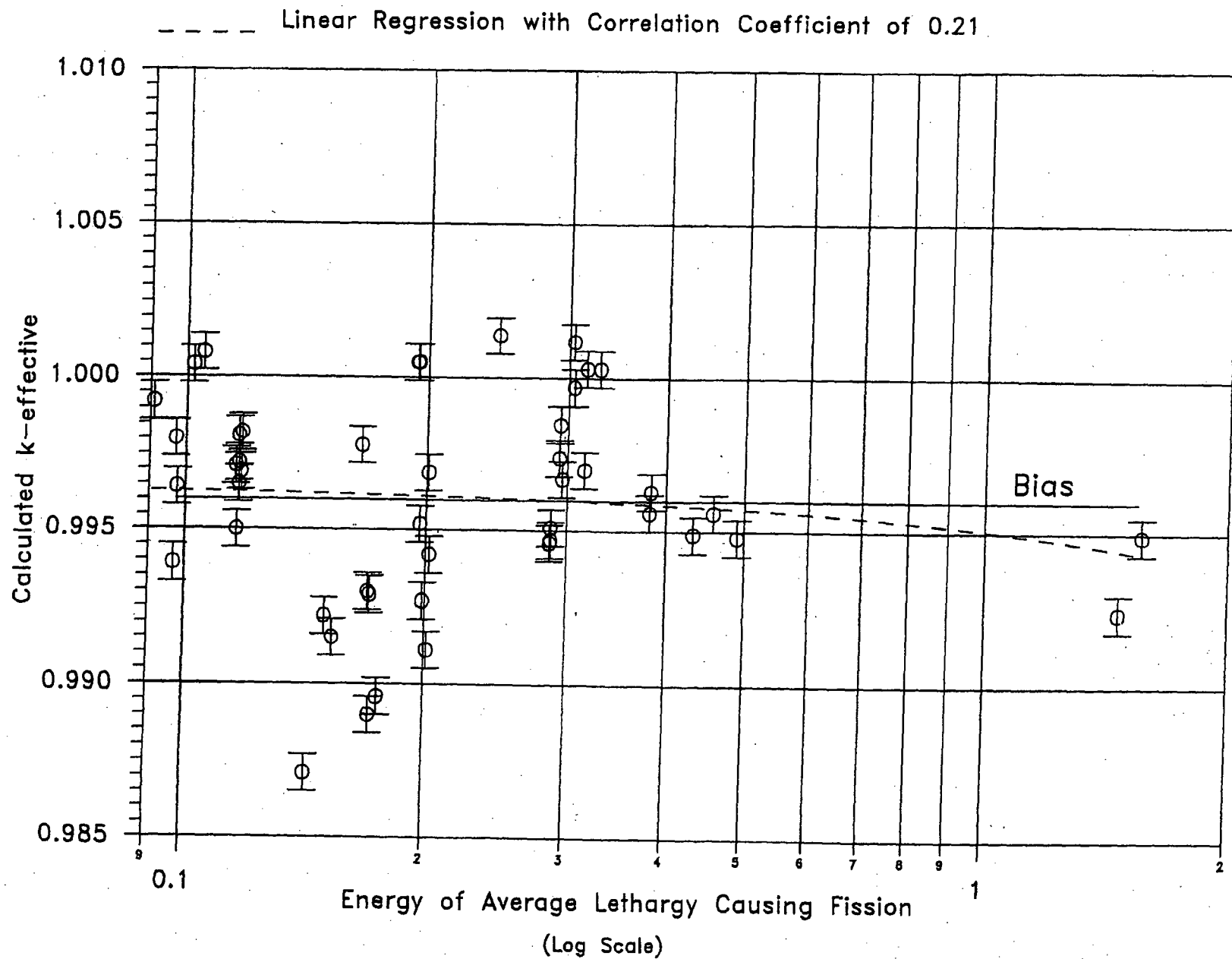VARIOUS VALUES OF THE SPECTRAL INDEX

