

St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
Spent Fuel Pool Soluble Boron Credit

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Enclosure 3

NON-PROPRIETARY
HOLTEC
LICENSE AMENDMENT
REPORT

(Bound Report)



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ST. LUCIE UNIT 1 SPENT FUEL POOL STORAGE RACK BORAFLEX DEGRADATION REMEDY

FOR

FPL

Holtec Report No: HI-2022940

Holtec Project No: 1237

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1.0 INTRODUCTION

The two-unit St Lucie Plant (PSL) is located on Hutchinson Island in St. Lucie County, Florida, south of the city of Fort Pierce. The plant consists of two Combustion Engineering Pressurized Water Reactor (PWR) nuclear units. Unit 1 has been in commercial operation since 1976 and Unit 2 since 1983.

The existing Unit 1 spent fuel storage racks credit Boraflex™ as a neutron absorber to ensure subcriticality of the stored fuel. It is known [1] that Boraflex degrades during service conditions within the Spent Fuel Pool (SFP). The existing PSL Unit 1 Technical Specifications provide a description of the racks, including Boraflex, and include storage limitations based on reliance on the Boraflex. FPL seeks to re-license the storage racks in Unit 1 by crediting soluble boron in the pool water coupled with specific rules on fuel positioning to ensure subcriticality in lieu of crediting Boraflex as a neutron absorber. This report provides the design basis, analysis methodology, and results for the re-evaluation of the fuel storage racks in the St. Lucie Unit 1 Spent Fuel Pool without consideration of the Boraflex neutron absorber.

Neglecting Boraflex in the fuel storage criticality evaluations does not require any physical changes to the pool storage or rack configuration or require replacement of any storage racks. Safe storage will continue to be assured through rules on positioning fuel and by the neutron absorption provided by soluble boron in the SFP coolant. The soluble boron concentration is controlled by Technical Specifications.

St. Lucie Unit 1 has a current licensed storage capacity of 1,706 fuel assemblies. The existing high-density racks were installed subsequent to a reracking analysis effort performed by Holtec in 1987. Holtec licensing report HI-87105 [2] provides a detailed summary of the evaluations performed to support the re-licensing effort. Since there will not be any physical changes required to the pool, storage racks, or fuel contained within the racks, the analyzed configurations and results documented in the previous Holtec report remain valid with respect to structural, thermal-hydraulic, radiological and accident conditions. However, the racks have been re-evaluated for the criticality considerations discussed in detail herein.

The methodologies employed to perform the rack criticality evaluations are a direct evolution of previous license applications. This report documents the analyses performed to demonstrate that the racks meet all governing requirements of the applicable codes and standards, in particular, 10CFR50.68(b)(4).

Section 2 of this report provides an abstract of the design and material information for the existing SFP storage racks. Section 3 provides an overview of the methodology used in an evaluation of postulated spent fuel pool boron dilution events and a summary of the results.

Section 4 provides a summary of the methods and results of the criticality evaluations performed for the spent fuel pool storage racks. The criticality safety analysis requires that the effective neutron multiplication factor (k_{eff}) be less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity and with the pool flooded with borated water at a temperature corresponding to the highest reactivity. In addition, the analysis requires that k_{eff} remains less than 1.0 following the assumed loss of soluble boron in the pool water, i.e. assuming unborated water in the spent fuel pool. The maximum calculated reactivities include a margin for uncertainty in reactivity calculations, including manufacturing tolerances, and are calculated with a 95% probability at a 95% confidence level [5].

Thermal-hydraulic considerations are discussed in Section 5. Rack module structural analysis considerations are presented in Section 6. The structural qualification also requires that subcriticality of the stored fuel array be maintained under all postulated accident scenarios. The structural consequences of these postulated accidents are addressed in Section 7 of this report.

Section 8 establishes the continued adequacy of the SFP structure. The radiological considerations are documented in Section 9. Section 10 summarizes a cost/benefit and environmental assessment prepared by FPL to address the Boraflex degradation remediation proposal.

All computer programs utilized to perform the criticality analyses documented in this report are benchmarked and verified. Holtec International has utilized these programs in numerous license applications over the past decade.

The analyses presented herein demonstrate that the Unit 1 SFP rack module arrays remain subcritical when soluble boron and specific rules on fuel assembly positioning are credited for reactivity control in lieu of Boraflex.

1.1 References

- [1] NRC Information Notice 95-38, Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Tacks," September 1995.
- [2] Holtec International Report HI-87105, "Licensing Report for Reracking St. Lucie Unit 1 Fuel Pool," Revision 3, dated April 1987.
- [3] Not Used.
- [4] American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel Code, Section III, 1989 Edition, Subsection NF, and Appendices.
- [5] M.G. Natrella, Experimental Statistics, National Bureau of Standards, Handbook 91, August 1963.

The existing PSL Unit 1 high-density fuel racks consist of individual cells with 8.65 inch (nominal) square cross-section, each of which accommodates a single fuel assembly. A total of 1706 cells are arranged in 17 distinct modules of varying sizes of which four are Region 1 design with water gaps between cells, and thirteen are Region 2 design with no water gaps (see Table 2.1). Figure 2.1 shows the arrangement of the rack modules in the spent fuel pool.

The high density racks are engineered to achieve maximum protection against structural loadings (arising from ground motion, thermal stresses, etc.), the maximum number of available storage locations, and to maintain fuel assemblies in a subcritical array. Each rack module is equipped (see Figures 2.2 and 2.3) with girdle bars measuring $\frac{3}{4}$ inches thick by 3- $\frac{1}{2}$ inches high. The girdle bar thickness on each rack ensures that a minimum gap of 1- $\frac{1}{2}$ inches is maintained between modules. Table 2.1 gives the relevant design data for each region. The modules of the two regions are of eight different types. Tables 2.2 and 2.3 summarize the physical data for each module type.

Table 2.1 RACK DESIGN DATA		
Region	Nominal Cell Pitch (inches)	Nominal Flux Trap Gap (inches)
1	10.12	1.12
2	8.86	0.0

Table 2.2
Table of Module Data

Module Identification	Number of Modules	Number of Cells in N-S Direction	Number of Cells in E-W Direction	Total Number of Cells per Module
Region 1 A1 to A2	2	9	9	81
Region 1 B1 to B2	2	9	10	90
Region 2 C1 to C4	4	13	9	117
Region 2 D1 to D3	3	13	8	104
Region 2 E1 to E2	2	11	8	88
Region 2 F1	1	12	8	96
Region 2 G1 to G2	2	12	9	108
Region 2 H1	1	13	8	96

Table 2.3

Module Dimensions and Weight

Module Identification	Nominal Cross Section Dimensions (inches)		Estimated Dry Weight per Module (lbs)
	N-S	E-W	
Region 1 A1 to A2	90-1/4	90-1/4	26,700
Region 1 B1 to B2	90-1/4	100-7/16	29,800
Region 2 C1 to C4	115-11/16	80-1/16	24,100
Region 2 D1 to D3	115-11/16	71-3/16	21,500
Region 2 E1 to E2	97-7/8	71-3/16	18,200
Region 2 F1	106-3/4	71-3/16	19,800
Region 2 G1 to G2	106-3/4	80-1/16	22,300
Region 2 H1	115-11/16	71-3/16	19,800

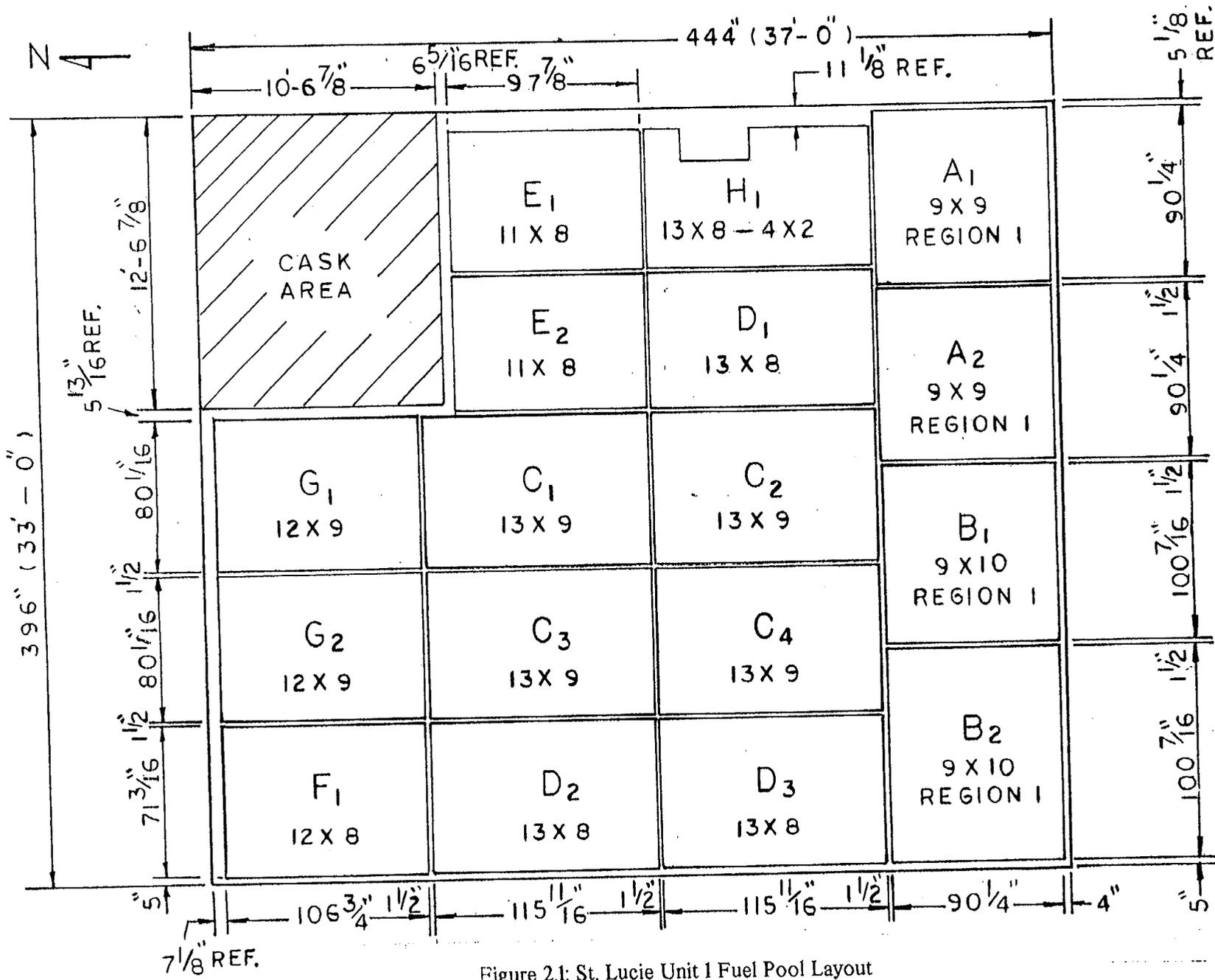


Figure 2.1: St. Lucie Unit 1 Fuel Pool Layout

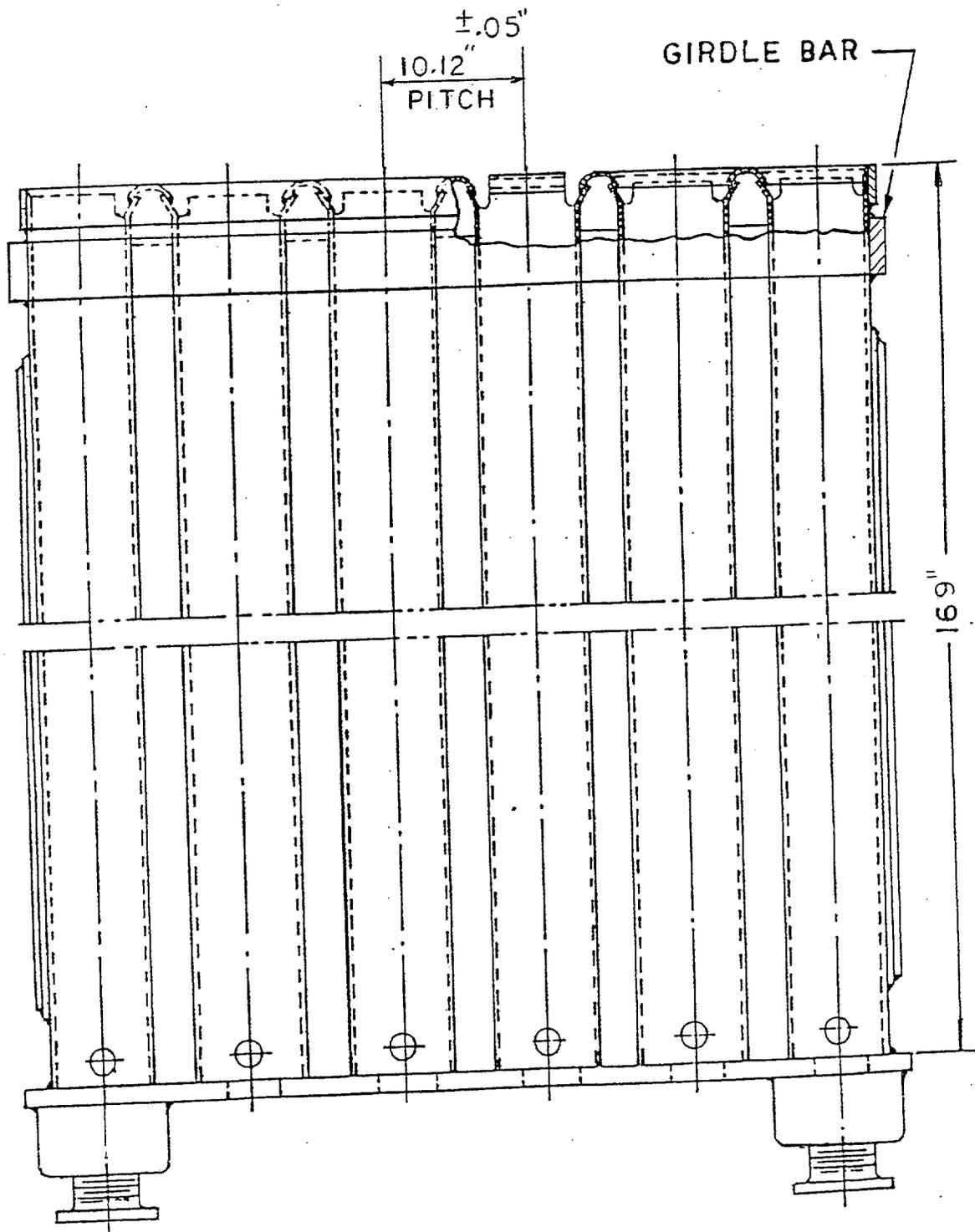


Figure 2.2 TYPICAL RACK ELEVATION-REGION 1

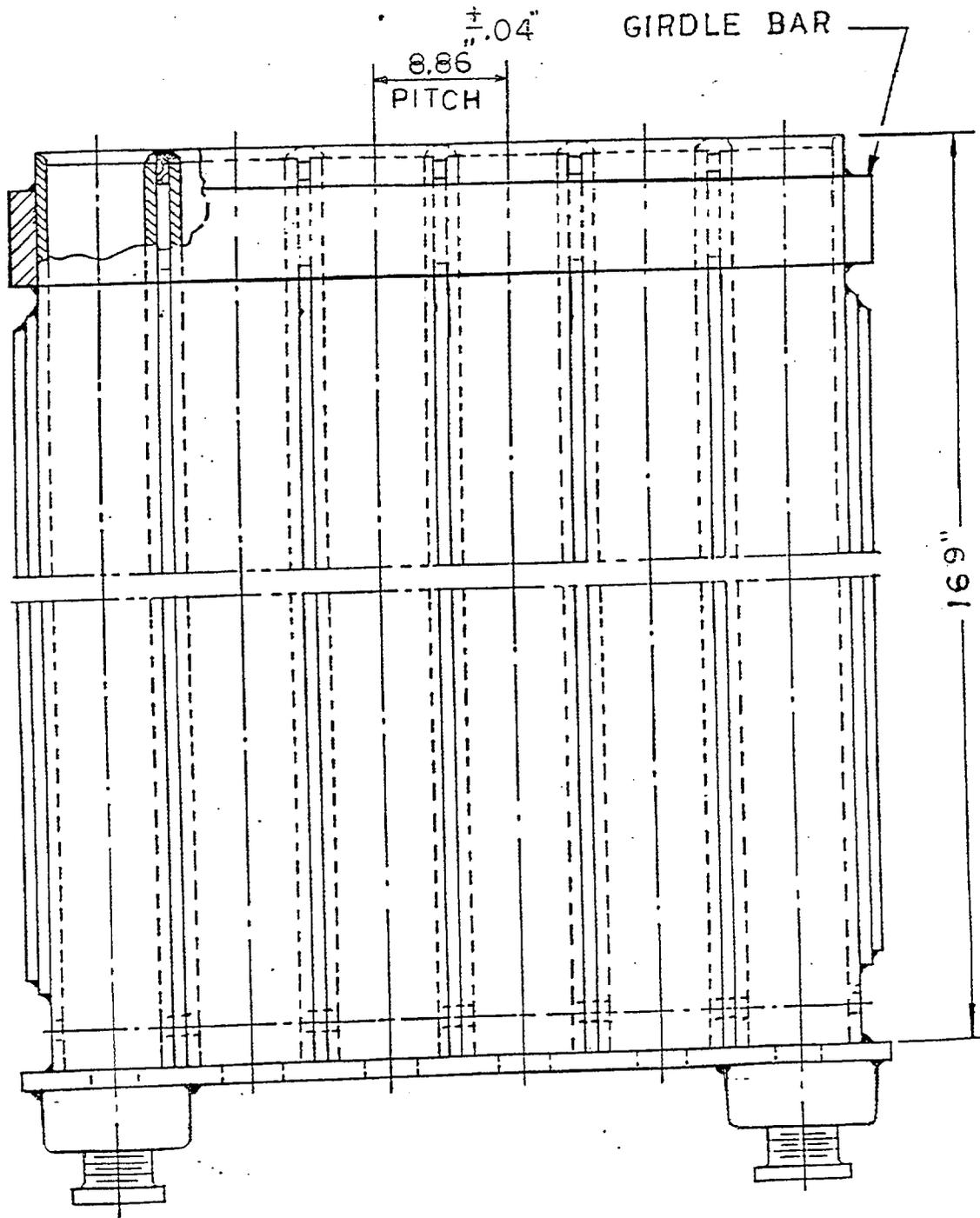


Figure 2.3 TYPICAL RACK ELEVATION-REGION 2

Florida Power and Light has prepared an evaluation that examines the potential for an inadvertent dilution of the St. Lucie Unit 1 spent fuel pool. The dilution scenarios presented in this report were developed after identifying the plant systems and components that interface with the Unit 1 fuel pool. Periodic activities performed by plant operators that involve the spent fuel pool or systems interfacing with the spent fuel pool were also considered. Time periods required for a loss of reactivity margin to an effective neutron multiplication factor (k_{eff}) of 0.95 have been quantified.

