10 CFR 50.90

Progress Energy

Serial: RNP-RA/03-0038

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
REGARDING CREDIT FOR SPENT FUEL STORAGE POOL DISSOLVED BORON

Ladies and Gentlemen:

In accordance with the provisions of the Code of Federal Regulations, Title 10, Part 50.90, Progress Energy Carolinas, Inc., also known as Carolina Power & Light (CP&L) Company, is submitting a request for an amendment to the Technical Specifications (TS) for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The proposed amendment would modify the TS requirements for spent fuel storage pool boron concentration and fuel storage.

The proposed amendment would eliminate the need to credit Boraflex neutron absorbing material for reactivity control in the HBRSEP, Unit No. 2, spent fuel storage pool. The new supporting analyses take credit for a combination of soluble boron and controlled fuel loading patterns within the spent fuel storage pool in order to maintain acceptable margins of subcriticality.

Attachment I provides an Affirmation pursuant to 10 CFR 50.30(b).

Attachment II provides a description of the current condition, a description of the proposed change, a technical justification of the proposed change, a No Significant Hazards Consideration Determination, and an Environmental Impact Consideration.

Attachment III provides a markup of the TS pages.

Attachment IV provides retyped pages of the proposed TS.

Attachment V provides a markup informational copy of the proposed TS Bases changes.

Attachment VI provides a retyped informational copy of the proposed TS Bases changes.

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

A001

Attachment VII provides a copy of the Holtec International criticality analysis report. This report includes the spent fuel storage pool dilution analysis. Although the cover page of the Holtec report indicates that it is proprietary, the proprietary aspects have been removed, such that Attachment VII is non-proprietary.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of the proposed license amendment.

In letter RNP-RA/96-0182, dated October 23, 1996, CP&L committed to continue to perform the Boraflex coupon surveillance program. That letter specified that approximately every four years, freshly discharged fuel assemblies would be moved near the Boraflex coupons, and a coupon would be removed for testing. Additionally, spent fuel storage pool silica concentrations would be trended. The commitment was implemented in procedures by specifying that the coupon testing would be performed at the refueling outage closest to the four year interval. The last time the surveillance was completed was in September 1999. Therefore, the next required test would occur during the next refueling outage, which is scheduled for April 2004. Approval of this proposed license amendment would eliminate any credit for Boraflex. Therefore, these commitments for performance of the coupon surveillance program and trending of silica would not be necessary and will terminate with the approval of this license amendment request.

Progress Energy Carolinas, Inc., requests approval of the proposed license amendment request by November 14, 2003, with the amendment being implemented within 30 days of approval. The requested approval date was selected to allow for effective planning for the refueling outage scheduled for April 2004. Approval of this license amendment will eliminate the need to schedule and perform another coupon surveillance test and will eliminate the need to locate a freshly discharged fuel assembly near the coupons.

In the interim, recognizing that the Boraflex in the high density storage racks may be continuing to degrade, HBRSEP, Unit No. 2, has established procedural controls to ensure fuel storage pool boron concentration will be sampled and analyzed once per 7 days (± 25%) and that immediate action will be taken if the sample analyses indicate a boron concentration of less than 1500 ppm.

If you have any questions concerning this matter, please contact me.

Sincerely,

C. T. Baucom
Supervisor – Licensing/Regulatory Programs

Attachments:

I. Affirmation
II. Request for Technical Specifications Change Regarding Credit for Spent Fuel Storage Pool Dissolved Boron
III. Markup of Technical Specifications Pages
IV. Retyped Technical Specifications Pages
V. Markup of Technical Specifications Bases Pages (Informational Copy)
VI. Retyped Technical Specifications Bases Pages (Informational Copy)
VII. Holtec International Report, Criticality Safety Analyses of the Robinson Spent Fuel Racks with Loss of Boraflex

RAC/rac

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/03-0038 is true and correct to the best of my information, knowledge, and belief; and the sources of my information are officers, employees, contractors, and agents of Progress Energy Carolinas, Inc., also known as Carolina Power & Light Company. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: MAY 2 8 2003

C L Burton

C. L. Burton
Director – Site Operations, HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING CREDIT FOR SPENT FUEL STORAGE POOL DISSOLVED BORON

Description of Current Condition

The spent fuel storage pool at H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, contains both low and high density racks. The low density spent fuel storage racks provide space for storage of 176 fuel assemblies and have a nominal 21-inch center-to-center cell spacing. The high density spent fuel storage racks provide space for storage of 368 fuel assemblies with a nominal 10.5-inch center-to-center cell spacing. Additionally, the high density storage racks contain Boraflex on each cell wall face.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, NRC guidelines (NUREG-0800, Standard Review Plan, Section 9.1.2) specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both the high and low density racks is based on the use of unborated water, which maintains the fuel in a subcritical condition during normal operation with the racks fully loaded. These analyses took credit for the Boraflex in the high density racks.

Reactivity calculations for the high density spent fuel racks indicated that fuel with enrichment greater than 4.6 w/o U^{235} would not meet the acceptance criterion of k_{eff} no greater than 0.95 without restrictions. Therefore, additional calculations were performed to establish required gadolinia limits for fuel enriched to 5.0 w/o U^{235}. These calculations indicated that fuel assemblies containing four UO_2-gadolinia bearing fuel rods with gadolinia loadings greater than 1.8 w/o would meet the k_{eff} criterion. Additionally, no empty rod locations are allowed for enrichments greater than 4.0 w/o U^{235}.

Reactivity calculations for the low density spent fuel storage racks concluded that k_{eff} remained less than 0.95 with no storage restrictions except that no empty fuel rod locations are permitted in fuel assemblies with enrichment greater than 4.25 w/o U^{235}.

Technical Specifications Limiting Condition for Operation (LCO) 3.7.14 specifies that fuel shall be stored in approved locations. The details on what constitutes an approved location are specified in the Updated Final Safety Analysis Report (UFSAR). The UFSAR describes the restrictions resulting from the criticality analyses as noted above.

There are two postulated events that could result in the addition of reactivity. The first involves the mis-positioning of a fuel assembly. The second involves dropping an assembly adjacent to a loaded fuel rack. Credit must be assumed for soluble boron in order to maintain k_{eff} less than 0.95 for these two events. Therefore, Technical Specifications LCO 3.7.13 specifies that the fuel storage pool boron concentration must be greater than 1500 ppm. Since these two events can only occur while moving fuel within the pool, this LCO is currently only applicable while moving fuel within the pool. The only action required if this LCO is not met is to immediately stop fuel movement activities. This eliminates the potential for a criticality accident.

Technical Specifications 4.3, "Fuel Storage," provides the limitations on enrichment and on k_{eff}. Technical Specifications 4.3.1.1.a limits the maximum U^{235} enrichment to 5.0 weight percent, and Technical Specifications 4.3.1.1.b ensures that the rack design limits k_{eff} to 0.95 when flooded with unborated water. Technical Specifications 4.3.1.1.e provides the requirements for burnable poison for enrichments greater than 4.6 w/o U^{235} as discussed above.

Description of the Proposed Change

The following changes to the HBRSEP, Unit No. 2, Technical Specifications are proposed:

- For Technical Specifications LCO 3.7.13, "Fuel Storage Pool Boron Concentration," the Applicability is revised to be "At All Times" instead of during new and spent fuel movement activities.

- For Technical Specifications 3.7.13, an additional action A.2 is added which requires immediate initiation of action to restore the fuel storage pool boron concentration to within limits. This action is in addition to ('AND' connector) action A.1, which requires suspending fuel movement.

- For Technical Specifications 4.3.1.1, the limitation on k_{eff} being less than or equal to 0.95 when flooded with unborated water is replaced with three separate limitations on k_{eff}. For the low density racks, the limitation remains the same, i.e., k_{eff} must be less than or equal to 0.95 when flooded with unborated water. For the high density racks, k_{eff} must be less than or equal to 0.95 when flooded with water with a boron concentration of 1500 ppm, and k_{eff} must be less than 1.0 when flooded with unborated water.

- For Technical Specifications 4.3.1.1, the requirement for burnable poison for specific enrichments is being deleted.

The Bases for Technical Specifications 3.7.13 and 3.7.14 provide information related to the criticality analyses and the various analysis assumptions, such as credit being taken for Boraflex for the high density racks. These Bases sections will require a number of changes to accurately represent the revised analyses and requirements. Therefore, an informational copy of the proposed Bases changes is also included with this submittal.

Technical Justification

The proposed changes to Technical Specifications Section 3.7.13 are more restrictive. The changes require a fuel storage pool boron concentration of at least 1500 ppm at all times, rather than during new and spent fuel movement activities. Additionally, action must be initiated immediately to return the boron concentration to within limits if determined to be below the limit. Since the surveillance requirement (SR 3.7.13.1) to determine the pool boron concentration will also be applicable at all times, the pool boron concentration must be determined once per 7 days continuously instead of the limited time during which fuel is being moved.

The HBRSEP, Unit No. 2, fuel storage pool boron concentration has typically been maintained above 2000 ppm. Therefore, there will be no change in actual plant practices in regard to pool boron concentration. However, including this requirement in the Technical Specifications and increasing the frequency of boron concentration surveillance provides assurance that the required minimum boron concentration will be maintained.

The changes to Technical Specifications 4.3.1.1 are less restrictive, but will continue to ensure that a criticality accident is not credible. With the expected condition of pool boron concentration in excess of 1500 ppm, k_{eff} will remain below 0.95. Should a low probability boron dilution event occur, k_{eff} could exceed 0.95, but even if the boron concentration were reduced to 0 ppm, k_{eff} would still remain less than 1.0 and hence a criticality accident is not credible. Attachment VII provides a discussion of the types of dilution events considered and an analysis of the potential for criticality.

The proposed limits on k_{eff} and the allowance for credit for soluble boron are consistent with the requirements of 10 CFR 50.68(b)(4). The allowance for credit for soluble boron is also consistent with approved license amendments for other plants, including McGuire Units 1 and 2, Oconee Units 1, 2 and 3, Ginna, Palisades, North Anna Units 1 and 2, and South Texas Project Units 1 and 2.

The technical bases for the conclusions on maintaining an acceptable subcriticality margin are provided in the Holtec report, "Criticality Safety Analyses of the Robinson Spent Fuel Racks with Loss of Boraflex." This report is provided in Attachment VII.

No Significant Hazards Consideration Determination

Progress Energy Carolinas, Inc., is proposing a change to the Appendix A, Technical Specifications, of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. This change revises Technical Specifications 3.7.13 and 4.3.1 and is related to requirements for ensuring adequate subcriticality margin in the spent fuel storage pool.

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed changes do not modify the facility. They apply additional administrative controls for maintaining the required boron concentration in the spent fuel storage pool. They also revise the acceptance criteria for the spent fuel storage pool criticality analyses. There will be a procedural change requiring increased frequency of spent fuel storage pool sampling for boron analysis. The sampling is performed in accordance with approved procedures and does not impact the probability or consequences of spent fuel storage pool accidents, which are a fuel handling accident and a loss of spent fuel storage pool cooling. The changes will allow for the further degradation of the Boraflex within the high density racks. The existence or degradation of the Boraflex has no relationship to the probability or consequences of a fuel handling accident or a loss of spent fuel storage pool cooling.

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

2. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

The proposed changes are related to the possibility of a criticality accident in the spent fuel storage pool. Detailed analyses have been performed to ensure a criticality accident in the spent fuel storage pool is not a credible event. The events that could lead to a criticality accident are not new. These events include a fuel mis-positioning event, a fuel drop event, and a boron dilution event. The proposed changes do not impact the probability of any of these events. The detailed criticality analyses performed demonstrate that criticality would not occur following any of these events. For the more likely events, such as a fuel mis-positioning event, k_{eff} remains less than or equal to 0.95. For the unlikely event that the spent fuel storage pool boron concentration was reduced to zero, k_{eff} remains less than 1.0. Since a criticality accident remains "not credible," the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed changes continue to provide the controls necessary to ensure a criticality event could not occur in the spent fuel storage pool. The acceptance criteria are consistent with the acceptance criteria specified in 10 CFR 50.68, which provide an acceptable margin of safety in regard to the potential for a criticality event. Therefore, the changes do not result in a significant reduction in the margin of safety.

Based on the above discussion, Progress Energy Carolinas, Inc., has determined that the requested change does not involve a significant hazards consideration.

Environmental Impact Consideration

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in a significant increase in individual or cumulative occupational radiation exposure. Progress Energy Carolinas, Inc., has reviewed this request and determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Proposed Change

Progress Energy Carolinas, Inc., is proposing a change to the Appendix A, Technical Specifications, of Facility Operating License No. DPR-23, for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. This change revises Technical Specifications 3.7.13 and 4.3.1 and is related to requirements for ensuring adequate subcriticality margin in the spent fuel storage pool.

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not involve a significant hazards consideration.

2. The only change in actual plant design or procedures will be an increased frequency in the sampling of spent fuel storage pool water for boron analyses. This increased number of samples will result in an insignificant increase in liquid radioactive waste volumes. The ability to meet all liquid effluent release limits will not be challenged as a result of the extra sample volume. Therefore, the proposed change does not result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

3. The only change in actual plant design or procedures will be an increased frequency in the sampling of spent fuel storage pool water for boron analyses. This increased number of samples will result in an insignificant increase in occupational exposure. The dose required to obtain and analyze a spent fuel storage pool sample is very small and the integrated additional dose will be insignificant. Therefore, the proposed change does not result in a significant increase in individual or cumulative occupational radiation exposures.