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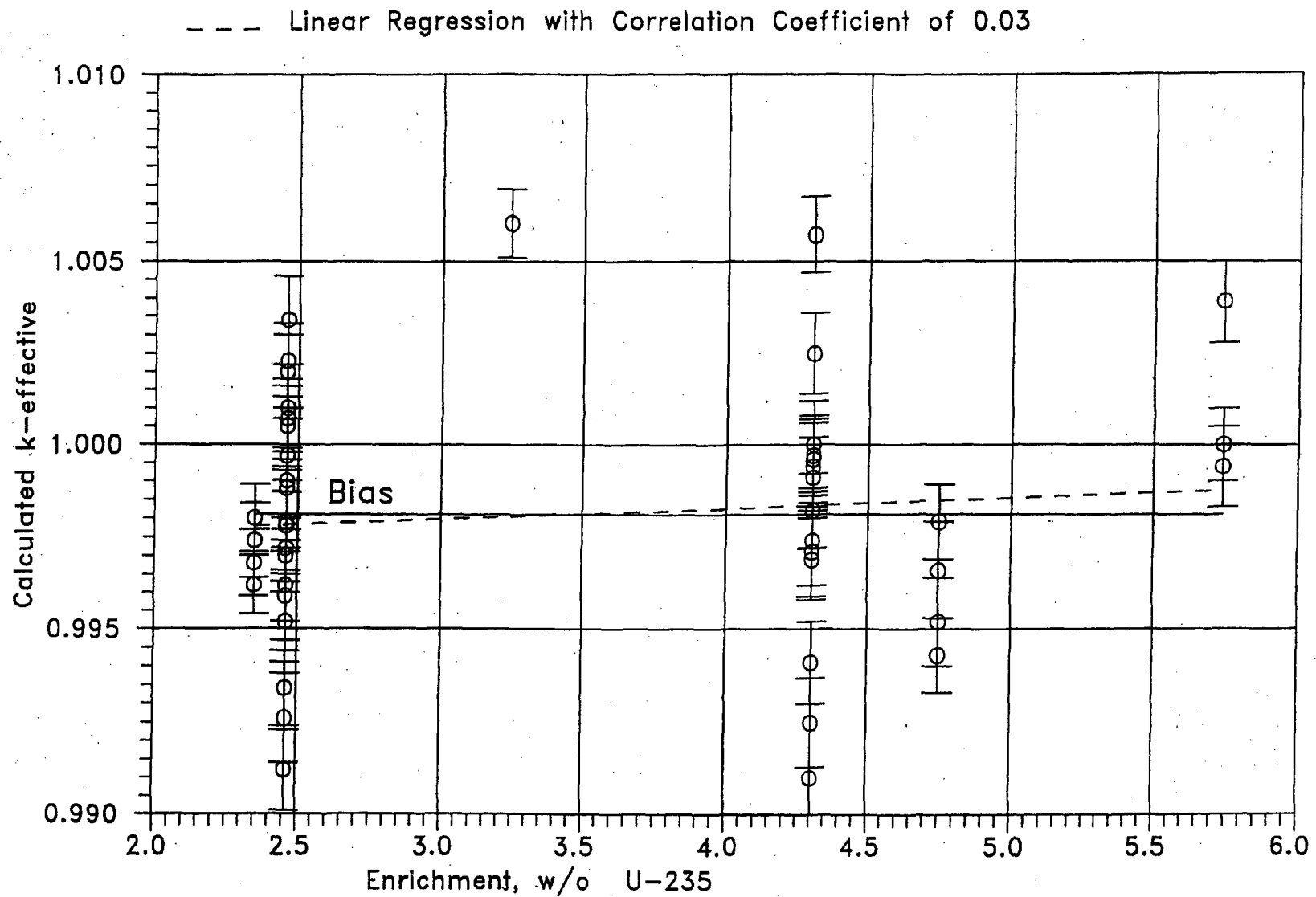VARIOUS VALUES OF THE SPECTRAL INDEX