Acceptance criteria are met if the evaluation concludes that sufficient time is available to detect and mitigate any credible dilution event before the k_{eff} design basis value is exceeded.

Typically, this analysis postulates the occurrence of multiple failures, as in the failure to correctly position a valve at the completion of an evolution coincident with a failure of an annunciator in the control room to alarm, or the failure of personnel to appropriately respond to an alarm. The evaluation did not consider the simultaneous occurrence of an inadvertent fuel pool dilution and a mis-positioned fuel assembly to be a credible scenario.

This analysis concludes that there are no credible spent fuel pool dilution events that could cause the soluble boron concentration to decrease from the assumed initial condition of 1720 ppm to a value such that k_{eff} equals 0.95.

The boron dilution analysis is provided as an enclosure to the license amendment request for soluble boron credit.

4. Criticality Safety Analyses

4.1 Introduction and Summary

Overview

This section documents a new criticality safety analysis for the storage of PWR nuclear fuel in existing Region 1 & 2 style fuel storage racks installed in the spent fuel pool (SFP) at the St. Lucie Unit 1 nuclear power plant. The spent fuel pool currently contains about 1350 fuel assemblies and is licensed to store up to 1706 assemblies. The analysis has been performed to qualify the existing racks from a criticality perspective under the assumption of a complete loss of the Boraflex™ neutron poison.

The existing spent fuel pool Region 1 & 2 style racks analyzed herein are used for the storage of irradiated fuel, and for fuel inspection, testing, and fuel reconstitution. This analysis excludes the new Region 1 cask pit rack, which is designed to accommodate fresh fuel and a portion of recently irradiated offload fuel.

The objective of the analysis is to qualify the existing SFP racks for the current spent fuel inventory and for future fuel discharges from Unit 1, without the need for additional neutron absorber inserts in the storage racks to offset an assumed loss of the Boraflex. This analysis credits the presence of soluble boron in the spent fuel pool, and the presence of control element assemblies (CEAs) placed in selected fuel assemblies. In order to achieve this analysis objective, it is necessary to group together fuel assemblies having similar reactivity characteristics and to establish different localized storage arrangements (i.e., checkerboard patterns) within the racks for assemblies with unique reactivity groupings.

Fuel Assembly Types Analyzed

A total of seven fuel assembly types were developed to reflect different reactivity groupings. The following table lists each type by number, the type description used in this report, and its minimum burnup requirement based on an initial enrichment of 4.5 weight percent:

Type	Description	Minimum Burnup @ 4.5 w/o U-235
1	Region 1 Case 2 "once burned"	17.51 GWd/MTU
2	Region 1 Case 3 "twice burned"	24.95 GWd/MTU
3	Region 1 Case 3 "lower reactivity" and Region 2 Case 4 "high reactivity"	34.66 GWd/MTU
4	Region 2 Case 1 "high reactivity"	42.98 GWd/MTU
5	Region 1 Case 2 "low reactivity"	44.00 GWd/MTU
6	Region 2 Case 5 "medium reactivity"	48.80 GWd/MTU
7	Region 2 Case 1 "low reactivity"	56.20 GWd/MTU

Fuel Storage Configurations Analyzed

A total of five fuel storage configurations (cases) with different fuel assembly types were analyzed, as follows:

- Case 1: Region 2, Checkerboard of high and low reactivity fuel assemblies
- Case 2: Region 1, Checkerboard of once burned and low reactivity fuel
- Case 3: Region 1, Checkerboard of twice burned and lower reactivity fuel
- Case 4: Region 2, Checkerboard of high reactivity fuel assemblies with and without CEAs
- Case 5: Region 2, Medium reactivity fuel assemblies only

Burnup vs Enrichment Curves

For each storage configuration above, and for each assembly type in a checkerboard array, the minimum required burnup has been determined as a function of the initial enrichment of the fuel.

These functions, also termed burnup versus enrichment curves, are established as polynomial functions in the form of:

$$BU = A * E^2 + B * E + C$$

with:

BU	Burnup in GWD/MTU
E	Initial Enrichment (wt %)
A,B,C	Coefficients

The current inventory of irradiated fuel at St. Lucie Unit 1 contains fuel assemblies with axial blankets, as well as fuel assemblies without axial blankets. Coefficients for all cases, for non-blanketed and blanketed assemblies, and for all relevant post-irradiation cooling times are listed in Table 4.1.1 and 4.1.2, respectively.

Special Fuel Loading Rules

A portion of the periphery of Region 2 storage racks faces the fuel pool wall. This part of the rack is analyzed for higher reactivity fuel, crediting the increased neutron leakage in this area. Also, a designated area is established in Region 2 racks for fuel inspection and reconstitution, allowing a limited number of fresh fuel assemblies to be placed in a predefined pattern surrounded by empty cells. Reactivity effects of interfaces between the adjacent, potentially dissimilar storage arrangements have also been evaluated to assure that under all credible conditions, the fuel pool reactivity will not exceed the regulatory limit of 0.95. These conditions lead to following requirements:

1. Normally, each rack module will contain only one of the above listed configurations, i.e., Cases 1, 4, or 5 for a Region 2 rack, and Case 2 or 3 for a Region 1 rack. However, a rack module may contain more than one permissible configuration if an empty row is used to separate fuel stored in one configuration from fuel stored in a different configuration.

2. Checkerboard patterns must be aligned across the gap between Region 1 rack modules, i.e., a high reactivity fuel assembly on one side of the gap must face a low reactivity assembly on the opposite side of the gap (i.e., “face-adjacent”).
3. Checkerboard patterns need not be aligned across the gap between Region 2 rack modules, i.e., a high reactivity assembly on one side of the gap can face a high reactivity assembly on the opposite side of the gap.
4. The outer row of cells of Region 2 racks facing the pool wall or the cask pit wall is qualified to accept assemblies meeting the burnup and enrichment requirements for Case 4 (Type 3 fuel assemblies), and need not contain a CEA, regardless of the fuel assembly characteristics in the remainder of the rack.
5. Up to 4 (four) fresh assemblies or fuel rod baskets can be placed in a storage rack module having a Case 1 or Case 5 configuration, as long as each fresh assembly or rod basket directly faces 4 empty cells, and each of the diagonal cells is either empty or contains a Type 4, 6, or 7 assembly. Empty cells may contain non-actinide material, such as an empty fuel assembly skeleton, or other hardware, so long as the material occupies no more than 75% of the cell volume.

Analysis Results

Analyses demonstrate that the effective neutron multiplication factor (k_{eff}) for all these cases is less than or equal to 0.95 when the storage racks are assumed to be fully loaded with fuel of the highest permissible reactivity and the pool is assumed to be flooded with borated water at a temperature corresponding to the highest reactivity. In addition, these analyses demonstrate that k_{eff} is less than 1.0 when the fuel pool is assumed to be flooded with unborated water. The maximum calculated values of the neutron multiplication factor include a margin for uncertainty in reactivity calculations, including manufacturing tolerances, and are calculated with a 95% probability at a 95% confidence level [4.7.1].

A minimum soluble boron concentration of 500 ppm must be maintained in the spent fuel pool to ensure that the effective neutron multiplication factor (k_{eff}) is less than or equal to 0.95 under all normal conditions.

Reactivity effects of accident conditions have also been evaluated. The most limiting accident condition involves the placement of a fresh fuel assembly between and directly adjacent to two other fresh fuel assemblies previously placed into a Region 2 rack module for inspection, testing or reconstitution. A minimum soluble boron concentration of 1090 ppm must be maintained in the spent fuel pool to ensure that the effective neutron multiplication factor (k_{eff}) is less than or equal to 0.95 under this condition.

St. Lucie Unit 1 Technical Specifications require that the fuel pool soluble boron concentration be maintained ≥ 1720 ppm at all times.

4.2 ACCEPTANCE CRITERIA

The objective of this analysis is to ensure that the effective neutron multiplication factor (k_{eff}) is less than or equal to 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity and with the pool flooded with borated water at a temperature corresponding to the highest reactivity. In addition, the analysis shall ensure that for all storage configurations considered, k_{eff} is less than 1.0 when the fuel pool is assumed to be flooded with unborated water. The maximum calculated values of the neutron multiplication factor shall include a margin for uncertainty in reactivity calculations, including manufacturing tolerances, and are calculated with a 95% probability at a 95% confidence level [4.7.1].

4.3 ASSUMPTIONS

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

- 1) Moderator is borated or unborated water at a temperature that results in the highest reactivity, as determined by the analyses.
- 2) Neutron absorption in minor structural members is neglected, i.e., spacer grids are replaced by water.
- 3) Absorber rods present in some fuel assemblies are conservatively assumed to be fuel rods.
- 4) The effective multiplication factor of an infinite radial array of fuel assemblies or assembly patterns was used in the analyses, except for the assessment of peripheral and interface effects, and for certain abnormal/accident conditions where neutron leakage is inherent.
- 5) For the moderator temperature during fuel depletion, the highest core average value found at any axial location is used. This is conservative, since depletion with a higher moderator temperature results in higher fuel reactivity.

4.4 DESIGN AND INPUT DATA

4.4.1 Fuel Assembly and Fuel Insert Specification

The design specifications for the Combustion Engineering (CE) and Framatome (FR) fuel assemblies, which were used for this analysis, are given in Table 4.4.1. Table 4.4.2 shows the specifications of the CEA fuel inserts used in the evaluations. Both tables also contain the applicable tolerances. The operating parameters used in the depletion analysis are given in Table 4.4.3.

4.4.2 Holtec Storage Rack Specification

Specifications of the storage racks used in the criticality evaluations are summarized in Table 4.4.4 for the Region 1 and the Region 2 racks. Figures 4.4.1 and 4.4.2 show sketches of the cells for the Region 1 and Region 2 racks, respectively, indicating all relevant nominal dimensions.

4.5 METHODOLOGY

The principal method for the criticality analysis of the storage racks is the three-dimensional Monte Carlo code MCNP4a [4.7.2]. MCNP4a is a continuous energy three-dimensional Monte

Carlo code developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been used previously and verified for criticality analyses and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V and ENDF/B-VI.

Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009 with an uncertainty of ± 0.0011 for MCNP4a, evaluated with a 95% probability at the 95% confidence level [4.7.1]. The calculations for this analysis utilized the same computer platform and cross-section libraries used for the benchmark calculations discussed in Appendix A.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information has been used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in storage rack criticality calculations. Based on these studies, the final calculations use a minimum of 10,000 histories per cycle, a minimum of 25 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). Further, the output was reviewed to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between precision and computation time for design basis calculations.

Analyses of fuel depletion during St. Lucie Unit 1 power operation were performed with CASMO-4 (using the 70-group cross-section library), a two-dimensional multigroup transport theory code based on capture probabilities [4.7.3-5]. CASMO-4 is used to determine the isotopic composition of the spent fuel. In addition, the CASMO-4 calculations are restarted in the storage rack geometry to yield the two-dimensional infinite multiplication factor (k_{inf}) for the storage rack. These restart calculations are used to determine the reactivity effect of fuel and rack

tolerances, and to perform various studies. For all calculations in the spent fuel pool racks, the Xe-135 concentration in the fuel is conservatively set to zero.

4.6 ANALYSIS

This section describes the calculations that were used to determine the acceptable storage criteria for both the Region 1 and Region 2 style racks and it summarizes their results. In addition, this section discusses the postulated abnormal and accident conditions applicable to St. Lucie Unit 1 fuel pool storage.

Unless otherwise stated, all calculations assumed nominal characteristics for the fuel and the fuel storage cells. The effect of manufacturing tolerances is accounted for with a reactivity adjustment as discussed below.

All calculations are made using an explicit model of the fuel and storage cell geometry. The MCNP models contain a 2-by-2 array of cells surrounded by periodic boundary conditions. This represents an infinite checkerboard array. In CASMO, only a single cell is modeled. Since CASMO-4 is a two-dimensional code, the fuel assembly hardware above and below the active fuel length is not represented. The three-dimensional MCNP4a models that included axial leakage assumed 30 cm of water above and below the active fuel length. Additional models with more than four cells and with different boundary conditions were developed for MCNP to investigate the effect of rack module interfaces and to evaluate accident conditions. These models are discussed in the appropriate sections below.

4.6.1 Bounding Fuel Assemblies

To determine the bounding assembly, calculations are performed for both assembly types listed in Table 4.4.1, and for both the upper bound and lower bound cladding thickness listed in that table. Further, calculations are performed for various enrichments, cooling times and burnups,

and for both Region 1 and Region 2 racks. Typical results are shown in Table 4.6.1, and demonstrate that for Region 1, the FR 14x14 assembly with a cladding thickness of 0.028 inches is the bounding assembly, whereas for Region 2, the CE 14x14 assembly with a cladding thickness of 0.026 inches is the bounding assembly. These assemblies are therefore used in all further calculations for the respective rack types.

4.6.2 Pool Water Temperature Effects

Pool water temperature effects on reactivity at 0 ppm soluble boron have been calculated with CASMO-4 and the results are presented in Table 4.6.2. The results in this table show that the spent fuel pool temperature coefficient of reactivity is positive for assemblies without CEAs (Region 1 and Region 2). In these cases, a higher temperature results in a higher reactivity, and the maximum normal pool temperature of 150 °F is therefore the bounding condition. However, for assemblies containing CEAs (only credited in Region 2 calculations), the temperature coefficient is negative, i.e. a lower temperature results in a higher reactivity. Consequently, all CASMO calculations for assemblies without CEAs are evaluated at 150 °F, whereas CASMO calculations for assemblies crediting CEAs are evaluated at 4 °C, which corresponds to the highest water density. For cases containing only assemblies without CEAs (cases 1, 2, 3 and 5), the tolerances for 150 °F are applied. For Case 4, which uses a checkerboard of assemblies with and without CEAs, conservatively the maximum of the tolerance effect is applied. Pool water temperature effects on reactivity have also been evaluated in the presence of soluble boron; these effects are reported on Tables 4.6.7 and 4.6.8.

In MCNP, the Doppler treatment and cross-sections are valid only at 300K (27 °C). Therefore, a conservative Δk value is determined in CASMO-4 from 20 °C (68 °F) to 150 °F, and is included in the final k_{eff} calculation as a bias. Conservatively, the maximum value of this bias for each rack type shown in Table 4.6.2 is used in the final k_{eff} calculations. Although Case 4 contains assemblies with CEAs, which have a negative temperature coefficient of reactivity in the storage racks, a bias value derived from assemblies without CEAs is applied. This is conservative, since the reactivity effect of a temperature change between 20 °C and 150 °F for assemblies without

CEAs is larger than the reactivity effect of the temperature change from 20 °C to 4 °C for assemblies containing CEAs.

Fuel pool water temperatures exceeding 150 °F are considered accident conditions, and are discussed in Section 4.6.14.1.

4.6.3 Uncertainties Due to Manufacturing Tolerances

In the calculation of the final k-infinity (k_{inf}), the effect of manufacturing tolerances on reactivity must be included. CASMO-4 was used to perform these calculations. Factors considered include tolerances of the rack dimensions (see Table 4.4.4), tolerances of the fuel dimensions (see Table 4.4.1) and tolerances of the CEA specifications (see Table 4.4.2). In addition to the tolerances specified in these tables, an enrichment tolerance of 0.05 wt% is analyzed. As was done to identify the bounding assembly, calculations are performed for Region 1 and Region 2 racks, and CEAs, at a variety of enrichments, cooling times and burnups. The reference condition is the condition with nominal dimensions and properties. To determine the Δk associated with a specific manufacturing tolerance, the k_{inf} calculated for the reference condition is compared to the k_{inf} from a calculation with the tolerance included. All of the Δk values from the various tolerances represent independent effects and may be statistically combined (square root of the sum of the squares) to determine the final reactivity allowance for manufacturing tolerances. Only the positive Δk values (signifying increasing reactivity) were used in the statistical combination.

Table 4.6.3 shows the individual reactivity effects of tolerances, which when statistically combined, result in the highest total reactivity effect for Region 1, Region 2 and Region 2 containing assemblies with CEAs.

4.6.4 Uncertainty in Depletion Calculations and Assembly Burnup

CASMO-4 was used to perform the depletion calculations. Since critical experiment data with spent fuel is not available for determining the uncertainty in burnup-dependent reactivity calculations, an allowance for uncertainty in reactivity was assigned based upon other considerations. This analysis assumes the uncertainty in depletion calculations is less than or equal to 5% of the total reactivity decrement, and it assigns a burnup dependent uncertainty in reactivity for burnup calculations on this basis [4.7.6]. Additionally, the uncertainty of the assembly burnup value is 2.5 %. The reactivity effect of this uncertainty in burnup is determined and then these uncertainties are statistically combined with the other reactivity allowances to determine the maximum k_{eff} for comparison with the limit of 0.95 for normal and accident conditions.

4.6.5 Isotopic Compositions

To perform the criticality evaluation for spent fuel in MCNP, the isotopic composition of the fuel is calculated with the depletion code CASMO and then this isotopic composition is specified as input data to MCNP. Three isotopes or grouped isotopes in CASMO do not have a corresponding cross section in the MCNP cross section library. These are Pm-148M, and the lumped fission products LFP1 and LFP2. To account for these isotopes in the MCNP calculations, an equivalent amount of B-10 is calculated for each, and this B-10 amount is used in the MCNP calculation instead. The B-10 amount is specified through a multiplier on the atom density for each isotope, i.e. the B-10 atom density is calculated to be the Pm-148M / LFP1 / LFP2 atom density calculated in CASMO multiplied by a constant factor. For each of the isotopes or isotope groups, a bounding factor is determined, and applied for the MCNP calculations.

The CASMO calculations to obtain the isotopic compositions for MCNP were performed generically, with one calculation for each rack type, enrichment and cooling time, using burnup

increments of 2.5 GWD/MTU or less. The isotopic composition for any given burnup is then determined by linear interpolation.

4.6.6 Effect of Gadolinium

At higher enrichments, assemblies contain up to 20 rods with Gadolinium (Gd) added to the fuel. These rods are in specific locations around the control rod guide tubes. Rods containing Gadolinium also have a lower U-235 enrichment than do rods without Gd in the same assembly. For a maximum assembly enrichment of 4.5 wt%, the highest U-235 enrichment in the rods with gadolinium will be approximately 2.6 wt%. A comparison of depletion calculations for fuel assemblies of equivalent enrichment, with and without Gd in these rods, shows that the assembly without Gd has a significantly higher reactivity for most conditions, and the presence of Gadolinium is therefore conservatively neglected in all further calculations.

4.6.7 Effect of Distributed Enrichments

As noted in the previous paragraph, some assemblies contain fuel rods with lower enrichments around the guide tubes. As an example, an assembly with a maximum fuel rod enrichment of 4.5 wt% can have up to 20 fuel rods with enrichments as low as 2.6 wt% (see previous section). In addition, an assembly can have up to 40 fuel rods, not containing Gd, at a reduced enrichment to control radial power peaking. In an assembly with a maximum fuel rod enrichment of 4.5 wt%, these additional 40 rods would typically be at an enrichment of 4.1 wt%. As a result, the planar average enrichment of a fuel assembly can be significantly lower than the maximum pellet enrichment. To show that it is acceptable to use the maximum planar average enrichment when determining the minimum required burnup from burnup vs. enrichment curves, calculations were performed for assemblies with radially distributed enrichments, and compared to calculations where all rods were set to a conservatively calculated planar average enrichment.

The calculations performed using the planar average enrichment result in slightly higher reactivity values than do the calculations performed using the actual assembly enrichment

distribution. It is therefore acceptable to use the maximum planar average enrichment of an assembly to determine the minimum required burnup.

4.6.8 Eccentric Fuel Assembly Positioning

The fuel assembly is assumed to be normally located in the center of the storage rack cell. Nevertheless, MCNP4a calculations assumed the fuel assemblies were positioned in the corner of the storage rack cell (a four-assembly cluster at closest approach). These calculations indicated that eccentric fuel positioning increases the reactivity of Region 1 by up to 0.0127 delta-k, and decreases the reactivity in Region 2. For Region 1 calculations, the maximum difference in reactivity of 0.0127 delta-k is included in the uncertainties in the final k_{eff} calculations.

4.6.9 Reactivity Effect of Axial Burnup and Enrichment Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As power operation progresses, the axial 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 (having less than average burnup) exists in a region of lower reactivity worth due to the ambient neutron leakage. Consequently, it would be expected that over most of their operating history, distributed burnup fuel assemblies would exhibit a slightly lower reactivity than that calculated for an assembly where all portions of fuel rods have the average burnup. As operation 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.

Generic analytic results of the axial burnup effect for assemblies without axial blankets were presented in [4.7.7]; these results are based upon comparisons of calculated and measured axial burnup distributions. These analyses confirm the minor and generally negative reactivity effect of the axially distributed burnup, which becomes positive at burnups greater than about 30 GWD/MTU. The trends observed [4.7.7] suggest the possibility of a small positive reactivity

effect above 30 GWD/MTU increasing to slightly over 1% Δk at 40 GWD/MTU. Since the required burnup for some enrichments and cases is greater than 30 GWD/MTU, the reactivity effect of the axially distributed burnup must be considered.

The St. Lucie Unit 1 plant also possesses fuel assemblies with natural (0.71 wt% ^{235}U) and low enriched (2.6 wt% ^{235}U) axial blankets on the ends, which effect the axial burnup distribution.

Calculations have been performed for the various axial burnup and enrichment variations, and the results were compared with a reference case, i.e. a case with an assumed axially constant burnup and enrichment. The results of this comparison indicate that, as expected, there is a positive reactivity effect from considering the axial burnup and enrichment distribution at higher burnup and cooling times for non-blanketed assemblies and for assemblies with enriched blankets. The effect of the axial burnup and enrichment distribution is considered in the calculations that establish burnup vs. enrichment curves, by conservatively performing calculations with both a uniform and non-uniform axial burnup and enrichment distribution, and selecting the higher of the resulting reactivity values as the representative reactivity value. Enriched blankets are used in all blanketed calculations for conservatism.

In addition, the spent fuel pool contains Vessel Flux Reduction assemblies, which contain depleted uranium at an axially constant initial enrichment of about 0.3 wt%. Although the reactivity of such assemblies initially increases slightly with burnup, the reactivity is still significantly below the reactivity of all other permissible assemblies in the pool. Therefore, these assemblies can be placed in any location in the racks designated for a fuel assembly, and no further evaluation is required with these assemblies for any of the cases.

4.6.10 B-10 Depletion in CEAs

CEAs are typically withdrawn from the active fuel region during full power operation of the core. A significant depletion of the B-10 in the CEAs therefore does not occur, and the initial B-10

loading of the CEA is used in the analyses. This lack of significant depletion of the B-10 in the CEAs is verified by measuring the CEA worth during startup physics testing after each reload.

However, to evaluate the effect of a conservative value for potential B-10 depletion, an additional calculation has been performed, wherein the B-10 concentration in the CEA was reduced by 30% in the lower 40 inches of each control rod finger. The results show that even this conservative reduction of the B-10 concentration does not lead to a significant difference in reactivity. It is therefore acceptable to model the CEA with the initial B-10 loading. Note that the dimensional tolerances of the CEA, including initial B-10 loading, were evaluated for their effects on reactivity and included in the total uncertainty calculation (Section 4.6.3).

4.6.11 Calculation of Burnup versus Enrichment Curves

This analysis considers the following parameters and parameter combinations:

- Two fuel storage rack styles, with a total of five different fuel loading configurations.
- Fuel enrichments between 1.9 and 4.5 wt% ^{235}U .
- Assemblies with and without axial blankets.
- Cooling times between 0 and 20 years

Not all combinations of enrichment and cooling time are of practical relevance. The parameter combinations which are required to ensure that all current and future discharged fuel assemblies can be safely loaded into the racks are summarized in Table 4.6.4, and burnup vs. enrichment curves are determined for these parameter combinations. Prior analysis has indicated that it is necessary to account for the presence of the axial blankets in fuel assemblies in order to demonstrate that all fuel assemblies can be loaded into the racks without credit for Boraflex, since these blankets reduce the reactivity of certain high burnup fuel assemblies. Currently, the minimum enrichment of blanketed assemblies in the pool is about 3.55 wt%. However, it is possible that blanketed assemblies with a lower enrichment could be used in the future as a replacement for a damaged assembly unloaded from the core. The enrichment range for blanketed assemblies has therefore been extended down to 2.5 wt%, as shown in Table 4.6.4, to cover such assemblies. This assembly average enrichment is close to the current target

enrichment for the blanketed region of 2.6 wt%. Replacement assemblies with average enrichments below 2.5 wt% are bounded by the evaluations for non-blanketed assemblies, which were analyzed down to an assembly average enrichment of 1.9 wt%.

All calculations to establish and validate the burnup versus enrichment curves are performed as full three-dimensional criticality calculations considering the axial burnup distribution of each assembly in the model.

The coefficients of the burnup vs. enrichment curves for all conditions listed in Table 4.6.4 are shown in Table 4.1.1 for non-blanketed assemblies, and in Table 4.1.2 for assemblies containing axial blankets. These tables also provide the required minimum burnup for selected values of initial enrichment. Figures 4.6.1 through 4.6.6 present this information in a graphical form.

Fuel specifications for the checkerboard arrays have been chosen to maximize the calculated reactivity. The results of one representative calculation of the effective neutron multiplication factor (k_{eff}) for each checkerboard storage arrangement is shown in Table 4.6.5 along with a tabulation of all biases and uncertainties applied to the calculated value prior to comparison with the 1.0 k_{eff} limit. This table shows that the total addition for each case, i.e. the sum of all the applicable biases and uncertainties varied between 0.0177 Δk (Cases 1 and 5) and 0.0315 Δk (Case 2). Additional results from selected calculations for each case are listed in Table 4.6.6; these results identify the fuel specifications for each side of the checkerboard array and present the maximum k_{eff} (after application of biases and uncertainties) for the array as a whole when analyzed at these conditions. Note that Case 5 is also treated as a checkerboard pattern, but with the same burnup vs. enrichment curves for both assemblies in the pattern. The highest maximum k_{eff} of any case with any analyzed combination of fuel parameters is below the regulatory limit of 1.000 applicable when considering no soluble boron to be present in the fuel pool water. It should be noted that the calculations contain a significant amount of additional safety margin as a result of the underlying conservative assumptions, such as:

- Maximum normal temperature in the pool
- Upper bound in-core moderator temperature
- Temperature bias and uncertainties calculated as maximum over the entire burnup / enrichment / cooling time range
- No interpolation of cooling times allowed between loading curves.

The selection of the fuel specifications for the confirmatory calculations, and the embedded conservatisms will ensure that the actual reactivity of the pool, under the assumed accident condition of the loss of the soluble boron in the pool, will always be below 1.0. All burnup vs. enrichment curves are therefore acceptable and result in reactivity values below the regulatory limit.

4.6.12 Interfaces

In general, only one of the five fuel checkerboard arrangements is planned in each storage rack module. Therefore, only interfaces between the five cases across the inter-module gap need to be considered. However, additional special situations are permitted as follows:

- Cells adjacent to the pool walls in Region 2 racks are qualified for a homogeneous loading of higher reactivity fuel, with the minimum burnup requirement as for Case 4.
- Fresh fuel assemblies may be placed in certain Region 2 rack module locations for inspection, testing or reconstitution, provided they are placed face adjacent to vacant cells and any diagonally adjacent fuel assemblies meet certain criteria noted below.
- A rack module may contain more than one permissible fuel storage configuration if an empty row is used to separate fuel in one configuration from fuel stored in another configuration. This condition is bounded by the evaluations of the interfaces across the inter-module gaps, since an empty row is much wider than any of the inter-module gaps.

The results for all calculations of the interface reactivity effect discussed in the following subsections are statistically equivalent to (i.e. agree within two standard deviations), or lower than the result of the corresponding reference calculation. This agreement demonstrates that these interface configurations are acceptable.

4.6.12.1 Region 1 to Region 1 Interfaces

The four Region 1 rack modules are separated from each other by a gap of 1.5 inches, which is only slightly larger than the cell to cell gap within each rack module. It is therefore required that the checkerboard storage patterns between Region 1 rack modules are aligned, i.e. that high reactivity assemblies on the rack module boundary face low reactivity assemblies across the inter-module gap. Rack modules facing each other with the same checkerboard pattern are bounded by the calculation for the individual module, since the inter-module gap is slightly larger than the cell-to-cell gap within the racks. However, a calculation has been performed for two adjacent racks with differing checkerboard storage characteristics, i.e. a Case 2 arrangement in one rack module and Case 3 in the other module; results of this calculation show that this configuration is acceptable.

4.6.12.2 Region 2 to Region 2 Interfaces

The bounding condition for Region 2 rack interfaces are at the corners of four rack modules, where each corner cell is occupied by a fuel assembly with the highest permissible reactivity for a Region 2 rack. This condition conservatively implies that checkerboard patterns in adjacent rack modules need not be aligned, i.e. it is permitted that higher reactivity assemblies face each other across rack module boundaries. The calculational model used to analyze this condition consists of a corner of a rack module with reflective boundary conditions on all four sides, thus effectively modeling an infinite array of racks with highest reactivity assemblies at all corners. The results show that this configuration is acceptable.

4.6.12.3 Region 1 to Region 2 Interface

Region 1 and Region 2 rack modules are separated by a gap of 1.5 inches. To model the interface with appropriate boundary conditions, a model was generated with 16 Region 2 cells on one side of the gap, and 14 Region 1 cells on the opposite side. The calculations show that this configuration is acceptable.

4.6.12.4 Cells facing the Pool Wall in Region 2 Racks

The peripheral row of Region 2 racks that face a pool wall or the cask pit wall is designated for storage of higher reactivity fuel assemblies, regardless of the checkerboard storage configuration used for the remainder of the rack. These higher reactivity assemblies correspond to the assemblies analyzed in Case 4, without CEAs. A number of variations for this interface have been analyzed, including:

- Rack to wall distance of 5 inches and 6 inches
- Stainless Steel liner thickness of 0.25 inches and 0.1875 inches
- Concrete wall (6 feet) or water layer (6 inches) behind the liner
- Side of the rack and corner of the rack

All variations of these parameters result in a reactivity value that is statistically equivalent to or lower than the reference case reactivity, with a Case 1, Case 4 or Case 5 configuration in the remainder of the rack. Placing higher reactivity fuel assemblies (Case 4, without CEAs) on the periphery of Region 2 racks so that they face the pool wall or the cask pit wall is therefore acceptable.

4.6.12.5 Fresh Fuel in Region 2 Racks

For fuel assembly inspection, testing and reconstitution, it is necessary to place up to 3 assemblies and a rod basket in close proximity to each other within a Region 2 rack module. As a bounding approach, these assemblies and the rod basket are modeled as fresh assemblies with an enrichment of 4.5 wt% in the calculations. To produce satisfactory results, it is required that the four cells face-adjacent to the cell with a fresh assembly be empty. Additionally, a fresh fuel assembly must not be placed in a cell diagonally adjacent to another cell containing a fresh assembly. However, it is acceptable to place spent fuel with the highest reactivity permitted for storage in Case 1 and Case 5 checkerboards in such a diagonal position. As a bounding approach, a configuration of 4 fresh assemblies is analyzed in an infinite array of a Case 1 checkerboard, with the fresh assemblies at the closest possible approach consistent with above

requirements. The pattern is shown in Figure 4.6.7. Analysis results confirm that this configuration is acceptable.

The empty cells were modeled with a water density of 25% of the normal water density. This assumption permits the placement of non-actinide material (i.e., hardware) in these cells as long as this non-fuel hardware does not occupy more than 75% of the cell volume.

No evaluation is performed considering the placement of fresh fuel assemblies within a Case 4 storage configuration, and this condition is therefore not permitted.

4.6.13 Soluble Boron Concentration for Maximum k_{eff} of 0.95

Calculations have been performed to determine the minimum soluble boron concentration in the spent fuel pool necessary to ensure that the reactivity of the fuel pool does not exceed 0.95. For each of the five fuel checkerboard storage configurations, calculations are performed at two soluble boron levels (100 ppm and 300 ppm for Region 1; 200 ppm and 500 ppm for Region 2), and the soluble boron concentration necessary to satisfy the regulatory requirement is then determined by linear interpolation. A target of 0.94 is used for the maximum k_{eff} values, which is lower, i.e. more conservative, than the regulatory limit. Note that the presence of borated water in the fuel pool results in a slightly higher delta-temperature reactivity bias for the Region 2 racks than would be calculated assuming the presence of pure water. The highest minimum soluble boron concentration calculated is 443 ppm, calculated for Case 4. The details for this calculation are shown in Table 4.6.7. For added conservatism, a minimum value of 500 ppm is specified for compliance purposes, which is larger than the calculated value of 443 ppm.

4.6.14 Abnormal and Accident Conditions

The effects on reactivity of credible abnormal and accident conditions are examined in this section. None of the abnormal or accident conditions that have been identified as credible cause the reactivity of St. Lucie Unit 1 fuel pool storage racks to exceed the limiting reactivity value of

$k_{\text{eff}} = 0.95$, considering the presence of soluble boron. The double contingency principle of ANSI N16.1-1975 (and the USNRC letter of April 1978) specifies that it shall require at least two unlikely, independent and concurrent events to produce a criticality accident. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions.

4.6.14.1 Temperature and Water Density Effects

The reactivity effect of fuel pool water temperatures exceeding 150 °F has been calculated. Temperatures up to 248 °F (120 C) are evaluated, as are local boiling conditions with void percentages up to 20%. The maximum reactivity increase compared to 150 °F is 0.0303 Δk for Region 1 and 0.0146 Δk for Region 2. It has been determined that a soluble boron concentration of 541 ppm is required to ensure a maximum k_{eff} of 0.95 is not exceeded under these conditions.

4.6.14.2 Dropped Assembly - Horizontal

In the event a fuel assembly is dropped on top of a storage rack module, the dropped assembly will come to rest horizontally on top of the rack with a minimum separation distance of at least 12 inches from the active region of stored fuel. This distance is sufficient to preclude neutron coupling (i.e., an effectively infinite separation). The maximum expected deformation under seismic or accident conditions will not reduce this minimum spacing to less than 12 inches. Consequently, the horizontal fuel assembly drop accident will not significantly increase reactivity in the fuel storage racks.

4.6.14.3 Dropped Assembly - Vertical

It is also possible to vertically drop an assembly into a location occupied by another assembly. Such a vertical impact would at most cause a small compression of the stored assembly, reducing the water-to-fuel ratio and thereby potentially increasing reactivity. However, the reactivity increase would be small compared to the reactivity increase created by the misloading of a fresh

assembly discussed in the following section. The vertical drop is therefore bounded by this misloading accident and no separate calculation is performed for the drop accident.

4.6.14.4 Abnormal Location of a Fuel Assembly

4.6.14.4.1 Misloaded Fresh Fuel Assembly

The misplacement of a fresh unburned fuel assembly could, in the absence of soluble poison, result in exceeding the regulatory limit (k_{eff} of 0.95). This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (4.5 wt%) were to be inadvertently misloaded into a Region 2 storage cell intended to be empty (see Section 4.6.12.5), or into a cell intended to hold a low reactivity assembly (Case 4, assembly with CEA). The reactivity consequences of these situations were investigated and it was determined that the misloading of a fresh assembly into a cell intended to remain empty is the bounding condition. The evaluation of this case is shown in Table 4.6.8. To assure that the regulatory limit of 0.95 for the maximum k_{eff} is not exceeded under this condition, a soluble boron level of 1090 ppm in the spent fuel pool is required.

4.6.14.4.2 Mislocated Fresh Fuel Assembly

The mislocation of a fresh unburned fuel assembly, i.e. the accidental placement of an assembly outside of the storage rack envelope but adjacent to other fuel assemblies, has also been considered. There is one area in the pool layout in which such an accident condition could be postulated to occur; this area is near the east wall of the pool in the cut-out of the Region 2 rack. However, the size of this cut-out is such that the mislocated assembly can face no more than 2 rack walls; an assembly positioned here would face a substantial water thickness on its other two sides. This condition is therefore bounded by the fuel misloading accident discussed earlier, since the misloading accident has a fresh assembly surrounded by two other fresh assemblies inside the Region 2 rack.

4.7 REFERENCES

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Table 4.1.1

Minimum Burnup as a Function of Enrichment for Non-Blanketed Assemblies

Case	Cooling Time	Coefficients [†]			Minimum Burnups (GWd/MTU) for various Enrichments			
		A	B	C	1.9%	2.5%	3.0%	3.8%
Case 1, Low Reactivity	0 years	-0.65	20.08	-16.52	19.29	29.62	37.87	50.40
	12 years	-0.65	17.76	-15.58	15.82	24.76	31.85	42.52
	15 years	-0.43	16.25	-13.84	15.48	24.10	31.04	41.70
	20 years	0.12	12.90	-9.61	15.33	23.39	30.17	41.14
Case 1, High Reactivity	0 years	-0.41	17.00	-21.39	9.43	18.55	25.92	37.29
	12 years	-0.54	16.22	-20.63	8.24	16.55	23.17	33.21
	15 years	-0.53	15.86	-20.07	8.15	16.27	22.74	32.54
	20 years	-0.46	15.11	-18.80	8.25	16.10	22.39	31.98
Case 2, Low Reactivity	0 years	-0.74	17.49	-19.72	10.84	19.38	26.09	36.06
	5 years	-0.56	15.64	-17.65	10.04	17.95	24.23	33.70
Case 2, High Reactivity	0 years	0.00	9.31	-24.39	0.00	0.00	3.54	10.99
Case 3, Low Reactivity	0 years	0.00	10.97	-14.71	6.13	12.72	18.20	26.98
Case 3, High Reactivity	0 years	0.00	10.51	-22.35	0.00	3.93	9.18	17.59
Case 4	0 years	0.00	10.97	-14.71	6.13	12.72	18.20	26.98
Case 5	0 years	-0.41	17.70	-17.97	14.18	23.72	31.44	43.37
	12 years	0.04	13.10	-12.56	12.47	20.44	27.10	37.80
	15 years	0.13	12.38	-11.83	12.16	19.93	26.48	37.09
	20 years	0.26	11.56	-11.16	11.74	19.37	25.86	36.52

[†] Coefficients for polynomial Function: $BU = A \cdot E^2 + B \cdot E + C$ with BU = Minimum Burnup in GWd/MTU; E = Initial Enrichment in wt% ²³⁵U; A, B, C = Coefficients

Table 4.1.2

Minimum Burnup as a Function of Enrichment for Blanketed Assemblies

Case	Cooling Time	Coefficients [†]			Minimum Burnups (GWd/MTU) for various Enrichments				
		A	B	C	2.5 %	3.0 %	3.5 %	4.0 %	4.5 %
Case 1, Low Reactivity	0 years	-0.84	19.25	-13.42	29.46	36.77	43.67	50.14	56.20
	5 years	-0.72	17.40	-12.03	26.97	33.69	40.05	46.05	51.69
	10 years	-0.66	16.32	-11.46	25.22	31.56	37.58	43.26	48.62
	15 years	-0.67	16.00	-11.73	24.08	30.24	36.06	41.55	46.70
	20 years	-0.76	16.45	-12.81	23.57	29.70	35.46	40.83	45.83
Case 1, High Reactivity	0 years	-0.98	18.97	-22.54	18.76	25.55	31.85	37.66	42.98
	5 years	-0.74	16.54	-19.10	17.63	23.86	29.73	35.22	40.35
	10 years	-0.57	14.73	-16.49	16.77	22.57	28.08	33.31	38.25
	15 years	-0.46	13.54	-14.70	16.28	21.78	27.06	32.10	36.92
	20 years	-0.41	12.98	-13.74	16.15	21.51	26.67	31.62	36.37
Case 2, Low Reactivity	0 years	-0.74	17.49	-19.72	19.38	26.09	32.43	38.40	44.00
	5 years	-0.56	15.64	-17.65	17.95	24.23	30.23	35.95	41.39
Case 2, High Reactivity	0 years	0.00	9.31	-24.39	0.00	3.54	8.20	12.85	17.51
Case 3, Low Reactivity	0 years	0.00	10.97	-14.71	12.72	18.20	23.69	29.17	34.66
Case 3, High Reactivity	0 years	0.00	10.51	-22.35	3.93	9.18	14.44	19.69	24.95
Case 4	0 years	0.00	10.97	-14.71	12.72	18.20	23.69	29.17	34.66
Case 5	0 years	-0.24	14.23	-10.38	23.70	30.15	36.49	42.70	48.80
	5 years	-0.20	13.10	-9.24	22.26	28.26	34.16	39.96	45.66
	10 years	-0.23	12.70	-9.27	21.04	26.76	32.36	37.85	43.22
	15 years	-0.32	13.02	-10.48	20.07	25.70	31.17	36.48	41.63
	20 years	-0.47	14.08	-12.85	19.41	25.16	30.67	35.95	40.99

[†] Coefficients for polynomial Function: $BU = A \cdot E^2 + B \cdot E + C$ with $BU =$ Minimum Burnup in GWD/MTU; $E =$ Initial Enrichment in wt% ^{235}U ; $A, B, C =$ Coefficients

Table 4.4.1
St. Lucie Unit 1 Fuel Assembly Specifications

Parameter	Value	
	CE 14x14	FR 14x14
Assembly type	CE 14x14	FR 14x14
Rod Array Size	14x14	14x14
Rod Pitch, Inches	0.580 [REDACTED]	0.580 [REDACTED]
Maximum Active Fuel Length, Inches	136.7	136.7
Stack Density (g/cm ³)	10.05 [REDACTED]	10.30 [REDACTED]
Total Number of Fuel Rods	176	176
Fuel Rod Outer Diameter, Inches	0.440 [REDACTED]	0.440 [REDACTED]
Cladding Thickness, Inches	0.026 – 0.028 [REDACTED]	0.028 – 0.031 [REDACTED]
Cladding Material	Zr-4	Zr-4
Maximum Pellet Diameter, Inches	0.3805 [REDACTED]	0.3770 [REDACTED]
Number of Guide Tubes	5	5
Guide Tube Outer Diameter, Inches	1.115 [REDACTED]	1.115 [REDACTED]
Guide Tube Wall Thickness, Inches	0.04 [REDACTED]	0.04 [REDACTED]
Guide Tube Material	Zr-4	Zr-4

Table 4.4.2
Control Element Assembly (CEA) Specifications

Parameter	Value
B4C Pellet Diameter, Inches	0.860 
Clad Thickness, Inches	0.040
Clad Outer Diameter, Inches	0.948
Clad Material	Ni-Cr-Fe 625
Tip of CEA to bottom of active fuel, Inches	Approx. 3.34
B-10 Isotopic Content, Atom %	19.75 
Pellet Density, % of theoretical	73 
Theoretical Pellet Density, g/cm ³	2.52
Boron Weight Percent	78 

Table 4.4.3
Core Operating Parameters for Depletion Analyses

Parameter	Value
Soluble Boron Concentration, ppm	750
Reactor Specific Power, MW/MtU	31.2
Core Average Fuel Temperature, °F	1275.1
Core Average Moderator Temperature at the Top of the Active Fuel Region, °F	600.63
In-Core Fuel Assembly Pitch, Inches	8.18

Table 4.4.4
St. Lucie Unit 1 Fuel Rack Dimensions

Parameter	Value	
	Region 1	Region 2
Cell ID	8.65 [REDACTED]	8.65 [REDACTED]
Wall Thickness	0.08 [REDACTED]	0.08 [REDACTED]
Cell Pitch	10.12 [REDACTED]	8.86 [REDACTED]
Boraflex Gap Thickness	0.075 [REDACTED]	0.05
Sheathing Thickness	0.02 [REDACTED]	n/a
Sheathing Width	7.5 [REDACTED]	n/a

Table 4.6.1

Comparison of k_{inf} for Various Fuel Assembly Types at Representative Fuel Conditions

Assembly Type	FR 14x14		CE 14x14	
Cladding Thickness	0.028	0.031	0.026	0.028
3.5 wt% Enrichment, 0 GWD/MTU Burnup				
Region 1	1.13298	1.13216	1.13290	1.13235
Region 2	1.31623	1.31497	1.31655	1.31569
3.5 wt% Enrichment, 0 years cooling time and 35 GWD/MTU Burnup				
Region 1	0.85445	0.85445	0.85423	0.85378
Region 2	0.98634	0.98607	0.98644	0.98575
3.5 wt% Enrichment, 10 years cooling time and 35 GWD/MTU Burnup				
Region 1	0.82240	0.82238	0.82217	0.82172
Region 2	0.94945	0.94915	0.94952	0.94884

Table 4.6.2

Effect of Pool Water Temperature on k_{inf}
for Fuel of 4.5 wt% Enrichment and 0 Years Cooling Time at 0 ppm Soluble Boron.

Burnup (GWD/MTU)	39 °F (4 °C)		68 °F (20 °C)	104 °F (40 °C)		150 °F	
	k_{inf}	Δk^\dagger	k_{inf}	k_{inf}	Δk^\dagger	k_{inf}	Δk^\dagger
<i>Region 1</i>							
5	1.1280	-0.0041	1.1321	1.1370	0.0049	1.1429	0.0109
15	1.0554	-0.0039	1.0593	1.0640	0.0047	1.0696	0.0103
25	0.9869	-0.0038	0.9907	0.9953	0.0046	1.0009	0.0102
35	0.9189	-0.0039	0.9227	0.9274	0.0046	0.9330	0.0103
45	0.8516	-0.0039	0.8555	0.8603	0.0048	0.8661	0.0106
<i>Region 2</i>							
5	1.3203	-0.0004	1.3206	1.3210	0.0003	1.3211	0.0004
15	1.2320	-0.0004	1.2324	1.2326	0.0002	1.2326	0.0002
25	1.1505	-0.0005	1.1510	1.1515	0.0004	1.1517	0.0007
35	1.0707	-0.0008	1.0715	1.0723	0.0008	1.0730	0.0015
45	0.9924	-0.0011	0.9936	0.9948	0.0012	0.9961	0.0025
55	0.9192	-0.0015	0.9207	0.9224	0.0017	0.9243	0.0037
<i>Region 2, Assemblies with CEAs</i>							
5	1.0516	0.0010	1.0507	1.0482	-0.0024	1.0450	-0.0057
15	0.9808	0.0009	0.9800	0.9777	-0.0023	0.9747	-0.0053
25	0.9143	0.0007	0.9136	0.9116	-0.0020	0.9089	-0.0047
35	0.8492	0.0005	0.8486	0.8470	-0.0017	0.8447	-0.0039
45	0.7855	0.0002	0.7853	0.7840	-0.0012	0.7823	-0.0030

[†] Relative to 20 °C

Table 4.6.3

Reactivity Effect of Rack and Fuel Tolerances

Tolerance	Region 1	Region 2	Region 2 with CEAs
<i>Rack Tolerances</i>			
Cell ID	0.0003	0.0011	0.0008
Wall Thickness	0.0031	0.0033	0.0024
Cell Pitch	0.0057	0.0051	0.0014
Sheathing Thickness	0.0020	n/a	n/a
Boraflex Gap Thickness	0.0001	n/a	n/a
<i>Fuel Tolerances</i>			
Density	0.0075	0.0058	0.0123
Enrichment	0.0083	0.0088	0.0038
Other (Statistically Combined)	0.0072	0.0045	0.0047
<i>CEA Tolerances</i>			
All CEA Tolerances (Statistically Combined)	n/a	n/a	0.0016
All Tolerances (Statistically Combined)	0.0149	0.0130	0.0141

Table 4.6.4:

Enrichment and Cooling Time Combinations for Burnup versus Enrichment Curves

Case	Non-Blanketed Assemblies		Blanketed Assemblies	
	Enrichment	Cooling Time	Enrichment	Cooling Time
1	1.9 – 3.8	0, 12, 15, 20	2.5 – 4.5	0, 5, 10, 15, 20
2	1.9 – 3.8	0, 5 (low reactivity only)	2.5 – 4.5	0, 5 (low reactivity only)
3	1.9 – 3.8	0	2.5 – 4.5	0
4	1.9 – 3.8	0	2.5 – 4.5	0
5	1.9 – 3.8	0, 12, 15, 20	2.5 – 4.5	0, 5, 10, 15, 20