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/03-0038
3 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING CREDIT FOR SPENT FUEL STORAGE POOL DISSOLVED BORON

MARKUP OF TECHNICAL SPECIFICATIONS PAGES

3.7 PLANT SYSTEMS

3.7.13 Fuel Storage Pool Boron Concentration

LCO 3.7.13 The fuel storage pool boron concentration shall be ≥ 1500 ppm.

APPLICABILITY: AT ALL TIMES ~~During new and spent fuel movement activities in the fuel storage pool~~.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	-----NOTE----- LCO 3.0.3 is not applicable. -----	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND	
	A.2 INITIATE ACTION TO RESTORE FUEL STORAGE POOL BORON CONCENTRATION TO WITHIN LIMIT	IMMEDIATELY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Verify the fuel storage pool boron concentration is within limit.	7 days

b. $K_{eff} \leq 0.95$ IN THE LOW DENSITY STORAGE RACKS IF FULLY FLOODED WITH UNBORATED WATER, WHICH INCLUDES AN ALLOWANCE FOR UNCERTAINTIES AS DESCRIBED IN SECTION 9.1 OF THE UFSAR.

c. $K_{eff} \leq 0.95$ IN THE HIGH DENSITY STORAGE RACKS IF FULLY FLOODED WITH WATER BORATED TO 1500 PPM, WHICH INCLUDES AN ALLOWANCE FOR UNCERTAINTIES AS DESCRIBED IN SECTION 9.1 OF THE UFSAR.

4.3 Fuel Storage (continued)

d ~~b~~. k_{eff} ~~≤ 0.95~~ LESS THAN 1.0 IN THE HIGH DENSITY STORAGE RACKS if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

e ~~c~~. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the high density fuel storage racks;

f ~~d~~. A nominal 21 inch center-to-center distance between fuel assemblies placed in low density fuel storage racks;

~~e. Fuel assemblies with maximum planar enrichments greater than 4.55 + 0.05 (4.55 nominal) weight percent U_{235} have requirements for minimum integral burnable absorber content.~~

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

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

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

c. $k_{eff} \leq 0.98$ in an optimum moderation event, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 <u>Drainage</u>

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below 18 feet above the fuel.

4.3.3 <u>Capacity</u>

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 544 assemblies.

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/03-0038
4 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING CREDIT FOR SPENT FUEL STORAGE POOL DISSOLVED BORON

RETYPED TECHNICAL SPECIFICATIONS PAGES

3.7 PLANT SYSTEMS

3.7.13 Fuel Storage Pool Boron Concentration

LCO 3.7.13 The fuel storage pool boron concentration shall be ≥ 1500 ppm.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	-----------NOTE----------- LCO 3.0.3 is not applicable. -------------------------	
	A.1 Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Verify the fuel storage pool boron concentration is within limit.	7 days

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

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

c. $k_{eff} \leq 0.95$ in the high density storage racks if fully flooded with water borated to 1500 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

d. k_{eff} less than 1.0 in the high density storage racks if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

e. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the high density fuel storage racks;

f. A nominal 21 inch center-to-center distance between fuel assemblies placed in low density fuel storage racks.

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

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

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

c. $k_{eff} \leq 0.98$ in an optimum moderation event, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.0 DESIGN FEATURES

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below 18 feet above the fuel.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 544 assemblies.

United States Nuclear Regulatory Commission
Attachment V to Serial: RNP-RA/03-0038
10 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING CREDIT FOR SPENT FUEL STORAGE POOL DISSOLVED BORON

MARKUP OF TECHNICAL SPECIFICATIONS BASES PAGES (INFORMATIONAL COPY)

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

The fuel storage pool contains both low and high density racks for spent fuel storage. The low density spent fuel storage racks provide space for storage of 176 fuel assemblies and have a nominal 21-inch center-to-center spacing. INSERT 1 The high density spent fuel storage racks provide space for storage of 368 fuel assemblies with a nominal 10.5-inch center-to-center cell spacing. Additionally, the high density storage racks contain Boraflex on each cell wall face. INSERT 2 ~~The fuel storage racks can accommodate storage of spent fuel with initial enrichments of up to 5.0 w/o U^{235}, regardless of the discharge burnup.~~

The water in the spent fuel storage pool normally contains A MINIMUM OF 1500 ppm soluble boron, which results in large subcriticality margins under actual operating conditions. ~~However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both the high and low density racks is based on the use of unborated water, which maintains the fuel in a subcritical condition during normal operation with the racks fully loaded~~.

The effective neutron multiplication factor, K_{eff}, was calculated for the most conservative conditions of temperature, fuel enrichment, fuel spacing, structural poisoning, and other parameters (Ref. 1). For both the high density and low density spent fuel racks 5.0 w/o (4.95 w/o nominal) enrichment was assumed as the maximum permissible. ~~The maximum K_{eff} for the high density spent fuel racks, including the above allowances for uncertainties, is less than 0.95. The maximum K_{eff} for the low density spent fuel racks is less than 0.95.~~

APPLICABLE SAFETY ANALYSES

~~Most accident conditions do not result in an increase in reactivity in the fuel storage pool. Examples of these accident conditions are the loss of cooling (reactivity decrease with decreasing water density) and the dropping of~~

CRITICALITY ANALYSES FOR THE HIGH DENSITY STORAGE RACKS TAKE CREDIT FOR SOLUBLE BORON AT 1500 ppm IN ORDER TO MAINTAIN Keff LESS THAN OR EQUAL TO 0.95.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~a fuel assembly on the top of the rack, both of which have negligible effect on reactivity. However, accidents~~ FOR SPECIFIC ACCIDENTS ACCIDENTS can be postulated that could increase the reactivity. This increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accidents ~~occurrences,~~ the presence of soluble boron in the storage pool prevents criticality. The postulated accidents are basically of two types. FIRST, A fuel assembly could be incorrectly stored ~~(e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The~~ second, ~~type of postulated accidents is associated with~~ a fuel assembly ~~which is~~ COULD BE dropped adjacent to the fully loaded storage rack. This could have a small positive reactivity effect. ~~However,~~ the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. ~~The accident analyses is provided in the UFSAR, Section 15.7.4 (Ref. 2).~~

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The fuel storage pool boron concentration is required to be ≥ 1500 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential critical accident scenarios AND IN MAINTAINING $K_{eff} \leq 0.95$ IN THE HIGH DENSITY STORAGE RACKS. as described in Reference 2. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies AT ALL TIMES. ~~whenever fuel assemblies are being moved in the spent fuel storage pool~~. INSERT 3 ~~The applicability takes credit for the dual verification performed during fuel movement in the fuel storage pool. The dual verification involves two individuals tracking the movement of each fuel assembly as it is moved into or out of the spent storage pool or transferred within the pool. This verification is equivalent to a complete spent fuel storage pool verification performed after the last fuel movement has ended. The only time a fuel assembly could be mispositioned is during the time it was removed from its original position until the dual verification is performed at the new location. Therefore, the boron concentration will be~~

(continued)

BASES

APPLICABILITY (continued)

~~monitored during the movement of any fuel movement activities in the fuel storage pool.~~

ACTIONS

A.1

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. Prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

INSERT 4

THE MOVEMENT OR STORAGE OF FUEL IN THE SPENT FUEL STORAGE POOL

~~If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement~~ is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies OR MAINTAIN THE FUEL STORAGE POOL BORON CONCENTRATION GREATER THAN 1500 PPM is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents AND CRITICALITY ANALYSES are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. UFSAR Section 9.1.2.

~~2. UFSAR, Section 15.7.4.~~

Insert 1

The low density storage racks can accommodate new or spent fuel assemblies with initial enrichments up to 5 weight percent U^{235} (nominal 4.95 ± 0.05 weight percent).

Insert 2

No credit is taken for the Boraflex in criticality analyses due to the potential for degradation over time. The high density storage racks can accommodate new or spent fuel assemblies with initial enrichments up to 5 weight percent U^{235} (nominal 4.95 ± 0.05 weight percent), with restrictions on loading patterns and fuel burnup as specified in Section 9.1 of the UFSAR.

Insert 3

The criticality analyses for the high density storage racks take credit for the soluble boron in order to maintain k_{eff} less than or equal to 0.95. It is assumed the fuel will remain in the spent fuel pool until the end of the Operating License, therefore, the specified boron concentration must be maintained at all times.

Insert 4

A.2

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to return the concentration to the required limit to ensure k_{eff} remains less than or equal to 0.95 in the high density storage racks.

B 3.7 PLANT SYSTEMS

B 3.7.14 New and Spent Fuel Assembly Storage

BASES

BACKGROUND

The new fuel storage racks are used for temporary storage capacity of 2/3 of the core inventory which is equivalent to 105 storage cells located on 21-inch centers. Of these 72 are available for fuel storage. The low density spent fuel storage racks provide space for storage of 176 fuel assemblies and have a nominal 21-inch center-to-center spacing. The high density spent fuel storage racks provide space for storage of 368 fuel assemblies with a nominal 10.5-inch center-to-center cell spacing. This capacity of 544 assemblies is equivalent to 3 1/3 cores. ~~The high density storage racks contain Boraflex on each cell wall face. The fuel storage racks can accommodate storage of new fuel with initial enrichments of up to 5.0 w/o U~~^{235}~~, or spent fuel, regardless of the discharge burnup. However, the criticality analysis (Ref. 1 and 2) has shown a need for certain restrictions in order to meet the NRC acceptance criteria (Ref. 3).~~

The new fuel storage racks are normally maintained in a dry condition, i.e., the new fuel is stored in air. However, the NRC acceptance criteria (REF. 2) for new fuel storage requires that the effective multiplication factor, k_{eff}, of the storage rack be no greater than 0.95 if accidentally flooded with pure water, and no greater than 0.98 if accidentally moderated with a low density hydrogenous material (optimum moderation). The new fuel storage racks have been analyzed for 5.0 w/o U^{235} enriched fuel for the full density flooding scenario and for the optimum moderation scenario. ~~The~~ (REF. 3) calculated worst-case k_{eff} for a full rack of 5.0 w/o U^{235} fuel does not meet the acceptance criteria stated above without the restrictions imposed on the storage configuration to prevent fuel from being placed in certain locations. For the fully flooded accident condition, the resulting k_{eff} is less than 0.95. The optimum moderation condition occurs at about 5 percent interspersed water volume and results in a k_{eff} of less than 0.98 (Ref. 1).

INSERT 5 →

~~Fuel stored in the high density spent fuel storage racks is normally flooded with water borated to at least 1500 ppm. However, the NRC acceptance criterion for spent fuel storage requires that the~~ k_{eff} ~~of the storage rack be no greater than~~

(continued)

Insert 5

The low density region in the spent fuel storage pool is flooded with water borated to at least 1500 ppm. However, criticality analyses (Ref. 3) demonstrate that k_{eff} remains less than or equal to 0.95 in this region with no credit taken for the dissolved boron. There are no restrictions on storage locations except that no empty fuel rod locations are permitted in fuel assemblies with enrichment greater than 4.25 weight percent U^{235}.

The high density region in the spent fuel storage pool is flooded with water borated to at least 1500 ppm. This region includes Boraflex neutron absorber material in the cell walls. However, no credit is taken for the Boraflex in criticality analyses (Ref. 4). The analyses assume water in the locations where Boraflex has been installed. The criticality analyses demonstrate that, should the concentration of dissolved boron go to zero, k_{eff} will remain less than 1.0. Taking credit for the dissolved boron results in a k_{eff} less than or equal to 0.95. In order to ensure the calculated k_{eff} criteria are met, there are loading restrictions in the high density racks. The details of these restrictions are given in Section 9.1 of the UFSAR, which specifies acceptable loading patterns as a function of enrichment and burnup.

BASES

BACKGROUND (continued)

~~0.95 when flooded with unborated water, including all appropriate uncertainties at a 95/95 probability/confidence level. Reactivity calculations for the high density spent fuel racks indicated that fuel with an enrichment greater than 4.6 w/o U^{235} would not meet the NRC acceptance criterion of k_{eff} no greater than 0.95 without restrictions. Therefore, additional calculations were performed to establish required gadolinia limits for fuel enriched to 5.0 w/o U^{235}. These calculations indicated that fuel assemblies containing four UO_2 gadolinia bearing fuel rods with gadolinia loadings greater than 1.8 w/o would meet the NRC k_{eff} criterion. Uncertainties (95/95) due to temperature, rack tolerances, and fuel tolerances, as well as the method bias and uncertainty were used with the most limiting arrangement of four gadolinia fuel rods and resulted in a maximum k_{eff} of less than 0.95 (Ref. 2)~~.

~~The results of calculations for the low density spent fuel storage racks showed that keff remained less than 0.95 with no storage restrictions except that no empty fuel rod locations are permitted in fuel assemblies with an enrichment greater than 4.25 w/o U^{235}~~.

APPLICABLE SAFETY ANALYSES

~~The hypothetical criticality accidents can only occur as a result of storage of a new or spent fuel assembly in a prohibited location or fuel assembly configuration (Ref. 4).~~ By closely controlling the manufacture of each fuel assembly, by controlling the movement of each fuel assembly, and by checking the location of each fuel assembly after movement, the potential for an inadvertent criticality becomes very small. THE RESTRICTIONS ON FUEL LOCATION ARE DESIGNED TO ENSURE THE ASSUMPTIONS OF THE CRITICALITY ANALYSES OF REFERENCES 3 AND 4 ARE MET.