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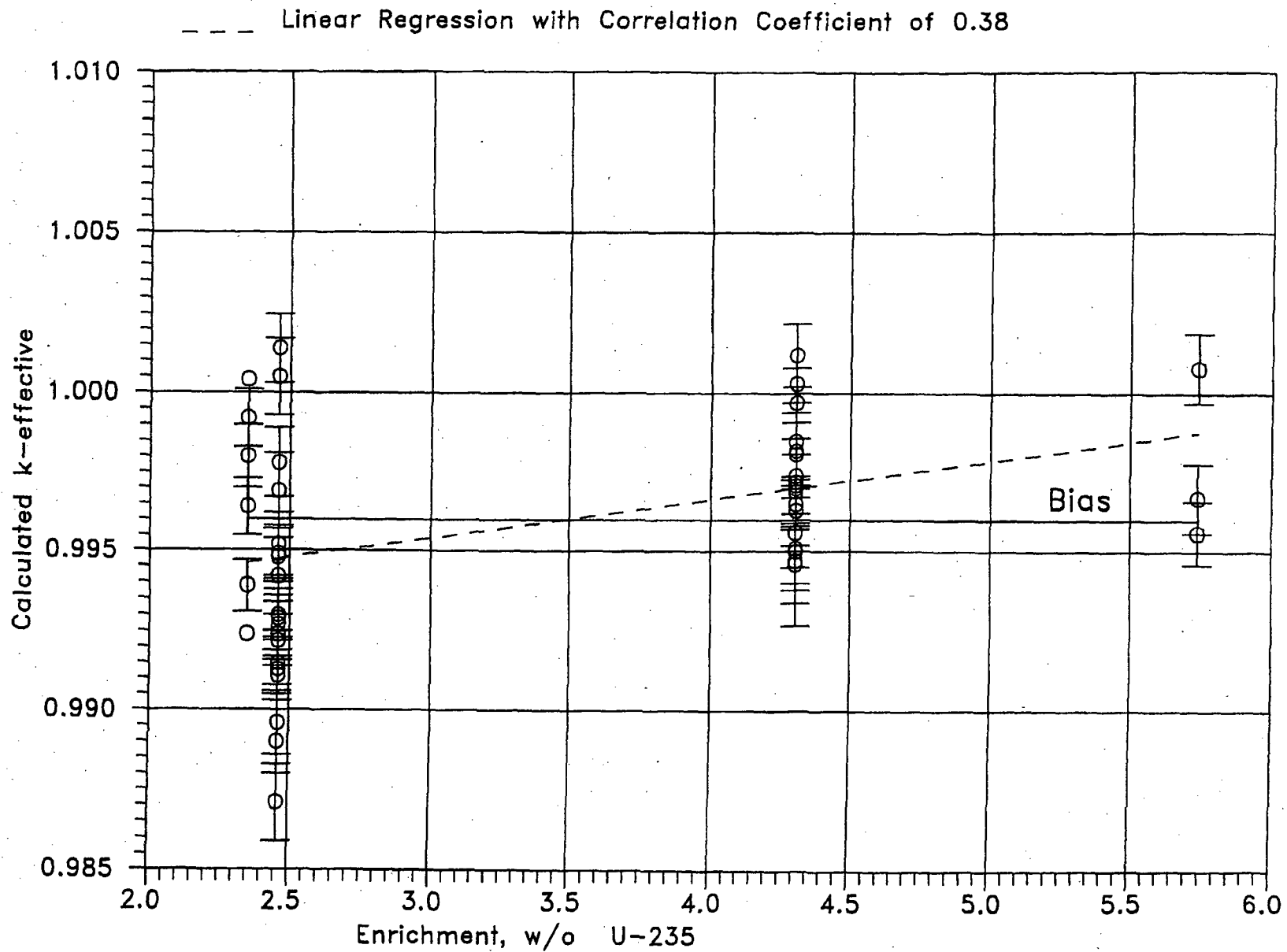AT VARIOUS U-235 ENRICHMENTS

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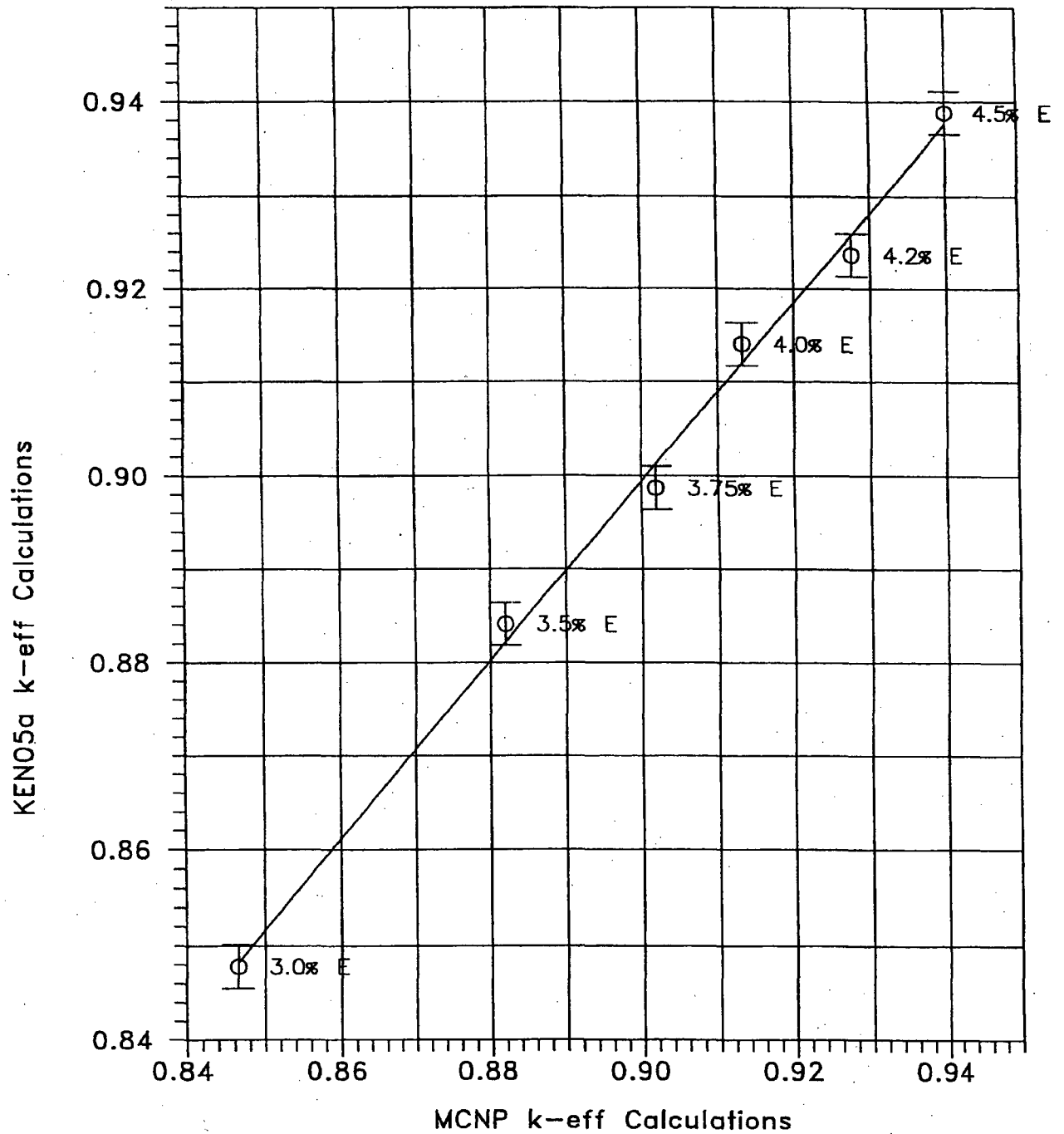AT VARIOUS U-235 ENRICHMENTS

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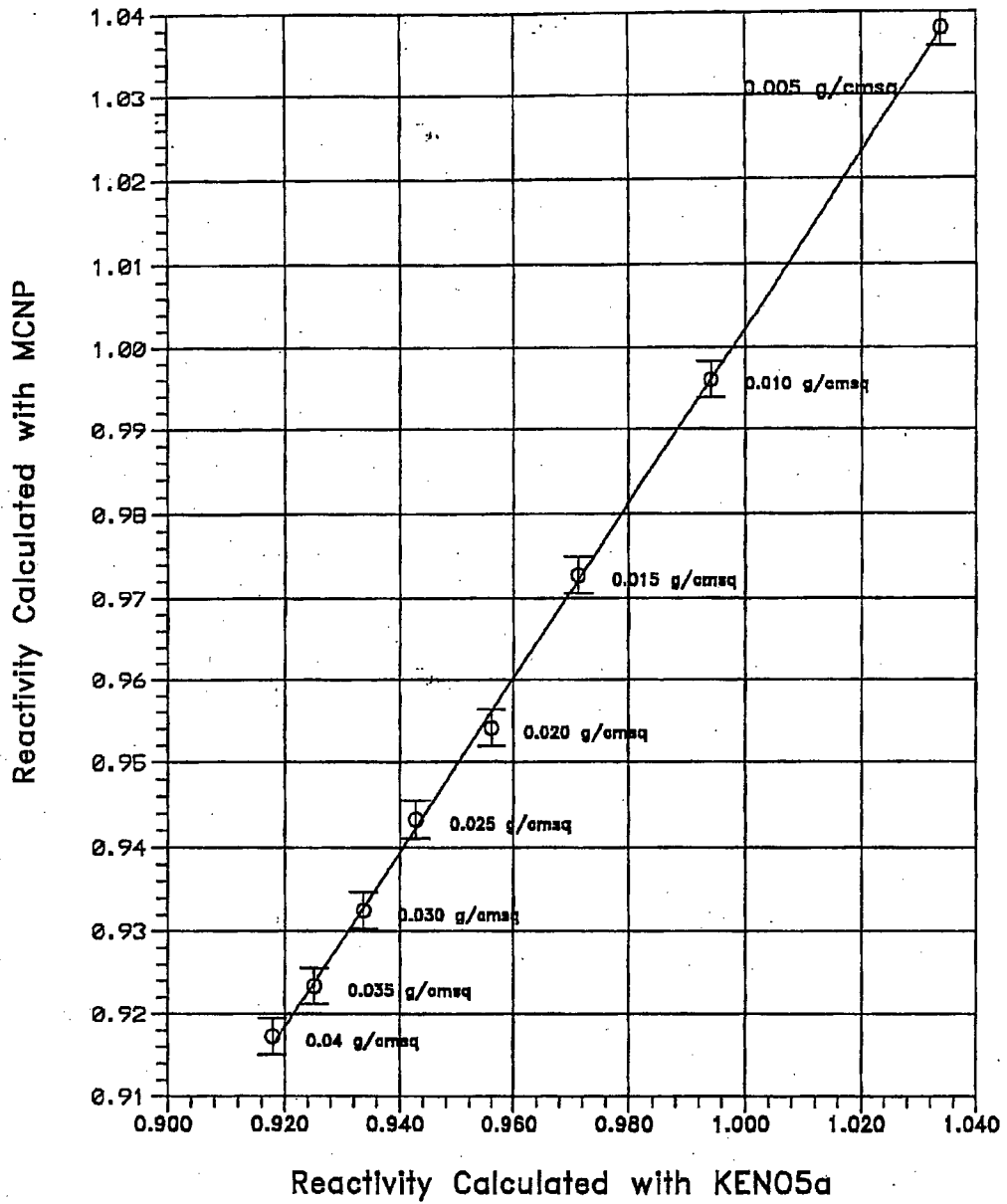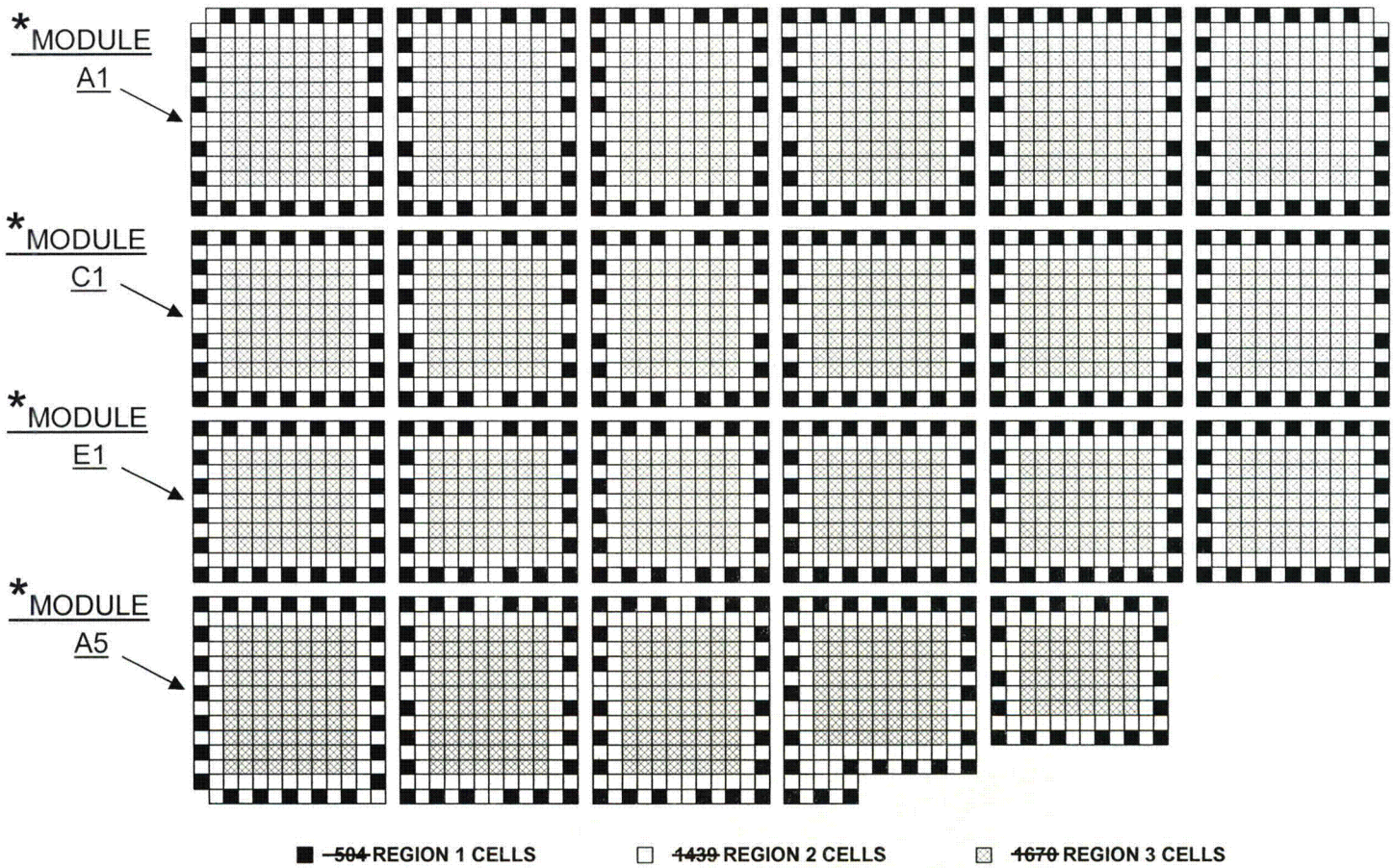CALCULATIONS FOR VARIOUS
FUEL ENRICHMENTS

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

Attachment 1 to AEP:NRC:8431

DONALD C. COOK NUCLEAR PLANT UNIT 1 TECHNICAL SPECIFICATION PAGES
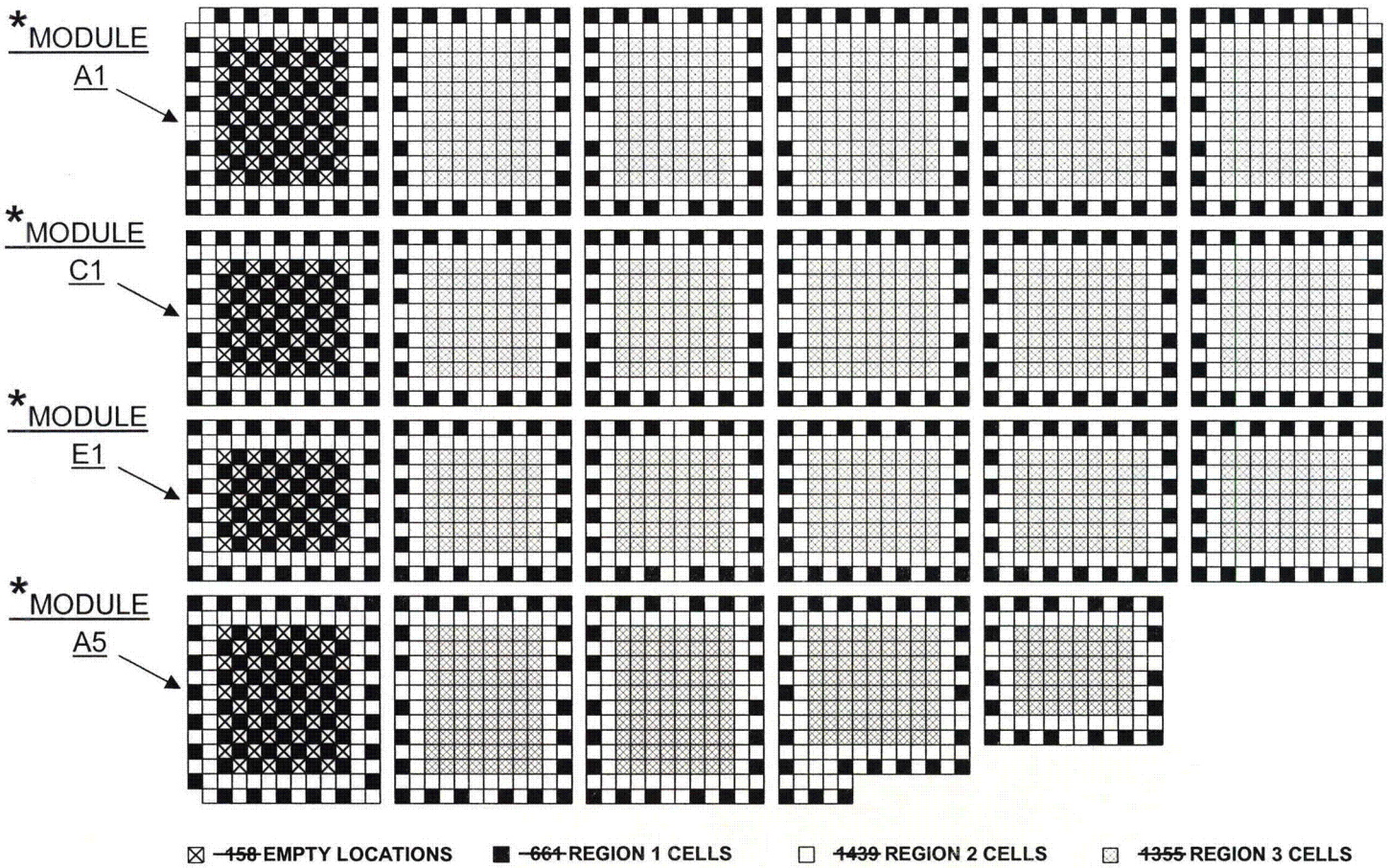MARKED TO SHOW CHANGES

4.0-4

4.0-5

*MODULE
A1

*MODULE
C1

*MODULE
E1

*MODULE
A5

■ ─504 REGION 1 CELLS   □ 4439 REGION 2 CELLS   ▨ 4670 REGION 3 CELLS

*The storage pattern for any of these individual modules may be as shown in this figure or Figure 4.3-2.

Figure 4.3-1 (page 1 of 1)
Normal Storage Pattern (Mixed Three Zone)

* MODULE A1

* MODULE C1

* MODULE E1

* MODULE A5

⊠ ~~158~~ EMPTY LOCATIONS   ■ ~~664~~ REGION 1 CELLS   ☐ ~~1439~~ REGION 2 CELLS   ⊠ ~~1355~~ REGION 3 CELLS

*The storage pattern for any of these individual modules may be as shown in this figure or Figure 4.3-1.
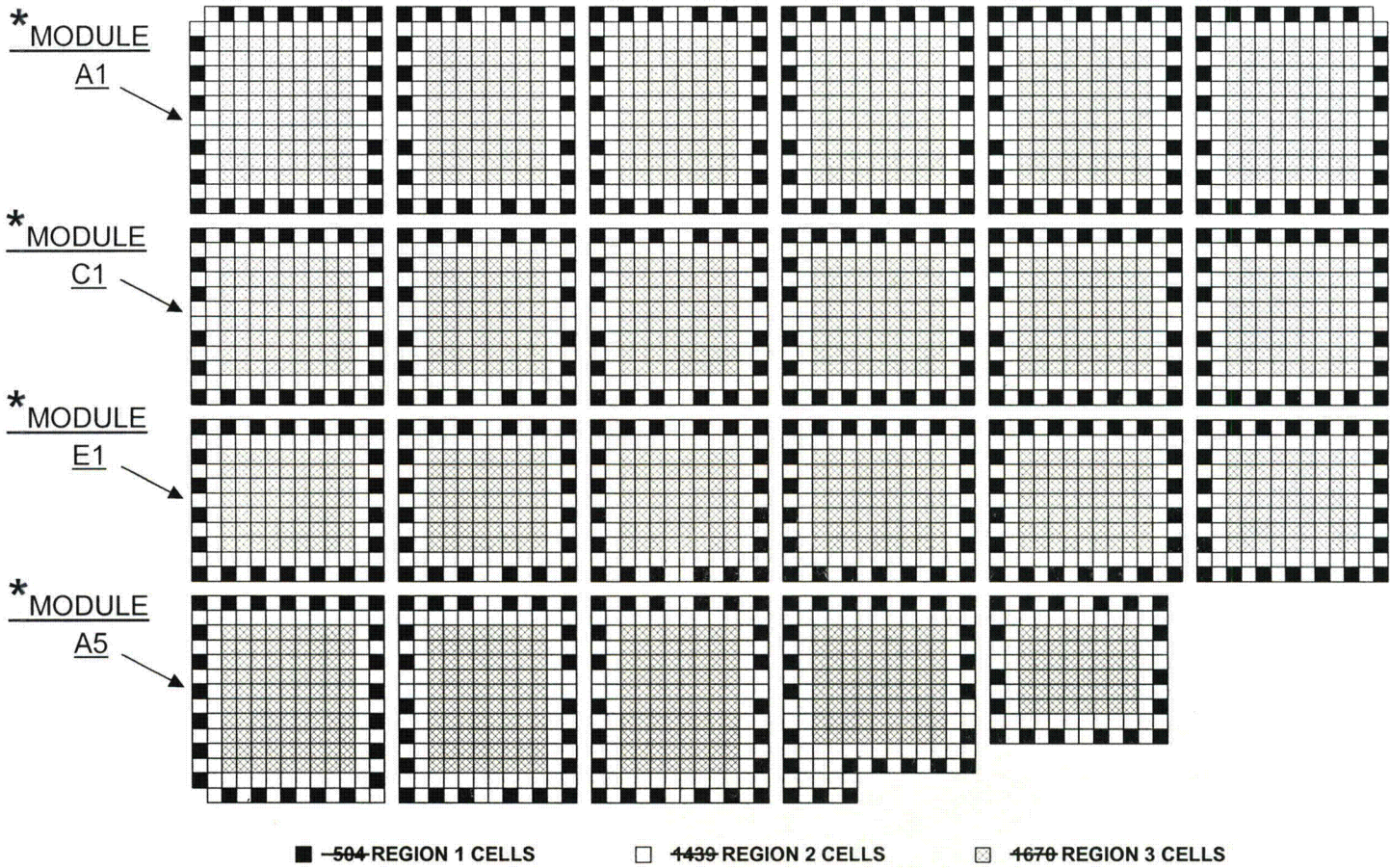
Figure 4.3-2 (page 1 of 1)
Interim Storage Pattern (Checkerboard)

Attachment 2 to AEP:NRC:8431

DONALD C. COOK NUCLEAR PLANT UNIT 2 TECHNICAL SPECIFICATION PAGES
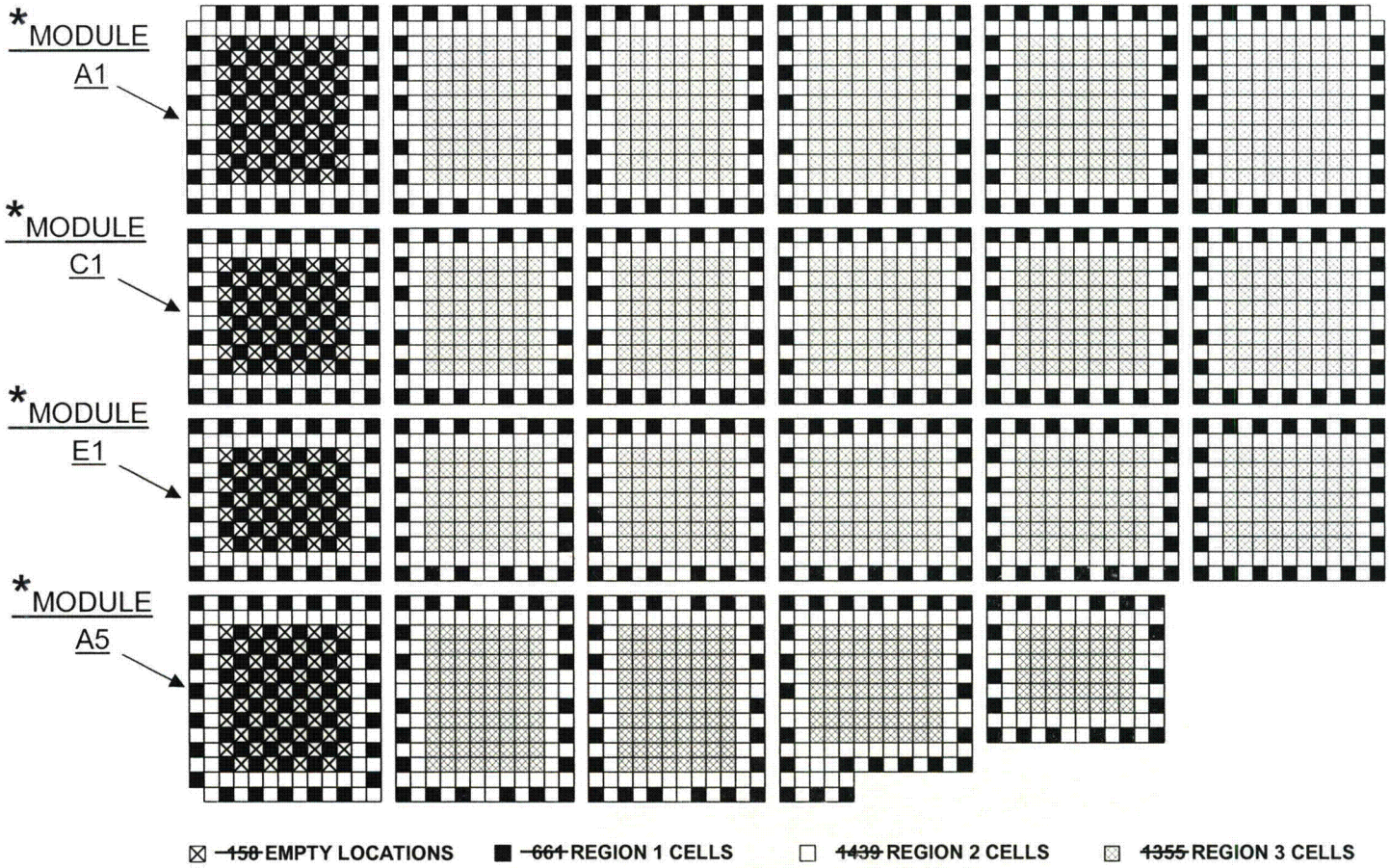MARKED TO SHOW CHANGES

4.0-4

4.0-5

*MODULE A1

*MODULE C1

*MODULE E1

*MODULE A5

■ 504 REGION 1 CELLS   □ 1439 REGION 2 CELLS   ⊠ 1670 REGION 3 CELLS

*The storage pattern for any of these individual modules may be as shown in this figure or Figure 4.3-2.

Figure 4.3-1 (page 1 of 1)
Normal Storage Pattern (Mixed Three Zone)

*MODULE
A1

*MODULE
C1

*MODULE
E1

*MODULE
A5

⊠ 158 EMPTY LOCATIONS   ■ 661 REGION 1 CELLS   ☐ 1439 REGION 2 CELLS   ⊠ 1355 REGION 3 CELLS

*The storage pattern for any of these individual modules may be as shown in this figure or Figure 4.3-1.

Figure 4.3-2 (page 1 of 1)
Interim Storage Pattern (Checkerboard)