Table 4.6.5

Representative Calculation for each Case

Case	1	2	3	4	5
Region	2	1	1	2	2
Assembly 1					
Enrichment	3.8	4.5	4.5	4.5	1.9
Burnup	50.4	44.0	34.7	34.7	14.1
Cooling Time	0	0	0	0	0
Assembly 2					
Enrichment	3.8	4.5	3.5	3.5	1.9
Burnup	32.0	17.5	14.4	23.7	11.7
Cooling Time	20	0	0	0	20
Calculated k-eff	0.9785	0.9615	0.9636	0.9780	0.9788
Bias	0.0009	0.0009	0.0009	0.0009	0.0009
Temperature Correction	0.0037	0.0109	0.0109	0.0037	0.0037
Uncertainties					
Bias	0.0011	0.0011	0.0011	0.0011	0.0011
Calculational [†]	0.0014	0.0014	0.0012	0.0012	0.0012
Eccentricity	0.0000	0.0127	0.0127	0.0000	0.0000
Tolerances	0.0130	0.0149	0.0149	0.0141	0.0130
Total Uncertainties	0.0131	0.0197	0.0196	0.0142	0.0131
Total Addition	0.0177	0.0315	0.0314	0.0188	0.0177
Maximum k-eff	0.9962	0.9930	0.9950	0.9968	0.9965

[†] Two times the standard deviation of the calculated k_{eff}

Table 4.6.6

Results of Additional Selected Calculations for each Case

<i>Case 1</i>				
Enr/Bu/Cool 1[†]	1.9/15.33/20	2.5/24.76/12	3.8/41.14/20	4.5/45.83/20
Enr/Bu/Cool 2	1.9/8.25/20	3.5/31.85/0	3.8/31.98/20	4.5/36.37/20
max. k_{eff}	0.9972	0.9900	0.9973	0.9954
<i>Case 2</i>				
Enr/Bu/Cool 1	2.5/19.38/0	3.5/32.43/0	3.8/30.06/0	4.5/44.0/0
Enr/Bu/Cool 2	2.5/0/0	4.5/17.51/0	3.8/10.99/0	3.5/8.2/0
max. k_{eff}	0.9858	0.9918	0.9951	0.9959
<i>Case 3</i>				
Enr/Bu/Cool 1	1.9/6.1/0	3.5/23.7/0	3.8/27.0/0	4.5/34.7/0
Enr/Bu/Cool 2	4.5/24.9/0	3.5/14.4/0	3.8/17.6/0	2.5/3.9/0
max. k_{eff}	0.9792	0.9943	0.9929	0.9954
<i>Case 4</i>				
Enr/Bu/Cool 1	1.9/6.1/0	3.5/23.7/0	3.8/27.0/0	4.5/34.7/0
Enr/Bu/Cool 2	3.8/27/0	3.5/23.7/0	4.5/34.7/0	1.9/6.1/0
max. k_{eff}	0.9859	0.9957	0.9983	0.9783
<i>Case 5</i>				
Enr/Bu/Cool 1	1.9/11.74/20	2.5/20.44/12	3.5/30.67/20	4.5/48.80/0
Enr/Bu/Cool 2	1.9/11.74/20	3.8/36.52/20	4.5/40.99/20	4.0/39.96/5
max. k_{eff}	0.9940	0.9974	0.9956	0.9948

[†] Enr = Enrichment in wt%; Bu = Burnup in GWD/MTU; Cool = Cooling Time in years;
1 & 2 = Two assemblies in Checkerboard Pattern

Table 4.6.7

Soluble Boron Concentration for a Maximum k_{eff} Value of 0.95 under Normal Conditions.

Case	4	4
Region	2	2
Soluble Boron Concentration	200	500
4.5% Fresh Fuel k_{inf}	1.3297	1.2684
Assembly 1		
Enrichment	4.5	4.5
Burnup	34.7	34.7
Cooling Time	0	0
Assembly 2		
Enrichment	3.5	3.5
Burnup	23.7	23.7
Cooling Time	0	0
Calculated k_{eff}	0.9453	0.9013
Bias	0.0009	0.0009
Temperature Correction	0.0057	0.0057
Uncertainties		
Bias	0.0011	0.0011
Calculational	0.0012	0.0016
Eccentricity	0	0
Depletion	0.0192	0.0184
Assembly Burnup	0.0049	0.0044
Tolerances	0.0141	0.0141
Total Uncertainties	0.0244	0.0236
Total Addition	0.0310	0.0302
Maximum k_{eff}	0.9763	0.9315
	Target k_{eff}	0.94
	Soluble Boron	443

Table 4.6.8

Soluble Boron Concentration for a Maximum k_{eff} Value of 0.95 under Accident Conditions.

Case (ppm)	1000	1500
Region	2	2
4.5% Fresh Fuel k-inf		
Calculated k-eff	1.1824	1.1098
Assembly 1		
Enrichment	1.9	1.9
Burnup	15.3	15.3
Cooling Time	20	20
Assembly 2		
Enrichment	1.9	1.9
Burnup	9.4	9.4
Cooling Time	0	0
Calculated k-eff	0.9223	0.8617
Bias	0.0009	0.0009
Temperature Correction	0.0068	0.0073
Uncertainties		
Bias	0.0011	0.0011
Calculational	0.0014	0.0012
Eccentricity	0.0000	0.0000
Assembly Burnup	0.0075	0.0075
Depletion	0.0130	0.0124
Tolerances	0.0145	0.0145
Total Uncertainties	0.0209	0.0206
Total Addition	0.0286	0.0288
Maximum k-eff	0.9509	0.8905
Target k-max		0.94
corresponding soluble boron level		1090

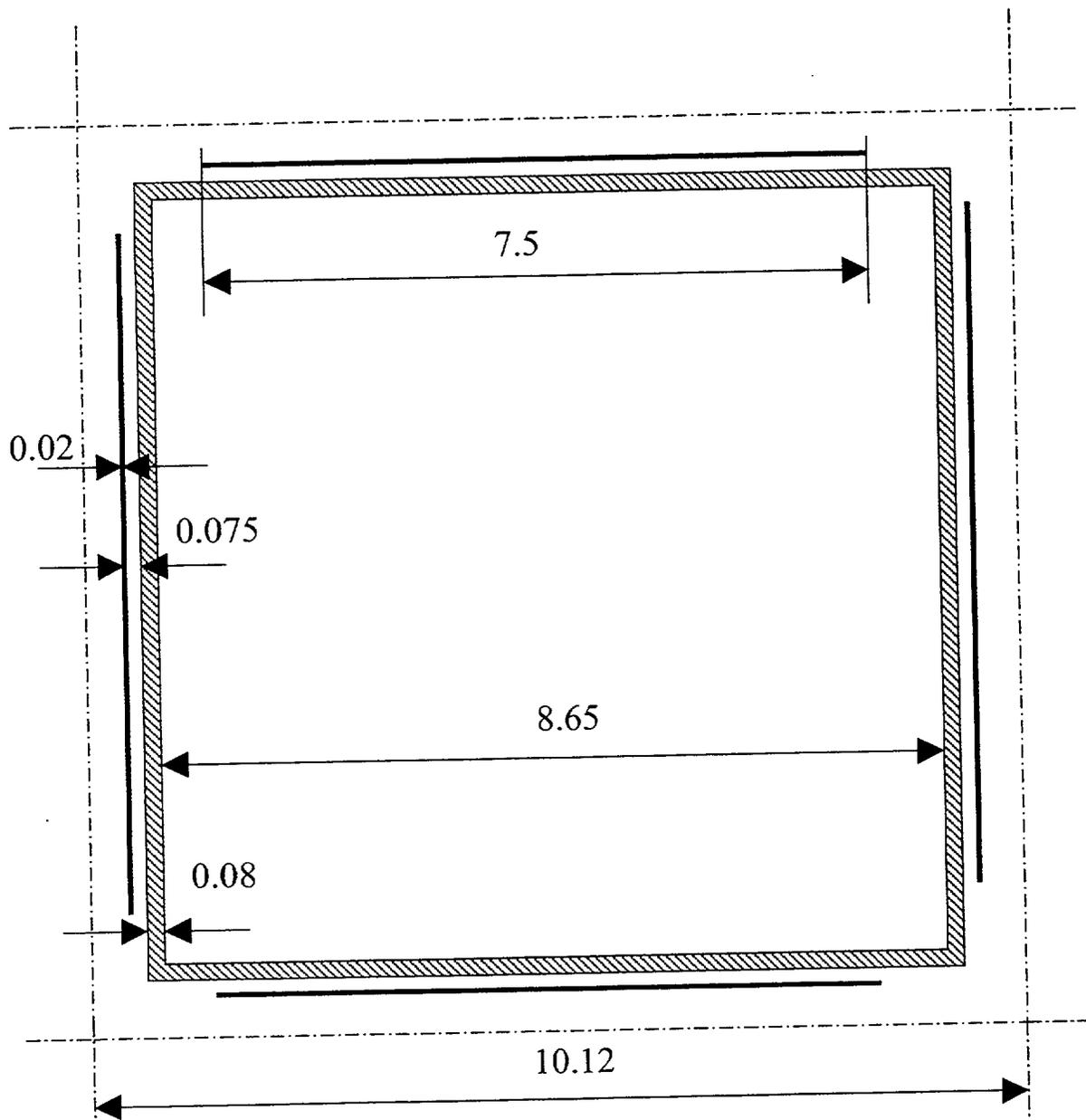


Figure 4.4.1: Schematic View of Region 1 Cell (not to scale)

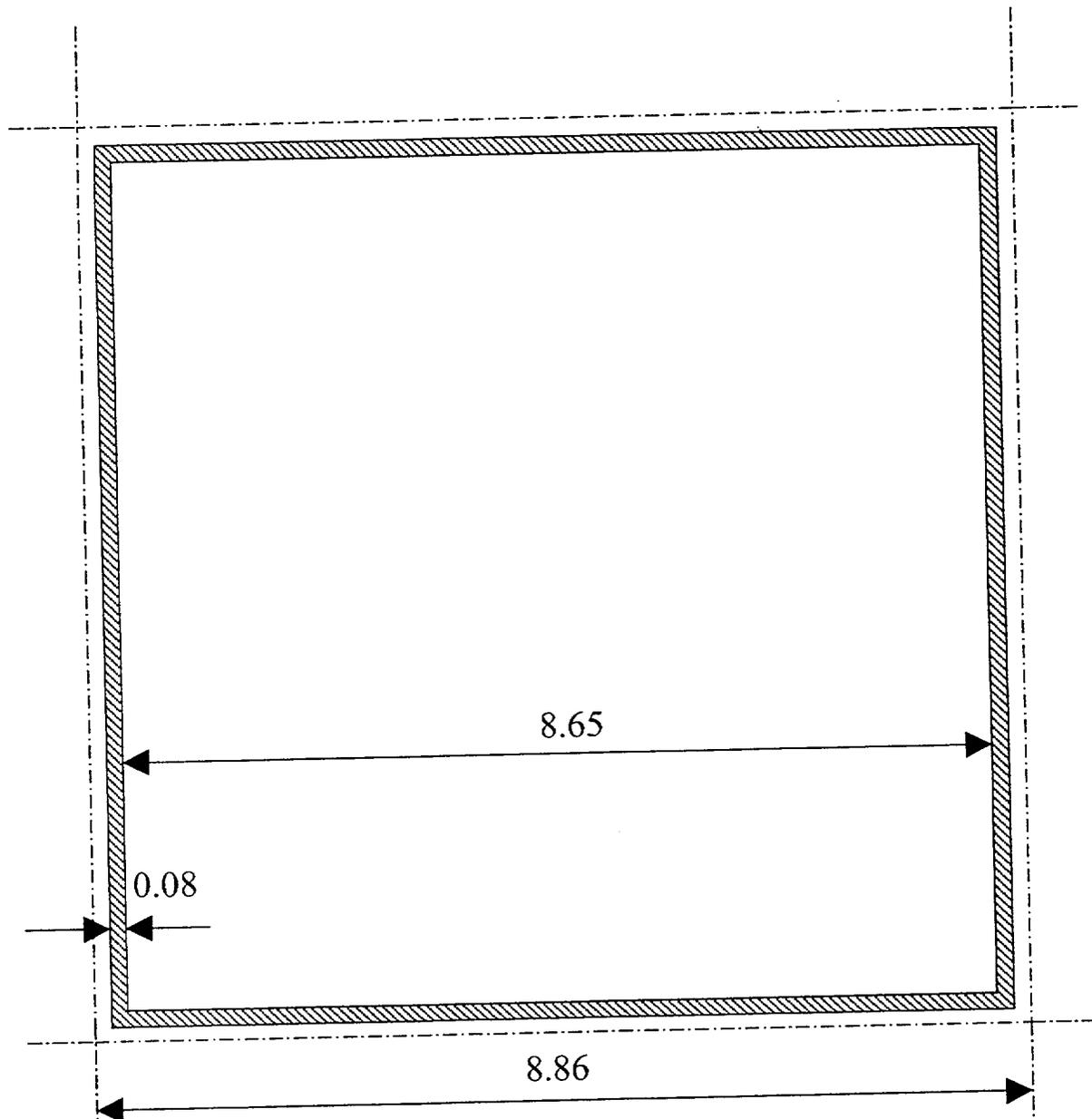


Figure 4.4.2: Schematic View of Region 2 Cell (not to scale)

Case 1, Low Reactivity

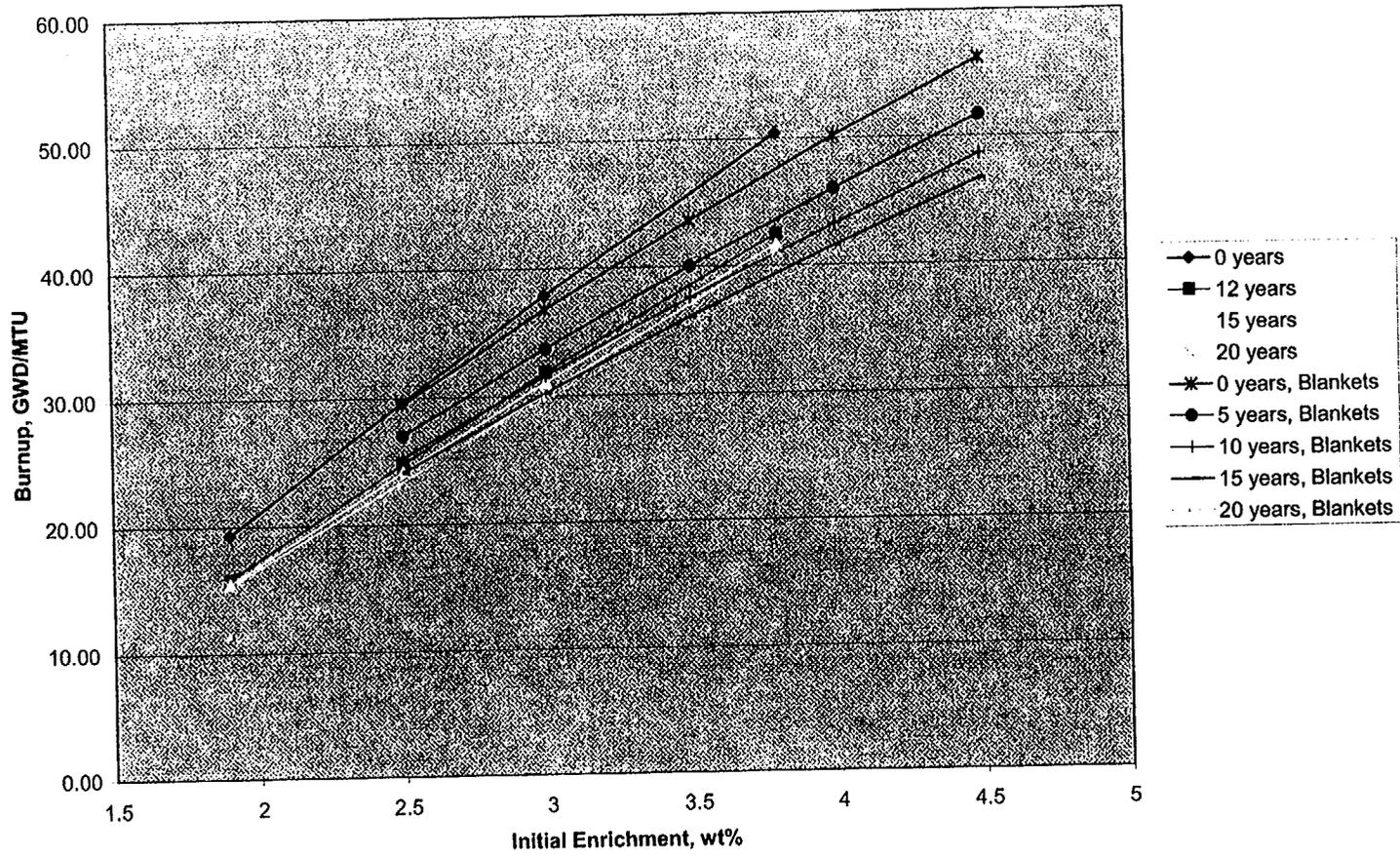


Figure 4.6.1 Minimum Burnup as a Function of Initial Enrichment for Case 1, Low Reactivity Assemblies

Case 1, High Reactivity

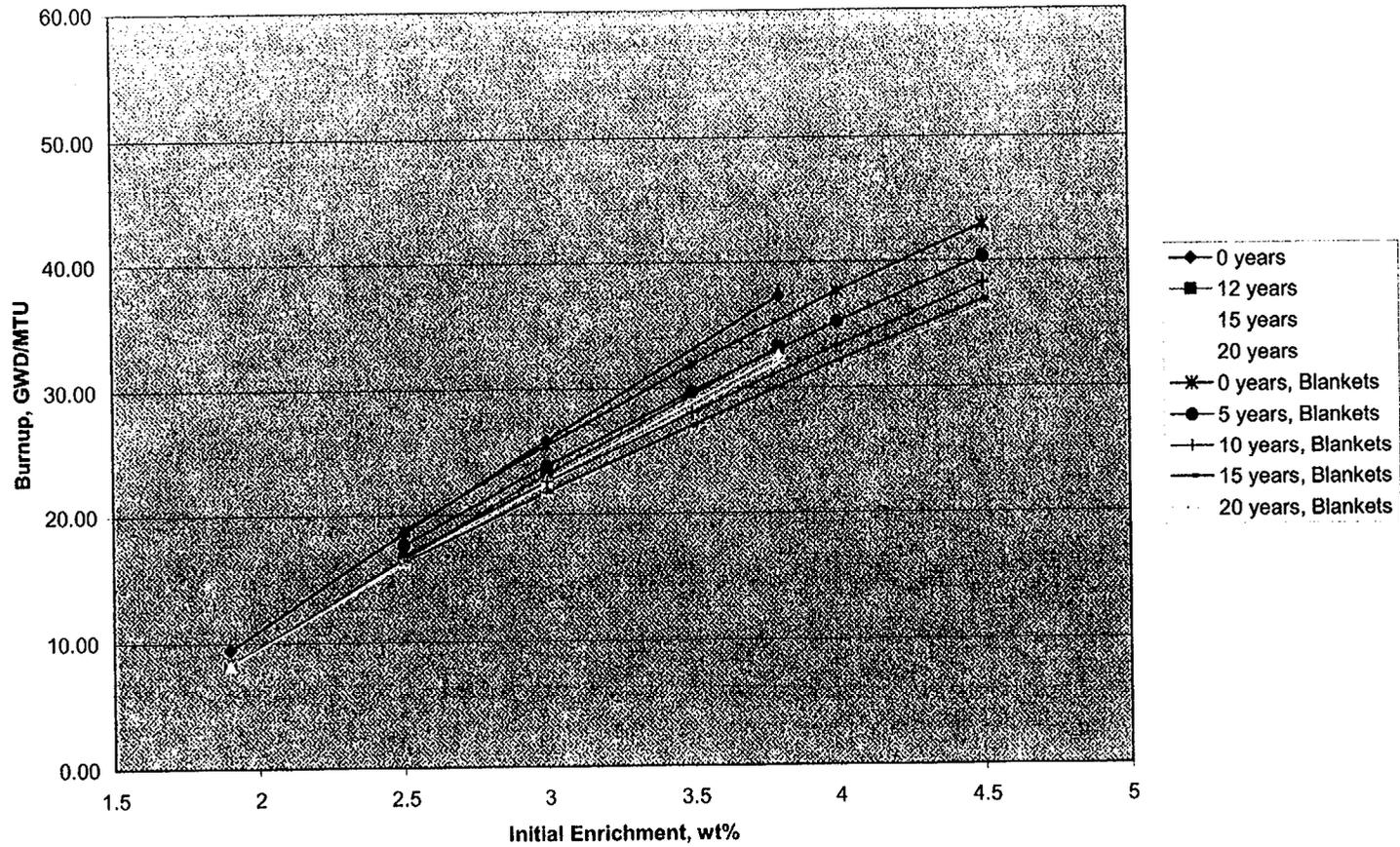


Figure 4.6.2 Minimum Burnup as a Function of Initial Enrichment for Case 1, High Reactivity Assemblies

Case 2

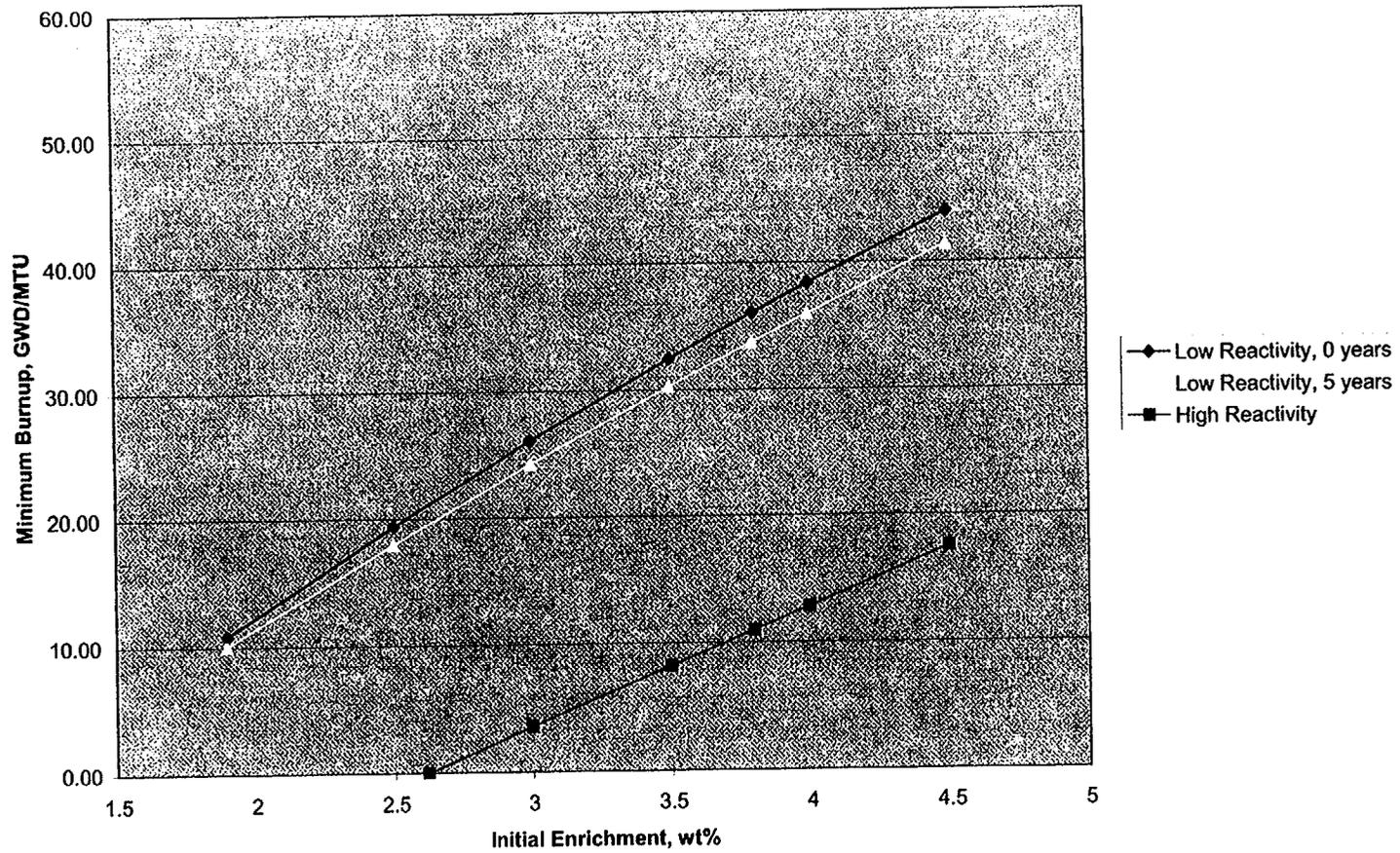


Figure 4.6.3 Minimum Burnup as a Function of Initial Enrichment for Case 2

Case 3

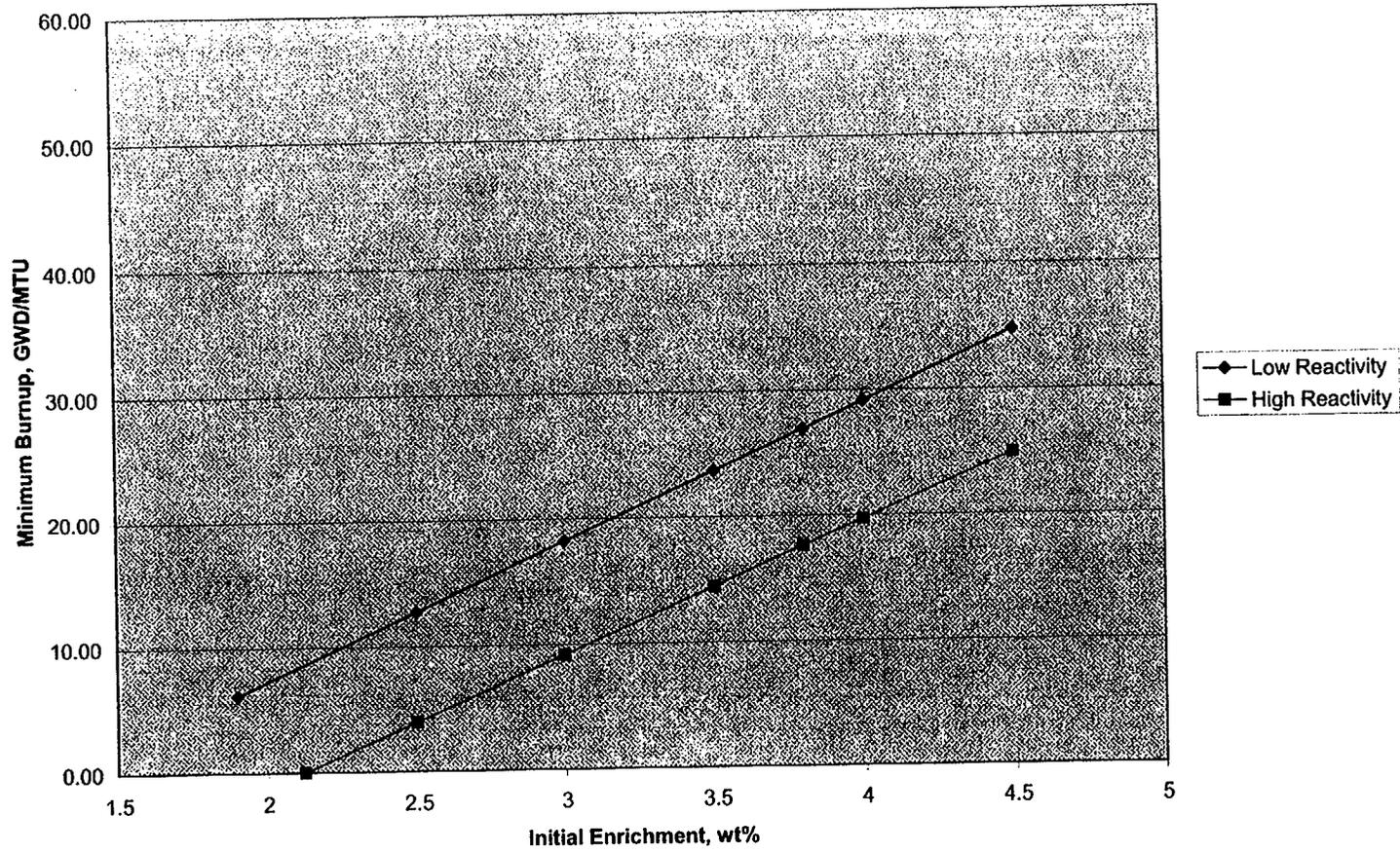


Figure 4.6.4 Minimum Burnup as a Function of Initial Enrichment for Case 3

Case 4

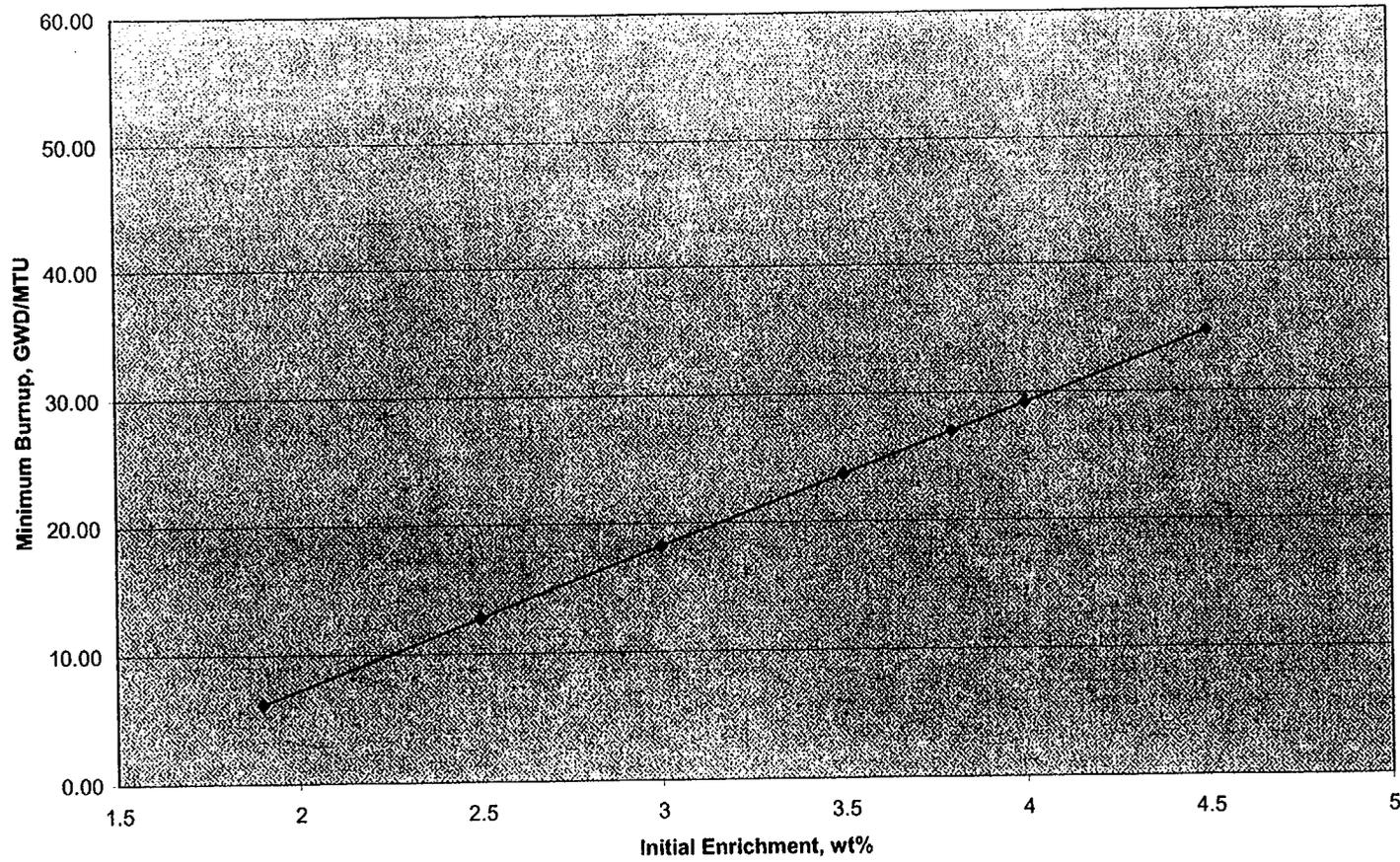


Figure 4.6.5 Minimum Burnup as a Function of Initial Enrichment for Case 4

Case 5

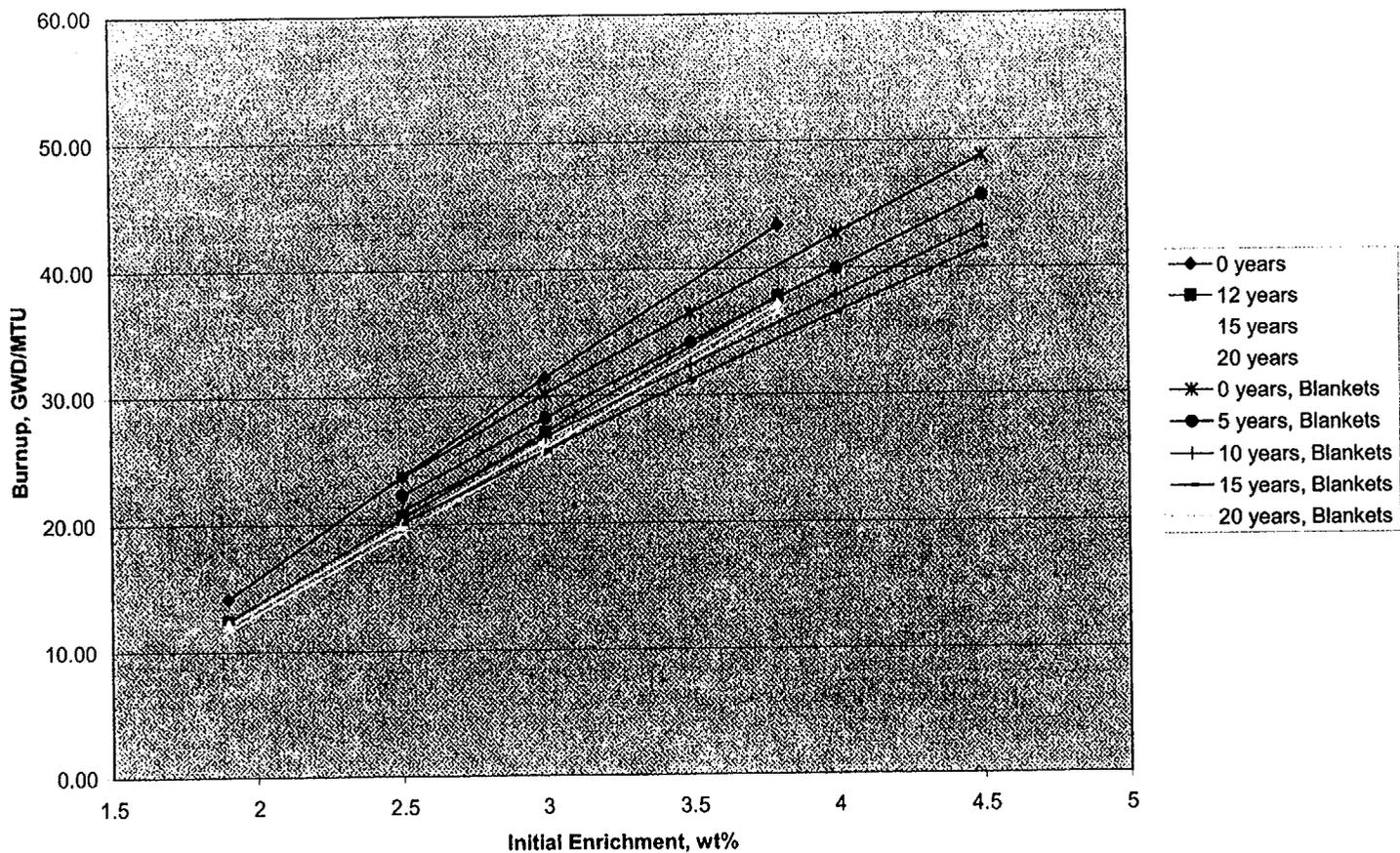


Figure 4.6.6 Minimum Burnup as a Function of Initial Enrichment for Case 5

4 Fresh Assemblies (all separated by at least one empty cell, closest approach)

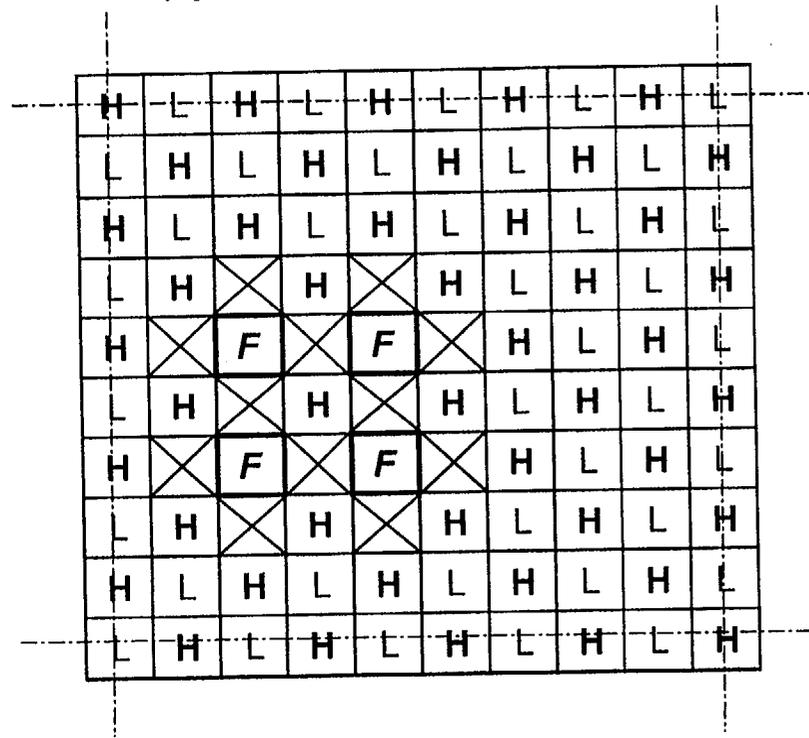
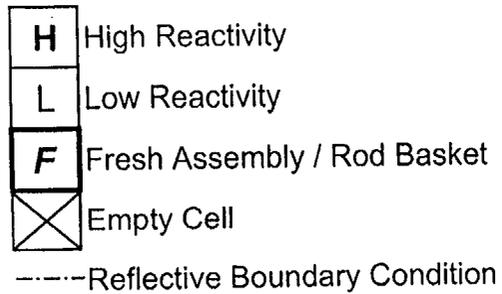


Figure 4.6.7 Schematic Configuration of the Calculational Model for Fresh Assemblies in Region 2 Racks for Inspection and Reconstitution

Appendix A
Benchmark Calculations

(total number of pages: 26 including this page)

Note: because this appendix was taken from a different report, the next page is labeled
“Appendix 4A, Page 1”.

APPENDIX 4A: BENCHMARK CALCULATIONS

4A.1 INTRODUCTION AND SUMMARY

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

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

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

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

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

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

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

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

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

$$\bar{k} = \frac{1}{n} \sum_i^n k_i \quad (4A.1)$$

† 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.

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

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

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

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

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

4A.2 Effect of Enrichment

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

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

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

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

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

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

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

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

4A.4 Miscellaneous and Minor Parameters

4A.4.1 Reflector Material and Spacings

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

4A.4.2 Fuel Pellet Diameter and Lattice Pitch

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

4A.4.3 Soluble Boron Concentration Effects

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

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

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).

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Table 4A.1

Summary of Criticality Benchmark Calculations

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

Table 4A.1
Summary of Criticality Benchmark Calculations

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

Table 4A.1

Summary of Criticality Benchmark Calculations

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

Table 4A.1
Summary of Criticality Benchmark Calculations

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

Table 4A.1

Summary of Criticality Benchmark Calculations

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

Notes: NC stands for not calculated.

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

†† These experimental results appear to be statistical outliers ($> 3\sigma$) 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 $k_{\text{eff}} \pm 1\sigma$	
	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 ¹⁰B LOADINGS

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

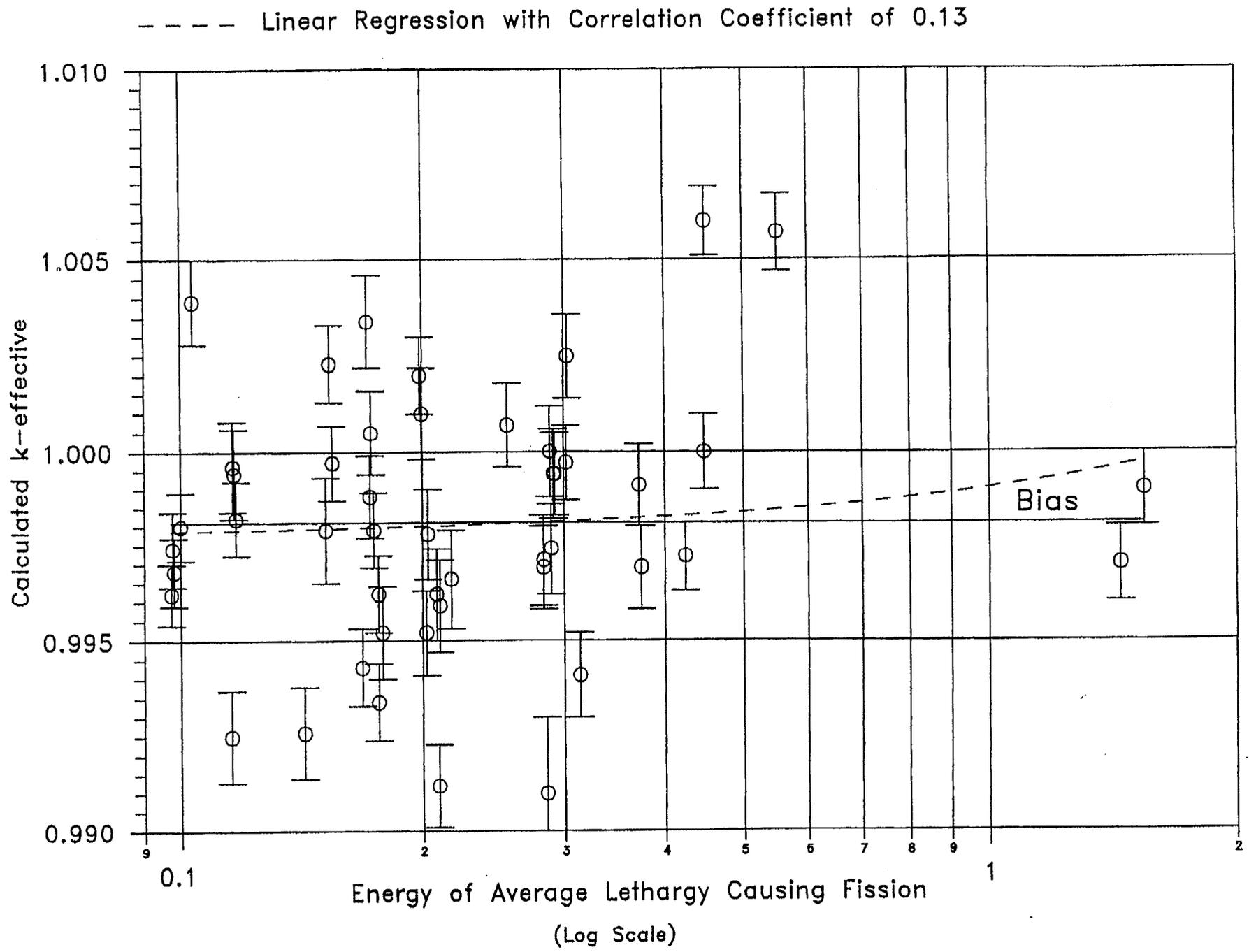


FIGURE 4A.1 MCNP CALCULATED k -eff VALUES for VARIOUS VALUES OF THE SPECTRAL INDEX

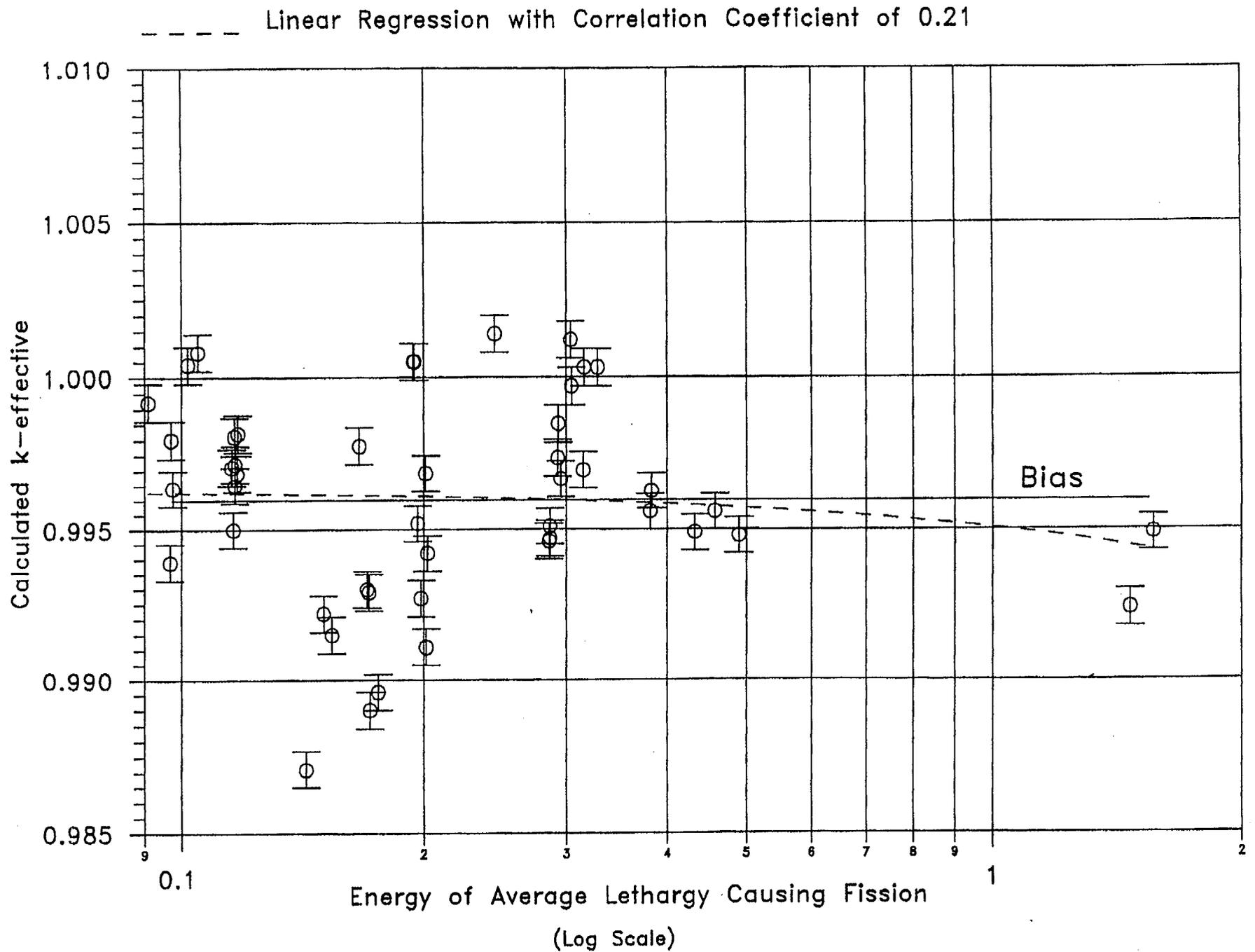


FIGURE 4A.2 KEN05a CALCULATED k-eff VALUES FOR VARIOUS VALUES OF THE SPECTRAL INDEX

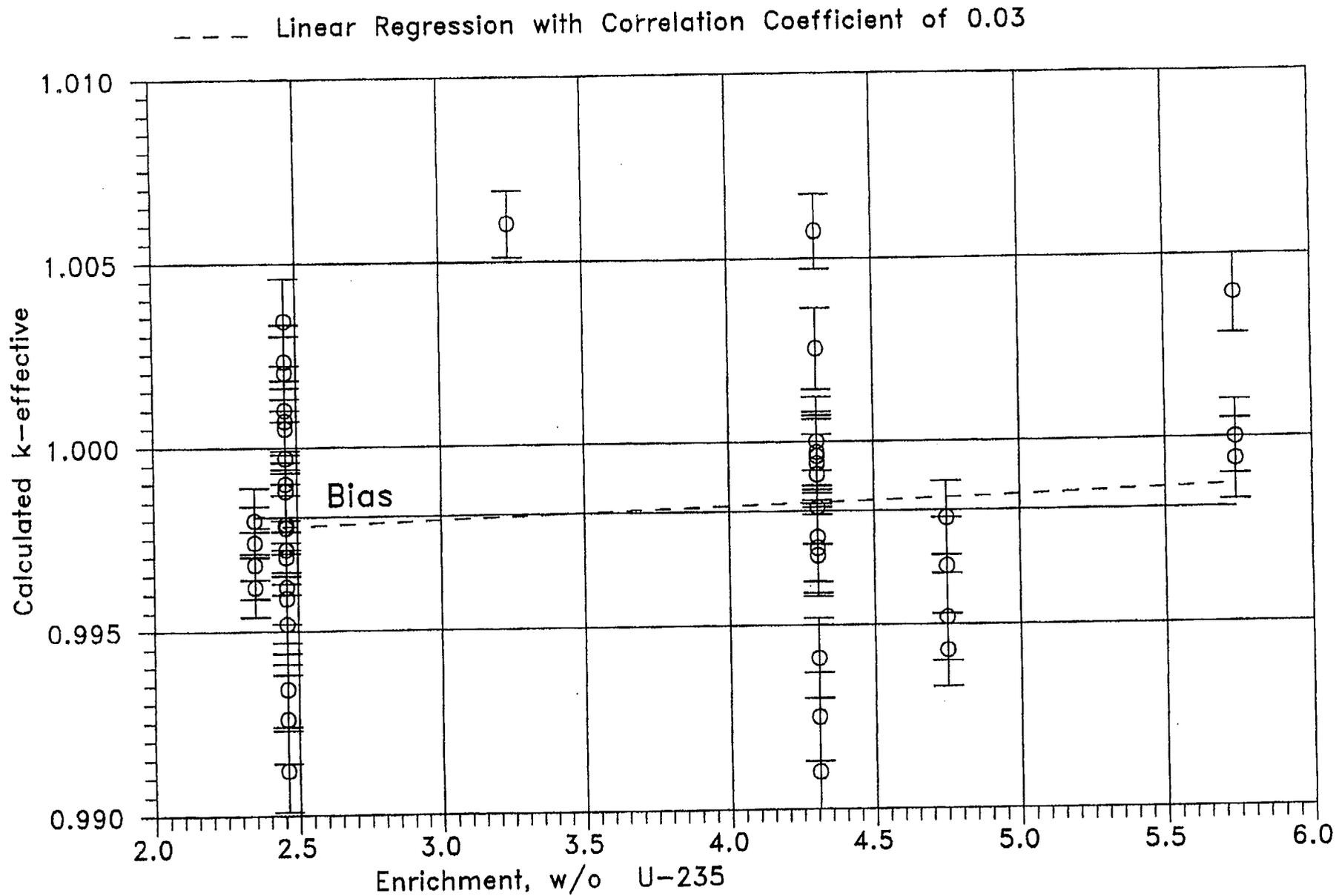


FIGURE 4A.3 MCNP CALCULATED k-eff VALUES AT VARIOUS U-235 ENRICHMENTS

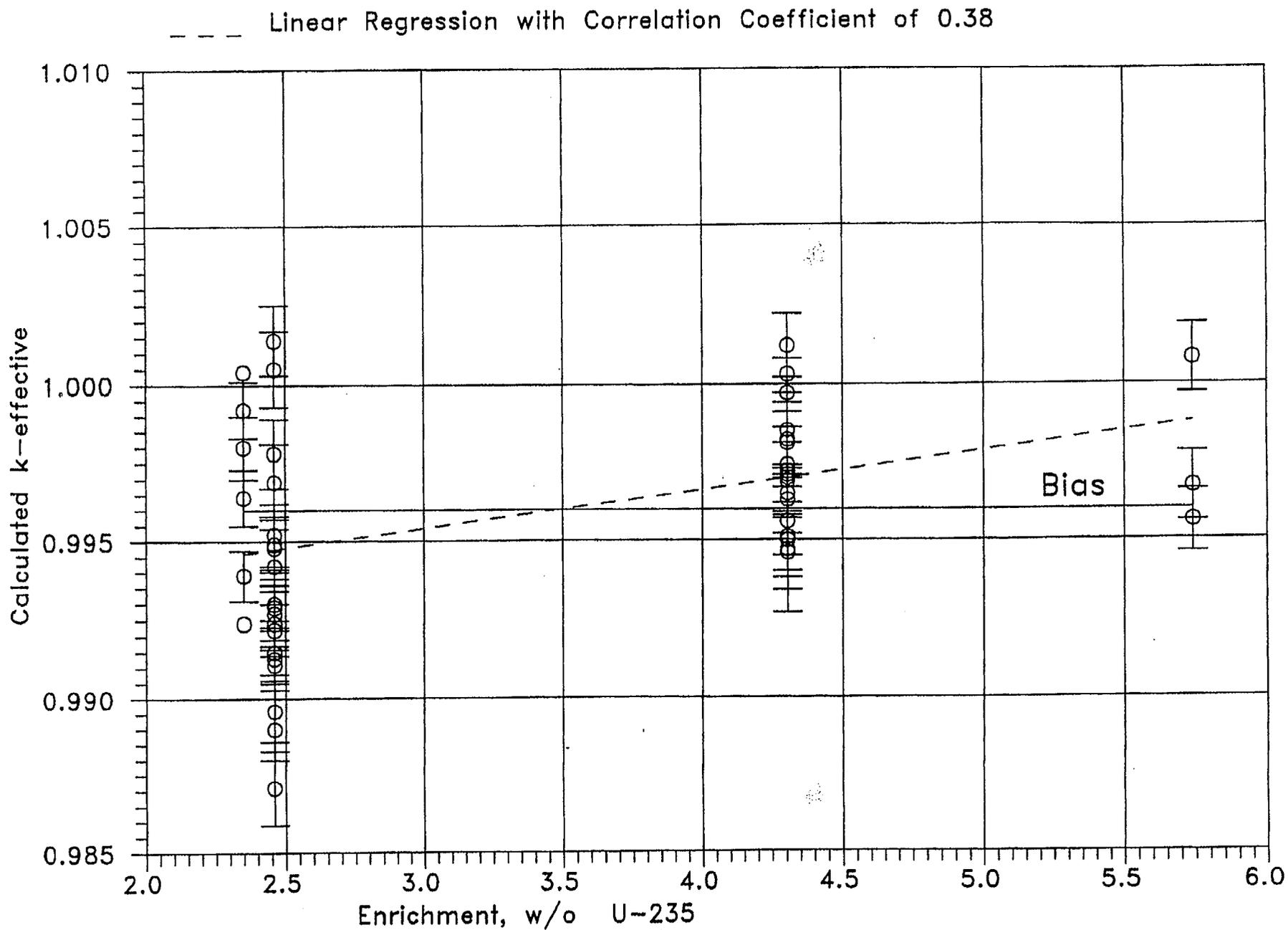


FIGURE 4A.4 KENO CALCULATED k -eff VALUES AT VARIOUS U-235 ENRICHMENTS

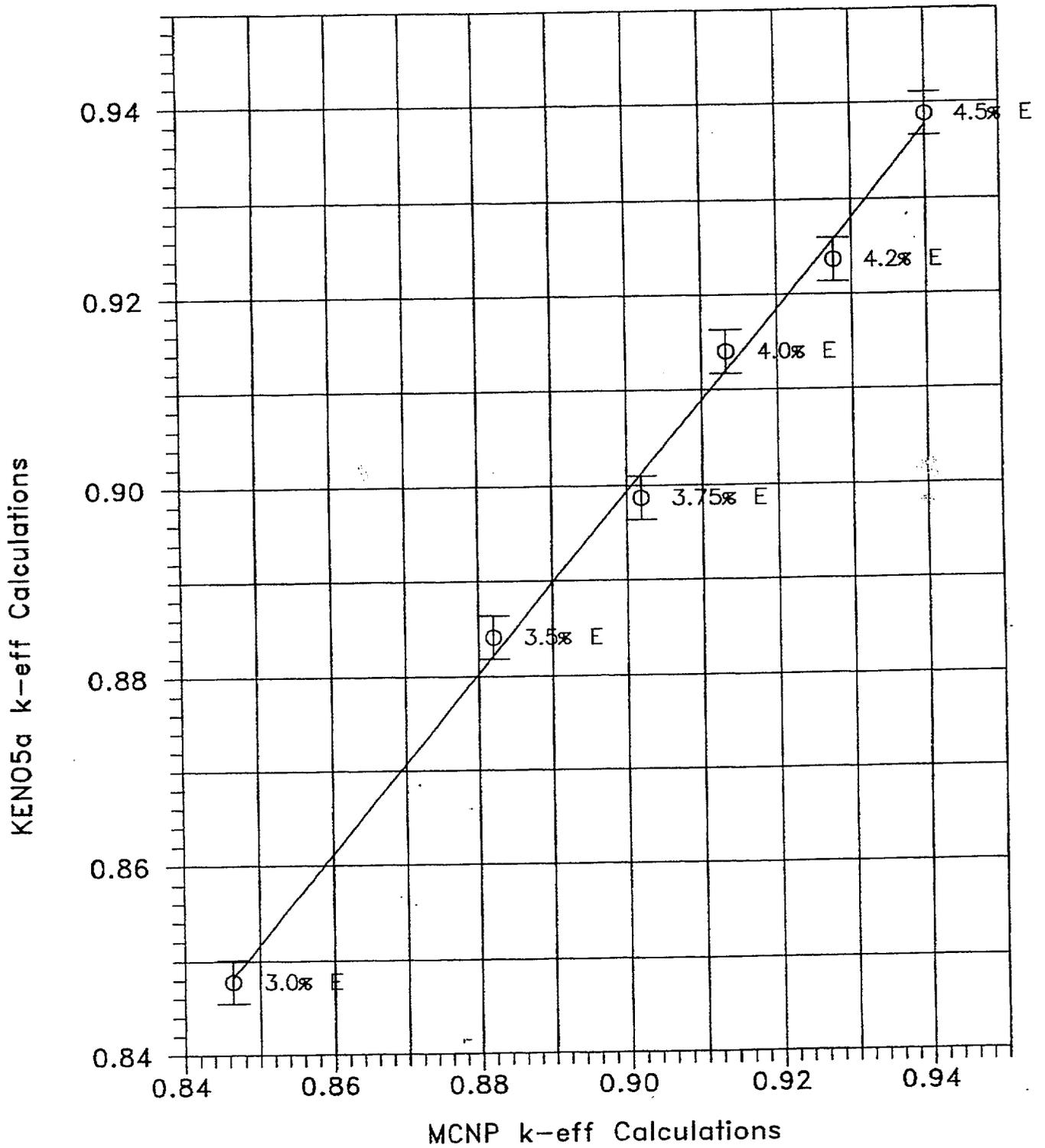


FIGURE 4A.5 COMPARISON OF MCNP AND KENO5A CALCULATIONS FOR VARIOUS FUEL ENRICHMENTS

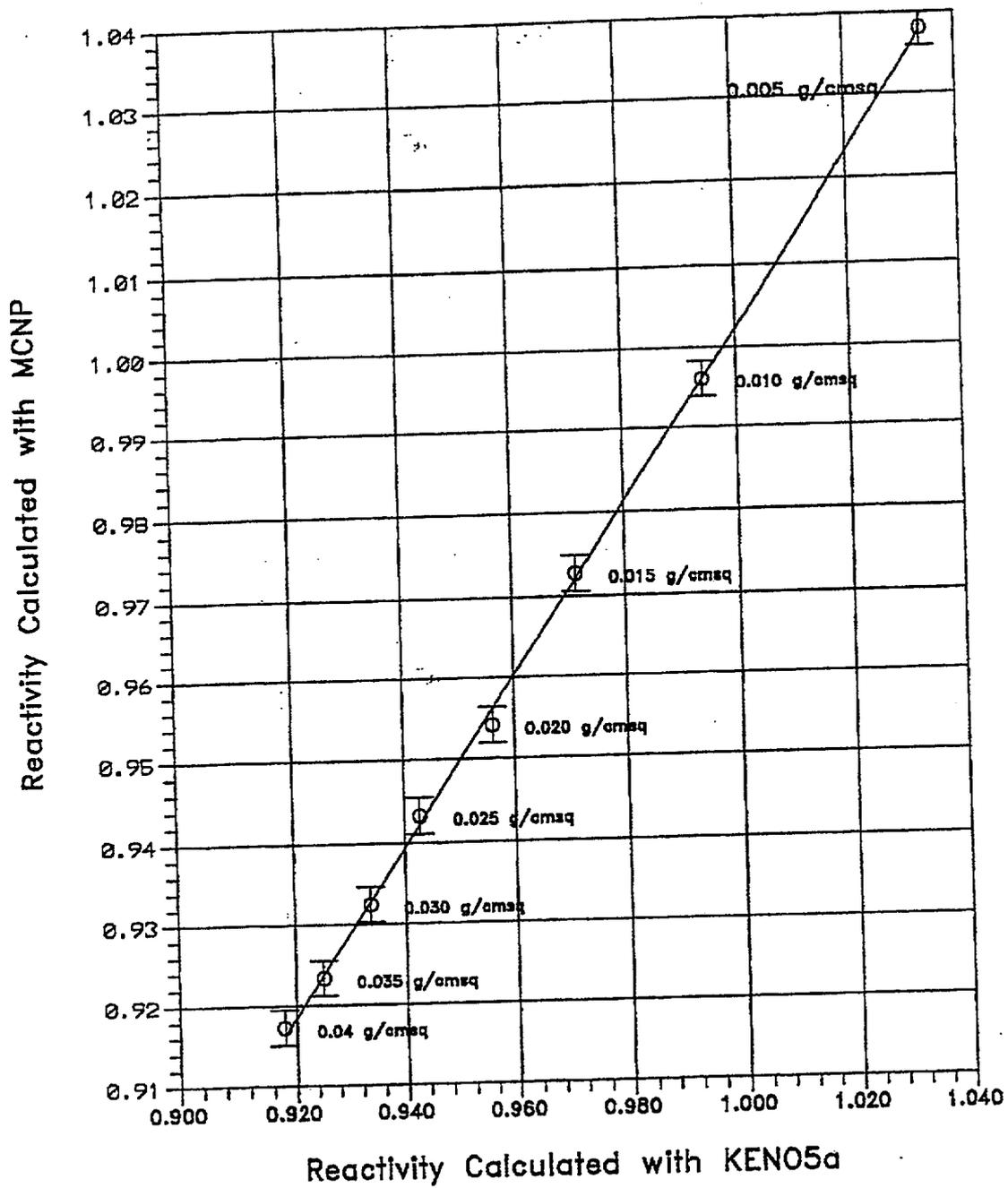


FIGURE 4A.6 COMPARISON OF MCNP AND KENO5a CALCULATIONS FOR VARIOUS BORON-10 AREAL DENSITIES

The proposed change does not entail any physical modifications to fuel, storage racks, or plant cooling systems. No changes in fresh fuel enrichment limits, constraints on maximum fuel rod burnup, or cooling time restrictions prior to the manipulation of irradiated fuel are being proposed. There will be no changes to the spent fuel decay heat load or to the SFP cooling system capabilities. Therefore, the previous thermal-hydraulic evaluations performed for the SFP remain valid.

The proposed change does not require any physical modifications to fuel, the fuel storage racks, or to plant structural systems. No new equipment is required to be installed. Changes in water coolant density will not significantly affect any of the evaluations previously performed for the racks. All loading conditions and load combinations previously considered remain valid. Therefore, the previously performed rack seismic/structural evaluations and reported results remain valid.

A spent fuel pool boron dilution analysis has been performed by Florida Power and Light and is included as an enclosure to the license amendment request. This dilution analysis includes a discussion of certain postulated accident conditions that can increase the pool water inventory (i.e., break in a makeup line) from the perspective of fuel pool reactivity.

The proposed change does not require any physical modifications to fuel, storage racks, or plant structures. The proposed change does not require installation of new equipment or require the removal of any existing plant equipment. The change does not produce any new potential accident conditions, because no changes to fuel handling techniques or fuel handling equipment are required to implement the proposed license amendment. The change does not produce any greater potential for previously postulated accident conditions to occur; fuel weight is not increased, the interface between fuel and the hoist grapple apparatus is not changed, and no other aspects of equipment used to perform fuel or control rod manipulation are changed.

The proposed license amendment change does not cause an increase in consequences of any postulated accident, because no changes in fresh fuel enrichment, the limitations on maximum fuel rod burnup, or minimum post-irradiation cooling times are proposed. Therefore, the previously performed mechanical accident evaluations for postulated fuel drops and the associated reported results remain valid.

8.0 POOL STRUCTURE ASSESSMENT

The proposed change does not require any physical modifications to fuel, storage racks, or plant structural systems. No new equipment is required to be installed. Changes in water coolant density will not significantly affect any of the evaluations previously performed for the pool structure. All loading conditions and load combinations previously considered remain valid. Therefore, the previously performed rack pool structure evaluations and reported results remain valid.

The proposed change does not require any physical modifications to fuel or to the fuel storage racks. The proposed change does not increase the amount of fuel stored in the fuel storage racks or cause the quantity of other activated material to increase. As a result, no new radiological source terms need to be considered. Changes in water coolant density will not significantly affect any of the radiological evaluations previously performed for the racks. It is noted that the credit taken for soluble boron does not in any way affect the soluble boron already present within the pool.

The maximum allowed fuel enrichment, fuel rod burnup, and the minimum allowed post-irradiation fuel cooling time remain unchanged. Thus, the spent fuel source terms remain unchanged. The revised fuel storage configuration will not significantly affect the location of source terms represented by spent fuel. The proximity of fuel to the pool water surface and exterior walls will not change. Thus, the radiation attenuation provided by the walls and the fuel pool water inventory remains unchanged. Therefore, the previously performed radiological evaluations and their reported results remain valid. Dose levels surrounding the SFP are not expected to change significantly after implementing the proposed change.

Florida Power and Light has prepared an environmental and cost benefit assessment (enclosure of the 10CFR50.92 evaluation) of the proposed license amendment. This assessment examines the underlying need for actions to mitigate the consequences of Boraflex degradation at St. Lucie Unit 1 and it considers the thermal and radiological impacts on the environment of the proposed change. This assessment also considers the occupational exposure that will be incurred as the proposed license amendment is implemented. In the assessment FPL identified several alternative methods of managing the storage of irradiated nuclear fuel and it examined the environmental and economic consequences of each candidate alternative. Finally, the assessment considered the ramifications of a "no action" alternative.

The conclusion of the environmental assessment is that none of the alternatives examined has a lower overall impact on the environment than the proposed alternative, which credits the presence of soluble boron in the fuel pool and the repositioning of stored irradiated fuel. The occupational exposure plant workers can expect to receive during the fuel repositioning campaign has been conservatively estimated and is a small fraction of the St. Lucie site's annual radiation exposure budget. Finally, the assessment concluded that none of the alternatives considered, including the "no action" alternative is economically superior to the chosen alternative.