The configuration of fuel assemblies in the new and spent fuel storage racks satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the new and spent fuel storage racks ensures the k_{eff} of the stored fuel will always remain WITHIN THE CRITERIA OF SECTION 4.3.1.1 OF THESE TECHNICAL SPECIFICATIONS ~~< 0.95, assuming the racks to be flooded with unborated water~~. The approved storage locations for fuel are identified in the fuel storage

(continued)

BASES

LCO (continued)	requirements contained in Updated Final Safety Analysis Report (UFSAR) ~~Table 9.1.2-2~~ Section 9.1 (Ref. ~~2~~ 1).
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the new or spent fuel storage racks.
ACTIONS	A.1 Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the configuration of fuel assemblies stored in the new and spent fuel storage racks is not in accordance with UFSAR Section 9.1 ~~Table 9.1.2-2~~, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with UFSAR ~~Table 9.1.2-2~~. Section 9.1. If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.14.1 This SR verifies by administrative means that fuel assembly storage is in accordance with UFSAR ~~Table 9.1.2-2~~ Section 9.1.
REFERENCES	1. UFSAR Section 9.1~~.1~~. ~~2. UFSAR Section 9.1.2.~~ 2 ~~3~~. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1987.

(continued)

BASES

REFERENCES (continued)

3 ~~A~~. EMF-94-113, "H. B. Robinson New and Spent Fuel Criticality Analysis," Siemens Power Corporation, July 1994 (transmitted to NRC by CP&L letter dated July 28, 1994).

4 ~~B~~. HOLTEC INTERNATIONAL REPORT HI-992350, "CRITICALITY SAFETY ANALYSES OF THE ROBINSON SPENT FUEL RACKS WITH LOSS OF BORAFLEX", REVISION 3

United States Nuclear Regulatory Commission
Attachment VI to Serial: RNP-RA/03-0038
7 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING CREDIT FOR SPENT FUEL STORAGE POOL DISSOLVED BORON

RETYPED TECHNICAL SPECIFICATIONS BASES PAGES (INFORMATIONAL COPY)

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

The fuel storage pool contains both low and high density racks for spent fuel storage. The low density spent fuel storage racks provide space for storage of 176 fuel assemblies and have a nominal 21-inch center-to-center spacing. The low density storage racks can accommodate new or spent fuel assemblies with initial enrichments up to 5 weight percent U^{235} (nominal 4.95 ± 0.05 weight percent). The high density spent fuel storage racks provide space for storage of 368 fuel assemblies with a nominal 10.5-inch center-to-center cell spacing. Additionally, the high density storage racks contain Boraflex on each cell wall face. No credit is taken for the Boraflex in criticality analyses due to the potential for degradation over time. The high density storage racks can accommodate new or spent fuel assemblies with initial enrichments up to 5 weight percent U^{235} (nominal 4.95 ± 0.05 weight percent), with restrictions on loading patterns and fuel burnup as specified in Section 9.1 of the UFSAR.

The water in the spent fuel storage pool normally contains a minimum of 1500 ppm soluble boron, which results in large subcriticality margins under actual operating conditions.

The effective neutron multiplication factor, K_{eff}, was calculated for the most conservative conditions of temperature, fuel enrichment, fuel spacing, structural poisoning, and other parameters (Ref. 1). For both the high density and low density spent fuel racks 5.0 w/o (4.95 w/o nominal) enrichment was assumed as the maximum permissible.

APPLICABLE SAFETY ANALYSES

Criticality analyses for the high density storage racks take credit for soluble boron at 1500 ppm in order to maintain K_{eff} less than or equal to 0.95.

BASES

APPLICABLE SAFETY ANALYSES (continued)	Accidents can be postulated that could increase the reactivity. For specific accidents, this increase in reactivity is unacceptable with unborated water in the storage pool. Thus, for these accidents, the presence of soluble boron in the storage pool prevents criticality. The postulated accidents are basically of two types. First, a fuel assembly could be incorrectly stored. Second, a fuel assembly could be dropped adjacent to the fully loaded storage rack. This could have a small positive reactivity effect. The negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.
LCO	The fuel storage pool boron concentration is required to be ≥ 1500 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 1 and in maintaining $K_{eff} \leq 0.95$ in the high density storage racks. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.
APPLICABILITY	This LCO applies at all times. The criticality analyses for the high density storage racks take credit for the soluble boron in order to maintain K_{eff} less than or equal to 0.95. It is assumed the fuel will remain in the spent fuel pool until the end of the Operating License, therefore, the specified boron concentration must be maintained at all times.

BASES

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. The movement or storage of fuel in the spent fuel storage pool is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies or maintain the fuel storage pool boron concentration greater than 1500 ppm is not sufficient reason to require a reactor shutdown

A.1

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies. Prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This does not preclude movement of a fuel assembly to a safe position.

A.2

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to return the concentration to the required limit to ensure K_{eff} remains less than or equal to 0.95 in the high density storage racks.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents and criticality analyses are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time.

REFERENCES

1. UFSAR Section 9.1.2.

B 3.7 PLANT SYSTEMS

B 3.7.14 New and Spent Fuel Assembly Storage

BASES

BACKGROUND

The new fuel storage racks are used for temporary storage capacity of 2/3 of the core inventory which is equivalent to 105 storage cells located on 21-inch centers. Of these 72 are available for fuel storage. The low density spent fuel storage racks provide space for storage of 176 fuel assemblies and have a nominal 21-inch center-to-center spacing. The high density spent fuel storage racks provide space for storage of 368 fuel assemblies with a nominal 10.5-inch center-to-center cell spacing. This capacity of 544 assemblies is equivalent to 3 1/3 cores.

The new fuel storage racks are normally maintained in a dry condition, i.e., the new fuel is stored in air. However, the NRC acceptance criteria (Ref. 2) for new fuel storage requires that the effective multiplication factor, k_{eff}, of the storage rack be no greater than 0.95 if accidentally flooded with pure water, and no greater than 0.98 if accidentally moderated with a low density hydrogenous material (optimum moderation). The new fuel storage racks have been analyzed for 5.0 w/o U^{235} enriched fuel for the full density flooding scenario and for the optimum moderation scenario (Ref. 3). The calculated worst-case k_{eff} for a full rack of 5.0 w/o U^{235} fuel does not meet the acceptance criteria stated above without the restrictions imposed on the storage configuration to prevent fuel from being placed in certain locations. For the fully flooded accident condition, the resulting k_{eff} is less than 0.95. The optimum moderation condition occurs at about 5 percent interspersed water volume and results in a k_{eff} of less than 0.98 (Ref. 1).

The low density region in the spent fuel storage pool is flooded with water borated to at least 1500 ppm. However, criticality analyses (Ref. 3) demonstrate that k_{eff} remains less than or equal to 0.95 in this region with no credit taken for the dissolved boron. There are no restrictions on storage locations except that no empty fuel rod locations are permitted in fuel assemblies with enrichment greater than 4.25 weight percent U^{235}.

BASES

BACKGROUND (continued)	The high density region in the spent fuel storage pool is flooded with water borated to at least 1500 ppm. This region includes Boraflex neutron absorber material in the cell walls. However, no credit is taken for the Boraflex in criticality analyses (Ref. 4). The analyses assume water in the locations where Boraflex has been installed. The criticality analyses demonstrate that, should the concentration of dissolved boron go to zero, k_{eff} will remain less than 1.0. Taking credit for the dissolved boron results in a k_{eff} less than or equal to 0.95. In order to ensure the calculated k_{eff} criteria are met, there are loading restrictions in the high density racks. The details of these restrictions are given in Section 9.1 of the UFSAR, which specifies acceptable loading patterns as a function of enrichment and burnup.
APPLICABLE SAFETY ANALYSES	By closely controlling the manufacture of each fuel assembly, by controlling the movement of each fuel assembly, and by checking the location of each fuel assembly after movement, the potential for an inadvertent criticality becomes very small. The restrictions on fuel location are designed to ensure the assumptions of the criticality analyses of References 3 and 4 are met. The configuration of fuel assemblies in the new and spent fuel storage racks satisfies Criterion 2 of the NRC Policy Statement.
LCO	The restrictions on the placement of fuel assemblies within the new and spent fuel storage racks ensures the k_{eff} of the stored fuel will always remain within the criteria of Section 4.3.1.1 of these Technical Specifications. The approved storage locations for fuel are identified in the fuel storage requirements contained in Updated Final Safety Analysis Report (UFSAR) Section 9.1 (Ref. 1).
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the new or spent fuel storage racks.

BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. When the configuration of fuel assemblies stored in the new and spent fuel storage racks is not in accordance with UFSAR Section 9.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with UFSAR Section 9.1.

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies by administrative means that fuel assembly storage is in accordance with UFSAR Section 9.1.

REFERENCES

1. UFSAR Section 9.1.
2. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1987.
3. EMF-94-113, "H. B. Robinson New and Spent Fuel Criticality Analysis," Siemens Power Corporation, July 1994 (transmitted to NRC by CP&L letter dated July 28, 1994).
4. Holtec International Report HI-992350, "Criticality Safety Analyses of the Robinson Spent Fuel Racks with Loss of Boraflex," Revision 3.

United States Nuclear Regulatory Commission
Attachment VII to Serial: RNP-RA/03-0038
68 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING CREDIT FOR SPENT FUEL STORAGE POOL DISSOLVED BORON

HOLTEC INTERNATIONAL REPORT

CRITICALITY SAFETY ANALYSES OF THE ROBINSON SPENT FUEL RACKS WITH LOSS OF BORAFLEX

HOLTEC
INTERNATIONAL

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CRITICALITY SAFETY ANALYSES OF THE ROBINSON SPENT FUEL RACKS WITH LOSS OF BORAFLEX

FOR

CP&L

Holtec Report No: HI-992350

Holtec Project No: 90725

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HOLTEC INTERNATIONAL

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DOCUMENT NAME :	CRITICALITY SAFETY ANALYSES OF THE ROBINSON SPENT FUEL RACKS WITH LOSS OF BORAFLEX
HOLTEC DOCUMENT I.D. NUMBER :	992350
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REVISION BLOCK

REVISION NUMBER *	AUTHOR & DATE !!	REVIEWER & DATE !!	QA & DATE !!	APPROVED & DATE !	DIST. x
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Holtec Project No: **90725**	

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Summary of Revisions

Holtec Report No. HI- 992330:

Revision 1:

Additional notes were added to the text and tables (Page B-8; Tables 6.5, 6.7, 6.9, 6.11) to clarify the effect of the storage of the proposed Advanced Siemens Fuel Assembly in the spent fuel racks on the criticality calculations. The conclusions remain unchanged from revision 0 of the report.

Revision 2:

The report was revised to incorporate the client comments on revision 1 of the report [E-mail from Scott Connelly (CP&L) to Debabrata Mitra-Majumdar (Holtec), dated 3/16/00]. Additional calculations were performed to investigate the effect of the storage of fuel specified for region 1, 2 and 3 storage in interface with each other.

Revision 3:

This document has been essentially rewritten. Per client specification, only two storage patterns for fresh fuel and spent fuel assemblies are retained – unrestricted storage of spent fuel assemblies and checkerboard arrangement of fresh fuel assemblies and empty water holes. All other checkerboard arrangements of fresh fuel assemblies and spent fuel assemblies have been deleted. Analyses were revised in response to the USNRC Regulatory Issue 2001-12.

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1.0 INTRODUCTION AND SUMMARY

The purpose of the present evaluation is to document the criticality safety of the high density racks in the spent fuel pool of the Robinson Plant of CP&L. The pool criticality analyses are performed under the very conservative assumption of the complete loss of Boraflex in the spent fuel rack panels. With the assumed loss of all Boraflex material, the temperature coefficient of reactivity is positive. Therefore, the calculations assumed a temperature of 171 ^{o}F which is slightly above the highest temperature expected. Higher temperatures are considered accident conditions for which the soluble boron normally present in the pool would assure maintaining the reactivity well below the regulatory limit of k_{eff} of 0.95. Under normal storage conditions, partial credit is taken for the soluble boron in the pool water, and for fuel burnup. The criticality analyses use the MCNP4a code, a Monte Carlo code developed by the Los Alamos National Laboratory, with an explicit modeling of actinide and fission product nuclide concentrations and KENO5a, a Monte Carlo code with the 238-group cross-section library developed by the Oak Ridge National Laboratory†. CASMO4 was used for calculation of the fuel depletion effects, manufacturing tolerances, and to determine the nuclide inventories used in the MCNP4a calculations. Not all fission product cross-sections are available in the MCNP4a libraries, and those, which could not be described in MCNP4a (totaling about 0.01Δk), were simulated by an equivalent Boron-10 concentration.

As permitted in the USNRC guidelines, parametric evaluations were performed for each of the manufacturing tolerances and the associated reactivity uncertainties were combined statistically. All calculations were made for an explicit modeling of the fuel and storage cell to define the limiting enrichment-burnup for spent fuel combinations that assure safe storage of spent fuel in the pool.

The criticality safety analyses have identified and evaluated two arrangements with criteria defined for acceptable storage of fresh and spent fuel. The two arrangements are:

† Reactivity-equivalent enrichments were used in these calculations, since KENO5a tends to under predict reactivity when using explicit actinide and fission product inventories.

Unrestricted Storage Arrangement

This defines a arrangement of all spent fuel, unrestricted with regard to storage locations, the only restraints being the minimum required burnups identified in Figure 1-1 as a function of the enrichment.

Checkerboard Arrangement of Fresh Fuel

The second alternative is a checkerboard arrangement of fresh (unburned) fuel of up to 4.95±0.05% enrichment alternating with cells containing only water. The fresh fuel assemblies in this arrangement may be substituted by fuel of any enrichment of 4.95±0.05% or less regardless of burnup.

For unrestricted storage, the maximum k-effective values were determined, assuming an infinite radial array of storage cells with a finite axial length, water reflected. For each initial enrichment, a minimum burnup value was determined that assures the maximum k-effective, including calculational and manufacturing uncertainties, remains less than 1.0 under the assumed accident condition of the loss of all soluble boron. Figure 1-1 summarizes the results of these analyses, showing the minimum acceptable burnup for fuel of various initial maximum planar average enrichments. All points on the curve have nearly the same maximum reactivity (all less than 1.0). Figure 1-1a shows the minimum acceptable burnup for fuel of various initial enrichments along with the burnup of the spent fuel currently in storage as a function of initial maximum planar average enrichment (the data points). This figure shows that all the spent fuel currently in storage can be safely stored in the unrestricted configuration. The soluble boron concentration required to maintain k_{eff} below 0.95, including all manufacturing and calculational tolerances, for unrestricted storage of fuel in the pool was determined to be 200 ppm.

In the spent fuel racks, there are some older Part Length Shield Assemblies (PLSA). The top 6 inches of these assemblies are natural Uranium, the next 96 inches are enriched to 2 w/o U-235 and the bottom 42 inches are stainless steel filler rod (0.350 in. diameter). If these assemblies were full-length enriched fuel, the required burnup would be less than 1000 MWD/MTU for unrestricted storage of fuel assemblies. Moreover, the stainless steel filler rods and the axial blanket would even further reduce reactivity.

It was also determined that the storage of fresh fuel assemblies and empty cells alternately (2 fresh fuel assemblies and 2 empty cells in the checkerboard array) in the pool meets the regulatory requirements, with the calculated k_{eff} being below 0.95, without any credit for the soluble boron in the pool water.

Accident scenarios, where a fresh fuel assembly replaces a spent fuel assembly or an empty cell, were also evaluated. In the unrestricted storage condition, the accident analyzed was the misloading of a fresh fuel assembly into a cell intended to contain spent fuel only. For this case, 425 ppm soluble boron is required to maintain k_{eff} below 0.95. In the checkerboard arrangement with fresh fuel and empty cells, the misplaced fresh fuel assembly was placed in an empty cell adjacent to four other fresh fuel assemblies. These calculations demonstrate that 800 ppm of soluble boron is adequate to protect against the most serious fuel misloading accident, assuring that the maximum reactivity remains below the regulatory limit. Recent USNRC Guidelines (10 CFR 50.68 and the Kopp Memorandum [1]) allow full credit for soluble boron under these accident conditions and this would be more than adequate to protect against the most serious fuel handling accident.

Results of the analyses confirm that the spent fuel storage racks can safely accommodate fuel with initial enrichments up to 5.0%, with assurance that the maximum reactivity, including calculational and manufacturing uncertainties, will be less than 0.95, with 95% probability at the 95% confidence level, provided the fuel conforms to the enrichment-burnup limits for the spent fuel as defined in Figure 1-1 or is placed in a checkerboard array with empty cells.

Either of the 2 arrangements may be used concurrently in the pool. For these 2 arrangements, the following restrictions and qualifications apply:

- A row separating these two arrangements shall be void of fuel assemblies, if the two arrangements are used concurrently in the spent fuel pool racks.

- In any location, fuel of a lower reactivity may be used in lieu of the fuel otherwise specified. This qualification includes the following:

- Fresh fuel with enrichments less than 4.95±0.05%, or spent fuel of any burnup may be used in lieu of the fresh 4.95% enriched fuel.

- Empty cells may be utilized in lieu of any specified spent fuel.

- Reconstituted fuel assemblies may be used in lieu of any specified spent fuel assembly, provided the average burnup of the remaining fuel pins satisfy the enrichment – burnup requirements for that arrangement and that water-displacement rods replace the fuel pins removed. These water-displacement rods may be either stainless steel or natural uranium rods to assure a reactivity less than that of the assembly otherwise specified.

- Consolidated fuel bundles may replace any fuel assembly provided only that the average enrichment-burnup combination is the same as that of the assembly being replaced. If necessary, separate analyses could be used to confirm that the reactivity of the consolidated assembly is less than that required for the assembly being replaced.

• The limiting burnup shown in Figure 1-1 is the assembly average burnup and, in their application, must be adjusted for the plant uncertainty in determining the actual burnup of the spent fuel assemblies.

• All enrichments cited refer to the initial maximum planar average enrichment. Manufacturing uncertainties for the fuel are included in the tolerance uncertainties identified in the summary tables.

2.0 ANALYSIS CRITERIA AND ASSUMPTIONS

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

- Criticality safety analyses were based upon an infinite radial array of cells; i.e., no credit was taken for radial neutron leakage, except for evaluating accident conditions where neutron leakage would be 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 Boraflex, the analyses assumed a temperature of 171°F. Higher temperatures would be an accident condition for which soluble boron credit is permitted.

- The criticality analyses assumed the Westinghouse 15x15 and Advanced Framatome-ANP 15x15 fuel assemblies, which were determined to be the most reactive.

- The axial burnup distribution calculations were performed assuming a conservative distribution, neglecting the effect of any axial blankets, and ignoring the presence of control rods in the fuel assemblies.

3.0 ACCEPTANCE CRITERIA

The primary acceptance criterion under normal conditions, when partial credit is taken for the soluble boron in the pool water, is that the maximum k_{eff} shall be less than 1.0, including calculation uncertainties and effects of mechanical tolerances under the postulated loss of all soluble boron. Partial credit is taken for the soluble boron in pool water to assure that the maximum k_{eff} shall be less than 0.95, including calculation uncertainties and effects of mechanical tolerances. Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- 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.

- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.

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

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

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

4.0 DESIGN AND INPUT DATA

4.1 Fuel Assembly Design Specifications

Three different fuel assembly designs were considered in the analyses; the Westinghouse 15x15, the Framatome-ANP 15x15 fuel and an advanced Framatome-ANP 15x15 fuel. Table 4.1 provides the pertinent design details for the three fuel assembly types.

4.2 Storage Racks

The storage cells are composed of stainless steel walls. Initially the design included neutron absorber, Boraflex, in 0.10 inches channels. These cells are located on a lattice spacing of 10.50 ± 0.06 inches. The box wall thickness is 0.0747 ± 0.007 inches. The box inside dimension is 8.75+0.025/-0.050 inches. The wrapper wall thickness is 0.035 inches. However, all the analyses presented in this report were performed under the assumption of the loss of all Boraflex in the racks. The Boraflex panels were assumed to be completely replaced by water. A cross-section of the storage cell and details of the fuel assembly is shown in Figure 1-2.

5.0 METHODOLOGY

The primary criticality analyses were performed with the three-dimensional MCNP4a code [2] (bias of 0.0009 ± 0.0011 as shown in Appendix A). CASMO4, a two-dimensional deterministic code [3] using transmission probabilities, was used to evaluate the small (differential) reactivity effects of manufacturing tolerances and nuclide concentrations. Validity of the CASMO4 code was established by comparison with KENO5a and MCNP4a calculations for comparable rack cases. The NITAWL-KENO5a code system [4] was used for independent check calculations and for the evaluation of the fresh fuel and empty cell checkerboard arrangement, using the 238-group SCALE cross-section set and the Nordheim treatment for U-238 resonance shielding. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0030 ±0.0012 (95%/95%) [5].

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

MCNP4a and KENO5a Monte Carlo calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To assure convergence and to minimize the statistical uncertainty of the calculated reactivities, a minimum of 4 million neutron histories were accumulated in each calculation, generally resulting in a statistical uncertainty of about ±0.0003 Δk (1σ).

6.0 ANALYSIS RESULTS

6.1 Bounding Fuel Assembly

Calculations were done, using CASMO4, to evaluate the reactivity of the Westinghouse 15x15 and the Framatome-ANP 15x15 fuel assemblies and the results are shown below.

Burnup, GWD/MTU	k_{inf} (W 15x15)	k_{inf} (Framatome-ANP 15x15)	k_{inf} (Advanced Framatome-ANP 15x15)
0	1.2042	1.1985	1.2037
10	1.1241	1.1191	1.1235
20	1.0587	1.0531	1.0583
30	0.9960	0.9891	0.9961
40	0.9337	0.9252	0.9343
50	0.8723	0.8615	0.8737

In the calculations for the checkerboard array, the Westinghouse fuel assembly is used, since it shows the highest reactivity for fresh fuel assembly. In the calculations for the unrestricted storage, the Advanced Framatome-ANP assembly is used, since it shows the highest reactivity at higher burnups. For lower burnups, the Advanced Framatome-ANP assembly shows a slightly lower reactivity than the Westinghouse assembly. However, the difference is small, with a maximum of 0.0006 Δk at a burnup of 10 GWD/MTU, which is negligible compared to the uncertainties and margins embedded in the calculations. It is therefore acceptable to use the Advanced Framatome-ANP assembly in all the calculations for the unrestricted storage.

6.2 Evaluation of Uncertainties

CASMO4 calculations were made to determine the uncertainties in reactivity associated with manufacturing tolerances. Tolerances that would increase reactivity were calculated; negative values are expected to be of equal magnitude but opposite in sign over the small tolerance variations. Results of these calculations are shown in Table 6.1. The reactivity effects were separately evaluated, in a sensitivity study for each independent tolerance, and the results combined statistically. Tolerances considered include the following.

6.2.1 Tolerance in Lattice Pitch and Box I.D.

For the Robinson racks, the reactivity effects of the tolerance in the lattice pitch and the tolerance in the Box I.D have been calculated. The reactivity uncertainties associated with these tolerances are given in Table 6.1. The minimum cell pitch is 10.44, which, for a nominal pitch of 10.50 inches, corresponds to a tolerance of ± 0.06 inches. The nominal box I.D. is 8.75 inches with a tolerance of +0.025 inches/ - 0.050 inches.

6.2.2 Stainless Steel Thickness

The nominal tolerance in steel thickness is 10% of both the box wall thickness and sheath thickness. The nominal box wall thickness is 0.0747 inches with a tolerance of ± 0.007 in. The nominal sheath thickness is 0.035 in with a tolerance of ± 0.003 in. The reactivity uncertainties associated with these tolerances are given in Table 6.1.

6.2.3 Tolerances in Fuel Enrichment and Density

For estimating the reactivity uncertainties associated with tolerances in fuel enrichment and density, conservative tolerances of ± 0.05% in enrichment and ± 0.200 g/cc in UO_2 density were assumed. The reactivity uncertainty associated with the fuel density tolerance is summarized in Table 6.1. The reactivity uncertainties associated with the tolerance in fuel enrichment are in Table 6.3.

To estimate the uncertainty due to the enrichment tolerance, the variation in reactivity, k, as a function of enrichment, E, was calculated at 171 °F and the results are tabulated in Table 6.3 for an enrichment tolerance of 0.05 wt%. At 5% enrichment the uncertainty is 0.0020.

6.2.4 Uncertainty in Depletion Calculations

The uncertainty in depletion calculations was taken as 5% of the reactivity decrement from beginning-of-life to the burnup of concern for the unrestricted storage cases.

6.2.5 Eccentric Location of Fuel Assemblies

The fuel assemblies are nominally stored in the center of the storage cells. Calculations were made with the fuel assemblies assumed to be in the corner of the storage rack cell (eccentric positioning of a four-assembly cluster at closest approach). Eccentric positioning of spent fuel assemblies in the unrestricted storage cells resulted in an increase in reactivity in these racks. For the checkerboard arrangement of fresh fuel and empty cells, the reactivity effect due to eccentric positioning of the fuel assembly is also positive. The increase in the reactivity, due to the eccentric positioning of fuel assemblies, have been conservatively included as an additional uncertainty in the calculations, as shown in Tables 6.5 and 6.6.

6.3 Abnormal and Accident Conditions

6.3.1 Temperature and Void Effects

Temperature effects were also evaluated using CASMO4 in the temperature range from 4°C to 120°C and the results are listed in Table 6.2. These results show that the temperature coefficient of reactivity is positive and that at 171°F (maximum expected spent fuel pool water temperature) a significantly higher reactivity is predicted. Any residual Boraflex that might remain would reduce the temperature penalty.

The calculations for the reactivities under different storage conditions were performed at a water density corresponding to a temperature of 20°C. The reactivity increment between the maximum expected water temperature and 20°C is taken into account as an additive term. The void coefficient of reactivity (boiling conditions) was found to be positive. However, since this condition would be encountered at beyond the design basis maximum spent fuel pool water temperature (171°F), it is considered as an accident condition and credit for soluble boron in the pool is taken into consideration to maintain the k_{eff} below the regulatory requirement of 0.95. The increment in the reactivity due to boiling (a maximum of 0.0155 at 0 burnup from Table 6.2) is small compared to that for other postulated accident conditions. The minimum soluble boron concentration required, to maintain the k_{eff} below 0.95, for those accident conditions have been calculated and the credit for the presence of the required soluble boron will mitigate the increased reactivity due to the presence of the void.

6.3.2 Mis-Loaded Fuel Assembly Accident

The potential effects of a fuel mis-loading accident condition were also considered in this study. For the fuel misloading accident scenarios, calculations were performed to determine the soluble boron concentration required to maintain k_{eff} below 0.95 in the pool under such scenarios. Two different fuel mis-loading accident scenarios were considered in this study:

a) For unrestricted storage of spent fuel of a certain burnup, a fresh fuel assembly was postulated to be placed in the location of a spent fuel assembly.

b) For a checkerboard pattern of storage of fresh fuel assembly and water cells (2 fresh fuel assemblies and 2 empty cells), a fresh fuel was postulated to be placed in the location of a water cell.

Results of these calculations are shown in Table 6.8.

For the most serious accident scenario, calculations show that credit for 800 ppm soluble boron will maintain the maximum reactivity below the regulatory limit of 0.95. This soluble boron concentration requirement bounds the requirements for all the scenarios described above.

A brief summary of the calculations of the reactivity effects of the abnormal and accident conditions is given in Table 6.4.

6.4 <u>Reactivity Effect of Axial Burnup Distribution</u>

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

The effect of the axial distribution in burnup was determined in 3-dimensional MCNP4a calculations. In these calculations, the axial height of the fuel assembly was divided into 10 axial zones. The methodology used for obtaining the axial burnup distribution is based on a generic study by Turner [6] and has been previously used for analysis of numerous plants as well as the CP&L Harris Pool. The zone dimensions (axial height) and burnups of the 10 axial zones for 4.95% enriched fuel at 34.75 MWD/KgU average burnup are listed below:

Axial Interval (cm)	**Burnup (MWD/Kg-U)**
0-15.24	19.06
15.24-30.48	29.46
30.48-60.96	37.43
60.96-121.92	38.40
121.92-182.88	38.16
182.88-243.84	37.50
243.84-304.80	36.49
304.80-335.28	33.38
335.28-350.52	25.50
350.52-365.76	16.23

Three dimensional MCNP4a calculations were necessary to describe the geometry of the storage racks. However, MCNP4a cannot perform depletion calculations. Depletion calculations were performed with CASMO4, which provided explicit actinide and fission product inventories at the various burnups. These inventories were then used to perform the 3-dimensional calculations with MCNP4a. Calculations indicate that for enrichments less than 4.5%, the axial burnup distribution penalty would become negative. No credit for a negative value was assumed and the

k_{eff} with uniform axial burnup was used for the lower enrichment cases.

Check calculations were made with NITAWL-KENO5a. In the reactivity-equivalent enrichment calculations, to correlate the reactivity and enrichments, the calculated data was fitted to the following:

@20 °C,

$$\ln(E) = -4.72758 + 11.9127*k - 10.825*k^2 + 4.39419*k^3 \qquad (1)$$

In addition, reactivity (k) variation with Burnup, Bu, was fitted to a polynomial as follows:
@4.95% enrichment and 20 °C,

$$k=1.19073 - 6.74218E\text{-}3 *Bu + 7.77856E\text{-}6 *Bu^2 \qquad (2)$$

These equations are used in the evaluation of the effect on reactivity of the axial distribution in burnup in the 3-dimensional calculations with KENO-5a. The KENO-5a calculations confirm the results of the MCNP calculation.

6.5 Criticality Analyses Results

A summary of the results of the criticality safety analysis for unrestricted storage† of spent fuel assemblies (initial enrichment 2.0% to 4.95%) in the spent fuel pool racks is given in Table 6.5. The results indicate that unrestricted storage of this spent fuel meets the regulatory requirements. For these normal acceptable storage configuration, it was calculated that 200 ppm of soluble boron would be required to maintain the k_{eff} below 0.95, including all bias and tolerances. For the accident scenario of the placement of a fresh fuel assembly in the intended location of a spent fuel assembly, it would require 425 ppm of soluble boron to maintain k_{eff} in the rack below the regulatory requirement of 0.95. This was compared against spent fuel of 4.95% enrichment burned to 34,752 MWD/MtU and against spent fuel of 4.0% enrichment burned to 24,467 MWD/MtU. In both cases, the maximum k_{eff} was less than 0.95, as illustrated in Table 6.8.

† In this context, "unrestricted" means that there are no restrictions on where the spent fuel may be placed other than the limiting enrichment-burnup combinations as shown in Figure 1-1.

Analyses were also performed to investigate the effects of storing fresh fuel assembly and water cells in a checkerboard pattern (2 fresh fuel assemblies and 2 empty cells). The results are summarized in Table 6.6 and indicate that the calculated k_{eff}, including all calculational biases and uncertainties, for this storage pattern is below the regulatory requirement of 0.95, without any credit for soluble boron. For the postulated accident scenario of the placement of a fresh fuel assembly into a water cell, adjacent to four other fresh fuel assemblies, calculations show that 800 ppm of soluble boron in the pool water would maintain k_{eff} in the racks below the regulatory requirement of 0.95 (See Table 6.8).

6.6 Code Comparison Calculations- CASMO4, KENO5a and MCNP4a

Independent calculations were made with both MCNP4a and KENO5a using both explicit actinide and fission product nuclides as well as the reactivity-equivalent enrichment methodology. In addition, these calculations also serve to verify the CASMO4 code, since CASMO4 is a two-dimensional code and cannot be directly validated against critical experiments. The USNRC guidelines, however, endorse CASMO4 and KENO5a as acceptable methods of criticality analysis. Results of these code comparison calculations are listed in Table 6.7, corrected for bias. These results are considered to be in good agreement, confirming the basic MCNP4a, KENO5a and CASMO calculations.

6.7 Boron Dilution Accident Evaluation

The soluble boron in the spent fuel pool water is normally 1500 ppm under operating conditions. Significant loss or dilution of the soluble boron concentration is extremely unlikely, if not incredible. The required minimum soluble boron concentration is 200 ppm under normal conditions and 800 ppm for the most serious credible accident scenario. The volume of water in the pool is 240,000 gallons (2,000,000 lbs.). Large amounts of unborated water would be necessary to reduce the boron concentration from 1500 ppm to 800 ppm or to 200 ppm. Abnormal or accident conditions are discussed below for either low dilution rates (abnormal conditions) or high dilution rates (accident conditions). The general equation for boron dilution is,

$$C_t = C_o e^{-\frac{F}{V}t},$$
where

C_t is the boron concentration at time t,

C_o is the initial boron concentration,

V is the volume of water in the pool, and

F is the flow rate of unborated water into the pool

This equation assumes the unborated water flowing into the pool mixes instantaneously with the water in the pool. This is a very conservative assumption, particularly under the accident scenarios, since the unborated water pouring into the top of the pool would be predominantly included in the water spilling over the sides of the pool.

For convenience, the above equation may be re-arranged to permit calculating the time required to dilute the soluble boron from its initial concentration to a specified minimum concentration, which is given below.

$$t = \frac{V}{F}\ln(C_o / C_t)$$

The soluble boron dilution accident evaluations are described below and the results are summarized in Table 6.9.

6.7.1 Low Flow Rate Dilution.

Small failures or mis-aligned valves could possibly occur in the normal soluble boron control system or related systems (e.g. leaking pump seals or in the pool liner). 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 some 168 days to reduce the boron concentration to the minimum required 200 ppm required under normal conditions or 52 days to reach the 800 ppm required for the most severe accident. Routine surveillance measurements of the soluble boron concentration would readily detect the reduction in soluble boron concentration with ample time for corrective action.

6.7.2 High Flow Rate Dilution

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 a demineralized 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 1330 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 228 minutes (3.8 hours) to dilute the soluble boron concentration to 425 ppm, which is the minimum required concentration to maintain k_{eff} below 0.95 for postulated fuel mis-loading accidents in the unrestricted spent fuel storage arrangement. In this dilution accident, some 302,700 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).

The maximum flow rate for a failure of the 2 inch demineralized water header would provide approximately 103 gpm (pump runout flow rate) 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 1,500 to 425 ppm is greater than 2900 minutes (approximately 2 days). The time to dilute the spent fuel pool soluble boron concentration from 1500 ppm to 800 ppm with this dilution flow rate of 103 gpm would be 1,465 minutes. An alarm on high Spent Fuel Pool level would be received approximately 1 hour into the event in the main control room, assuming that the Spent Fuel Pool level started at the low alarm. In this scenario, there is sufficient time to isolate the failure and to prevent the spilling of some 302,700 gallons of water.

[†] Maximum flow rate for the open-ended break of the 2" diameter fire protection header.

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 enunciator 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 level. The enunciator 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.

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.

7.0 CONCLUSIONS

- Fuel assemblies with spent fuel having the burnup-enrichment combination as depicted in Figure 1-1 and Table 6.5 may be safely accommodated in the storage racks, with no constraints on their placements in the pool. The limiting burnup at any initial enrichment is obtained from equation in Figure 1-1.

- Storage of fresh fuel assemblies and empty cells alternately (2 fresh fuel assemblies and 2 empty cells in the checkerboard array) in the pool meets the regulatory requirements without any credit for the soluble boron in the pool water.

- 800 ppm soluble boron in the spent fuel pool is adequate to mitigate the effects of the most serious fuel mis-handling accidents analyzed within this study.

8.0 REFERENCES

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

6. 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 4.1 Design Basis Fuel Assembly Specifications[†].

FUEL ROD DATA	W 15X15	FRAMATOME-ANP[+] 15X15	ADVANCED FRAMATOME-ANP 15X15*
Outside diameter, in.	0.422	0.424	0.424
Cladding inside diameter, in.	0.373	0.364	0.374
Cladding material	Zr-4	Zr-4	Zr-4
Stack density, gms UO_2/cc	10.41±0.20	10.41±0.20	10.41±0.20
Pellet diameter, in.	0.3659	0.357	0.367
Maximum enrichment, wt. % U-235	5.0%	5.0%	5.0%
Fuel rod array	15x15	15 x 15	15 x 15
Number of fuel rods	204	204	204
Fuel rod pitch, in.	0.563	0.563	0.563
Number of guide tubes/Inst. Tubes	21	21	21
Guide/Inst. tubes O.D., in.	0.546	0.544	0.544
Guide/Inst. tubes I.D., in.	0.512	0.511	0.511

† No axial blankets were assumed to be present in these calculations.

\+ Formerly known as Siemens 15x15

* Proposed future design.Formerly known as Advanced Siemens 15x15.

Table 6. 1. Reactivity Effects of Manufacturing Tolerances for Westinghouse 15x15 Fuel Assembly in CP&L Robinson High Density Spent Fuel Racks.

BURNUP, GWD/MTU	REFERENCE[1]	MINIMUM BOX ID		MINIMUM PITCH		MIN. BOX WALL		MIN. SHEATH		FUEL DENSITY		STATISTICAL SUM
	k_{inf}	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	
0	1.2174	1.2180	0.0006	1.2254	0.0080	1.2217	0.0043	1.2194	0.0020	1.2187	0.0013	0.0094
10	1.1358	1.1365	0.0007	1.1433	0.0075	1.1399	0.0041	1.1377	0.0019	1.1368	0.0010	0.0088
20	1.0700	1.0707	0.0007	1.0770	0.0070	1.0740	0.0040	1.0720	0.0020	1.0713	0.0013	0.0084
30	1.0074	1.0080	0.0006	1.0140	0.0066	1.0113	0.0039	1.0093	0.0019	1.0091	0.0017	0.0081
40	0.9454	0.9458	0.0004	0.9515	0.0061	0.9491	0.0037	0.9472	0.0018	0.9477	0.0023	0.0077
50	0.8845	0.8850	0.0005	0.8903	0.0058	0.8880	0.0035	0.8862	0.0017	0.8876	0.0031	0.0076

Note: All calculations are at 171 F and 4.95% initial enrichment.

Table 6.2 Reactivity Effects of Temperature and Void for Westinghouse 15x15 Fuel Assembly in CP&L Robinson High Density Spent Fuel Racks.

BURNUP, GWD/MTU	T = 4 °C	T = 20 °C		T = 77.4 °C (171 °F)		T = 120 °C		T = 120 °C + VOID	
	k_{inf}	k_{inf}	Δk^+	k_{inf}	Δk^*	k_{inf}	Δk^*	k_{inf}	Δk^{**}
0	1.2003	1.2042	0.0039	1.2174	0.0132	1.2276	0.0234	1.2329	0.0053
10	1.1206	1.1241	0.0035	1.1358	0.0117	1.1450	0.0209	1.1490	0.0040
20	1.0552	1.0587	0.0035	1.0700	0.0113	1.0790	0.0203	1.0823	0.0033
30	0.9926	0.9960	0.0034	1.0074	0.0115	1.0166	0.0206	1.0193	0.0027
40	0.9302	0.9337	0.0035	0.9454	0.0117	0.9549	0.0212	0.9575	0.0026
50	0.8687	0.8723	0.0036	0.8845	0.0122	0.8946	0.0223	0.8972	0.0026

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

Note: Initial enrichment was 4.95% in these calculations

Table 6.3 Reactivity Effects of Fuel Enrichment Tolerance for Westinghouse 15x15 Fuel Assembly in CP&L Robinson High Density Spent Fuel Racks.

ENRICHMENT (%)	REFERENCE (@ 171 ^{o}F)	ENRICHMENT TOLERANCE (@ 171 ^{o}F)	
	k_{inf}	k_{inf}	Δk
2	0.9887	0.9960	0.0074
2.5	1.0531	1.0586	0.0055
3.0	1.1017	1.1059	0.0042
3.5	1.1398	1.1431	0.0034
4.0	1.1705	1.1732	0.0027
4.5	1.1958	1.1981	0.0023
5.0	1.2151	1.2171	0.0020

Table 6.4 Reactivity Effects of Abnormal and Accident Conditions

ACCIDENT/ABNORMAL CONDITIONS	REACTIVITY EFFECT
Temperature increase (See Table 6.2)	Positive
Void (Boiling) (See Table 6.2)	Positive
Misplacement of a fresh fuel assembly (2 fresh and 2 water hole configuration)	Worst input requires minimum 800 ppm soluble boron

Table 6.5 Summary of the Criticality Safety Analyses for Unrestricted Storage of Spent Fuel, with No Credit for Boraflex or Soluble Boron in Robinson High Density Storage Racks.

INITIAL ENRICHMENT, WT % U235	2%	2.5%	3%	3.5%	4%	4.5%	4.95%
Design Basis Burnup, MWD/MtU	952	7,722	13,518	19,116	24,467	29,642	34,752
MCNP4a Calculated k_{eff}	0.9625	0.9632	0.9624	0.9612	0.9614	0.9627	0.9636
MCNP Calculational Bias	0.0009	0.0009	0.0009	0.0009	0.0009	0.0009	0.0009
Uncertainties							
MCNP Statistics	±0.0005	±0.0007	±0.0007	±0.0005	±0.0007	±0.0007	±0.0007
In Bias	±0.0011	±0.0011	±0.0011	±0.0011	±0.0011	±0.0011	±0.0011
Enrichment Tolerance (±0.05%) Uncertainty	±0.0074	±0.0055	±0.0042	±0.0034	±0.0027	±0.0023	±0.0020
Depletion Uncertainty	±0.0006	±0.0036	±0.0060	±0.0079	±0.0094	±0.0107	±0.0119
Fuel Eccentricity	±0.0045	±0.0045	±0.0045	±0.0045	±0.0045	±0.0045	±0.0045
Manufacturing Tolerances+	±0.0094	±0.0094	±0.0094	±0.0094	±0.0094	±0.0094	±0.0094
Statistical Combination of Uncertainties	±0.0129	±0.0124	±0.0128	±0.0136	±0.0144	±0.0152	±0.0160
Effect of Temperature to 171 °F	0.0087	0.0106	0.0112	0.0117	0.0119	0.0122	0.0123
k_{eff}	0.9721 ±0.0129	0.9747 ±0.0124	0.9745 ±0.0128	0.9738 ±0.0136	0.9742 ±0.0144	0.9758 ±0.0152	0.9768 ±0.0160
Maximum k_{eff}	0.9850	0.9871	0.9873	0.9874	0.9886	0.9910	0.9928
Regulatory Limiting k_{eff}	1.00	1.00	1.00	1.00	1.00	1.00	1.00

\+ Conservatively the maximum value at zero burnup, from Table 6.1, is used.

** Advanced Framatome ANP fuel design is used.

Table 6.6 Summary of the Criticality Safety Analyses for Checkerboard Storage of Two Fresh Fuel Assemblies and Two Water Filled Cells in CP&L Robinson High Density Storage Racks**.

STORAGE ARRANGEMENT	2 Fresh Fuel and 2 Empty Cell Checkerboard
Initial Enrichment	4.95%
Reference k_{eff}	0.9228
Uncertainties	
Enrichment Tolerance (±0.05%) Uncertainty	±0.0019
Fuel Eccentricity	±0.0011
Manufacturing Tolerances	±0.0094
Bias Uncertainty (95%/95%)	±0.0012
Calculational Statistics (95%/95%, 1.7xσ)	±0.0005
Statistical Combination of Uncertainties	±0.0097
Axial Burnup Distribution Penalty	Negative
Effect of Temperature to 171 °F	+0.0132
Calculational Bias (see Appendix A)	0.0030
Maximum k_{eff}	0.9487
Regulatory Limiting k_{eff}	0.9500

** The use of the advanced Framatome-ANP fuel assembly will not affect the results in this storage configuration. These calculations used the bounding fuel assembly (Westinghouse 15x15).

Table 6.7 Comparison of Calculations by Different Codes*.

CASE	CASMO-4	KENO**	MCNP**
1) 34 GWD/MTU Burnup Fuel, Unrestricted, 1.924% E, Infinite Axial Length	0.9700	0.9688± 0.0014	0.9698±0.0012
2) 34 GWD/MTU Burnup, in Rack, E=1.924% E, Finite Axial Length	-	0.9658± 0.0014	0.9669 ± 0.0012
3) Normal Storage with 200 ppm B	-	0.9208±0.0013	0.9216±0.0013+
4) Checkerboard in Rack, Fresh Fuel Assembly and Empty Cell	-	0.9258±0.0013	0.9253±0.0013
5) Accident condition with 400 ppm B*	-	0.9214±0.0013	0.9216±0.0013+

* The calculations are for comparative purposed only and independent verification. The results are not used in any calculations.
** KENO and MCNP results have calculational bias added.
\+ These MCNP calculations were performed using explicit actinides and fission products.

Table 6.8 Soluble Boron Requirements for Normal and Accident Conditions.

CASE I.D.	MINIMUM SOLUBLE BORON	CALCULATED k_{eff}	MAXIMUM k_{eff}
1) Normal Storage 4.0% Initial Enrichment @ 24,467 MWD/MtU	200	0.9158	0.9429
2) Normal Storage 4.95% Initial Enrichment @ 34,752 MWD/MtU	200	0.9207	0.9499
3) Mis-loaded Fuel Accident with 4.0% Enrichment @ 24,467 MWD/MtU	425	0.9218	0.9489
4) Mis-loaded Fuel Accident with 4.95% Enrichment @ 34,752 MWD/MtU	425	0.9183	0.9475
5) Fresh Fuel Assembly mis-loaded into an empty cell in a 2x2 Checkerboard Array	800	0.9121	0.9380

Table 6.9 Boron Dilution Times from 1500 ppm Boron

CASE	DILUTION FLOW RATE, GPM	TIME, MINUTES	TOTAL DILUTION FLOW, GALLONS
Normal Storage dilution to 200 ppm			
a) System Leakage	2	242,000	483,600
b) Demin. Water Pipe Rupture	103	4,695	483,600
c) Fire Header Pipe Rupture	1330	364	483,600
Accident Condition Dilution to 425 ppm			
a) System Leakage	2	151,000	302,700
b) Demin. Water Pipe Rupture	103	2939	302,700
c) Fire Header Pipe Rupture	1330	228	302,700
Accident with Checkerboard Arrangement dilution to 800 ppm Boron			
a) System Leakage	2	75,400	150,900
b) Demin. Water Pipe Rupture	103	1,465	150,900
c) Fire Header Pipe Rupture	1330	113	150,900

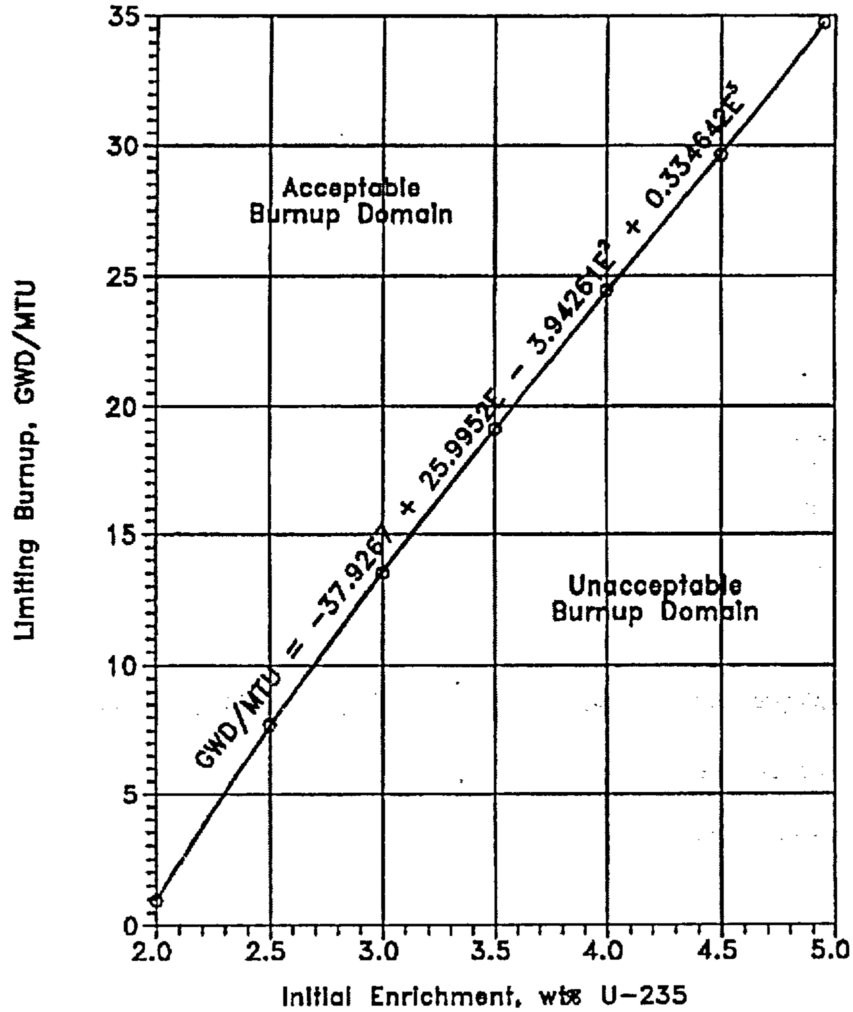

Figure 1-1 Limiting Fuel Burnup For Unrestricted Storage of Spent Fuel in the High Density Storage Racks.

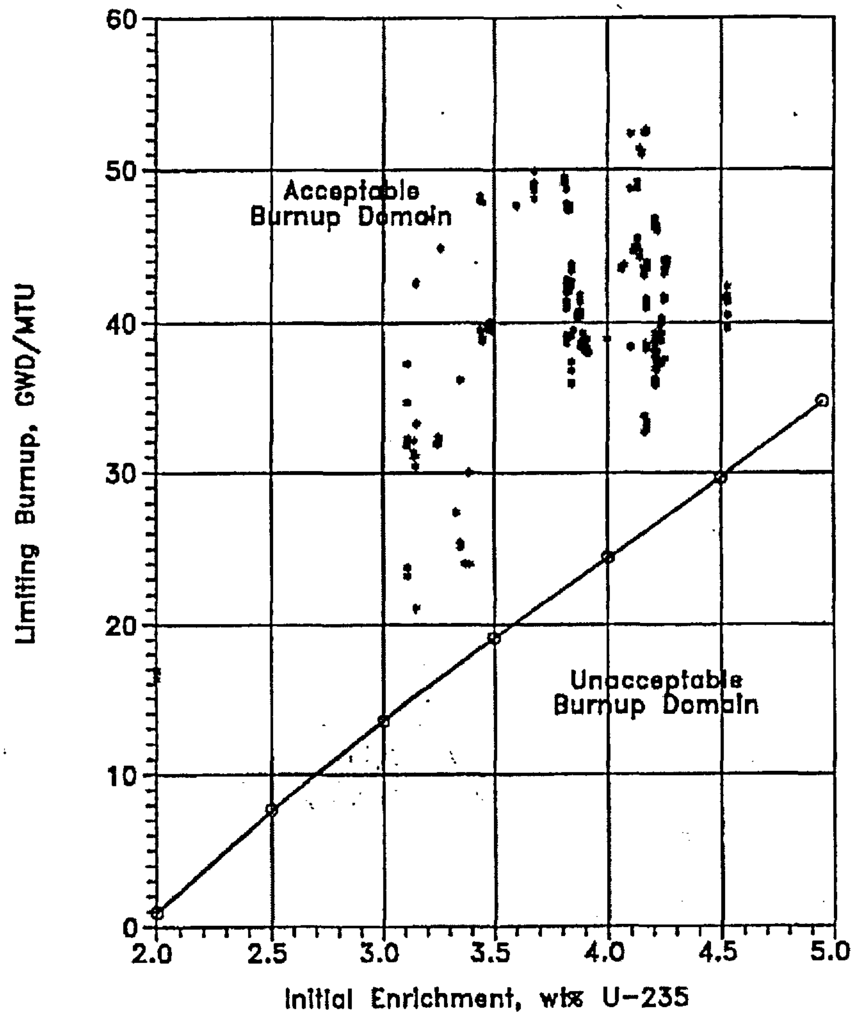

Figure 1-1a Minimum Burnup Requirements For Unrestricted Storage of Spent Fuel in the CP&L Robinson High Density Storage Racks with Burnup-Enrichment Combinations for Spent Fuel Currently in Storage.

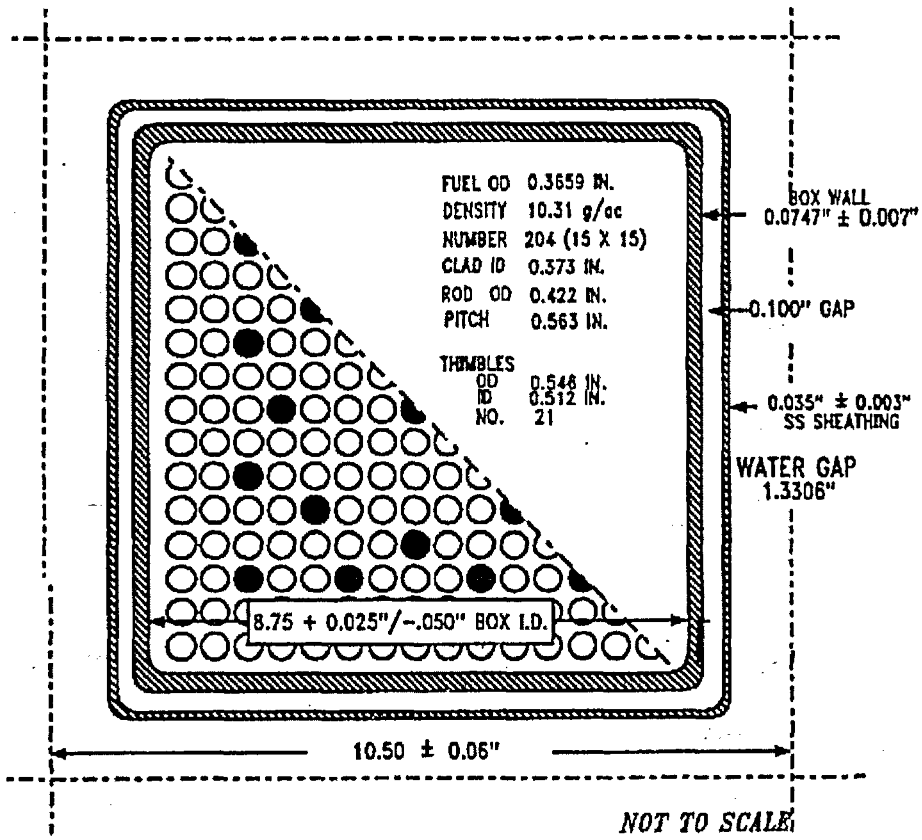

Figure 1-2 Cross-section of the Storage Cell and Details of the Fuel Assembly.

APPENDIX A: BENCHMARK CALCULATIONS

(Total of 26 Pages Including This Page)

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

APPENDIX 4A: BENCHMARK CALCULATIONS

4A.1 INTRODUCTION AND SUMMARY

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

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

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

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

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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).

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

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

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

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

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

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

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

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

4A.2 Effect of Enrichment

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

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

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

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

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

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

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

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

4A.4 Miscellaneous and Minor Parameters

4A.4.1 Reflector Material and Spacings

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

4A.4.2 Fuel Pellet Diameter and Lattice Pitch

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

4A.4.3 Soluble Boron Concentration Effects

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

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

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

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[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

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

Notes: NC stands for not calculated.

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

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

Table 4A.2

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

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

† Based on the GE 8x8R fuel assembly.

Table 4A.3

MCNP4a CALCULATED REACTIVITIES FOR
CRITICAL EXPERIMENTS WITH NEUTRON ABSORBERS

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

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

Table 4A.4

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

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

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

Table 4A.5

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH THICK LEAD AND STEEL REFLECTORS†

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

† Arranged in order of increasing reflector-fuel spacing.

Table 4A.6

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH VARIOUS SOLUBLE BORON CONCENTRATIONS

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

Table 4A.7

CALCULATIONS FOR CRITICAL EXPERIMENTS WITH MOX FUEL

Reference	Case[†]	MCNP4a		KENO5a	
		k_{eff}	EALF[††]	k_{eff}	EALF[††]
PNL-5803 [4A.16]	MOX Fuel - Exp. No. 21	1.0041±0.0011	0.9171	1.0046±0.0006	0.8868
	MOX Fuel - Exp. No. 43	1.0058±0.0012	0.2968	1.0036±0.0006	0.2944
	MOX Fuel - Exp. No. 13	1.0083±0.0011	0.1665	0.9989±0.0006	0.1706
	MOX Fuel - Exp. No. 32	1.0079±0.0011	0.1139	0.9966±0.0006	0.1165
WCAP-3385-54 [4A.17]	Saxton @ 0.52" pitch	0.9996±0.0011	0.8665	1.0005±0.0006	0.8417
	Saxton @ 0.56" pitch	1.0036±0.0011	0.5289	1.0047±0.0006	0.5197
	Saxton @ 0.56" pitch borated	1.0008±0.0010	0.6389	NC	NC
	Saxton @ 0.79" pitch	1.0063±0.0011	0.1520	1.0133±0.0006	0.1555

Note: NC stands for not calculated

† Arranged in order of increasing lattice spacing.

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

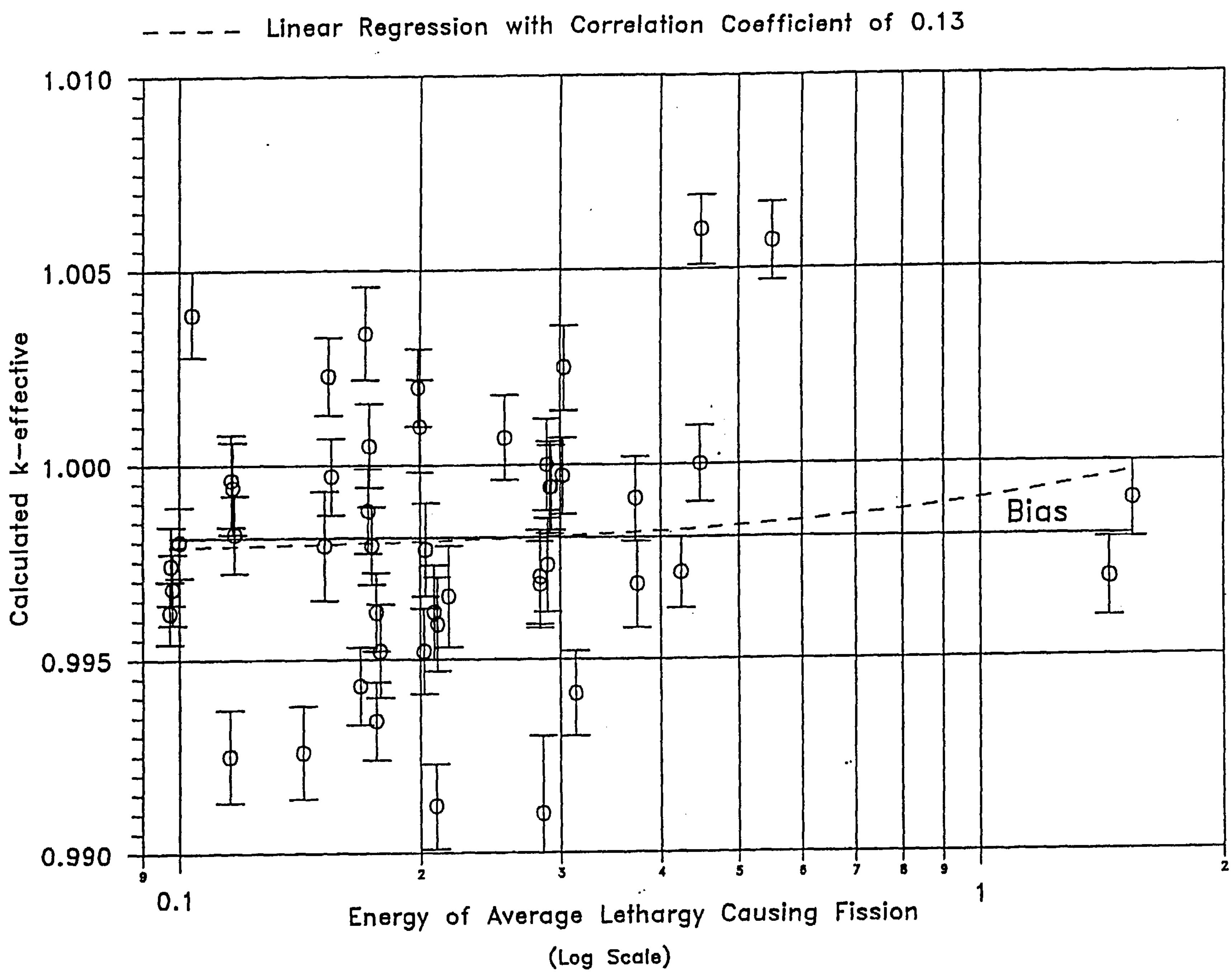

FIGURE 4A.1 MCNP CALCULATED k-eff VALUES for VARIOUS VALUES OF THE SPECTRAL INDEX

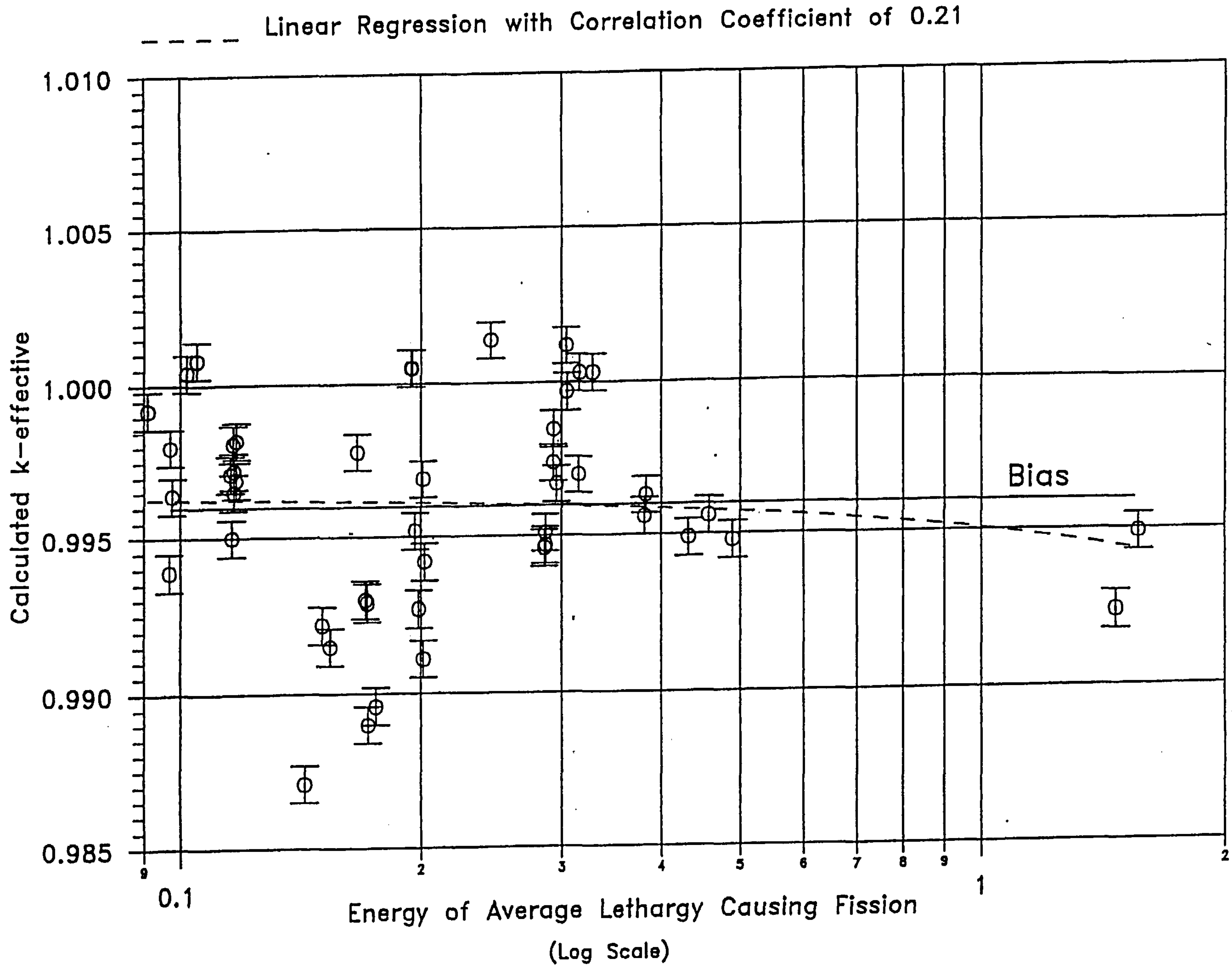

FIGURE 4A.2 KENO5a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

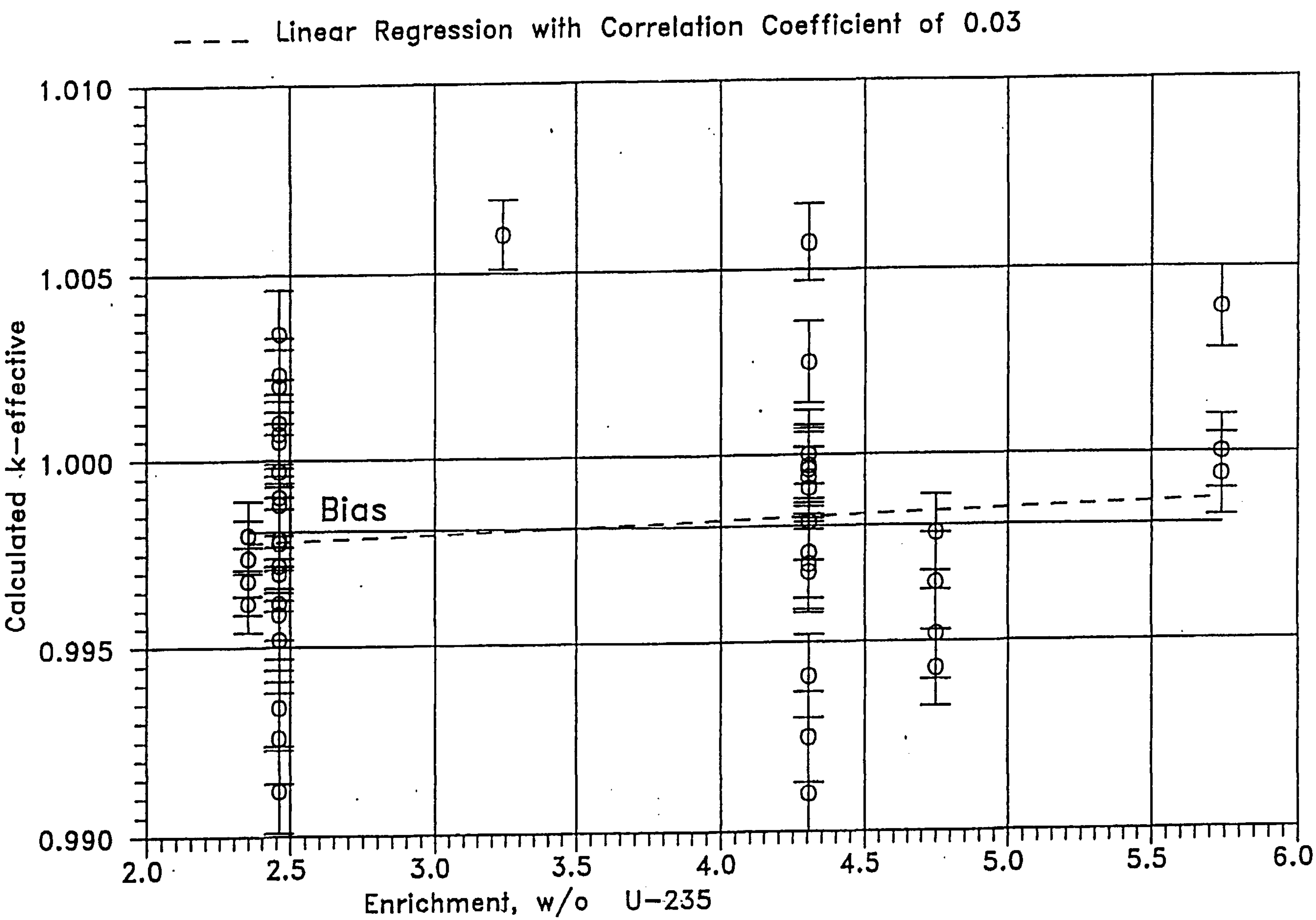

FIGURE 4A.3 MCNP CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

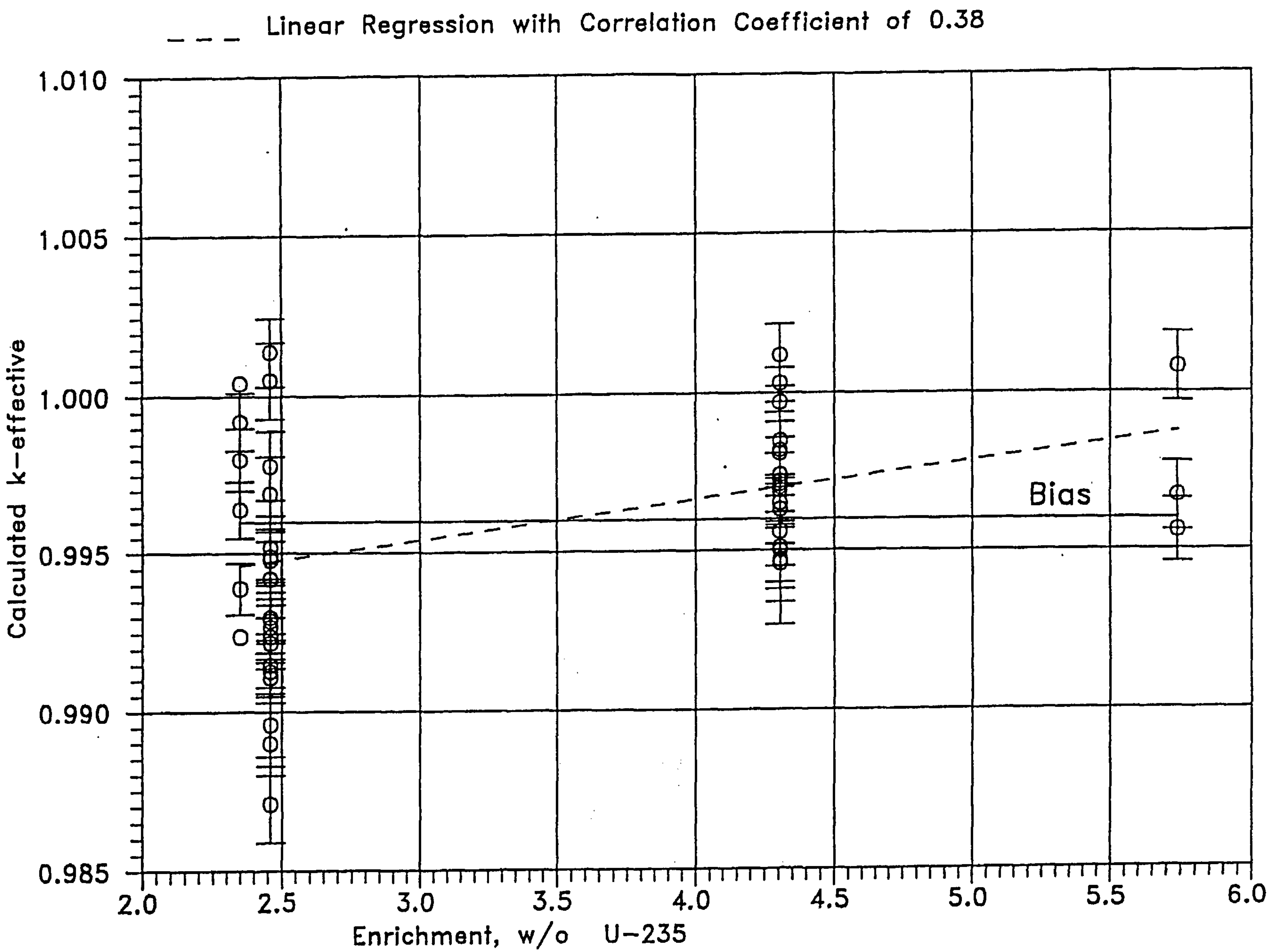

FIGURE 4A.4 KENO CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

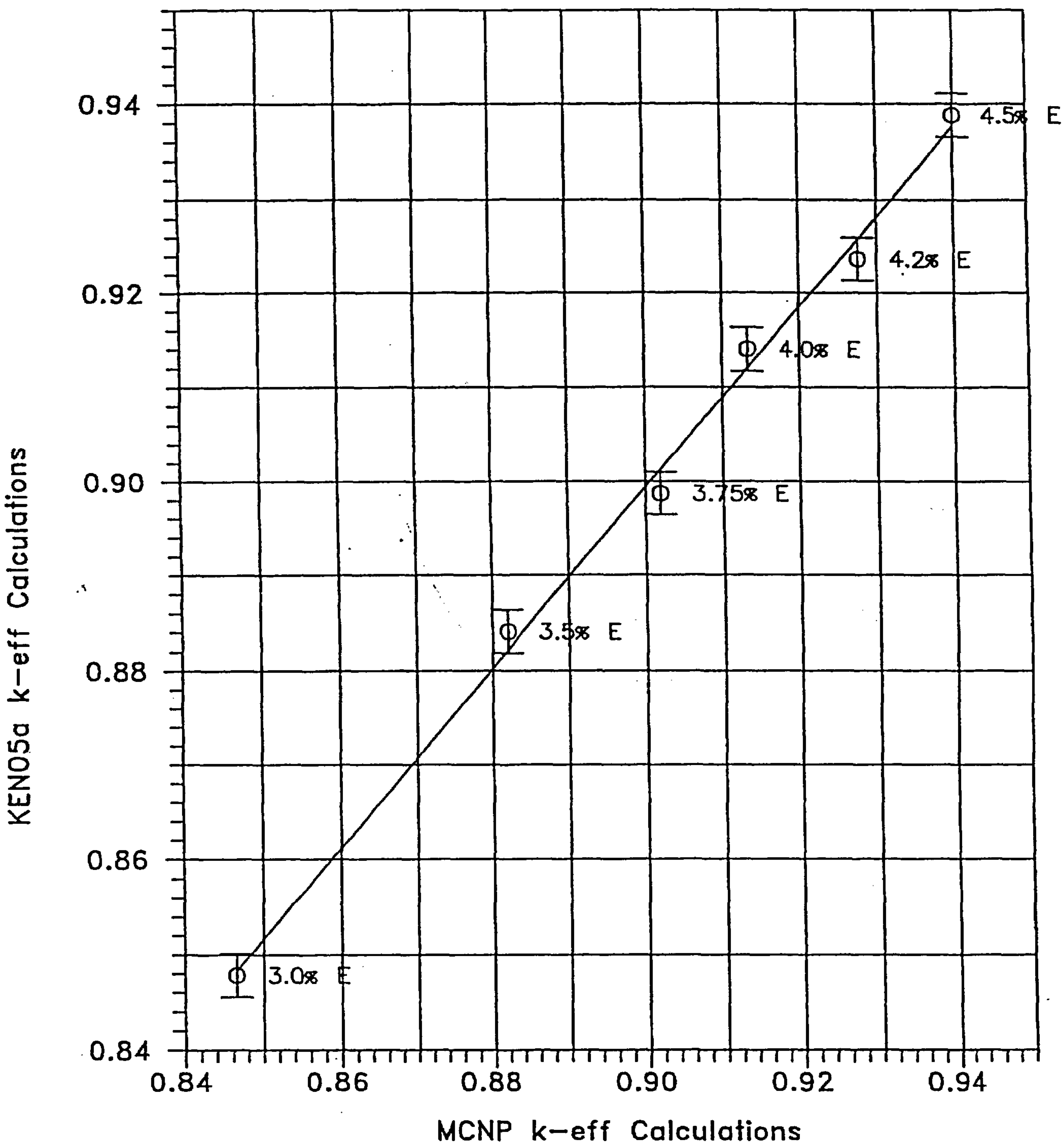

FIGURE 4A.5 COMPARISON OF MCNP AND KENO5A CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

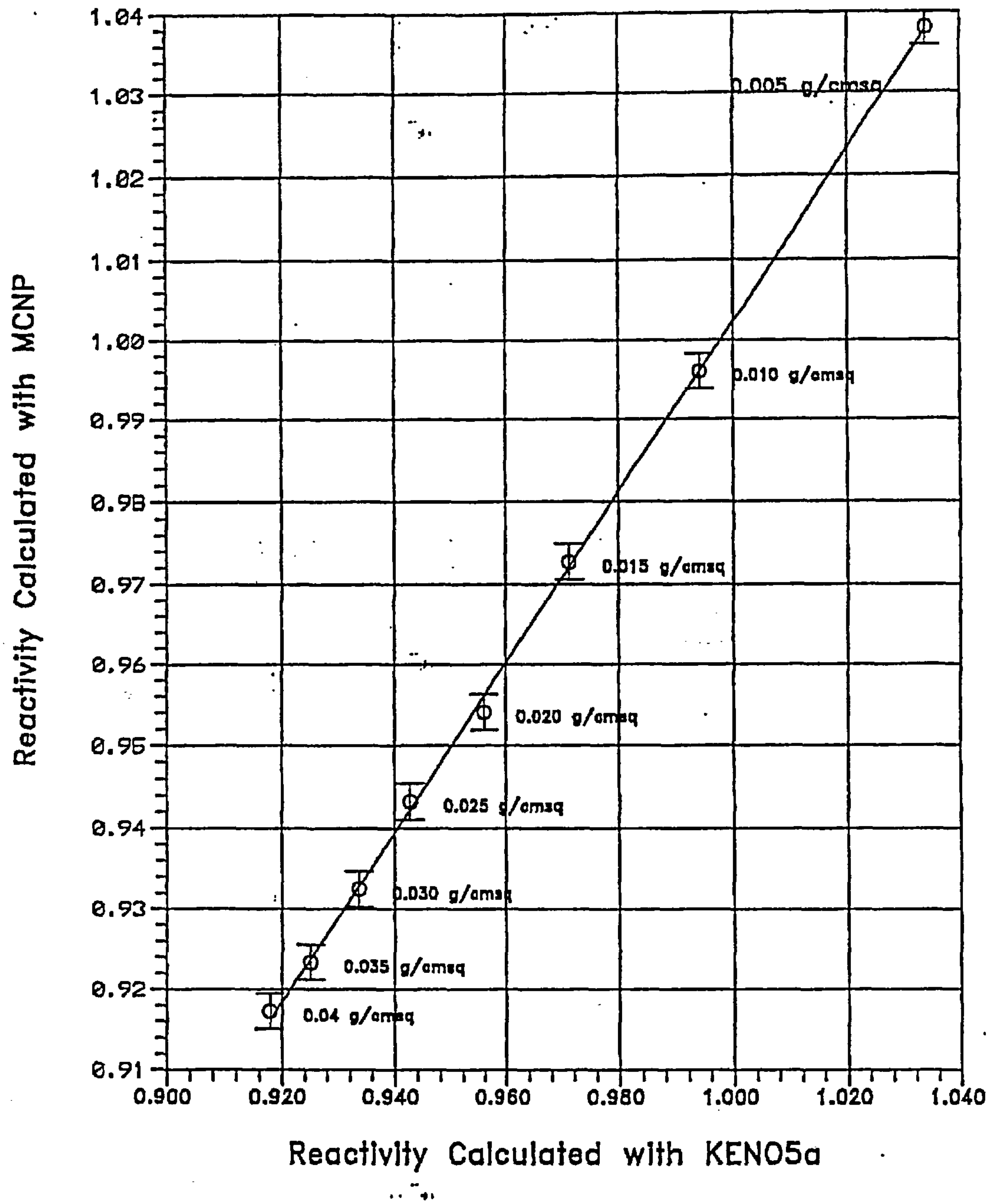

FIGURE 4A.6 : COMPARISON OF MCNP AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES