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2CAN060306

June 30, 2003

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

SUBJECT: License Amendment Request
To Revise the Spent Fuel Pool Loading Pattern
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests an amendment for Arkansas Nuclear One, Unit 2 (ANO-2). The proposed amendment will: 1) eliminate credit for the Boraflex neutron absorbing material used for reactivity control in Region 1 of the Spent Fuel Pool (SFP); 2) credit a combination of soluble boron and several defined fuel loading patterns within the storage racks to maintain SFP reactivity within the effective neutron multiplication factor (k_{eff}) limits of 10 CFR 50.68; 3) increase the minimum boron concentration in the SFP to 2000 parts per million; and 4) reduce the fresh fuel assembly initial enrichment to less than or equal to 4.55±0.05 weight percent (wt%) uranium-235 (U-235).

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

The proposed change includes a new commitment which is summarized in Attachment 4.

Entergy requests approval of the proposed amendment by September 3, 2003 in order to support a full core offload during the ANO-2 fall 2003 refueling outage. Once approved, the amendment shall be implemented within 30 days. Although this request is neither exigent nor emergency, your prompt review is requested.

If you have any questions or require additional information, please contact Dana Millar at 601-368-5445.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 30, 2003.

Sincerely,

CGA/dm

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Changes to Technical Specification Bases Pages – For Information Only
4. List of Regulatory Commitments
5. Holtec License Amendment Report

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Attachment 1

2CAN060306

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2).

The proposed change will revise the ANO-2 Technical Specifications (TSs) to: 1) eliminate credit for the Boraflex neutron absorbing material used for reactivity control in Region 1 of the Spent Fuel Pool (SFP); 2) credit a combination of soluble boron and several defined 2 x 2 fuel patterns within the storage racks to maintain SFP reactivity within the effective neutron multiplication factor (k_{eff}) limits of 10 CFR 50.68; 3) increase the minimum soluble boron concentration in the SFP to 2000 parts per million (ppm); and 4) reduce the fresh fuel assembly initial enrichment to less than or equal to 4.55±0.05 weight percent (wt%) uranium-235 (U-235). Approval of this amendment is needed for the fall 2003 refueling outage in order to allow full core offload capability.

2.0 PROPOSED CHANGE

The proposed amendment will: 1) eliminate credit for the Boraflex neutron absorbing material used for reactivity control in Region 1 of the SFP; 2) credit a combination of soluble boron and several defined 2 x 2 fuel patterns within the storage racks to maintain SFP reactivity within the k_{eff} limits of 10 CFR 50.68; 3) increase the minimum soluble boron concentration in the SFP to 2000 ppm; and 4) reduce the fresh fuel assembly initial enrichment to less than or equal to 4.55±0.05 wt% U-235.

TS 3.9.12.a currently allows storage of fuel assemblies having an initial enrichment less than or equal to 5.0 wt% U-235 in the spent fuel pool. The fuel assembly initial enrichment will be changed based on the reactivity analysis performed to support the proposed change. The new fuel assembly initial enrichment will be limited to less than or equal to 4.55±0.05 wt% U-235.

Changes are proposed to TS 3.9.12.b which describes the fuel loading restrictions in the currently defined Regions 1 and 2. Five 2 x 2 storage patterns along with their interfacing relationships are proposed that will allow credit for a combination of soluble boron and fuel positioning within the storage racks to maintain SFP reactivity. The five patterns include a combination of various fuel enrichments, fuel burnup, fuel cooling times, assemblies with Control Element Assemblies (CEAs) inserted and without CEAs inserted, and water cells. Different fuel types or vacant spaces as defined below will be specified within the 2 x 2 patterns.

- "F" – fresh fuel assemblies with an initial enrichment less than or equal to 4.55±0.05 wt%.
- "B" – an assembly with a maximum initial enrichment of 4.1 wt%, a minimum burnup of 11.3 GWD/MTU and zero years cooling time. Typically "B" type assemblies have only one cycle of burnup.
- "Fc" – fresh fuel assemblies with an initial enrichment of less than or equal to 4.55±0.05 wt% with a CEA inserted. The CEA must be one that is of the new design.
- "A" – an assembly that meets specific burnup requirements that will be included in the proposed change.
- "BC" – a "B" type assembly with a CEA inserted. This CEA may be of the old or new design.

- "C" – a high burnup assembly that meets specific burnup requirements that will be included in the proposed change. Initial enrichment is less than or equal to 4.50±0.05 wt%.
- "AC" – an "A" type assembly with a CEA inserted. This CEA may be of the old or new design.
- "CC" – a "C" type assembly with a CEA inserted. This CEA may be of the old or new design. Initial enrichment is less than or equal to 4.50±0.05 wt%.
- "X" – a water cell with no fuel.

The ANO-2 CEAs incorporate in their design five "fingers." In the old design, the middle finger of the CEAs is of a different design, using a 8 ½ inch inconel slug in the bottom of the fingers. The outer fingers use a 12 ½ inch silver, indium, cadmium (Ag-In-Cd) slug in the bottom of the fingers. The bottom of the fingers of the new CEAs all have a12 ½ inch Ag-In-Cd tip.

The five loading patterns, which will be reflected in TS Figure 3.9-2, have been evaluated to consider the reactivity effects based on various interfacing combinations. The allowable pattern interfaces will also be reflected in TS Figure 3.9-2.

A change to TS 3.9.12.c is proposed that will increase the SFP boron concentration to 2000 ppm. Since soluble boron will be credited to maintain SFP reactivity, the increase in the TS boron concentration will provide additional margin.

Supporting changes are also proposed to the Action and the Surveillance Requirements (SRs) associated with TSs 3.9.12.a, 3.9.12.b and 3.9.12.c. In addition, Note 1 will be deleted. The new criticality analysis does not support use of the vacant spaces defined in the Note.

The current minimum burnup verses initial assembly average U-235 loading curve depicted in TS Figure 3.9.2 will be deleted. New information reflecting the allowable 2 x 2 loading patterns, the acceptable interfaces between the five patterns, the minimum burnup requirements of type "A" and "C" assemblies at varying enrichment and cooling times, and bounding polynomial equations used to determine the minimum acceptable burnup for type "A" and "C" assemblies will be added as the proposed TS Figure 3.9-2.

A change to TS 5.3.1.b is proposed stating that k_{eff} will be maintained less than or equal to 0.95 when credit for 240 ppm of boron is taken. The current TS 5.3.1.c will become 5.3.1.d and a new TS 5.3.1.c will be added stating that k_{eff} will remain below 1.0 if the pool is flooded with unborated water.

A change is also proposed to TS 5.3.2.a to limit the new fuel assembly U-235 enrichment to 4.55±0.05 wt%. Due to restrictions considered in the spent fuel storage criticality analyses, the maximum initial fuel enrichment will be restricted to 4.55±0.05 wt%. This impacts the discussion of the new fuel storage rack design in TS 5.3.2. This change is a conservative restriction on fuel enrichment and does not have any adverse impact on the new fuel rack design.

The font will be changed on all pages, which is considered an editorial change and will not be discussed further.

A new criticality analysis has been performed to support this amendment request. A License Amendment Report was prepared by Holtec International to summarize the methodology,

assumptions, and results of the new analysis based on partial credit for soluble boron and other neutron absorbers, other than Boraflex. The report is included as Attachment 5 to this amendment request.

A boron dilution analysis was also prepared for the proposed change to demonstrate that an inadvertent dilution event would not reduce the SFP boron concentration to a value less than the minimum requirement of the criticality analysis. The conclusions of the boron dilution analysis are included in Attachment 5.

In summary, the proposed change will allow the use of several defined 2 x 2 loading patterns in the ANO-2 SFP, while taking credit for boron, as a means of maintaining SFP reactivity within the k_{eff} limits of 10 CFR 50.68.

Changes are also proposed to the TS bases which are included for information only in Attachment 3.

3.0 BACKGROUND

The proposed change is requested to allow an alternate loading pattern for fresh and spent fuel assemblies in the ANO-2 SFP. Currently, storage of fuel assemblies in the SFP is governed by ANO-2 Technical Specifications (TS) 3.9.12.a, 3.9.12.b, 3.9.12.c, 5.3.1, 5.3.2, and 5.3.4. As is depicted in TS Figure 3.9.1, the ANO-2 SFP is divided into two regions. Region 1 contains Boraflex, while Region 2 has no poison panel inserts. The proposed change will eliminate reliance on Boraflex and take credit for soluble boron in the SFP water.

On June 26, 1996, the Nuclear Regulatory Commission office of Nuclear Reactor Regulation issued Generic Letter (GL) 96-04, *Boraflex Degradation in Spent Fuel Pool Storage Racks*. The GL addressed issues concerning the use of Boraflex in SFP storage racks and requested an assessment of the capability of the Boraflex to maintain a 5 percent subcriticality margin and the submittal of a plan describing the utility's proposed actions if the subcriticality margin cannot be maintained by the Boraflex material because of current or projected future Boraflex degradation. ANO responded to the GL by letters dated October 24, 1996 (Arkansas Nuclear One – Units 1 and 2, 120 Day Response to Generic Letter 96-04 (0CAN109605)), January 29, 1998 (Arkansas Nuclear One- Unit 1, Revised Commitment Made in Response to Generic Letter 96-04 (1CAN019803)), and March 18, 2002 (Degradation of Boraflex in ANO-1 Spent Fuel Pool (1CAN030203)).

In the original response to GL 96-04 (letter dated October 24, 1996), Entergy outlined ANO's Boraflex monitoring program. The purpose of the monitoring program is to ensure that the 5 percent subcriticality margin can be maintained for the life of the spent fuel storage racks. The program provides an allowance should an assessment indicate that the 5 percent subcriticality margin cannot be maintained, as follows:

> "In the event this assessment determines the 5 percent subcriticality margin cannot be maintained, immediate steps will be taken to maintain soluble boron levels to insure this margin is sustained. Additionally, analyses could be performed to justify continued use of the racks in unborated water based on more realistic assumptions such as credit for integral poisons, higher burnup requirements, or additional restrictions on fuel storage configurations."

Section 9.1.2.3 of the ANO-2 Safety Analysis Report (SAR) describes the assumptions used to perform the criticality analysis for the spent fuel storage racks. For Region 1, credit was taken for the minimum design area Boron-10 (B-10) loading in the Boraflex poison panels. In March 2003 an analysis of the Boraflex was performed that indicated the Boraflex content had dropped below the analysis assumptions. The ANO-2 SFP Region 1 racks are considered operable but degraded as allowed by Generic Letter 91-18, *"Information To Licensees Regarding Two NRC Inspection Manual Sections On Resolution of Degraded and Nonconforming Conditions and On Operability."* The degraded condition has been documented in the plant corrective action program. Entergy is currently administratively controlling fuel storage configurations in Region 1. This is consistent with the actions presented in our initial response to GL 96-04.

By letter dated January 29, 2003 (License Amendment Request to Change the Spent Fuel Pool (SFP) Loading Restrictions (2CAN010304)), Entergy submitted a request to modify a portion of the Region 2 SFP racks by inserting Metamic poison panels. The proposed change took no credit for Boraflex, created spent fuel loading restrictions in Regions 1 and 2, and took credit for soluble boron. Due to unexpected manufacturing challenges, the Metamic poison panels have not been produced. Thus, in order to accommodate the known degradation of Boraflex and the ability to perform a complete core offload, an alternate loading pattern for the ANO-2 SFP is proposed.

3.1 Spent Fuel Pool Racks

The ANO-2 SFP provides 988 storage locations for spent fuel assemblies or other items (e.g. incore detectors) which require long term submerged storage. The racks are comprised of twelve free standing structures. Four modules contain 81 fuel storage locations each in a 9x9 array; four modules contain 90 fuel storage locations each in a 9x10 array; two modules contain 80 fuel storage locations in an 8x10 array; and, two modules contain 72 fuel storage locations in an 8x9 array. The SFP is lined with type 304L stainless steel and is designed to seismic Category I criteria.

Each fuel storage module is made up of rectangular storage cells which are capable of accepting one fuel assembly. The cells are open at the top and bottom to provide a flow path for convective cooling of spent fuel assemblies through natural circulation. The fuel storage cells are structurally connected to form storage modules which provide the assurance that the required minimum fuel assembly spacing is maintained for all design conditions including a design basis earthquake (DBE).

All welded construction is used in fabrication of the fuel storage cells and in the interconnection of cells to form modules. The fuel storage modules are constructed of type 304 stainless steel. The welded construction ensures the structural integrity of the storage modules and provides assurance of smooth snag-free paths in the storage cells so that it is highly improbable that a fuel assembly could become stuck in the racks.

ANO-2 SAR Section 9.1.3 contains a detailed description of the ANO-2 SFP, the associated structural and seismic considerations, the fuel rack structural analysis, the pool structural analysis, the criticality analysis for Regions 1 and 2, the postulated spent fuel storage criticality analysis, and the testing and inspection requirements. Based on approval of the proposed changes, the appropriate section of the SAR will be revised.

TS 3.9.12.b currently defines two storage regions (Region 1 and Region 2), each of which has specified loading restrictions based on assembly average burnup in GWD/MTU and the initial assembly average U-235 loading per unit length (g/inch). TS Figure 3.9.2 was initially included in the ANO-2 TSs with the approval of TS Amendment 43 (NRC Safety Evaluation Report (SER) dated April 15, 1983). The amendment allowed modification of the SFP racks which resulted in increased storage capacity in the pool and the addition of Region 1 which contains Boraflex.

TS Figure 3.9.2 was subsequently modified with the approval of TS Amendment 178 (NRC SER dated January 14, 1997). This amendment allowed an increase in the initial fuel enrichment from 4.1 weight percent (w/o) to 5.0 w/o. The criticality analysis performed based on the higher initial enrichment resulted in changes to TS Figure 3.9.2. TS 3.9.12.a. was also modified.

TS Amendment 224 (NRC SER dated October 24, 2000) allowed an alternate storage configuration of fuel assemblies adjacent to the walls within Region 1 of the SFP provided they were less reactive than the area of the graph enclosed by Curve A on TS Figure 3.9.2. This change provided 17 additional storage locations. This allowance was incorporated with the insertion of Note 1.

The controls used in determining the storage location for new and irradiated fuel in the SFP are governed by procedure. The procedure currently contains guidelines pertaining to restricted and unrestricted fuel storage as reflected by TS Figure 3.9.2. The new loading pattern restrictions will continue to be governed by procedure. Checkerboard storage configurations will be procedurally controlled with the vacant spaces administratively controlled by procedure.

3.2 Spent Fuel Pool System

The fuel pool system is designed to:

- maintain the pool temperatures less than or equal to approximately 150°F during a full core discharge. The cooling system's heat removal capacity is a function of service water temperature. Refueling operations are administratively controlled in order to minimize the potential of exceeding a pool temperature of 150°F during a full core discharge whenever service water system temperature is elevated.

- maintain purity and optical clarity of the fuel pool water.

- maintain purity of the water in the refueling cavity and in the refueling water tank.

- maintain the water level a minimum of 9.5 feet above the top of the active fuel during fuel handling and storage operations.

The cooling portion of the fuel pool system is a closed loop system consisting of two half-capacity pumps for normal duty and one full-capacity heat exchanger. The fuel pool water is drawn from the fuel pool near the surface and is circulated by the fuel pool pumps through the fuel pool heat exchanger where heat is rejected to the service water system. From the outlet of the fuel pool heat exchanger, the cooled fuel pool water is returned to the top of the fuel pool via a distribution header at the end of the pool opposite from the intake.

The clarity and purity of the water in the fuel pool, refueling cavity, and refueling water tank are maintained by the purification portion of the fuel pool system. The purification loop consists of the fuel pool purification pump, ion exchanger, filters, and strainers. The purification flow is drawn from the bottom of the fuel pool. A basket strainer is provided in the purification line to the pump suction to remove any relatively large particulate matter. The fuel pool water is circulated by the pump through a filter which removes particulates and through an ion exchanger to remove ionic material. Connections to the refueling water tank and refueling water cavity are provided for purification and makeup.

Makeup to the fuel pool is provided from the Chemical and Volume Control System via the blending tee, the refueling water tank via the purification pumps, or the Boron Management System (BMS) holdup tanks if chemistry specifications are met. In an emergency, Seismic Category I makeup is available from either service water system loop. The boric acid makeup tanks are also available for boration of the SFP. Overflow protection is provided by transferring the fuel pool water to the refueling water tank or one of the BMS holdup tanks via the purification pump on high level alarm.

A detailed description of the SFP system is included in Section 9.1.3 of the ANO-2 SAR. No modifications are proposed to the SFP system in order to support the proposed change.

4.0 TECHNICAL ANALYSIS

Attachment 5 of this letter provides a detailed technical analysis in support of the proposed changes. Below is a brief summary of the attachment.

4.1 Criticality Considerations

A criticality analysis was performed for the ANO-2 SFP racks and spent fuel currently in the SFP and the core. Five 2 x 2 fuel patterns and their interfacing relationship were established. These patterns and interfaces were analyzed as acceptable loading configurations to ensure k_{eff} is less than or equal to 0.95 within the storage racks when credit is taken for soluble boron. The analysis also demonstrated that k_{eff} is maintained less than 1.0 under the assumed accident of the loss of soluble boron in the pool water.

4.2 Accident Analysis

Temperature and water density effects and lateral rack movement were analyzed. The maximum bulk pool water temperature was included in the analysis as a bias in the calculation to assure the true reactivity will be lower over the expected range of water temperatures. The analysis evaluated the effect of lateral motion of the storage racks under postulated seismic conditions. Under these conditions credit will be taken for soluble boron.

The analysis also considered a dropped assembly; a misplaced assembly, between the racks and the SFP walls; and a misplaced assembly within any of the five 2 x 2 patterns. The most limiting of these is the misplaced assembly within one of the five patterns. Assuming this accident condition, a soluble boron concentration of at least 825 ppm is required to ensure k_{eff} remains less than 0.95.

A boron dilution evaluation was also performed. If a dilution event were to occur, sufficient time is available to take corrective actions that will assure a 5% criticality margin is maintained.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

By letter dated October 6, 1998, the Nuclear Regulatory Commission granted ANO-1 and ANO-2 an exemption from the requirements of 10 CFR 70.24, *"Criticality accident requirements."* With the approval of the proposed change this exemption is no longer required. Arkansas Nuclear One, Unit 2 (ANO-2) currently complies with the requirements of 10 CFR 50.68, *"Criticality accident requirements."* There are eight criteria that must be satisfied in the regulation. ANO-2 complies with these as follows.

1) 10 CFR 50.68, (b) (1) - Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

   ANO-2 Response - ANO-2's fuel handling procedures ensure that subcriticality is maintained in the reactor vessel and the SFP under the most adverse moderation conditions feasible by unborated water. Storage of fuel assemblies is procedurally controlled to assure k_{eff} remains below 1.0, at a 95% probability, 95% confidence level, when flooded with unborated water.

2) 10 CFR 50.68, (b) (2) - The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

   ANO-2 Response - Previously performed criticality calculations of the new fuel vault fully loaded with Combustion Engineering (CE) 16x16 fresh fuel assemblies and filled with the most reactive unborated water showed that reactivity did not exceed 0.95, at a 95% probability, 95% confidence level.

3) 10 CFR 50.68, (b) (3) - If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

   ANO-2 Response - Criticality calculations were performed on the new fuel vault fully loaded with CE 16x16 fresh fuel assemblies and filled with the most reactive low density

hydrogenous fluid. The results of these calculations showed that reactivity did not exceed 0.98, at a 95% probability, 95% confidence level.

4) 10 CFR 50.68, (b) (4) - If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

   ANO-2 Response - Soluble boron credit will be taken in the spent fuel pool storage racks. Reactivity will not exceed 0.95 at a 95% probability with a 95% confidence level when credit is taken for 240 ppm boron. The criticality calculations included in the proposed change show that k_{eff} remains below 1.0, at a 95% probability, 95% confidence level, when flooded with unborated water.

5) 10 CFR 50.68, (b) (5) - The quantity of SNM, other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

   ANO-2 Response - ANO does not currently have a quantity of SNM, other than the nuclear fuel, stored on site to ascertain a critical mass.

6) 10 CFR 50.68, (b) (6) - Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

   ANO-2 Response - Radiation monitors are located in the ANO-2 SFP area which alarm in the control room. When fuel movement is in progress fuel handling procedures require placement of additional radiation monitors directly on the fuel handling bridges to provide an additional audible indication of excessive radiation levels.

7) 10 CFR 50.68, (b) (7) - The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

   ANO-2 Response - Per ANO-2 TS 5.3.2 and 3.9.12.a, the maximum U-235 fuel enrichment limit is 5.0 weight percent including enrichment uncertainties. The proposed change will reduce the maximum U-235 enrichment to 4.55±0.05 wt% U-235.

8) 10 CFR 50.68, (b) (8) - The FSAR is amended no later than the next update which 50.71(e) of this part requires, indicating that the licensee has chosen to comply with 50.68(b).

   ANO-2 Response - The ANO-2 SAR will be amended no later than the next required update after the proposed TS change is approved and implemented. This SAR update will indicate that ANO-2 has chosen to comply with 50.68(b).

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any General

Design Criterion (GDC) differently than described in the Updated Final Safety Analysis Report (UFSAR.)

5.2 No Significant Hazards Consideration

The proposed change will modify the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TSs) to allow several defined 2 x 2 fuel loading patterns in the ANO-2 spent fuel pool (SFP). The loading patterns and their associated interfaces have been analyzed to assure a subcritical margin of five percent is maintained when credit is taken for soluble boron. The loading patterns and their associated interfaces have also been analyzed to assure the effective neutron multiplication factor (k_{eff}) remains below 1.0 (subcritical) with no credit for boron. The proposed change will also modify the current TS allowed value for fuel assembly initial enrichment and the minimum SFP soluble boron concentration.

Entergy Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) 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 or consequences of an accident previously evaluated?

   Response: No.

   The fuel handling accidents described below can be postulated to increase reactivity. However, for these accident conditions, the double contingency principle of ANS N16.1-1975 is applied. This states that it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since its absence would be a second unlikely event.

   Three types of drop accidents have been considered: a vertical drop accident, a horizontal drop accident, and an inadvertent drop of an assembly between the outside periphery of the rack and the pool wall.

   - A vertical drop directly upon a cell will cause damage to the racks in the active fuel region. The proposed 2000 ppm soluble boron concentration will ensure that K_{eff} does not exceed 0.95.
   - A fuel assembly dropped on top of the rack that comes to rest horizontally will not deform the rack structure such that criticality assumptions are invalidated. The rack structure is such that an assembly positioned horizontally on top of the rack results in a minimum separation distance from the upper end of the active fuel region of the stored assemblies. This distance is sufficient to preclude interaction between the dropped assembly and the stored fuel.
   - An inadvertent drop of an assembly between the outside periphery of the rack and the pool wall is bounded by the worst case fuel misplacement accident condition.

   The fuel assembly misplacement accident was considered for all storage configurations. An assembly with high reactivity is assumed to be placed in a storage location which

requires a fuel assembly with a lower reactivity. The presence of soluble boron in the pool water assumed in the analysis has been shown to offset the worst case reactivity effect of a misplaced fuel assembly for any configuration. This soluble boron requirement is less than the proposed 2000 ppm that will be required by the ANO-2 TS. Thus, a five percent subcriticality margin can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

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 will define several acceptable 2x2 loading patterns and acceptable interfaces between the patterns. In addition the proposed change will credit soluble boron to assure a five percent subcriticality margin is maintained during normal conditions and in the event of a postulated accident. The soluble boron concentration assumed in the analyses for a postulated accident is less than the proposed TS change of 2000 ppm. Thus, a five percent subcriticality margin can easily be met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

   No new or different types of fuel assembly drop scenarios are created by the proposed change. The presence of soluble boron in the SFP water assures a subcriticality margin is maintained in the event of fuel assembly misplacement.

   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.

   With the presence of a nominal boron concentration, the fuel storage patterns are designed to assure that fuel assemblies of less than or equal to 455±0.05 weight percent U-235 enrichment when loaded in accordance with the proposed loading patterns will be maintained within a subcritical array with a five percent subcritical margin (95% probability at the 95% confidence level). This has been verified by criticality analyses.

   Credit for soluble boron in the SFP water is permitted under accident conditions as well as in non-accident conditions. Criticality analyses have been performed to determine the required boron concentration that would ensure a subcriticality margin of at least five percent. By increasing the minimum boron concentration to greater than 2000 ppm, the margin of safety currently defined by taking credit for soluble boron will be maintained.

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

Based on the above, Entergy concludes that the proposed amendment present 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.

5.3 Environmental Considerations

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.

Attachment 2

2CAN060306

Proposed Technical Specification Changes (mark-up)

REFUELING OPERATIONS

FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12.a Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to ~~5.0~~4.55±0.05 w/o U-235. The provisions of Specification 3.0.3 are not applicable.

3.9.12.b Storage in ~~Region 1 or Region 2 (as shown on Figure 3.9.1) of~~ the spent fuel pool shall be further restricted by the limits specified in Figure 3.9.2. ~~In the event a cross-hatch storage configuration is deemed necessary for a portion of either Region 1 or Region 2, vacant spaces diagonal to the four corners of any fuel assembly or vacant spaces on two opposite faces of any fuel assembly shall be physically blocked before any such fuel assembly may be placed in that region (Note 1). Also, the Region 1 storage cells adjacent to the Region 2 interface are restricted to fuel assemblies that are outside of the area of the graph enclosed by Curve A on Figure 3.9.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the four faces of any fuel assembly shall be physically blocked before any such fuel assembly may be placed in Region 2.~~ The provisions of Specification 3.0.3 are not applicable.

3.9.12.c The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than ~~1600~~ 2000 parts per million.

APPLICABILITY: During storage of fuel in the spent fuel pool.

ACTION:

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined a fuel assembly has been placed in an incorrect location until such time as the correct storage location is determined. Move the assembly to its correct location before resumption of any other fuel movement.

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than ~~1601~~ 2001 ppm, until such time as the boron concentration is increased to ~~1601~~ 2001 ppm or greater.

SURVEILLANCE REQUIREMENTS

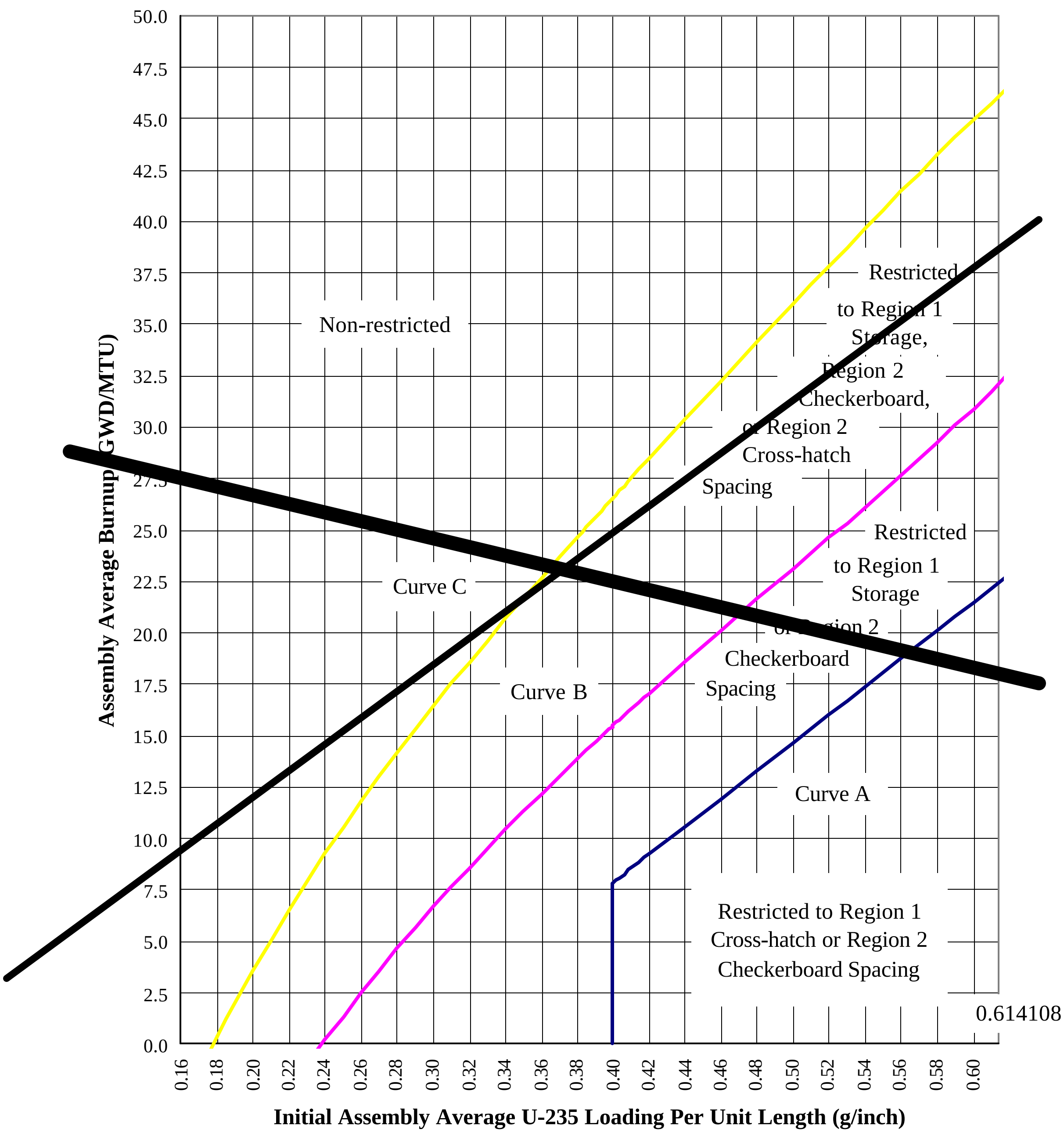
4.9.12.a Verify all fuel assemblies to be placed in the spent fuel pool have an initial enrichment of less than or equal to ~~5.0~~4.55±0.05 w/o U-235 by checking the assemblies' design documentation.

4.9.12.b Verify all fuel assemblies to be placed in the spent fuel pool are within the limits of Figure 3.9.2 by checking the assemblies' design and burnup documentation.

4.9.12.c Verify at least once per 31 days the spent fuel pool boron concentration is greater than ~~1600~~ 2000 ppm.

~~Note 1: If the most peripheral row/column of the Region I contains vacant spaces in a cross-hatch storage configuration, these vacant spaces may be filled with fuel assemblies that are outside of the area of the graph enclosed by Curve A on Figure 3.9.2, provided that the most southwest and southeast corner locations remain empty.~~

FIGURE 3.9.2
MINIMUM BURNUP VS. INITIAL ASSEMBLY AVERAGE U-235 LOADING



~~Curve A = 68.008x-19.366 when x > 0.399181~~
~~Curve B = 239.01x^3-347.75x^2+243.18x-41.452~~
~~Curve C = -714.35x^4+1335.80x^3-946.44x^2+394.52x-47.040~~

Figure 3.9-2

Pattern 1

BC or A	BC or A
BC or A	BC or A

Pattern 2

F	X
X	F

Pattern 3

F	CC
CC	CC

Pattern 4

Fc	C
C	Fc

Pattern 5

B	C
C	C

Legend:

1. Type "F" – fresh fuel assemblies with an initial enrichment less than or equal to 4.55±0.05 wt%.
2. Type "B" – an assembly with a maximum initial enrichment of 4.1 wt%, a minimum burnup of 11.3 GWD/MTU and zero years cooling time. Typically "B" type assemblies have only one cycle of burnup.
3. "Fc" – fresh fuel assemblies with an initial enrichment of less than or equal to 4.55±0.05 wt% with a CEA inserted. The CEA must be of the new design.
4. "A" – an assembly that meets specific burnup requirements that are included in Table 1 below.
5. "BC" – a "B" type assembly with a CEA inserted. This CEA may be of the old or new design.
6. "C" – a high burnup assembly that meets specific burnup requirements that are included in Table 2 below. Initial enrichment is less than or equal to 4.50±0.05 wt%.
7. "AC" – an "A" type assembly with a CEA inserted. This CEA may be of the old or new design.
8. "CC" – a "C" type assembly with a CEA inserted. This CEA may be of the old or new design. Initial enrichment is less than or equal to 4.50±0.05 wt%.
9. "X" – a water cell with no fuel.

Note: Any less restrictive assembly may be substituted for another assembly within the pattern. Fuel assembly types are ranked 1 through 9, with the highest reactivity ranking being 1.

Figure 3.9-2

Allowable Pattern Interfaces

Pattern 1

BC or A	BC or A
BC or A	BC or A

Pattern 2

F	X
X	F

Pattern 3, Alternate

CC	F
CC	CC

Pattern 4

Fc	C
C	Fc

Pattern 5

B	C
C	C

Figure 3.9-2

Allowable Pattern Interfaces

Pattern 2 Alternate

X	F
F	X

Pattern 3, Alternate

CC	CC
F	CC

Pattern 3, Alternate

CC	F
CC	CC

Pattern 4

Fc	C
C	Fc

Pattern 4, Alternate

C	Fc
Fc	C

Pattern 5

B	C
C	C

Pattern 5, Alternate

C	C
B	C

Pattern 5, Alternate

C	B
C	C

Figure 3.9-2

Allowable Pattern Interfaces

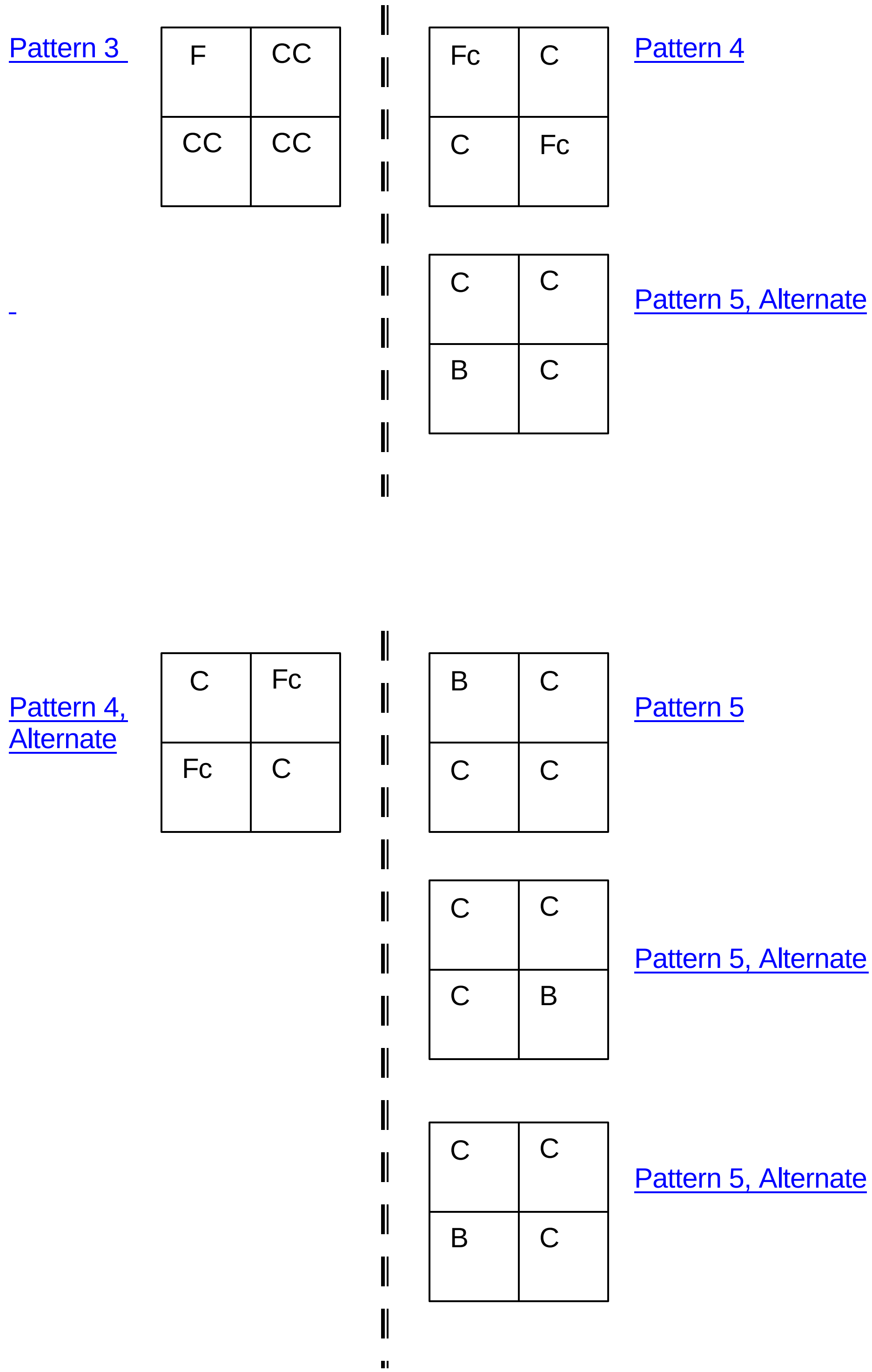


Figure 3.9-2

Table 1
Minimum Burnup Requirements for “A” Assemblies at Varying Enrichment and Cooling Time

BURNUP, MWD/KgU					
Average Enrichment, wt% U-235	0 Years Cooling Time	5 Years Cooling Time	10 Years Cooling Time	15 Years Cooling Time	20 Years Cooling Time
2	7.56	7.56	7.56	7.56	7.56
2.5	14.94	14.21	13.47	12.74	12.00
3.0	21.30	20.30	19.30	18.30	17.30
3.5	27.10	25.95	24.80	23.65	22.50
4.0	32.67	31.38	30.09	28.79	27.50
4.5	39.30	37.73	36.15	34.58	33.00
4.95	45.00	43.33	41.65	39.98	38.30

Table 1-1

Bounding Polynomial Fits to Determine Minimum Acceptable Burnup for “A” Type Assemblies Storage as a Function of Initial Enrichment

Decay Time, Years	Burnup, MWD/KgU
0	$0.68 * E^3 – 7.449 * E^2 + 38.56 * E - 45.20$
5	$0.5489 * E^3 – 5.9344 * E^2 + 32.496 * E - 38.05$
10	$0.4153 * E^3 – 4.3948 * E^2 + 26.356 * E – 30.75$
15	$0.2867 * E^3 – 2.9045 * E^2 + 20.367 * E – 23.80$
20	$0.153 * E^3 – 1.3649 * E^2 + 14.227 * E – 16.60$

Note: E = Initial average enrichment in wt% U-235

Figure 3.9-2

Table 2
Minimum Burnup Requirements for "C" Assemblies at Varying Enrichment and Cooling Time

BURNUP, MWD/KgU					
Average Enrichment, wt% U-235	0 Years Cooling Time	5 Years Cooling Time	10 Years Cooling Time	15 Years Cooling Time	20 Years Cooling Time
2	17.00	15.00	14.00	13.30	13.00
2.5	25.00	22.00	21.00	20.30	20.00
3.0	32.00	29.00	27.00	26.00	25.00
3.5	37.00	35.00	32.50	31.00	30.00
4.0	43.00	40.00	39.00	38.00	37.00
4.5	49.75	46.00	45.00	44.00	43.00

Table 2-1

Bounding Polynomial Fits to Determine Minimum Acceptable Burnup for "C" Type Assemblies Storage as a Function of Initial Enrichment

Decay Time, Years	Burnup, MWD/KgU
0	$1.3148 * E^3 - 13.552 * E^2 + 57.491 * E - 53.6$
5	$0.3704 * E^3 - 4.5397 * E^2 + 29.59 * E - 28.8$
10	$0.6667 * E^3 - 6.5714 * E^2 + 32.976 * E - 30.3$
15	$0.7111 * E^3 - 6.919 * E^2 + 33.634 * E - 31.3$
20	$0.8148 * E^3 - 7.7302 * E^2 + 35.169 * E - 32.0$

Note: E = Initial average enrichment in wt% U-235

5.3 Fuel Storage

5.3.1 Spent Fuel Storage Rack Criticality

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

a. Fuel assemblies stored in the spent fuel pool in accordance with Specification 3.9.12;

b. $k_{eff} \leq 0.95$ if fully flooded with 240 ppm ~~un~~borated water, which includes an allowance for uncertainties as described in Section 9.1 of the SAR; and

c. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the SAR; and

d. A nominal 9.8 inch center to center distance between fuel assemblies placed in the storage racks.

5.3.2 New Fuel Storage Rack Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of ~~5.0~~4.55±0.05 weight percent;

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

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

d. A nominal 26 inch center to center distance between fuel assemblies placed in the storage racks.

5.3.3 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 399' 10½".

5.3.4 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 988 fuel assemblies.

Attachment 3

2CAN060306

Changes to Technical Specification Bases Pages
For Information Only

3/4.9.9 and 3/4.9.10 WATER LEVEL-REACTOR VESSEL AND SPENT FUEL POOL WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 12% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 FUEL HANDLING AREA VENTILATION SYSTEM

The limitations on the fuel handling area ventilation system ensure that all radioactive materials released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The operation of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

Acceptable removal efficiency is shown by methyl iodide penetration of less than 5.0% when tests are performed in accordance with ASTM D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon," at a temperature of 30°C and a relative humidity of 95%. The penetration acceptance criterion is determined by the following equation:

$$\text{Allowable Penetration} = \frac{[100\% - \text{methyl iodide efficiency for charcoal credited in accident analysis}]}{\text{safety factor of 2}}$$

Applying a safety factor of 2 is acceptable because ASTM D3803-1989 is a more accurate and demanding test than older tests.

3/4.9.12 FUEL STORAGE

~~Region 1 and Region 2 of the~~ The spent fuel storage racks are designed to assure that when fuel assemblies of less than or equal to ~~5.0~~4.55±0.05 w/o U-235 enrichment ~~that~~ are stored ~~are~~ within the limits of Figure 3.9.2 ~~will be maintained in~~ a subcritical array with $K_{eff} \leq 0.95$ will be maintained when a concentration of 240 ppm of soluble boron is present in the spent fuel pool water.~~in unborated water.~~ These conditions have been verified by criticality analyses.

The requirement for ~~1600~~ 2000 ppm boron concentration is to assure the fuel assemblies will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in the event of a postulated accident. Analysis has shown that, during a postulated accident with the fuel stored within the limits of this specification, that a K_{eff} of ≤ 0.95 will be maintained when the boron concentration is at or above ~~1000~~ 825 ppm.

~~Normally, fuel stored in a cross-hatch storage configuration must have all four diagonal spaces or at least two adjacent faces remain vacant to meet the criticality safety analysis mentioned above. However, the spent fuel pool walls may be credited as a neutron leakage path. Therefore, vacant spaces face adjacent to the walls of the Region I cross-hatch configured assemblies may be used to store fuel assemblies that are outside of the area of the graph enclosed by Curve A on Figure 3.9.2, excluding the most southeast and southwest corner spaces of Region 1 which must remain empty.~~
The following restrictions apply to fuel assembly storage in the SFP:

Racks must be loaded using one of the five patterns shown in Figure 3.9-12. The type "A" and "C" assemblies have restrictions which are designated in Tables 1, 1-1, 2, and 2-1 of Figure 3.9-2. In any pattern, a less restrictive assembly may be substituted for another assembly within the pattern.

For adjacent patterns, both inside and between racks in the SFP, only the allowed interface configurations shown in Figure 3.9-2 may be used.

A minimum soluble boron concentration of 825 ppm must be maintained in the SFP to ensure that K_{eff} is less than or equal to 0.95 under all credible accident conditions.

Attachment 4

2CAN060306

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The ANO-2 SAR will be amended no later than the next required update after the proposed TS change is approved and implemented. This SAR update will indicate that ANO-2 has chosen to comply with 50.68(b).	x		6 months after the end of the ANO-2 fall 2003 refueling outage.

Attachment 5

2CAN060306

Holtec License Amendment Report

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1.0 INTRODUCTION

Arkansas Nuclear One, Unit 2 (ANO-2) operated by Entergy Operations and in commercial operation since 1980, is located approximately 70 miles northwest of Little Rock, Arkansas and about five miles west of Russellville, Arkansas. ANO-2 is a Combustion Engineering pressurized water reactor (PWR) with a licensed thermal power level of 3026 megawatts. The reactor core contains 177 fuel assemblies and the spent fuel pool (SFP) is licensed for the storage of 988 fuel assemblies.

The racks in the SFP of ANO-2 are free-standing and self supporting racks of Westinghouse design. The principal fabrication materials are ASTM A-240, Type 304 stainless steel for the structural members and shapes. "Boraflex", a product of BISCO (a division of Brand, Inc.) was originally used to augment reactivity control.

The SFP was designed to hold spent fuel assemblies (or rod cluster control assemblies) in underwater storage for long-term decay after their removal from the reactor core. The structure is seismic Category I, heavy walled, reinforced concrete pool, located on grade outside the containment structure. The interior of the pool is lined with stainless steel plate (Type 304L).

This report documents the criticality safety evaluation for the ANO Unit 2 Spent Fuel Pool (SFP) to support storage of spent and fresh fuel assemblies in various defined patterns in the racks. The ANO Unit 2 SFP contains both Region 1 and Region 2 style storage racks. Although the Region 1 racks contain Boraflex poison material, analysis has indicated that Boraflex has degraded below the original analytical assumptions. This evaluation is performed to qualify the spent fuel and fresh fuel storage in the racks from a criticality perspective taking no credit for the Boraflex neutron poison absorbers.

This analysis is used to qualify the existing racks for the existing spent fuel currently in the pool and in the core. Additionally, fresh fuel assemblies must be able to be stored in the pool racks during the reload. Credit is taken for the presence of soluble boron in the spent fuel pool, fuel burnup, cooling time and for the presence of control element assemblies (CEAs) in selected fuel assemblies. In order to achieve this objective, it is necessary to establish different patterns for assemblies with different reactivities. Five different fuel storage configurations (patterns) were established for use during the upcoming fall 2003 refueling outage and following refueling outages. Figure 1.1 shows each loading configuration pattern and Table 1.1 provides a description of each of the assembly types with its reactivity ranking. The significance of the reactivity ranking is to determine which assemblies can be replaced with other assembly types in a given pattern. As a rule, only a less reactive assembly may substitute a given assembly for a pattern to be valid (i.e. an assembly with a higher ranking can be substituted within a given pattern for an assembly with a lower ranking).

Pattern 1 is based on the Region 2 burnup/enrichment/cooling time curves generated when the “A” type assembly is used. Any less reactive assembly may be substituted for the “A” assemblies. Pattern 2 is a 2 x 2 Region 2 configuration with two fresh fuel assemblies and two empty cells. Pattern 3 is a 2 x 2 Region 2 configuration with one “F” assembly with 3 “C” type assemblies, which contain CEAs. Pattern 4 is a 2 x 2 Region 2

configuration with two fresh fuel assemblies, containing CEAs, and two “C” type assemblies. Pattern 5 is a 2 x 2 Region 2 configuration with one “B” assembly with 3 “C” type assemblies. The definitions for the “A” and “C” type spent fuel assemblies are determined using iterative processes, in terms of burnup, enrichment, and cooling time. The analyses performed for the Region 2 racks and the burnup requirements determined for the storage of spent fuel assemblies in the Region 2 type racks are also applicable to the Region 1 type racks.

The racks are evaluated for Combustion Engineering (CE) 16x16 fresh and spent fuel assemblies. Credit is taken for control elements (CEAs), fuel burnup, cooling time, and soluble boron in pool water as applicable per 10 CFR 50.68.

The objective of this analysis is to ensure that, per 10 CFR 50.68, the racks shall remain subcritical under normal conditions with no credit for soluble boron and less than or equal to 0.95 when partial credit is taken for soluble boron in the pool water, including calculation uncertainties and reactivity effects of mechanical tolerances. Reactivity effects of abnormal and accident conditions have also been evaluated to determine the required soluble boron concentration in the pool to assure that under all credible abnormal and accident conditions, the reactivity will not exceed the regulatory limit of 0.95. The required soluble boron concentrations are summarized in Table 2.6. In this context "abnormal" refers to conditions, which may reasonably be expected to occur during the lifetime of the plant and "accident" refers to conditions, which are not expected to occur but nevertheless must be protected against. The double contingency principle of ANSI N-16.1-1975 and of the April 1978 NRC letter allows full credit for soluble boron under other abnormal or accident conditions, since only a single independent accident need be considered at one time.

Applicable codes, standards, and regulations or pertinent sections thereof, include the following:

- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, “Prevention of Criticality in Fuel Storage and Handling.”

- Code of Federal Regulation 10CFR50.68, Criticality Accident Requirements.

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3 - July 1981.

- 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.

- L. Kopp, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.

- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December 1981.

- ANSI ANS-8.17-1984, “Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.”

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

- The Region 2 type racks, which are more reactive than the Region 1 racks, are used in the analyses. The burnup requirements determined the storage of spent fuel assemblies in the Region 2 type racks will also be applicable to the Region 1 type racks.

- The effective multiplication factor (k_{eff}) of an infinite radial array of fuel assemblies or assembly patterns was used in the analyses, except for the assessment of peripheral and interface effects, and for certain abnormal/accident conditions where neutron leakage is inherent.

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

- Because the temperature coefficient of reactivity is positive in the absence of neutron absorber panels, the analyses assumed a temperature of 150 °F. This is the design basis maximum pool water temperature. Higher temperatures would be an accident condition for which full soluble boron credit is permitted and the reactivity effects would be mitigated by the presence of the large amounts of soluble boron in the pool water.

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

- In-core depletion calculations assume conservative operating conditions, highest average fuel and moderator temperature, and an allowance for the average soluble boron concentrations during in-core operations.

- Absorber rods present in some fuel assemblies are conservatively assumed to be fuel rods.

- The densities of Boron Carbide (B_4C) in the fuel rod and the Silver-Indium-Cadmium (Ag-In-Cd) tip absorbers are reduced by approximately 5% of nominal to conservatively account for burnout.

2.0 SUMMARY OF CRITICALITY ANALYSES FOR NORMAL OPERATING CONDITIONS

The criticality analyses for each of the five separate storage patterns are summarized in Table 2.1. For the acceptance criteria defined in the previous section, the maximum effective multiplication factor (k_{eff}) values are shown to be less than 1.0 (95% probability at the 95% confidence level) in each of the regions when no credit is taken for the presence of soluble boron in the pool. Credit for soluble boron is required to ensure k_{eff} is maintained less than 0.95 and the required soluble boron concentrations are summarized in Table 2.6.

The maximum k_{eff} values for storage of fuel assemblies, in the five different patterns, were determined assuming an infinite radial array of storage cells with a finite axial length, water reflected. For each spent fuel cooling time and initial enrichment, minimum burnup values were determined that assure the maximum k_{eff}, including calculational and manufacturing uncertainties, remains subcritical under the assumed accident condition of the loss of all soluble boron. The calculated maximum reactivity includes the reactivity effect of the axial distribution in burnup and provides an additional margin of uncertainty for the calculations. Table 2.1 summarizes the results of these analyses at zero cooling time for fresh fuel assemblies with an initial enrichment of 4.55 ±0.05 wt% ^{235}U and spent fuel assemblies with an initial enrichment of 4.50 ±0.05 wt% ^{235}U. The results show that the five storage patterns are acceptable for fuel storage.

Table 2.2 shows the minimum acceptable burnup for the “A” type assemblies. Table 2.3 provides the bounding polynomial fits to determine minimum acceptable burnup for “A” type assemblies as a function of initial enrichment. Table 2.4 shows the minimum acceptable burnup for the “C” type assemblies. Table 2.5 provides the bounding polynomial fits to determine minimum acceptable burnup for “C” type assemblies as a function of initial enrichment. Fuel assemblies with initial enrichments less than 2.0 wt% ^{235}U will conservatively be required to meet the burnup requirements of 2.0 wt% ^{235}U assemblies, which are reflected in Tables 2.2 and 2.4. The minimum soluble boron concentration required to maintain k_{eff} below 0.95 for normal storage of fresh and spent fuel assemblies, including all manufacturing and calculational tolerances, for the storage of spent fuel in the racks is 240 ppm.

3.0 REFERENCE DESIGN INPUT DATA

3.1 Reference Fuel Assembly Design

The spent fuel storage racks are designed to accommodate Combustion Engineering (CE) 16x16 fuel assemblies. The design specifications for the CE fuel assemblies, as used for this analysis, are given in Table 3.1.

3.2 Region 1 Fuel Storage Cells

Figure 3.1 shows the Region 1 spent fuel storage cell. The Region 1 storage cells are composed of stainless steel boxes separated by a gap. The 0.075±0.0040 inch thick steel walls define the storage cells, which have a 8.58 +0.050/-0.025 inch nominal inside dimension. A 0.020 inch stainless steel sheath is around the gap and defines the boundary of the flux-trap water-gap used to augment reactivity control. The cells are located on a lattice spacing of 9.80 inches in both directions. Stainless steel channels connect the storage cells in a rigid structure and define the flux-trap of 0.806 ±0.010 inches, between the sheathing of adjacent cells.

3.3 Region 2 Fuel Storage Cells

Figure 3.2 shows the Region 2 spent fuel storage cell. The Region 2 storage cells area also have a flux trap between adjacent cells and are composed of stainless steel boxes separated by a gap. The straight portion of the flux trap is 7.2 inches. The measured flux trap water gap of 0.97 +0.12/-0.08 inches was used in the analysis. The 0.075 ±0.0040 thick steel walls define the storage cells, which have a 8.58 +0.050/-0.025 inch nominal inside dimension. The measured value of the flux trap water gap corresponded to a Box ID of 8.68 inches because of the bow in the cell walls. This value of the Box ID was used in the analysis. The cells are located on a lattice spacing of 9.80 inches in both directions. No additional water gaps exist between adjacent Region 2 cells in a rack.

4.0 ANALYTICAL METHODOLOGY

4.1 Reference Design Calculations

The principal methods for the criticality analyses of the storage racks include the following codes: (1) MCNP4a [1], and (2) CASMO-4 [2-4]. MCNP4a is a continuous energy three-dimensional Monte Carlo code developed at the Los Alamos National Laboratory. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009 with an uncertainty of ±0.0011 for MCNP4a evaluated with the 95% probability at the 95% confidence level [5].

Fuel depletion analyses during core operation were performed with CASMO-4, a two-dimensional multi-group transport theory code based on capture probabilities [2-4]. Restarting the CASMO-4 calculations in the storage rack geometry yields the two-dimensional infinite multiplication factor (k_∞) for the storage rack. CASMO-4 was also used to determine the reactivity uncertainties (differential calculations) of manufacturing tolerances and the reactivity effects of variations in the water temperature and density.

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. Three-dimensional MCNP calculations were necessary to describe the geometry of the 2 x 2 cases. However, MCNP cannot perform depletion calculations, thus depletion calculations were performed with CASMO-4. Explicit description of the fission product nuclide concentrations in the spent fuel was determined from the CASMO-4 calculations and

used in the MCNP calculations. This methodology explicitly incorporates approximately 40 of the most important fission products, accounting for all but about 1% in k. The remaining ~1% in k was accounted for in one of two ways. For patterns 1 and 2, to compensate for the few fission product nuclides that are not in the MCNP library, an equivalent boron-10 concentration in the fuel was determined which produced the same reactivity in MCNP as the CASMO-4 result. For patterns 3, 4 and 5, appropriate cross-sections were used in MCNP to produce the same reactivity effect as in the results generated in CASMO-4.

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, the final calculations use a minimum of 10,000 histories per cycle, a minimum of 25 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).

4.2 Fuel Burnup Calculations and Uncertainties

CASMO-4 was used for burnup calculations in the hot operating condition. To the extent possible, CASMO-4 has been benchmarked [3, 4] against cold, clean, critical experiments (including plutonium-bearing fuel) and also by comparison with Monte Carlo calculations.

In the CASMO-4 geometric models, each fuel rod and its cladding were described explicitly and reflective boundary conditions were used in the axial direction and between storage cells. These boundary conditions have the effect of creating an infinite array of storage cells in both the radial and axial directions.

Conservatively bounding moderator and fuel temperatures and the average core operating soluble boron concentrations (900ppm) were used to assure the highest plutonium production and hence conservatively high values of reactivity. Since critical experiment data with spent fuel is not available for determining the uncertainty in calculations, an allowance for uncertainty in reactivity was assigned based upon other considerations [6]. Calculations are performed to determine uncertainties for depletion for each of patterns 1, 3, 4, and 5. The depletion uncertainty is determined using two MCNP criticality calculations for each pattern. The reference case for each pattern is the case with the highest minimum spent fuel burnup requirement, which is the 4.50% initial enrichment , 0 year cooling time 49.75 GWD/MTU case. Then a new calculation is run where every assembly in the pattern is modeled as a fresh assembly ("F" or "Fc"). The assemblies that contain CEAs in the reference case will keep the CEAs in the depletion (fresh assembly) case. The depletion uncertainty is equal to 5% of the Δk_{eff}.

4.3 Effect of 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 regions. At high burnups, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of high neutron leakage. Consequently, it is expected that over most of the burnup history, fuel assemblies with distributed burnups will exhibit a slightly lower reactivity than that calculated for the uniform 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.

Among others, Turner [7] has provided generic analytic results of the axial burnup effect based upon calculated and measured axial burnup distributions. These analyses confirm the minor and generally negative reactivity effect of the axially distributed burnups at values less than about 30 GWD/MTU with small positive reactivity effects at higher burnup values. Since the required burnup for some enrichments and cases is greater than 30 GWD/MTU, the reactivity effect of the axially distributed burnup must be considered. These calculations were performed in MCNP4a with 10 zone axial calculations, using specific (CASMO-4) concentrations of actinides and fission product nuclides in each zone. For all patterns, the burnup level of higher reactivity was determined and the more conservative profile (flat or distributed) was used. Calculations were performed based upon a burnup distribution provided by ANO.

4.4 MCNP4a Temperature Correction

The reactivity for non-poisoned racks in the spent fuel pool increases with pool water temperature. The maximum bulk pool water temperature is administratively controlled at ANO Unit 2 as 150 °F. However, since the Doppler treatment and cross-sections in MCNP4a are valid only at 20 °C, the Δk determined in CASMO-4 calculations from 20 °C to 150 °F is included as a bias in the final k_{eff} calculation.

4.5 Long-Term Changes in Reactivity

At reactor shutdown, the reactivity of the fuel initially decreases due to the growth of Xe-135. Subsequently, the Xenon decays and the reactivity increases to a maximum at about a hundred hours after the Xenon has decayed. Therefore, for conservatism, the Xenon is set to zero in the calculations to assure maximum reactivity. During the next 50 years, the reactivity continuously decreases due primarily to ^{241}Pu decay and ^{241}Am growth. Credit for this decay and for changes in fission product concentration is included in calculations of the decrease in reactivity in long term storage (up to 20 years). The CASMO-4 code includes the capability of tracking the decay of the actinides and the most significant fission product nuclides during long term storage.

5.0 CRITICALITY ANALYSES

5.1 Nominal Design Case

For the nominal storage cell design in Region 2, the criticality safety analyses for the five different storage patterns are summarized in Table 2.1. These data confirm that the maximum reactivity in all five storage patterns remain subcritical (less than the regulatory limit $k_{eff} < 1.0$) under the assumed condition of the loss of all soluble boron in the pool water. Tables 2.2 and 2.4 show the limiting burnup values for fuel of other enrichments and cooling times.

5.2 Uncertainties Due to Tolerances

A listing of the tolerance values is given in Table 5.1. Base calculations are performed using nominal values of the rack dimensions and fuel density and initial enrichment of the fuel assemblies. The reactivity effects of the tolerances in fuel and rack dimensions and the fuel initial enrichments are determined in the analysis by performing sensitivity studies of the reactivity effects of the tolerance variations.

The reactivity effects of the rack manufacturing and fuel density tolerance uncertainties, except depletion tolerance, for pattern 1 and pattern 2 are shown in Table 5.2a. Table 5.2b shows the reactivity effects of fuel enrichment tolerance for pattern 1. The temperature bias was also calculated in CASMO-4 for patterns 1 and 2 and is shown in Table 5.2c.

Table 5.3a shows the reactivity effects of the rack manufacturing and fuel density tolerances for patterns 3, 4, and 5, which are statistically combined. Table 5.3b shows the reactivity effects of fuel enrichment tolerance for patterns 1, 3, 4 and 5. The temperature bias was also calculated in CASMO-4 for patterns 3, 4, and 5 and is shown in Table 5.3c.

Each of the reactivity effects shown in Table 5.2a-5.2c and 5.3a-5.3c is calculated, one at a time, with CASMO-4 by setting each tolerance at its most reactive value in a parametric study. All of the individual reactivity uncertainties are then combined using the standard conventional statistical methodology, i.e., square root of the sum of the squares of the independent uncertainties. The maximum k_{eff} for a storage pattern is then determined using the following formula:

$$\text{Maximum } k_{eff} = k_{eff}\text{ (reference)} + \text{biases} + [\Sigma i\,(UNC_i)^2]^{1/2}$$

where "UNC" values are the reactivity effects of individual uncertainties, each evaluated at the 95% probability, 95% confidence level or greater. Since each of the reactivity uncertainties are independently calculated at the 95% probability/ 95% confidence level (or greater), the final maximum k_{eff} is at the 95% probability/ 95% confidence level, in accordance with the specifications of the 1978 Grimes letter and the 1998 Kopp memorandum. The reactivity effects of the tolerances are also tabulated in Table 2.1.

5.3 Eccentric Fuel Positioning

The fuel assemblies are assumed to be normally located in the center of the storage rack cell. However, for determining the reactivity effects of the eccentric positioning of the fuel assemblies, calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). The k_{eff} of each pattern's eccentric positioning analysis is compared to the k_{eff} of the corresponding reference case and the uncertainty is set equal to the difference between the two. The uncertainty for eccentricity is included in the calculations for the final k_{eff} in Table 2.1.

5.4 Interface Calculations

In these evaluations, interfaces between two different storage patterns placed adjacent to each other within a rack are analyzed. This conservatively bounds rack-to-rack interfaces for acceptable configurations since no credit is taken for the gap between racks. The interface configurations that are deemed unacceptable for storage within a rack will also be considered unacceptable for adjacent storage patterns between racks. Figures 5.1a to 5.1w show each of the analyzed configurations.

6.0 ABNORMAL AND ACCIDENT CONDITIONS IN THE SPENT FUEL POOL RACKS

6.1 Temperature and Water Density Effects

The moderator temperature coefficient of reactivity in both Region 1 and Region 2 is positive. Therefore, a moderator temperature of 20 °C (68 °F) was assumed for the reference MCNP4a calculations and the increase in reactivity to the maximum bulk pool water temperature of 150 °F is included (CASMO-4 calculation) as a bias in the calculation of the maximum k_{eff}. This assures that the true reactivity will always be lower over the expected range of water temperatures. The reactivity effects of the pool water temperature effects on reactivity have been evaluated using CASMO-4.

6.2 Lateral Rack Movement

Lateral motion of the storage racks under postulated seismic conditions could potentially alter the spacing between racks. Under these conditions, credit for the soluble boron (permitted under accident conditions) would maintain the k_{eff} at a value well below the maximum allowable. Nevertheless, the separation (water-gap) between rack modules is sufficiently large that even for the maximum movement expected under seismic excitation, the water gap remains larger than the water gap within the Region 1 modules. In the Region 2 racks, the k_{eff} is independent of the inter-module water gap and is not sensitive to any potential seismic induced movement of the modules. The water gap structure in each cell is included in the analysis and precludes any closer proximity between modules.

6.3 Abnormal Location of a Fuel Assembly

The misplacement of a fresh unburned fuel assembly of the highest permissible reactivity could, in the absence of soluble poison, result in exceeding the regulatory limit (k_{eff}< 1.0). This could occur if a fresh fuel assembly of the highest permissible initial enrichment (4.55 ±0.05 wt% ^{235}U) were to be inadvertently loaded into a Region 1 or Region 2 storage cell, which is intended to store spent fuel assemblies or remain empty. Calculations confirmed that the highest reactivity, including uncertainties, for the worst case postulated accident condition (fresh fuel assembly in Region 2 cell intended to remain empty) would exceed the limit on reactivity in the absence of soluble boron. Soluble boron in the spent fuel pool water, for which credit is permitted under these accident conditions, would assure that the reactivity is maintained substantially less than the design limitation. Calculations indicate that a soluble boron concentration of 825 ppm is adequate, under accident conditions, to assure that the maximum k_{eff} does not exceed 0.95. Proposed ANO-2 Technical Specifications will require that a concentration of at least 2000 ppm boron is maintained in the SFP.

There is also no opportunity for an accidental drop of a fuel assembly outside and adjacent to the racks in the rack-to-rack gaps because there is insufficient clearance between the racks. Accidental placement of a fuel assembly between the racks and the spent fuel pool walls is bounded by the internal misplacement of fuel assemblies. This is an area of high neutron leakage and the reactivity effect would be bounded by that of a fuel assembly accidentally mis-loaded internal to a rack.

6.4 Dropped Fuel Assembly

For the case in which a fuel assembly is assumed to be dropped on top of a rack and the fuel assembly comes to rest horizontally on top of the rack the minimum separation distance between the active fuel regions is more than 12 inches, which is sufficient to preclude neutron coupling (with fuel in the storage rack). Consequently, the horizontal fuel assembly drop accident will not result in a significant increase in reactivity. Furthermore, the soluble boron in the spent fuel pool water assures that the true reactivity is always less than the limiting value for this dropped fuel accident.

It is also possible to vertically drop an assembly into a location occupied by another assembly. Such a vertical impact, would, at most cause a small compression of the stored assembly, reducing the water-to-fuel ratio and thereby reducing reactivity. In addition the distance between the active fuel regions of both assemblies will be more than sufficient to ensure no neutron interaction between the two assemblies.

Dropping of an assembly into an unoccupied cell could result in a localized deformation of the base plate of the rack. The immediate eight surrounding fuel cells could also be affected. However, the amount of deformation for these cells would be considerably less. The resultant effect would be the lowering of a few fuel assemblies in the area near the deformation. The presence of the required soluble Boron concentration of 2000 ppm in the pool water assures the maximum k_{eff} is well below the 0.95 acceptance criteria.

7.0 SOLUBLE BORON DILUTION EVALUATION

The soluble boron in the spent fuel pool water is normally a minimum of 2000 ppm under operating conditions. Significant loss or dilution of the soluble boron concentration is extremely unlikely, if not incredible. Nonetheless, an evaluation was performed based on the ANO-2 spent fuel pool data. The minimum required soluble boron concentration in the spent fuel pool water for various conditions are summarized in Table 2.6.

The required minimum soluble boron concentration is 240 ppm under normal conditions and 825 ppm for the most serious credible accident scenario. The volume of water in the pool is 199,200 gallons. Large amounts of unborated water would be necessary to reduce the boron concentration from 2000 ppm to 825 ppm or 240 ppm. Abnormal or accident conditions are discussed below for either low dilution rates (abnormal conditions) or high dilution rates (accident conditions). It should be noted that routine surveillances to measure the soluble boron concentrations in the pool water are required by Technical Specifications.

Small failures or mis-aligned valves could possibly occur in the normal soluble boron control system or related systems. Such failures might not be immediately detected. These flow rates would be of the order of 2 gpm (comparable to normal evaporative loss) and the increased frequency of makeup flow might not be observed. However, an assumed loss flow-rate of 2 gpm dilutions flow rate would require 146 days to reduce the boron concentration to the minimum required 240 ppm required under normal conditions or 61 days to reach the 825 ppm required for the most severe fuel handling accident. Routine surveillance measurements of the soluble boron concentration would readily detect the reduction in soluble boron concentration with ample time for corrective action.

Under certain accident conditions, it is conceivable that a high flow rate of unborated water could flow onto the top of the pool. Such an accident scenario could result from rupture of an unborated water supply line or possibly the rupture of a fire protection system header, both events potentially allowing unborated water to spray onto the pool. A flow rate of up to 2500 gpm could possibly flow onto the spent fuel pool as a result of a rupture of the fire protection line. This would be the most serious condition and bounds all other accident scenarios. Conservatively assuming that all the unborated water from the break poured onto the top of the pool and further assuming instantaneous mixing of the unborated water with the pool water, it would take approximately 169 minutes to dilute the soluble boron concentration to 240 ppm, which is the minimum required concentration to maintain k_{eff} below 0.95 under normally operating conditions. In this dilution accident, approximately 422,000 gallons of water would spill on the auxiliary building floor and into the air-conditioning duct system. Well before the spilling of such a large volume of water, multiple alarms would have alerted the control room of the accident consequences (including the fuel pool high-level alarm, the fire protection system pump operation alarm, and the floor drain receiving tank high level alarm). For this high flow rate condition, 71 minutes would be required to reach the 825 ppm required for the most severe fuel handling accident.

A flow rate of 9.5 gallons per minute (gpm) would be required to dilute the soluble boron concentration to 240 ppm in 31 days and a total volume loss of approximately 422,000 gallons. Administrative controls require a measurement of the soluble boron concentration in the pool water at least weekly. Thus, the longest time period that a

potential boron dilution might exist without a direct measurement of the boron concentration is 7 days. In this time period, an undetected dilution flow rate of 30.6 gpm would be required to reduce the boron concentration to 825 ppm. No known dilution rate of this magnitude has been identified. Furthermore, a total of more than 300,000 gallons of unborated water would be associated with this dilution event and such a large flow of unborated water would be readily evident by high-level alarms and by visual inspection during daily walk-downs of the storage pool area.

For the fire control line break, upon the initial break, the fire protection system header pressure would drop to the auto start setpoint of the fire protection pumps. The start is accompanied with an alarm in the main control room. The annunciator response is to dispatch an operator to find the source of the pump start. Approximately 5 minutes into the event, a Spent Fuel Pool high level alarm would be received in the main control room, assuming that the Spent Fuel Pool level started at the low alarm. The annunciator response for high Spent Fuel Pool level is to investigate the cause. The coincidence of the 2 alarms would quickly lead to the discovery of the failure of the fire protection system and sufficient time to isolate the failure.

The analysis assume that for a double-ended break in the fire control piping, the stream of water will arch through the air some 40 feet falling on top of the pool. This is virtually an incredible event. Should the stream of water fall upon the pool deck, a 3 inch high curb would channel some of the water to the pool drain and prevent all of the water from reaching the pool. Furthermore, the evaluation also assumes at least 3 independent and concurrent accidents occur simultaneously:

- Large amount of water flowing from the double-ended pipe break would remain un-detected and is ignored.

- Pool water high level alarms either fail or are ignored.

- Alarms indicating large amounts of water flowing into the floor drain have failed or are ignored.

It is not considered credible that multiple alarms would fail or be ignored or that the spilling of large volumes of water would not be observed. Therefore, such a major failure would be detected in sufficient time for corrective action to avoid violation of an administrative guideline and to assure that the health and safety of the public is protected.

The maximum flow rate for a failure of the demineralized water header would provide approximately 400 gpm into the Spent Fuel Pool. Failure of the demineralized water header is not accompanied with an alarm; however, the time to dilute the Spent Fuel Pool from 2000 ppm to 240 ppm is greater than the bounding case described above. In this scenario, there is sufficient time to isolate the failure and to prevent the spilling of some 320,000 gallons of water.

Instantaneous mixing of pool water with the water from the rupture of the demineralized water supply line is an extremely conservative assumption. Water falling on to the pool surface would mix with the top layers of pool water and the portions of the mixed volumes would continuously spill out of the pool. The density difference between water

at 150 °F (maximum permissible pool bulk water temperature) and at the temperature of the demineralizer water supply is small. This density difference will not cause the water falling on to the pool surface to instantaneously sink down into the racks overcoming the principal driving force for the flow in the pool, which is the buoyancy force generated in the spent fuel pool racks region due to the heat generation from the spent fuel in the racks. This would further enhance the mixing process between the pool water and spilled water above the racks.

8.0 STORAGE RESTRICTIONS

A criticality safety evaluation has been performed for the ANO Unit 2 SFP to support the fall 2003 refueling outage and subsequent refueling outages using the existing racks without credit for Boraflex. The following restrictions apply:

Racks must be loaded using one of the 5 patterns shown in Figure 1.1, following the assembly description in Table 1.1. The "A" and the "C" assemblies are defined by burnup/enrichment curves shown in Tables 2.3 and 2.5. In any pattern, only a less reactive assembly (as dictated by the ranking in Table 1.1) may be substituted for another assembly within a pattern.

AND

For adjacent patterns both inside and between racks in the SFP, only allowed interface configurations shown in figures 5.1a to 5.1w of this report may be used.

AND

A minimum soluble boron level of 825 ppm must be maintained in the SFP to ensure that the effective neutron multiplication factor (k_{eff}) is less than or equal to 0.95 under all credible accident conditions.

9.0 REFERENCES

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

6. L.I. Kopp, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” NRC Memorandum from L. Kopp to T. Collins, August 19, 1998.

7. S.E. Turner, "Uncertainty Analysis - Burnup Distributions", presented at the DOE/SANDIA Technical Meeting on Fuel Burnup Credit, Special Session, ANS/ENS Conference, Washington, D.C., November 2, 1988.

Table 1.1

Assembly Types, Rank and Description

Rank	Assembly	Description
1	F	Fresh Fuel, 4.55± 0.05 wt% U-235
2	B	4.1% maximum enrichment, 11.3 GWD/MTU minimum burnup, 0 years cooling time. Low burnup, typically one cycle. Does not meet the burnup requirements from Tables 2.2 and 2.4.
3	Fc	Fresh 4.55±0.05 wt% fuel with a new CEA (Note 1) inserted
4	A	An assembly that meets the minimum burnup requirements as defined in Tables 2.2 and 2.3.
5	BC	A "B" type assembly with a CEA inserted (Note 2)
6	C	High burnup fuel, as defined in Tables 2.4 and 2.5.
7	AC	An "A" type assembly with a CEA inserted (Note 2)
8	CC	A "C" type assembly with a CEA inserted (Note 2)
9	X	Water cell, no fuel.

Note 1 - current CEA means the new CEA type. The old CEAs have a slightly different design.

Note 2 - The BC, AC, and CC types may have either CEA design inserted.

Table 2.1

Summary of the Criticality Evaluations for Each Pattern

	Pattern 1	Pattern 2*	Pattern 3	Pattern 4	Pattern 5
Reference k_{eff}	0.9706	0.9151	0.9758	0.9742	0.9706
MCNP4a Bias	0.0009	0.0009	0.0009	0.0009	0.0009
Temperature Effect Bias	0.0092	0.0101	0.0094	0.0094	0.0094
MCNP4a Bias Uncertainty	0.0011	0.0011	0.0011	0.0011	0.0011
MCNP4a Statistics (95/95) Uncertainty	0.0007	0.0007	0.0007	0.0007	0.0007
Manufacturing Tolerance Uncertainty	0.0097	0.0129	0.0096	0.0096	0.0096
Enrichment Tolerance Uncertainty	0.0033	0.0019	0.0033	0.0033	0.0033
Depletion Uncertainty	0.0136	N/A	0.0070	0.0092	0.0136
Fuel Eccentric Positioning Uncertainty	0.0021	0.0002	0.0010	0.0010	0.0022
Statistical Combination of Uncertainties	0.0172	0.0131	0.0124	0.0138	0.0171
Total k_{eff}	**0.9979**	**0.9392**	**0.9985**	**0.9983**	**0.9980**

* This pattern was conservatively analyzed at a fresh fuel enrichment of 4.95± 0.05 wt%.

Table 2.2

Minimum Burnup Requirements for “A” Assemblies at Varying Enrichment and Cooling Times

BURNUP, MWD/KgU					
Average Enrichment, wt% ^{235}U	**0 Years Cooling Time**	**5 Years Cooling Time**	**10 Years Cooling Time**	**15 Years Cooling Time**	**20 Years Cooling Time**
2	7.56	7.56	7.56	7.56	7.56
2.5	14.94	14.21	13.47	12.74	12.00
3.0	21.30	20.30	19.30	18.30	17.30
3.5	27.10	25.95	24.80	23.65	22.50
4.0	32.67	31.38	30.09	28.79	27.50
4.5	39.30	37.73	36.15	34.58	33.00
4.95	45.00	43.33	41.65	39.98	38.30

Table 2.3

Bounding Polynomial Fits to Determine Minimum Acceptable Burnup for "A" Type Assemblies Storage as a Function of Initial Enrichment

Decay Time, Years	Burnup, MWD/KgU
0	$0.68*E^3 - 7.449*E^2 + 38.56*E - 45.20$
5	$0.5489*E^3 – 5.9344*E^2 + 32.496*E - 38.05$
10	$0.4153*E^3 – 4.3948*E^2 +26.356*E - 30.75$
15	$0.2867*E^3 – 2.9045*E^2 +20.367*E - 23.80$
20	$0.153*E^3 - 1.3649*E^2 +14.227*E - 16.60$

Note: E = Initial average enrichment in wt% ^{235}U

Table 2.4

Minimum Burnup Requirements for "C" Assemblies at Varying Enrichment and Cooling Times.

BURNUP, GWD/MTU					
Enrichment, wt% U^{235}	**0 Years Cooling Time**	**5 Years Cooling Time**	**10 Years Cooling Time**	**15 Years Cooling Time**	**20 Years Cooling Time**
2	17.00	15.00	14.00	13.30	13.00
2.5	25.00	22.00	21.00	20.30	20.00
3	32.00	29.00	27.00	26.00	25.00
3.5	37.00	35.00	32.50	31.00	30.00
4	43.00	40.00	39.00	38.00	37.00
4.5	49.75	46.00	45.00	44.00	43.00

Table 2.5

Bounding Polynomial Fits to Determine Minimum Acceptable Burnup for "C" Type Assemblies Storage as a Function of Initial Enrichment.

Decay Time, Years	Burnup, MWD/KgU
0	$1.3148*E^3 – 13.552*E^2 + 57.491*E – 53.6$
5	$0.3704*E^3 – 4.5397*E^2 + 29.59*E – 28.8$
10	$0.6667*E^3 - 6.5714*E^2 +32.976*E – 30.3$
15	$0.7111*E^3 – 6.919*E^2 +33.634*E – 31.3$
20	$0.8148*E^3 –7.7302*E^2 +35.169*E – 32.0$

Note: E = Initial average enrichment in wt% ^{235}U.

Table 2.6

Required Soluble Boron Concentrations in the SFP Water

Condition	Soluble Boron Required for k<1	Soluble Boron Required for k<0.95 (ppm)
Normal Storage of Fresh and Spent Fuel Assemblies	0	240
Accident condition of 1 fresh fuel assembly mis-placed into a cell intended to remain empty	-	825

Table 3.1

Fuel Assembly Specifications and Core Operating Parameters

Assembly Data	
Rod Array Size	16x16
Rod Pitch (inches)	0.506
Total Width (inches)	8.130–8.149
Total Length (inches)	176.803
Active Fuel Length (inches)	149.61–150
Fuel Rod Data	
Total Number of Fueled Rods	Up to 236
Fuel Rod Total Length (inches)	161.318–161.868
Fuel Rod Outer Diameter (inches)	0.382
Fuel Rod Inner Diameter (inches)	0.332
Cladding Thickness (inches)	0.025
Cladding Material	Zircaloy
Pellet Diameter (inches)	0.325
UO_2 Stack Density, gms/cc	10.522
Guide Tube Data	
Number of Tubes	5
Tube Outer Diameter (inches)	0.980
Tube Thickness (inches)	0.040
Tube Material	Zircaloy

Table 5.1

Manufacturing Tolerances

Data	**Value**
Box I.D.	See Figures 3.1 and 3.2
Box Wall Thickness	±0.004 inches
SS Sheath Thickness	±0.004 inches
Water Gap (Region 1)	±0.01 inches
Water Gap (Regions 2)	±0.08 inches

Table 5.2a

Reactivity Effects of Manufacturing Tolerances for Patterns 1 and 2

Burnup, GWD/MtU	Reference	Minimum Box ID		Min. Water Gap		Min. Box Wall Thickness		Fuel Density		Statistical Sum
	k_{inf}	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	
0	1.2722	1.2759	0.0037	1.2837	0.0115	1.2768	0.0046	1.2728	0.0006	0.0129
10	1.1867	1.1901	0.0034	1.1973	0.0106	1.1911	0.0044	1.1872	0.0005	0.0120
20	1.1198	1.1230	0.0032	1.1296	0.0098	1.1240	0.0042	1.1203	0.0005	0.0111
30	1.0578	1.0608	0.0030	1.0670	0.0092	1.0618	0.0040	1.0585	0.0007	0.0105
40	0.9976	1.0004	0.0028	1.0063	0.0087	1.0014	0.0038	0.9987	0.0011	0.0100
50	0.9396	0.9422	0.0026	0.9478	0.0082	0.9432	0.0036	0.9411	0.0015	0.0094

Table 5.2b

Reactivity Effect of Enrichment Tolerance for Pattern 2

BURNUP, GWD/MTU	REFERENCE	ENRICHMENT TOLERANCE	
	k_{inf}	k_{inf}	?k
0	1.2621	1.2640	0.0019
35	1.0187	1.0217	0.0030
37	1.0065	1.0095	0.0030
39	0.9945	0.9975	0.0030
41	0.9825	0.9856	0.0031
43	0.9706	0.9737	0.0031

Table 5.2c

Temperature Bias for Patterns 1 and 2

BURNUP, GWD/MTU	T = 20 ^{0}C	T = 150 ^{0}F	
	k_{inf}	k_{inf}	?k
0	1.2621	1.2722	0.0101
10	1.1780	1.1868	0.0088
20	1.1113	1.1198	0.0085
30	1.0491	1.0578	0.0087
40	0.9887	0.9977	0.0090
50	0.9302	0.9396	0.0094

Table 5.3a

Reactivity Effects of Manufacturing Tolerances for Patterns 3, 4, and 5

Reference	Minimum Box Id		Min. Box Wall Thickness		Min. Water Gap		Fuel Density		Statistical Sum
k_{inf}	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	
0.8988	0.9013	0.0025	0.9022	0.0034	0.9072	0.0084	0.9005	0.0017	0.0096

Table 5.3b

Reactivity Effect of Enrichment Tolerance for Patterns 1, 3, 4, and 5

4.5 wt **%**	**4.55** wt **%**	
k_{inf}	k_{inf}	Δk
0.8988	0.9021	0.0033

Table 5.3c

Temperature Bias for Patterns 3, 4, and 5

T = 20 ^{0}C	**T = 150 ^{0}F**	
k_{inf}	k_{inf}	Δk
0.8988	0.9082	0.0094

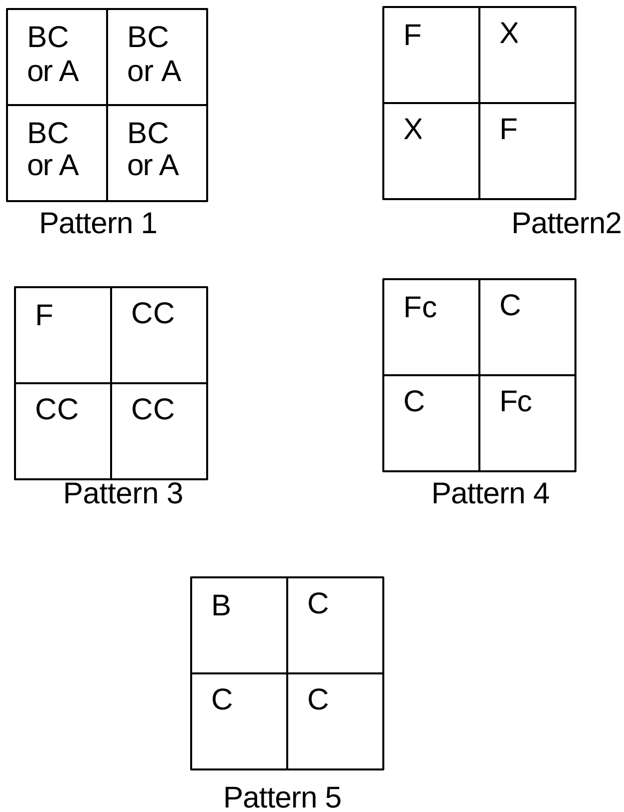

Figure 1.1: Loading Configurations for ANO-2 SFP.

(Alternate arrangements of these patterns, as applicable to the interfaces between patterns, are shown in Figures 5.1a to 5.1w.)

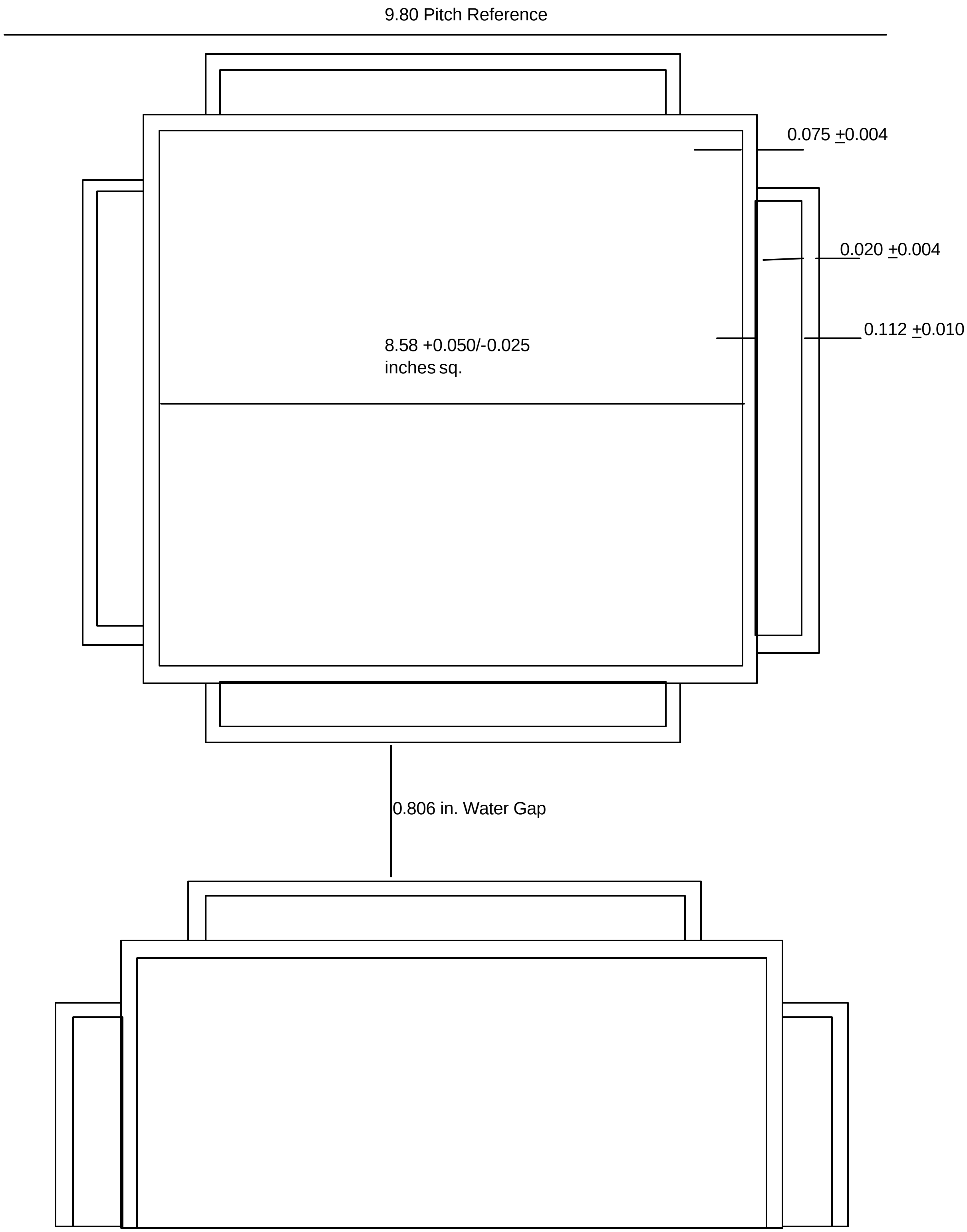

Figure 3.1: A Cross-Sectional View of the Calculational Model Used for the Region 1 Rack Analysis (NOT TO SCALE).

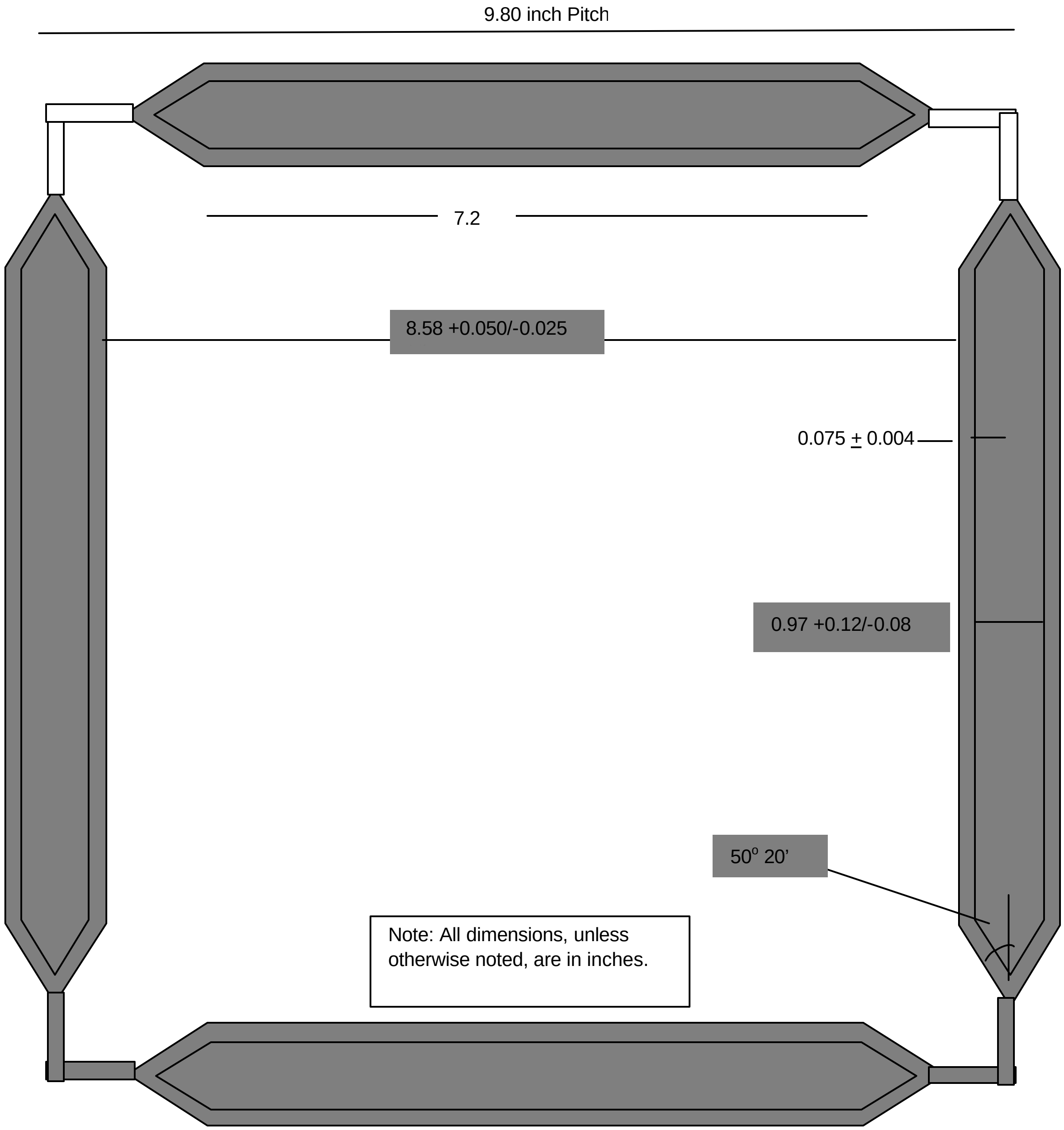

Figure 3.2: A Cross-Sectional View of the Calculational Model Used for the Region 2 Rack Analysis (NOT TO SCALE).

Note: In order to preserve the pitch due to a conservative reduction of the flux trap gap width from a design reference value of 1.07 inches to 0.97 inches (based on measurements), the cell ID was modeled as 8.68+0.050/-0.025inches.

A	A	F	X
A	A	X	F

Figure 5.1a: Pattern 1 and Pattern 2 Interface.
(This configuration is acceptable)

A	A	F	CC
A	A	CC	CC

Figure 5.1b: Pattern 1 and Pattern 3 Interface.
(This configuration is unacceptable)

A	A	CC	F
A	A	CC	CC

Figure 5.1c: Pattern 1 and Pattern 3 Interface, Alternate Configuration.
(This configuration is acceptable)

A	A	Fc	C
A	A	C	Fc

Figure 5.1d: Pattern 1 and Pattern 4 Interface.
(This configuration is acceptable)

A	A	B	C
A	A	C	C

Figure 5.1e: Pattern 1 and Pattern 5 Interface.
(This configuration is acceptable)

X	F	F	CC
F	X	CC	CC

Figure 5.1f: Pattern 2 and Pattern 3 Interface.
(This configuration is unacceptable)

X	F	CC	CC
F	X	F	CC

Figure 5.1g: Pattern 2 and Pattern 3 Interface, Alternate Configuration.
(This configuration is acceptable)

X	F	CC	F
F	X	CC	CC

Figure 5.1h: Pattern 2 and Pattern 3 Interface, Alternate Configuration.
(This configuration is acceptable)

X	F	Fc	C
F	X	C	Fc

Figure 5.1i: Pattern 2 and Pattern 4 Interface.
(This configuration is acceptable)

X	F	C	Fc
F	X	Fc	C

Figure 5.1j: Pattern 2 and Pattern 4 Interface, Alternate Configuration.
(This configuration is acceptable)

X	F	C	C
F	X	B	C

Figure 5.1k: Pattern 2 and Pattern 5 Interface.
(This configuration is acceptable)

X	F	B	C
F	X	C	C

Figure 5.1l: Pattern 2 and Pattern 5 Interface, Alternate Configuration.
(This configuration is acceptable)

X	F	C	B
F	X	C	C

Figure 5.1m: Pattern 2 and Pattern 5 Interface, Alternate Configuration.
(This configuration is acceptable)

CC	F	Fc	C
CC	CC	C	Fc

Figure 5.1n: Pattern 3 and Pattern 4 Interface.
(This configuration is unacceptable)

CC	CC	Fc	C
CC	F	C	Fc

Figure 5.1o: Pattern 3 and Pattern 4 Interface, Alternate Configuration.
(This configuration is unacceptable)

F	CC	Fc	C
CC	CC	C	Fc

Figure 5.1p: Pattern 3 and Pattern 4 Interface, Alternate Configuration.
(This configuration is acceptable)

CC	F	B	C
CC	CC	C	C

Figure 5.1q: Pattern 3 and Pattern 5 Interface.
(This configuration is unacceptable)

CC	F	C	C
CC	CC	B	C

Figure 5.1r: Pattern 3 and Pattern 5 Interface, Alternate Arrangement.
(This configuration is unacceptable)

CC	F	C	B
CC	CC	C	C

Figure 5.1s: Pattern 3 and Pattern 5 Interface, Alternate Configuration.
(This configuration is unacceptable)

F	CC	C	C
CC	CC	B	C

Figure 5.1t: Pattern 3 and Pattern 5 Interface, Alternate Configuration.
(This configuration is acceptable)

C	Fc	C	C
Fc	C	C	B

Figure 5.1u: Pattern 4 and Pattern 5 Interface.
(This configuration is acceptable)

C	Fc	B	C
Fc	C	C	C

Figure 5.1v: Pattern 4 and Pattern 5 Interface, Alternate Configuration.
(This configuration is acceptable)

C	Fc	C	C
Fc	C	B	C

Figure 5.1w: Pattern 4 and Pattern 5 Interface, Alternate Configuration.
(This configuration is acceptable)

Appendix A: Criticality Benchmark Package
(Total 25 Pages Including This Cover 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).

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† 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).

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

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† 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.

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

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

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).

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† 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.

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† 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.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.

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

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[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 † (eV) MCNP4a	EALF † (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

				Calculated k_{eff}		EALF † (eV)	
	Reference	Identification	Enrich.	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[†] (eV) MCNP4a	EALF[†] (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 † (eV) MCNP4a	EALF † (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}		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.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

		MCNP4a		KENO5a	
Reference	Case†	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.

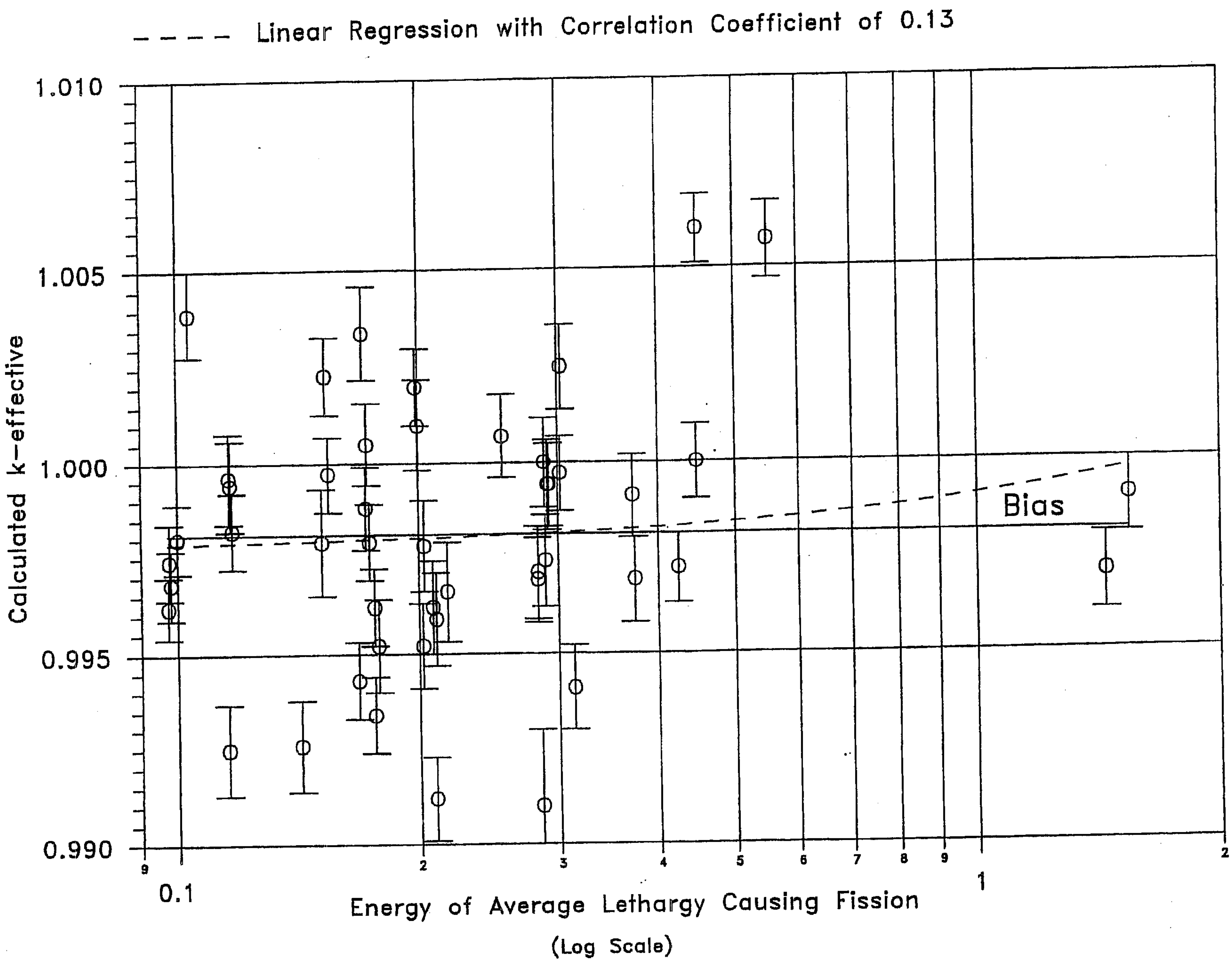

FIGURE 4A.1 MCNP CALCULATED k−eff VALUES for VARIOUS VALUES OF THE SPECTRAL INDEX

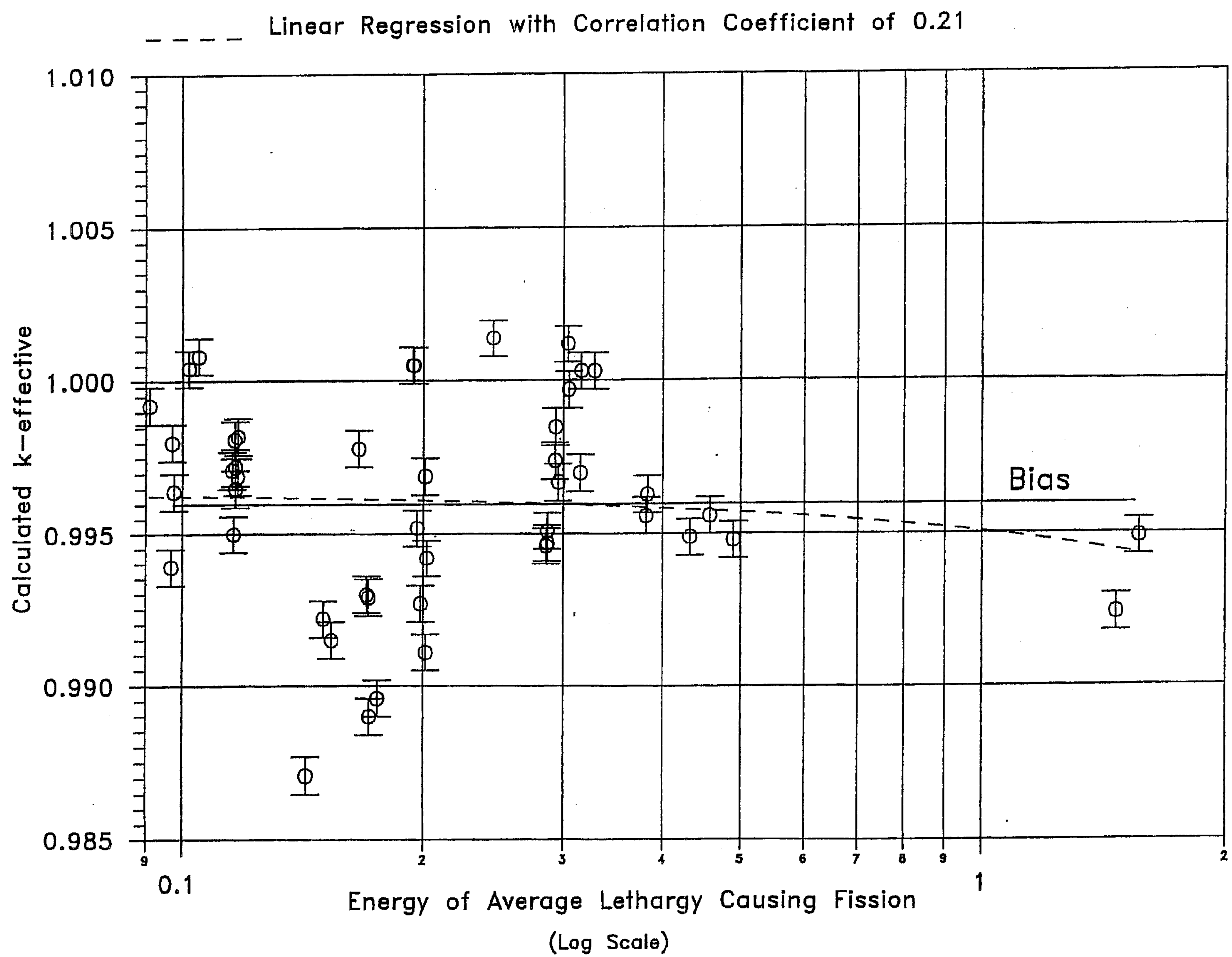

FIGURE 4A.2 KENO5a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

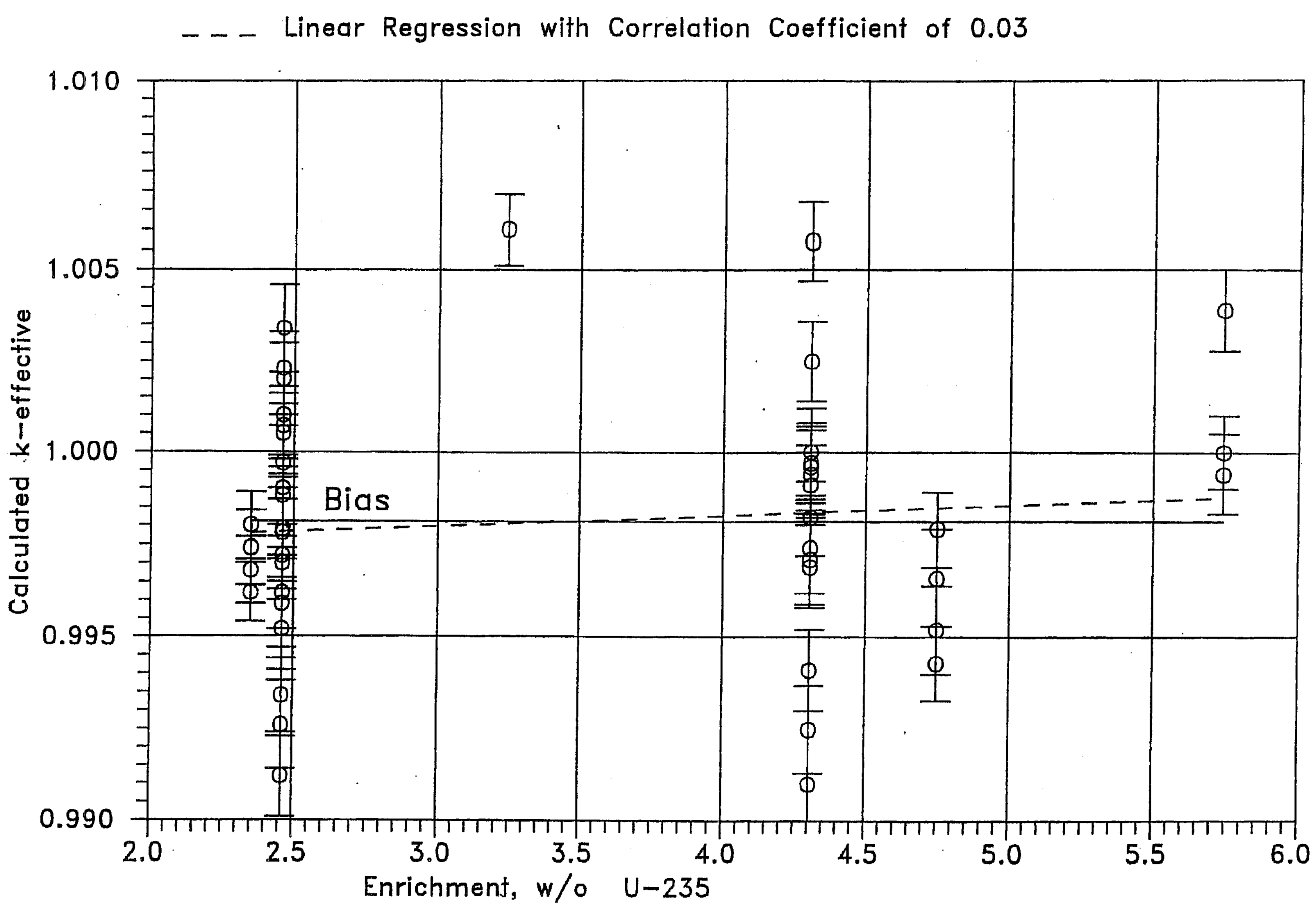

FIGURE 4A.3 MCNP CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

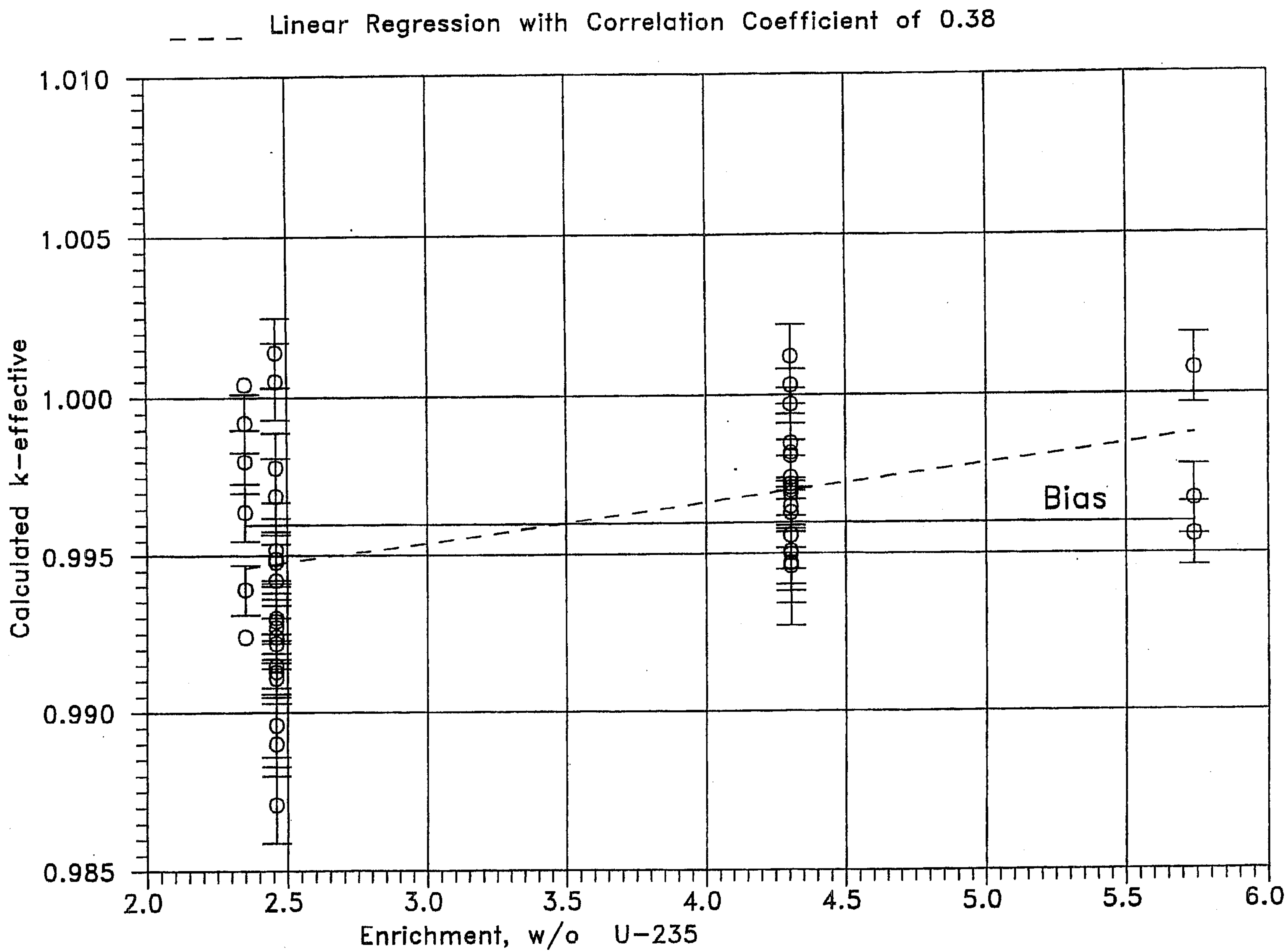

FIGURE 4A.4 KENO CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

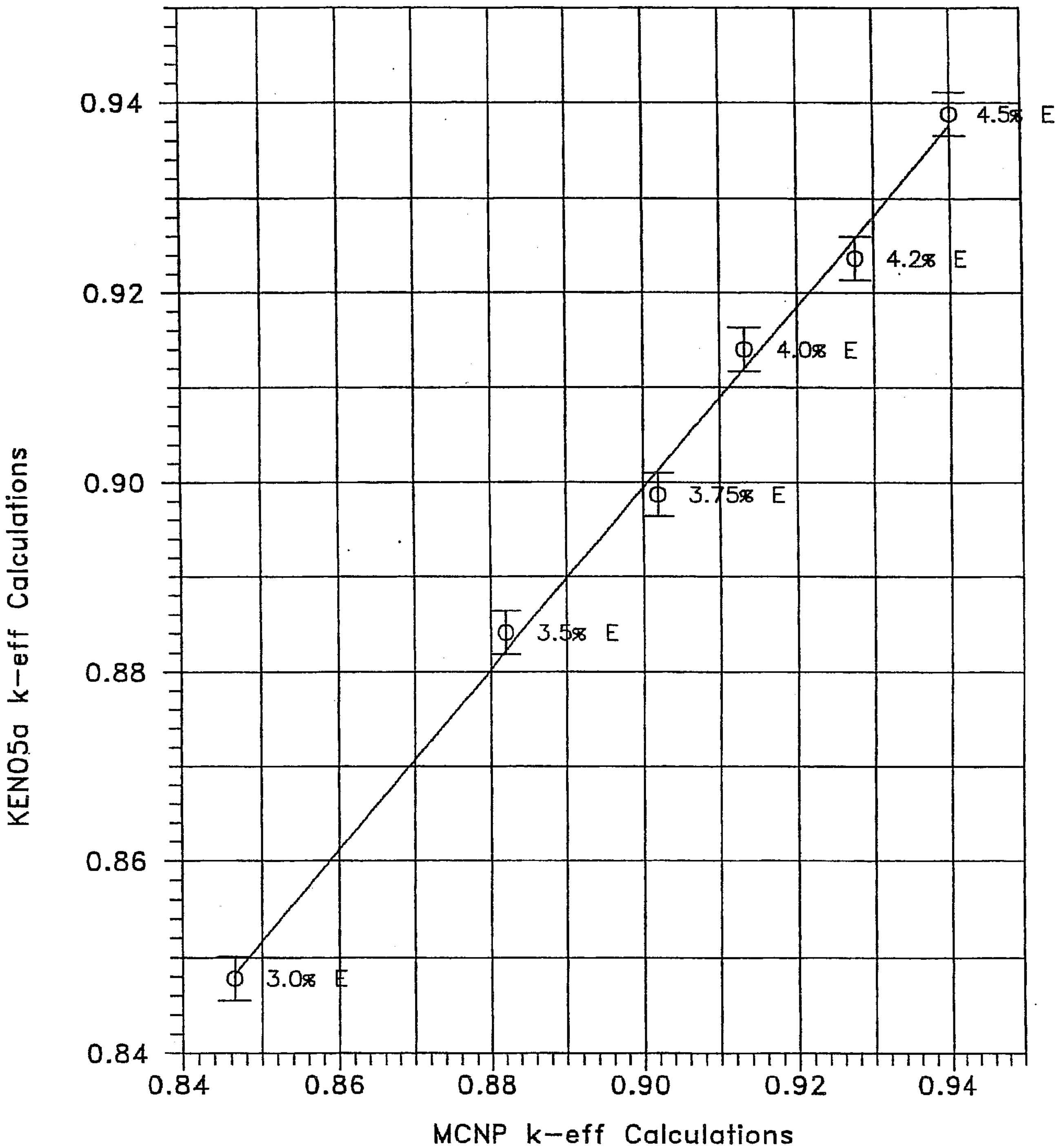

FIGURE 4A.5 COMPARISON OF MCNP AND KENO5A CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

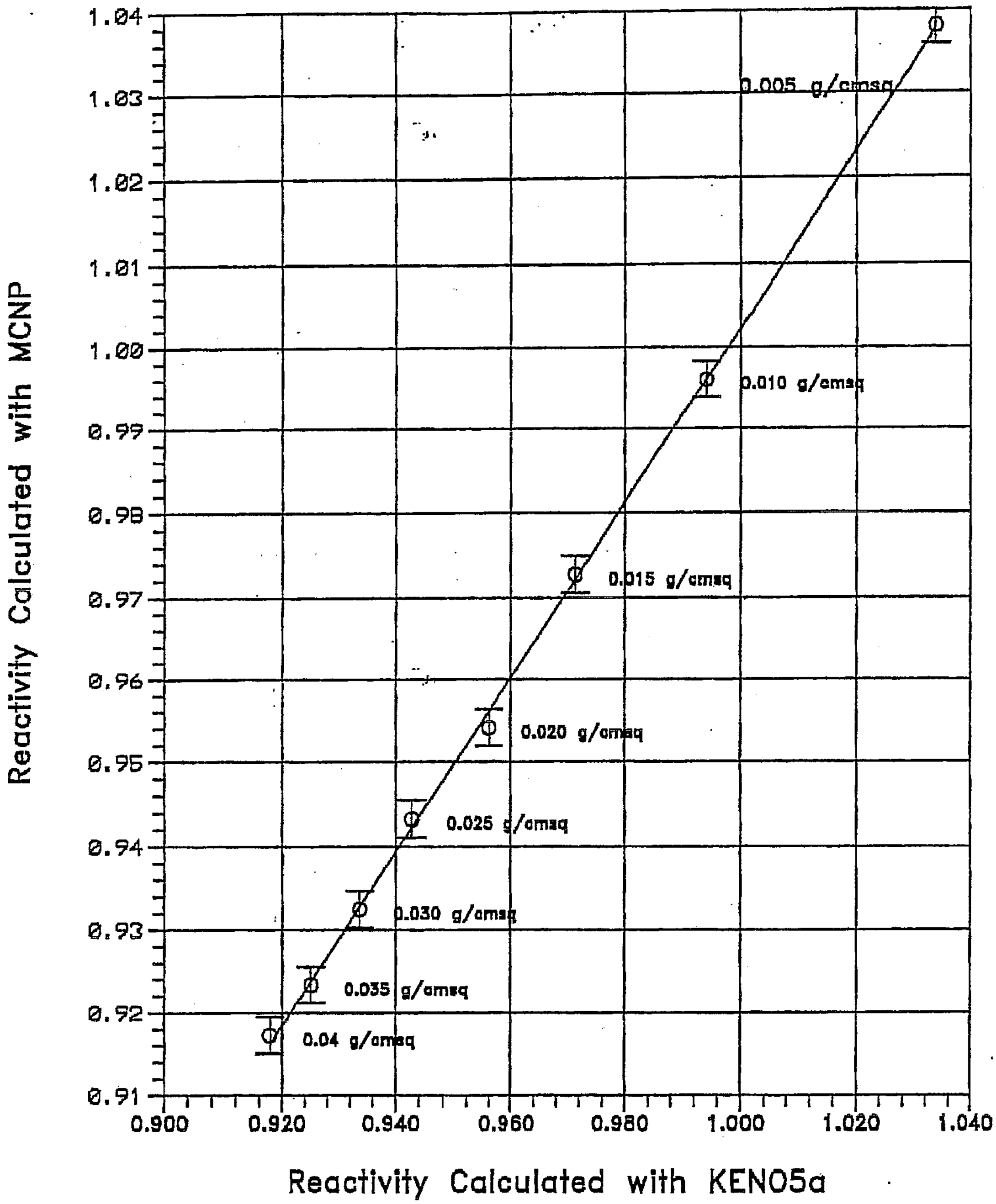

FIGURE 4A.6 COMPARISON OF MCNP AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES