# CRITICALITY ANALYSIS OF THE DONALD C. COOK NUCLEAR PLANT FUEL RACKS REGION 1 CHECKERBOARD OF BURNED & FRESH FUEL ASSEMBLIES

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# 1.0 INTRODUCTION

This report presents the results of a criticality analysis of the Donald C. Cook Nuclear Plant Spent Fuel Pool (SFP) storage rack for the storage of Westinghouse and ANF (Advanced Nuclear Fuel) 15x15 and 17x17 fuel assemblies. This analysis considers a checkerboard arrangement of burned and fresh fuel assemblies within the SFP Region 1 area.

The Donald C. Cook Nuclear Plant SFP storage rack was previously analyzed[1] for the storage of Westinghouse 15x15 and 17x17 fuel as two separate spent fuel arrays or regions. Region 1 was analyzed for nominal enrichments up to 4.95 w/o $U^{235}$ using a three out of four fuel assembly storage arrangement. Region 2 was analyzed for nominal enrichments up to 3.95 w/o $U^{235}$ using all available storage cells. Enrichments greater than 3.95 w/o were also allowed in the Region 2 area, provided restrictions on burnup were met.

This report describes an alternate fuel assembly storage arrangement for the Region 1 area which utilizes all available storage locations. This arrangement regains the twenty-five percent storage cell loss of the previous three out of four storage arrangement, by checkerboarding burned and fresh fuel assemblies together within the same region. Burnup credit, which takes into consideration the changes in fuel and fission product inventory resulting from depletion in the reactor core, will be used to establish the burnup requirements of the burned fuel assemblies in the checkerboard. The criticality and burnup credit analysis of the checkerboard arrangement is presented in Section 3 of this report.

This Donald C. Cook Nuclear Plant SFP criticality analysis is based on maintaining $k_{eff} \leq 0.95$ for storage of Westinghouse 15x15 STD (Standard) and OFA (Optimized Fuel Assembly), 17x17 STD, OFA and VANTAGE 5, and ANF 15x15 and 17x17 fuel assemblies. The fuel parameters relevant to this analysis are given in Table 1 on page 12.

## 1.1 DESIGN DESCRIPTION

The Donald C. Cook Nuclear Plant SFP storage cell design is depicted schematically in Figure 1 on page 18 with nominal dimensions given on the figure. The Region 1 checkerboard arrangement of burned and fresh fuel assemblies is shown in Figure 2 on page 19 and an example of the interface boundary be-

tween the Region 1 checkerboard and existing Region 2 storage areas is given in Figure 3 on page 20.

The total number of SFP locations designated as Region 1 or 2 is left to the utility to determine. The boundary between the two regions can be drawn anywhere within the SFP racks, but the interface must be configured such that there is a one row carryover of the pattern established in Region 1 into Region 2. This assures that the pattern of fuel assemblies at the interface will not be more reactive than the patterns allowed on either side of the boundary.

Figure 3 on page 20 illustrates the Region 1 to 2 boundary interface. In this figure, the horizontally-lined boxes represent Region 1 "fresh" fuel storage locations which are qualified for fresh fuel enrichments up to 5.0 w/o $U^{235}$. The diagonally-lined boxes represent Region 1 "burned" fuel storage locations which are qualified for storage of fuel assemblies which satisfy the Region 1 burnup-enrichment requirements of Figure 4 on page 21. The blank boxes represent Region 2 fuel assembly storage locations which are qualified for storage of fuel assemblies which satisfy the Region 2 enrichment-burnup requirements as reported in the previous criticality report[1]. In this figure, the checkerboard pattern of Region 1 burned assemblies (diagonally-lined boxes) has been carried into the Region 2 area by one row. The remaining cells in the first row of Region 2 are then filled with fuel assemblies which meet the requirements for storage in Region 2 (blank boxes). In this way, the requirements for fuel assembly storage in both regions are satisfied, and any grouping of four assemblies at the interface boundary will always be less than or equal in reactivity to similar groups of four assemblies allowed on either side of the boundary.

## 1.2 DESIGN CRITERIA

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between fuel assemblies and inserting neutron poison between fuel assemblies.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective neutron multiplication factor, $k_{eff}$, of the fuel assembly array will be less than 0.95 as recommended by ANSI 57.2-1983 and Reference 2.

[illegible]

# 2.0 ANALYTICAL METHODS

## 2.1 CRITICALITY CALCULATION METHODOLOGY

The criticality calculation method and cross-section values are verified by comparison with critical experiment data for fuel assemblies similar to those for which the racks are designed. This benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

The design method which insures the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX[3, 4] system of codes for cross-section generation and KENO IV[5] for reactivity determination.

The 227 energy group cross-section library that is the common starting point for all cross-sections used for the benchmarks and the storage rack is generated from ENDF/B-V[3] data. The NITAWL[4] program includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM[4] program which is a one-dimensional $S_n$ transport theory code. These multigroup cross-section sets are then used as input to KENO IV[5] which is a three dimensional Monte Carlo theory program designed for reactivity calculations.

A set of 33 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty. The experiments range from water moderated, oxide fuel arrays separated by various materials (B4C, steel, water, etc) that simulate LWR (Light Water Reactor) fuel shipping and storage conditions[6] to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials[7] (Plexiglas and air) that demonstrate the wide range of applicability of the method. Table 2 on page 13 summarizes these experiments.

The average $k_{eff}$ of the benchmarks is 0.992[8]. The standard deviation of the bias value is 0.0008 $\Delta k$. The 95/95 one sided tolerance limit factor for 33 values is 2.19. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.0018 $\Delta k$.

## 2.2 REACTIVITY EQUIVALENCING METHODOLOGY

Spent fuel storage, in the Region 1 area of the Donald C. Cook Nuclear Plant SFP, is achievable by means of the concept of reactivity equivalencing. The concept of reactivity equivalencing is predicated upon the reactivity decrease associated with fuel depletion. A series of reactivity calculations is performed to generate a set of enrichment-fuel assembly discharge burnup ordered pairs which all yield the equivalent $k_{eff}$ when the fuel is stored in the Region 1 racks.

The data points on the reactivity equivalence curve are generated with a transport theory computer code, PHOENIX[9]. PHOENIX is a depletable, two-dimensional, multigroup, discrete ordinates, transport theory code. A 25 energy group nuclear data library based on a modified version of the British WIMS[10] library is used with PHOENIX.

A study was done to examine fuel reactivity as a function of time following discharge from the reactor. Fission product decay was accounted for using CINDER[11]. CINDER is a point-depletion computer code used to determine fission product activities. The fission products were permitted to decay for 30 years after discharge. The fuel reactivity was found to reach a maximum at approximately 100 hours after discharge. At this time, the major fission product poison, $Xe^{135}$, has nearly completely decayed away. Furthermore, the fuel reactivity was found to decrease continuously from 100 hours to 30 years following discharge. Therefore, the most reactive time for a fuel assembly after discharge from the reactor can be conservatively approximated by removing the $Xe^{135}$.

The PHOENIX code has been validated by comparisons with experiments where the isotopic fuel composition has been examined following discharge from a reactor. In addition, an extensive set of benchmark critical experiments has been analyzed with PHOENIX. Comparisons between measured and predicted uranium and plutonium isotopic fuel compositions are shown in Table 3 on page 14. The measurements were made on fuel discharged from Yankee Core 5[12]. The data in Table 3 on page 14 shows that the agreement between PHOENIX predictions and measured isotopic compositions is good.

The agreement between reactivities computed with PHOENIX and the results of 81 critical benchmark experiments is summarized in Table 4 on page 15. Key parameters describing each of the 81 experiments are given in Table 5 on page 16. These reactivity comparisons again show good agreement between experiment and PHOENIX calculations.

Since the burnup history of fuel assemblies which will be discharged in the future is not known exactly, a reactivity uncertainty is applied to the burnup-dependent reactivities computed with PHOENIX. An uncertainty which increases linearly with burnup to 0.01 $\Delta k$ at 30,000 MWD/MTU is applied to the PHOENIX calculational results in the development of the Region 1 burnup requirements.

This uncertainty is considered to be very conservative and is based on consideration of the good agreement between PHOENIX predictions and measurements (comparison results with the Yankee Core experiments and 81 benchmark experiments are given in Table 3 on page 14 and Table 4 on page 15) and on conservative estimates of fuel assembly isotopic buildup variances. For the Donald C. Cook Nuclear Plant SFP Region 1 analysis, the PHOENIX calculations for the maximum burnup of 31,000 MWD/MTU include a reactivity uncertainty of 0.0103 $\Delta k$.

# 3.0 CRITICALITY ANALYSIS OF REGION 1 CHECKERBOARD ARRANGEMENT

This section develops and describes the analytical techniques and models employed to perform the criticality and reactivity equivalencing analysis for storage of fuel in the Donald C. Cook Nuclear Plant Spent Fuel Pool (SFP) Region 1 area. This analysis considers a checkerboard arrangement of burned and fresh fuel assemblies within the SFP Region 1 area.

Section 3.1 describes the KENO reactivity calculations for the checkerboard arrangement of burned and fresh fuel assemblies. For the KENO analysis, the "burned" fuel assemblies will be represented by fresh fuel assemblies with a low enrichment. Section 3.2 will then describe the PHOENIX reactivity equivalencing analysis which will establish the burnup requirements of the "burned" fuel assemblies in the checkerboard. Section 3.3 will discuss postulated accidents and Section 3.4 will present the results of the PHOENIX sensitivity calculations for enrichment, cell center-to-center spacing, and poison loading.

## 3.1 KENO REACTIVITY CALCULATIONS

The following assumptions are used to develop the nominal case KENO model for checkerboard storage of burned and fresh fuel assemblies in the Region 1 area (refer to Figure 2 on page 19 for layout).

1. Westinghouse 17x17 OFA fuel assemblies with nominal enrichments of 5.0 w/o $U^{235}$ are modelled in the "fresh" fuel storage locations of the checkerboard. The enrichment of 5.0 w/o $U^{235}$ was chosen to conservatively bound all present and future fuel assembly enrichments. Evaluation of the Westinghouse 15x15 and 17x17 and ANF 15x15 and 17x17 fuel assemblies shows that the Westinghouse 17x17 OFA fuel assembly is the most reactive fuel type at this enrichment. The Westinghouse 17x17 VANTAGE 5 fuel design parameters relevant to the criticality analysis are the same as the OFA parameters and will yield equivalent results. Therefore, only the Westinghouse 17x17 OFA fuel assembly is analyzed in the "fresh" fuel checkerboard locations (see Table 1 on page 12 for fuel parameters).

2. Westinghouse 17x17 STD fuel assemblies with nominal enrichments of 2.3 w/o $U^{235}$ are modelled in the "burned" fuel storage locations of the checkerboard. Evaluation of the Westinghouse 15x15 and 17x17 and ANF 15x15

and 17x17 fuel assemblies shows that the Westinghouse 17x17 STD fuel assembly is more reactive than other Westinghouse 17x17 and ANF 15x15 and 17x17 fuel types and approximately equivalent (within 0.0015 $\Delta k$) to the Westinghouse 15x15 fuel assembly at this enrichment. Therefore, only the Westinghouse 17x17 STD fuel assembly is analyzed in the "burned" fuel checkerboard locations (see Table 1 on page 12 for fuel parameters).

3. All fuel assemblies contain the highest authorized enrichment over the finite 144 inch length of each rod, are at the most reactive point in life, and no credit is taken for any burnable absorber in the fuel rods or any natural enrichment axial blankets. These assumptions result in conservative calculations of reactivity.

4. The fuel pellets are assumed to be at 96% of theoretical density, and no credit is taken for dishing or chamfering.

5. No credit is taken for any $U^{234}$ or $U^{236}$ in the fuel, nor is any credit taken for the build up of fission product poison material.

6. The moderator is pure water at a temperature of 68°F. A conservative value of 1.0 gm/cm$^3$ is used for the density of water.

7. No credit is taken for any spacer grids or spacer sleeves.

8. All available fuel cells are utilized. Fuel assemblies are arranged in a checkerboard pattern, as depicted in Figure 2 on page 19.

9. The array is infinite in lateral extent and finite in axial extent which allows neutron leakage from only the axial direction.

10. The minimum poison material loading of 0.02 grams $B^{10}$ per square centimeter is used throughout the array.

The KENO calculation for the nominal case resulted in a $k_{eff}$ of 0.8932 with a 95 percent probability/95 percent confidence level uncertainty of ±0.0045. The nominal case result can be compared to the worst case result to determine the relative impact of applying the worst case assumptions. The nominal case is also used as the center point for the sensitivity analysis discussed in Section 3.4.

The maximum $k_{eff}$ under normal conditions arises from consideration of mechanical and material thickness tolerances resulting from the manufacturing process in addition to asymmetric positioning of fuel assemblies within the storage cells. Westinghouse internal studies of asymmetric positioning of fuel assemblies within the storage cells have shown that symmetrically placed fuel assemblies yield equal or conservative results in rack $k_{eff}$. The sheet metal tolerances are considered along with construction tolerances related to the cell I.D., and cell center-to-center spacing. For the Region 1 racks, this results in the reduction of the nominal center to center spacings to their minimum values.

Furthermore, fuel enrichments are increased by 0.05 w/o $U^{235}$ to conservatively account for enrichment variability. Enrichments are assumed to be 5.05 w/o $U^{235}$ for the "fresh" fuel cells and 2.35 w/o $U^{235}$ for the "burned" fuel cells. Thus, the "worst case" KENO model of the Region 1 storage racks contains minimum center to center spacings, symmetrically placed fuel assemblies, and maximum fuel assembly enrichments.

Based on the analysis described above, the following equation is used to develop the maximum $k_{eff}$ for the Donald C. Cook Nuclear Plant SFP Region 1 racks:

$$k_{eff} = k_{worst} + B_{method} + B_{part} + B_{asm} + \sqrt{[\ (ks)^2_{worst} + (ks)^2_{method}\ ]}$$

where:

| | | |
|---|---|---|
| $k_{worst}$ | = | worst case KENO $k_{eff}$ that includes material tolerances, and mechanical tolerances which can result in spacings between fuel assemblies less than nominal |
| $B_{method}$ | = | method bias determined from benchmark critical comparisons |
| $B_{part}$ | = | bias to account for poison particle self-shielding. This standard term accounts for the increased neutron transmission through the poison plate due to the inherent effects of poison particle self-shielding, and has been analytically determined for poison plates similar to those used in this analysis. |
| $B_{asm}$ | = | bias to account for the reactivity difference between the Region 1 SFP "burned" checkerboard locations loaded with Westinghouse 17x17 STD fuel assemblies at 2.3 w/o $U^{235}$ versus Westinghouse 15x15 fuel assemblies at 2.3 w/o $U^{235}$. |
| $ks_{worst}$ | = | 95/95 uncertainty in the worst case KENO $k_{eff}$ |
| $ks_{method}$ | = | 95/95 uncertainty in the method bias |

Substituting calculated values in the order listed above, the result is:

$$k_{eff} = 0.9295 + 0.0083 + 0.0014 + 0.0015 + \sqrt{[(0.0044)^2 + (0.0018)^2\ ]} = 0.9455$$

Since $k_{eff}$ is less than 0.95 including uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality is met for the Region 1 checkerboard arrangement of "fresh" fuel cells at a nominal enrichment of 5.0 w/o $U^{235}$ and "burned" fuel cells at a nominal enrichment of 2.3 w/o $U^{235}$.

## 3.2 PHOENIX REACTIVITY EQUIVALENCING

Spent fuel storage, in the Region 1 checkerboard area, is achievable by means of the concept of reactivity equivalencing. The concept of reactivity equivalencing is predicated upon the reactivity decrease associated with fuel depletion. A series of reactivity calculations are performed to generate a set of enrichment-fuel assembly discharge burnup ordered pairs which all yield the equivalent $k_{eff}$ when the fuel is stored in the Region 1 "burned" fuel storage cells.

The maximum $k_{eff}$ for storage of spent fuel in the Region 1 checkerboard area is determined using the methods described in Section 2. Figure 4 on page 21 represents combinations of fuel enrichment and discharge burnup yielding the same rack multiplication factor ($k_{eff}$) as the rack loaded with a checkerboard of 5.0 w/o $U^{235}$ fuel in the "fresh" fuel cells and 2.3 w/o $U^{235}$ fuel (at zero burnup) in the "burned" fuel cells. This curve was obtained by first calculating the equivalent reactivity points using PHOENIX and then normalizing the points to the nominal KENO calculation described in Section 3.1. The uncertainty associated with the reactivity equivalence methodology is included in the development of the burnup requirements. This uncertainty was discussed in Section 2.2.

Figure 4 on page 21 shows the constant $k_{eff}$ contour generated for the Donald C. Cook Nuclear Plant Region 1 checkerboard. Note in Figure 4 on page 21 the endpoint at 0 MWD/MTU where the enrichment is 2.3 w/o, and at 31,000 MWD/MTU where the enrichment is 5.0 w/o. The interpretation of the endpoint data is as follows: the reactivity of the Region 1 checkerboard rack containing 5.0 w/o $U^{235}$ fuel at zero burnup in the "fresh" fuel cells and 5.0 w/o $U^{235}$ fuel at 31,000 MWD/MTU burnup in the "burned" fuel cells is equivalent to the reactivity of the Region 1 checkerboard rack containing 5.0 w/o $U^{235}$ fuel at zero burnup in the "fresh" fuel cells and 2.3 w/o $U^{235}$ fuel at zero burnup in the "burned" fuel cells. It is important to recognize that the curve in Figure 4 on page 21 is based on a constant rack reactivity for that region and not on a constant fuel assembly reactivity. In this way, the environment of the storage rack and its influence on assembly reactivity is implicitly considered.

## 3.3 POSTULATED ACCIDENTS

Most accident conditions will not result in an increase in $k_{eff}$ of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not excessively deformed and the dropped fuel assembly has more than twelve inches of water separating it from the active fuel height of stored fuel assemblies which precludes interaction).

However, accidents can be postulated which would increase reactivity (i.e., misloading a fuel assembly with a burnup and enrichment combination outside

of the acceptable area in Figure 4 on page 21, or dropping a fuel assembly between the rack and pool wall). For these accident conditions, the double contingency principle of ANSI N16.1-1975 is applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

The presence of approximately 2000 ppm boron in the pool water will decrease reactivity by about 0.25 $\Delta k$. In perspective, this is about five times more negative reactivity than could be added if every cell in the SFP were filled with fresh 5.0 w/o $U^{235}$ fuel assemblies. Thus, for postulated accidents, should there be a reactivity increase, $k_{eff}$ would be less than or equal to 0.95 due to the effect of the dissolved boron. Since the Donald C. Cook Nuclear Plant SFP will be maintained at a boron concentration of 2400 ppm, additional margin will exist to the 0.95 limit.

## 3.4 SENSITIVITY ANALYSIS

To show the dependence of $k_{eff}$ on fuel and storage cells parameters as requested by the NRC[21], the variation of the $k_{eff}$ with respect to the following parameters was developed using the PHOENIX computer code:

1. Fuel enrichment, with a 0.50 w/o $U^{235}$ delta about the nominal case enrichment. For this sensitivity, both the "fresh" and "burned" fuel assembly enrichments were adjusted simultaneously.

2. Center-to-center spacing of storage cells, with a half inch delta about the nominal case center-to-center spacing.

3. Poison loading, with a 0.01 gm-$B^{10}$/$cm^2$ delta about the nominal case poison loading.

Results of the sensitivity analysis for the Region 1 checkerboard storage arrangement are shown in Figure 5 on page 22 through Figure 7 on page 24.

## 4.0 SUMMARY OF CRITICALITY RESULTS

The acceptance criteria for criticality requires the effective neutron multiplication factor, $k_{eff}$, to be less than or equal to 0.95, including uncertainties, under all conditions for the storage of fuel assemblies in the Spent Fuel Pool (SFP).

This report shows that the acceptance criteria for criticality is met for the Donald C. Cook Nuclear Plant Spent Fuel Pool (SFP) Region 1 checkerboard of burned and fresh fuel assemblies. This conclusion is valid for the storage of Westinghouse 15x15 STD and OFA, and 17x17 STD, OFA and VANTAGE 5, and ANF 15x15 and 17x17 fuel assemblies with the following nominal enrichment limits:

| | | |
|---|---|---|
| SFP Region 1 "Fresh" Fuel | ≤ | 5.0 w/o $U^{235}$ |
| SFP Region 1 "Burned" Fuel | ≤ | 5.0 w/o $U^{235}$, with burnup restrictions given by Figure 4 on page 21 |

The checkerboard arrangement to be used for the storage of burned and fresh fuel assemblies in the Region 1 area is shown on Figure 2 on page 19. An example of the interface boundary between the Region 1 checkerboard and the Region 2 area is given by Figure 3 on page 20.

The analytical methods employed herein conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI 57.2-1983, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 6.4.2; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety"; and with the NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage."

Table 1. Fuel Parameters Employed in Criticality Analysis

| PARAMETER | W STD/OFA 15x15 | W OFA/V5 17x17 | W STD 17x17 | ANF 15x15 | ANF 17x17 |
|---|---|---|---|---|---|
| Number of Fuel Rods per Assembly | 204 | 264 | 264 | 204 | 264 |
| Rod Zirc-4 Clad O.D. (inch) | 0.422 | 0.360 | 0.374 | 0.424 | 0.360 |
| Clad Thickness (inch) | 0.0243 | 0.0225 | 0.0255 | 0.0300 | 0.0250 |
| Fuel Pellet O.D. (inch) | 0.3659 | 0.3088 | 0.3225 | 0.3565 | 0.3030 |
| Fuel Pellet Density (% of Theoretical) | 96 | 96 | 96 | 96 | 96 |
| Fuel Pellet Dishing Factor (%) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Rod Pitch (inch) | 0.563 | 0.496 | 0.496 | 0.563 | 0.496 |
| Number of Guide Tubes | 20 | 24 | 24 | 20 | 24 |
| Guide Tube O.D. (inch) | 0.533 | 0.474 | 0.482 | 0.545 | 0.480 |
| Guide Tube Thickness (inch) | 0.017 | 0.016 | 0.016 | 0.017 | 0.016 |
| Number of Instrument Tubes | 1 | 1 | 1 | 1 | 1 |
| Instrument Tube O.D. (inch) | 0.533 | 0.474 | 0.482 | 0.545 | 0.480 |
| Instrument Tube Thickness (inch) | 0.017 | 0.016 | 0.016 | 0.017 | 0.016 |

Table 2. Benchmark Critical Experiments [5,6]

| | General Description | Enrichment w/o U235 | Reflector | Separating Material | Soluble Boron ppm | Keff |
|---|---|---|---|---|---|---|
| 1. | UO2 rod lattice | 2.46 | water | water | 0 | 0.9857 +/- .0028 |
| 2. | UO2 rod lattice | 2.46 | water | water | 1037 | 0.9906 +/- .0018 |
| 3. | UO2 rod lattice | 2.46 | water | water | 764 | 0.9896 +/- .0015 |
| 4. | UO2 rod lattice | 2.46 | water | B4C pins | 0 | 0.9914 +/- .0025 |
| 5. | UO2 rod lattice | 2.46 | water | B4C pins | 0 | 0.9891 +/- .0026 |
| 6. | UO2 rod lattice | 2.46 | water | B4C pins | 0 | 0.9955 +/- .0020 |
| 7. | UO2 rod lattice | 2.46 | water | B4C pins | 0 | 0.9889 +/- .0027 |
| 8. | UO2 rod lattice | 2.46 | water | B4C pins | 0 | 0.9983 +/- .0025 |
| 9. | UO2 rod lattice | 2.46 | water | water | 0 | 0.9931 +/- .0028 |
| 10. | UO2 rod lattice | 2.46 | water | water | 143 | 0.9928 +/- .0025 |
| 11. | UO2 rod lattice | 2.46 | water | stainless steel | 514 | 0.9967 +/- .0020 |
| 12. | UO2 rod lattice | 2.46 | water | stainless steel | 217 | 0.9943 +/- .0019 |
| 13. | UO2 rod lattice | 2.46 | water | borated aluminum | 15 | 0.9892 +/- .0023 |
| 14. | UO2 rod lattice | 2.46 | water | borated aluminum | 92 | 0.9884 +/- .0023 |
| 15. | UO2 rod lattice | 2.46 | water | borated aluminum | 395 | 0.9832 +/- .0021 |
| 16. | UO2 rod lattice | 2.46 | water | borated aluminum | 121 | 0.9848 +/- .0024 |
| 17. | UO2 rod lattice | 2.46 | water | borated aluminum | 487 | 0.9895 +/- .0020 |
| 18. | UO2 rod lattice | 2.46 | water | borated aluminum | 197 | 0.9885 +/- .0022 |
| 19. | UO2 rod lattice | 2.46 | water | borated aluminum | 634 | 0.9921 +/- .0019 |
| 20. | UO2 rod lattice | 2.46 | water | borated aluminum | 320 | 0.9920 +/- .0020 |
| 21. | UO2 rod lattice | 2.46 | water | borated aluminum | 72 | 0.9939 +/- .0020 |
| 22. | U metal cylinders | 93.2 | bare | air | 0 | 0.9905 +/- .0020 |
| 23. | U metal cylinders | 93.2 | bare | air | 0 | 0.9976 +/- .0020 |
| 24. | U metal cylinders | 93.2 | bare | air | 0 | 0.9947 +/- .0025 |
| 25. | U metal cylinders | 93.2 | bare | air | 0 | 0.9928 +/- .0019 |
| 26. | U metal cylinders | 93.2 | bare | air | 0 | 0.9922 +/- .0026 |
| 27. | U metal cylinders | 93.2 | bare | air | 0 | 0.9950 +/- .0027 |
| 28. | U metal cylinders | 93.2 | bare | plexiglass | 0 | 0.9941 +/- .0030 |
| 29. | U metal cylinders | 93.2 | paraffin | plexiglass | 0 | 0.9928 +/- .0041 |
| 30. | U metal cylinders | 93.2 | bare | plexiglass | 0 | 0.9968 +/- .0018 |
| 31. | U metal cylinders | 93.2 | paraffin | plexiglass | 0 | 1.0042 +/- .0019 |
| 32. | U metal cylinders | 93.2 | paraffin | plexiglass | 0 | 0.9963 +/- .0030 |
| 33. | U metal cylinders | 93.2 | paraffin | plexiglass | 0 | 0.9919 +/- .0032 |

**Table 3. Comparison of PHOENIX Isotopics Predictions to Yankee Core 5 Measurements**

| Quantity (Atom Ratio) | % Difference |
|---|---|
| U235/U | -0.67 |
| U236/U | -0.28 |
| U238/U | -0.03 |
| Pu239/U | +3.27 |
| Pu240/U | +3.63 |
| Pu241/U | -7.01 |
| Pu242/U | -0.20 |
| Pu239/U238 | +3.24 |
| Mass(Pu/U) | +1.41 |
| FISS-Pu/TOT-Pu | -0.02 |

Table 4. Benchmark Critical Experiments PHOENIX Comparison

| Description of Experiments | Number of Experiments | PHOENIX $k_{eff}$ Using Experiment Bucklings |
|---|---|---|
| $UO_2$ | | |
| Al clad | 14 | 0.9947 |
| SS clad | 19 | 0.9944 |
| Borated $H_2O$ | 7 | 0.9940 |
| Subtotal | 40 | 0.9944 |
| U-Metal | | |
| Al clad | 41 | 1.0012 |
| TOTAL | 81 | 0.9978 |

Table 5. Data for U Metal and $UO_2$ Critical Experiments (Part 1 of 2)

| Case Number | Cell Type | A/O U-235 | H2O/U Ratio | Fuel Density (G/CC) | Pellet Diameter (CM) | Material Clad | Clad OD (CM) | Clad Thickness (CM) | Lattice Pitch (CM) | Boron PPM |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | Hexa | 1.328 | 3.02 | 7.53 | 1.5265 | Aluminum | 1.6916 | .07110 | 2.2050 | 0.0 |
| 2 | Hexa | 1.328 | 3.95 | 7.53 | 1.5265 | Aluminum | 1.6916 | .07110 | 2.3590 | 0.0 |
| 3 | Hexa | 1.328 | 4.95 | 7.53 | 1.5265 | Aluminum | 1.6916 | .07110 | 2.5120 | 0.0 |
| 4 | Hexa | 1.328 | 3.92 | 7.52 | .9855 | Aluminum | 1.1506 | .07110 | 1.5580 | 0.0 |
| 5 | Hexa | 1.328 | 4.89 | 7.52 | .9855 | Aluminum | 1.1506 | .07110 | 1.6520 | 0.0 |
| 6 | Hexa | 1.328 | 2.88 | 10.53 | .9728 | Aluminum | 1.1506 | .07110 | 1.5580 | 0.0 |
| 7 | Hexa | 1.328 | 3.58 | 10.53 | .9728 | Aluminum | 1.1506 | .07110 | 1.6520 | 0.0 |
| 8 | Hexa | 1.328 | 4.83 | 10.53 | .9728 | Aluminum | 1.1506 | .07110 | 1.8060 | 0.0 |
| 9 | Square | 2.734 | 2.18 | 10.18 | .7620 | SS-304 | .8594 | .04085 | 1.0287 | 0.0 |
| 10 | Square | 2.734 | 2.92 | 10.18 | .7620 | SS-304 | .8594 | .04085 | 1.1049 | 0.0 |
| 11 | Square | 2.734 | 3.86 | 10.18 | .7620 | SS-304 | .8594 | .04085 | 1.1938 | 0.0 |
| 12 | Square | 2.734 | 7.02 | 10.18 | .7620 | SS-304 | .8594 | .04085 | 1.4554 | 0.0 |
| 13 | Square | 2.734 | 8.49 | 10.18 | .7620 | SS-304 | .8594 | .04085 | 1.5621 | 0.0 |
| 14 | Square | 2.734 | 10.38 | 10.18 | .7620 | SS-304 | .8594 | .04085 | 1.6891 | 0.0 |
| 15 | Square | 2.734 | 2.50 | 10.18 | .7620 | SS-304 | .8594 | .04085 | 1.0617 | 0.0 |
| 16 | Square | 2.734 | 4.51 | 10.18 | .7620 | SS-304 | .8594 | .04085 | 1.2522 | 0.0 |
| 17 | Square | 3.745 | 2.50 | 10.27 | .7544 | SS-304 | .8600 | .04060 | 1..0617 | 0.0 |
| 18 | Square | 3.745 | 4.51 | 10.37 | .7544 | SS-304 | .8600 | .04060 | 1.2522 | 0.0 |
| 19 | Square | 3.745 | 4.51 | 10.37 | .7544 | SS-304 | .8600 | .04060 | 1.2522 | 0.0 |
| 20 | Square | 3.745 | 4.51 | 10.37 | .7544 | SS-304 | .8600 | .04060 | 1.2522 | 456.0 |
| 21 | Square | 3.745 | 4.51 | 10.37 | .7544 | SS-304 | .8600 | .04060 | 1.2522 | 709.0 |
| 22 | Square | 3.745 | 4.51 | 10.37 | .7544 | SS-304 | .8600 | .04060 | 1.2522 | 1260.0 |
| 23 | Square | 3.745 | 4.51 | 10.37 | .7544 | SS-304 | .8600 | .04060 | 1.2522 | 1334.0 |
| 24 | Square | 3.745 | 4.51 | 10.37 | .7544 | SS-304 | .8600 | .04060 | 1.2522 | 1477.0 |
| 25 | Square | 4.069 | 2.55 | 9.46 | 1.1278 | SS-304 | 1.2090 | .04060 | 1.5113 | 0.0 |
| 26 | Square | 4.069 | 2.55 | 9.46 | 1.1278 | SS-304 | 1.2090 | .04060 | 1.5113 | 3392.0 |
| 27 | Square | 4.069 | 2.14 | 9.46 | 1.1278 | SS-304 | 1.2090 | .04060 | 1.4500 | 0.0 |
| 28 | Square | 2.490 | 2.84 | 10.24 | 1.0297 | Aluminum | 1.2060 | .08130 | 1.5113 | 0.0 |
| 29 | Square | 3.037 | 2.64 | 9.28 | 1.1268 | SS-304 | 1.1701 | .07163 | 1.5550 | 0.0 |
| 30 | Square | 3.037 | 8.16 | 9.28 | 1.1268 | SS-304 | 1.2701 | .07163 | 2.1980 | 0.0 |
| 31 | Square | 4.069 | 2.59 | 9.45 | 1.1268 | SS-304 | 1.2701 | .07163 | 1.5550 | 0.0 |
| 32 | Square | 4.069 | 3.53 | 9.45 | 1.1268 | SS-304 | 1.2701 | .07163 | 1.6840 | 0.0 |
| 33 | Square | 4.069 | 8.02 | 9.45 | 1.1268 | SS-304 | 1.2701 | .07163 | 2.1980 | 0.0 |
| 34 | Square | 4.069 | 9.90 | 9.45 | 1.1268 | SS-304 | 1.2701 | .07163 | 2.3810 | 0.0 |
| 35 | Square | 2.490 | 2.84 | 10.24 | 1.0297 | Aluminum | 1.2060 | .08130 | 1.5113 | 1677.0 |
| 36 | Hexa | 2.096 | 2.06 | 10.38 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.1737 | 0.0 |
| 37 | Hexa | 2.096 | 3.09 | 10.38 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.4052 | 0.0 |
| 38 | Hexa | 2.096 | 4.12 | 10.38 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.6162 | 0.0 |
| 39 | Hexa | 2.096 | 6.14 | 10.38 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.9891 | 0.0 |
| 40 | Hexa | 2.096 | 8.20 | 10.38 | 1.5240 | Aluminum | 1.6916 | .07112 | 3.3255 | 0.0 |
| 41 | Hexa | 1.307 | 1.01 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.1742 | 0.0 |
| 42 | Hexa | 1.307 | 1.51 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.4054 | 0.0 |
| 43 | Hexa | 1.307 | 2.02 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.6162 | 0.0 |

Table 5. Data for U Metal and $UO_2$ Critical Experiments (Part 2 of 2)

| Case Number | Cell Type | A/O U-235 | H2O/U Ratio | Fuel Density (G/CC) | Pellet Diameter (CM) | Material Clad | Clad OD (CM) | Clad Thickness (CM) | Lattice Pitch (CM) | Boron PPM |
|---|---|---|---|---|---|---|---|---|---|---|
| 44 | Hexa | 1.307 | 3.01 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.9896 | 0.0 |
| 45 | Hexa | 1.307 | 4.02 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 3.3249 | 0.0 |
| 46 | Hexa | 1.160 | 1.01 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.1742 | 0.0 |
| 47 | Hexa | 1.160 | 1.51 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.4054 | 0.0 |
| 48 | Hexa | 1.160 | 2.02 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.6162 | 0.0 |
| 49 | Hexa | 1.160 | 3.01 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.9896 | 0.0 |
| 50 | Hexa | 1.160 | 4.02 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 3.3249 | 0.0 |
| 51 | Hexa | 1.040 | 1.01 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.1742 | 0.0 |
| 52 | Hexa | 1.040 | 1.51 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.4054 | 0.0 |
| 53 | Hexa | 1.040 | 2.02 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.6162 | 0.0 |
| 54 | Hexa | 1.040 | 3.01 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.9896 | 0.0 |
| 55 | Hexa | 1.040 | 4.02 | 18.90 | 1.5240 | Aluminum | 1.6916 | .07112 | 3.3249 | 0.0 |
| 56 | Hexa | 1.307 | 1.00 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.4412 | 0.0 |
| 57 | Hexa | 1.307 | 1.52 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.5926 | 0.0 |
| 58 | Hexa | 1.307 | 2.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.7247 | 0.0 |
| 59 | Hexa | 1.307 | 3.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.9609 | 0.0 |
| 60 | Hexa | 1.307 | 4.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 2.1742 | 0.0 |
| 61 | Hexa | 1.160 | 1.52 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.5926 | 0.0 |
| 62 | Hexa | 1.160 | 2.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.7247 | 0.0 |
| 63 | Hexa | 1.160 | 3.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.9609 | 0.0 |
| 64 | Hexa | 1.160 | 4.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 2.1742 | 0.0 |
| 65 | Hexa | 1.160 | 1.00 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.4412 | 0.0 |
| 66 | Hexa | 1.160 | 1.52 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.5926 | 0.0 |
| 67 | Hexa | 1.160 | 2.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.7247 | 0.0 |
| 68 | Hexa | 1.160 | 3.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 1.9609 | 0.0 |
| 69 | Hexa | 1.160 | 4.02 | 18.90 | .9830 | Aluminum | 1.1506 | .07112 | 2.1742 | 0.0 |
| 70 | Hexa | 1.040 | 1.33 | 18.90 | 19.050 | Aluminum | 2.0574 | .07620 | 2.8687 | 0.0 |
| 71 | Hexa | 1.040 | 1.58 | 18.90 | 19.050 | Aluminum | 2.0574 | .07620 | 3.0086 | 0.0 |
| 72 | Hexa | 1.040 | 1.83 | 18.90 | 19.050 | Aluminum | 2.0574 | .07620 | 3.1425 | 0.0 |
| 73 | Hexa | 1.040 | 2.33 | 18.90 | 19.050 | Aluminum | 2.0574 | .07620 | 3.3942 | 0.0 |
| 74 | Hexa | 1.040 | 2.83 | 18.90 | 19.050 | Aluminum | 2.0574 | .07620 | 3.6284 | 0.0 |
| 75 | Hexa | 1.040 | 3.83 | 18.90 | 19.050 | Aluminum | 2.0574 | .07620 | 4.0566 | 0.0 |
| 76 | Hexa | 1.310 | 2.02 | 18.88 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.6160 | 0.0 |
| 77 | Hexa | 1.310 | 3.01 | 18.88 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.9900 | 0.0 |
| 78 | Hexa | 1.159 | 2.02 | 18.88 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.6160 | 0.0 |
| 79 | Hexa | 1.159 | 3.01 | 18.88 | 1.5240 | Aluminum | 1.6916 | .07112 | 2.9900 | 0.0 |
| 80 | Hexa | 1.312 | 2.03 | 18.88 | .9830 | Aluminum | 1.1506 | .07112 | 1.7250 | 0.0 |
| 81 | Hexa | 1.312 | 3.02 | 18.88 | .9830 | Aluminum | 1.1506 | .07112 | 1.9610 | 0.0 |

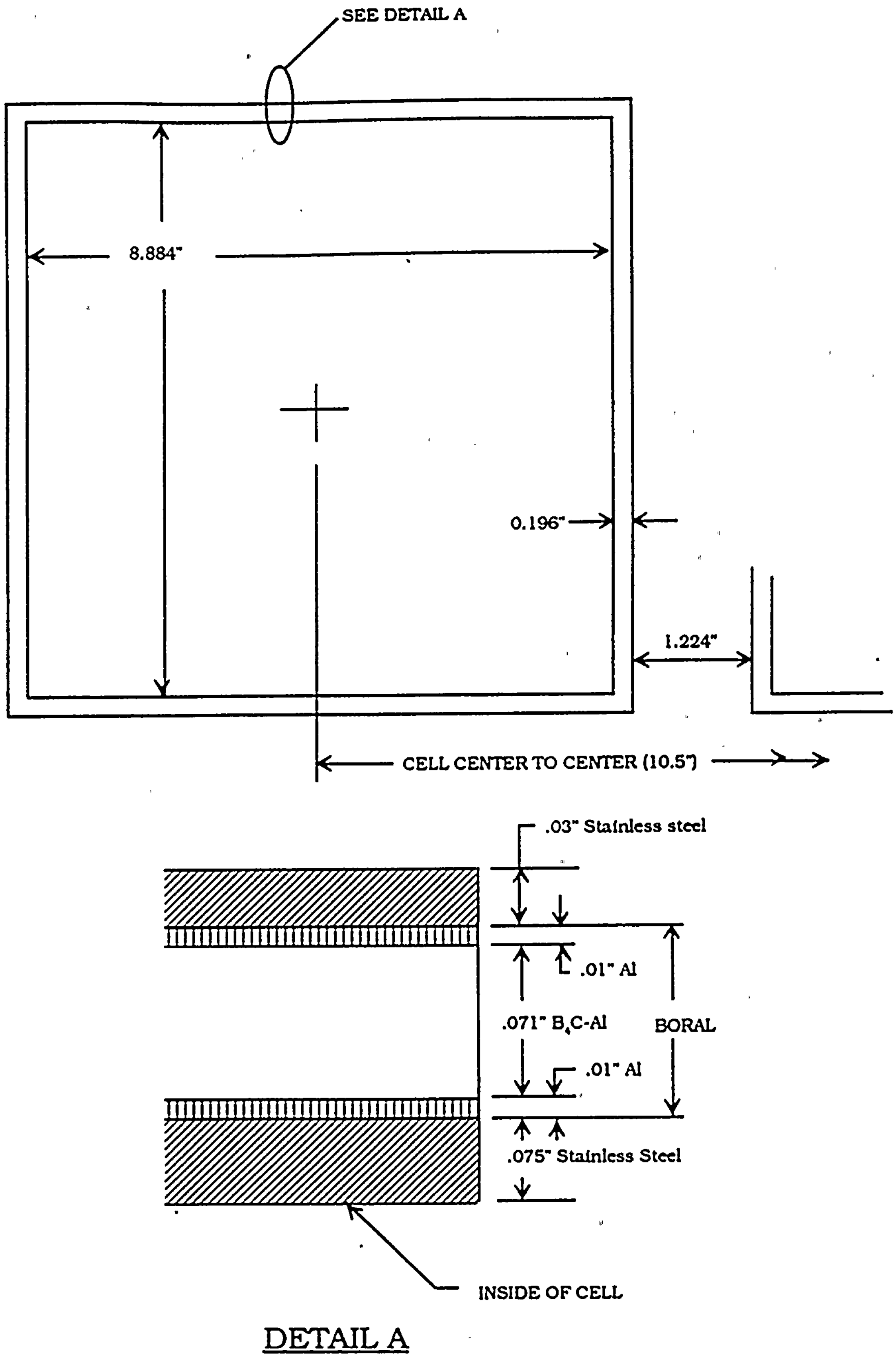
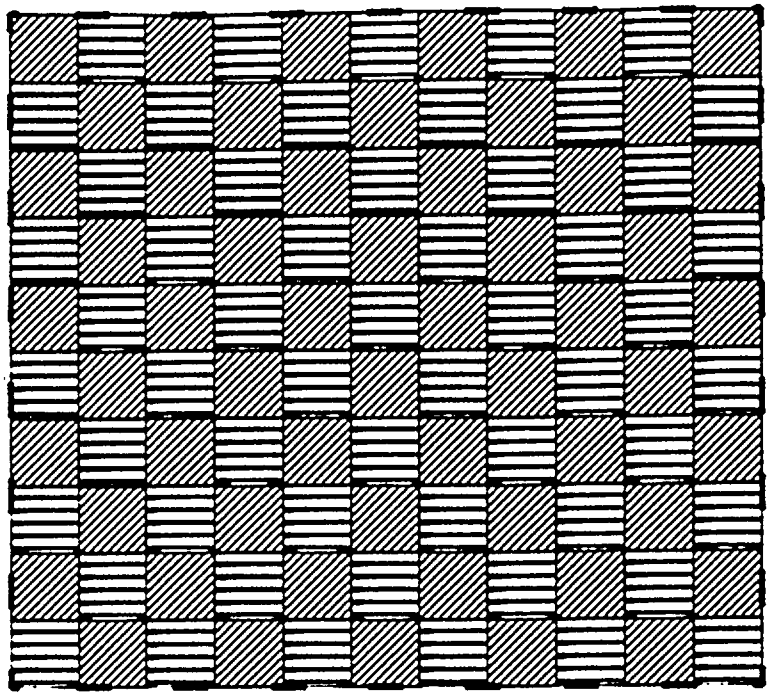



**Figure 1. Donald C. Cook Nuclear Plant Spent Fuel Pool Storage Cell Nominal Dimensions**

[illegible]



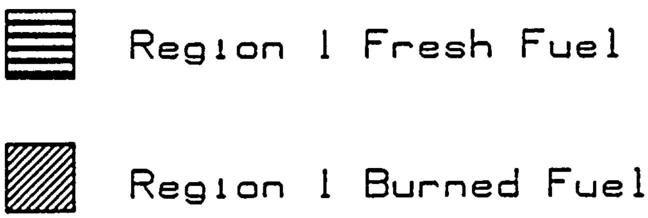



**Figure 2. Donald C. Cook Nuclear Plant SFP Region 1 Checkerboard Fuel Assembly Loading Schematic**

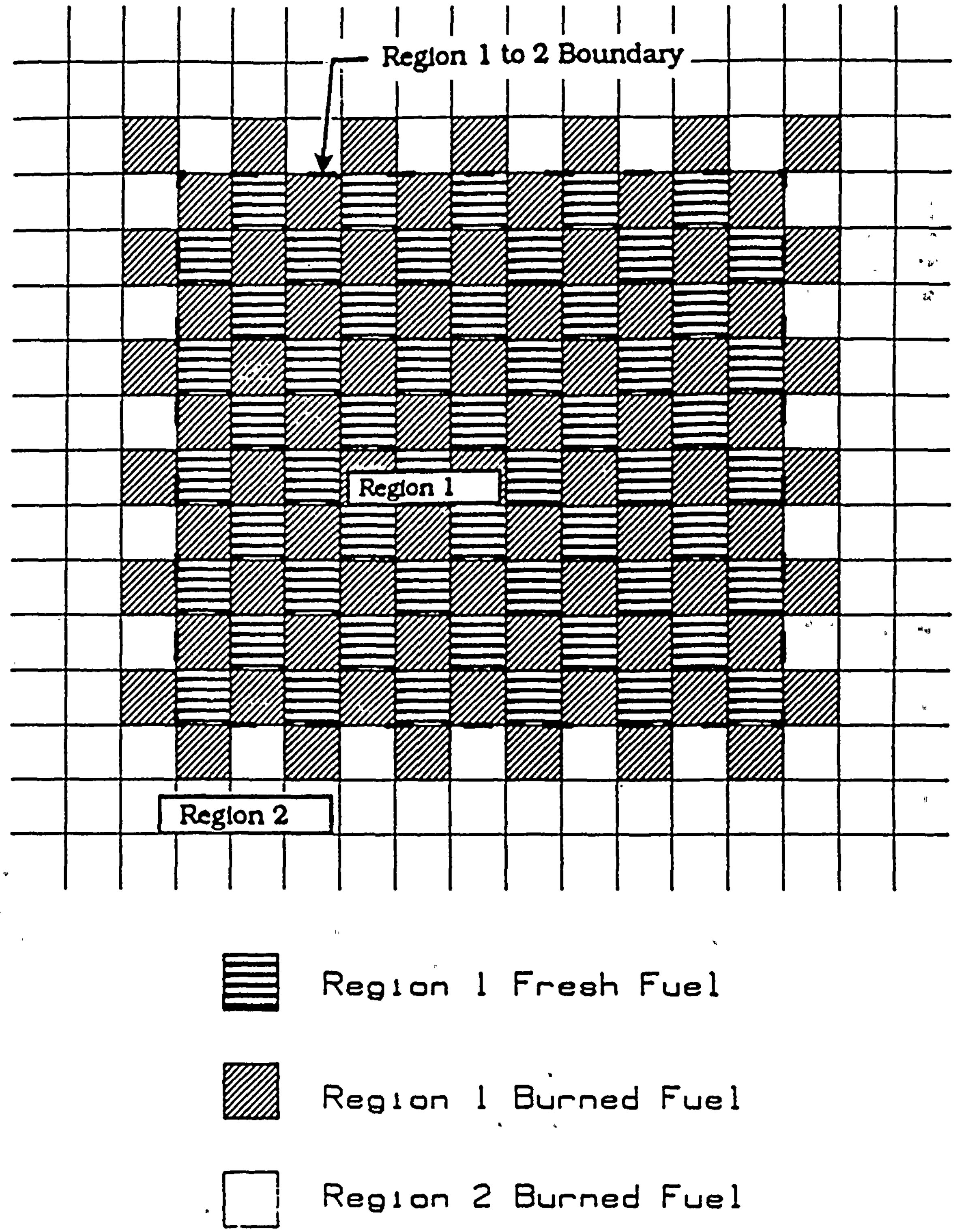



**Figure 3.** Donald C. Cook Nuclear Plant Schematic for SFP Interface Boundary Between Regions 1 and 2

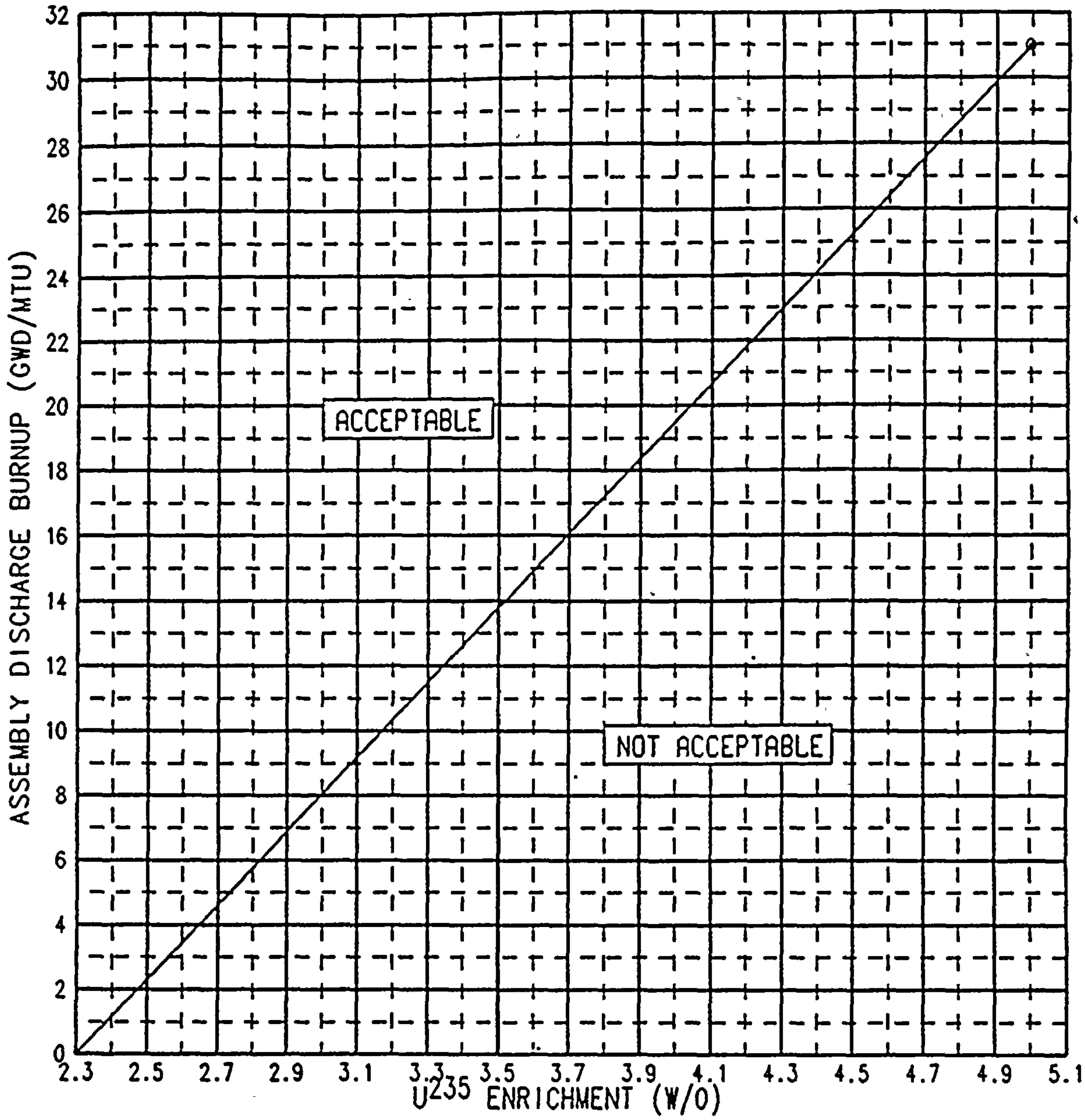
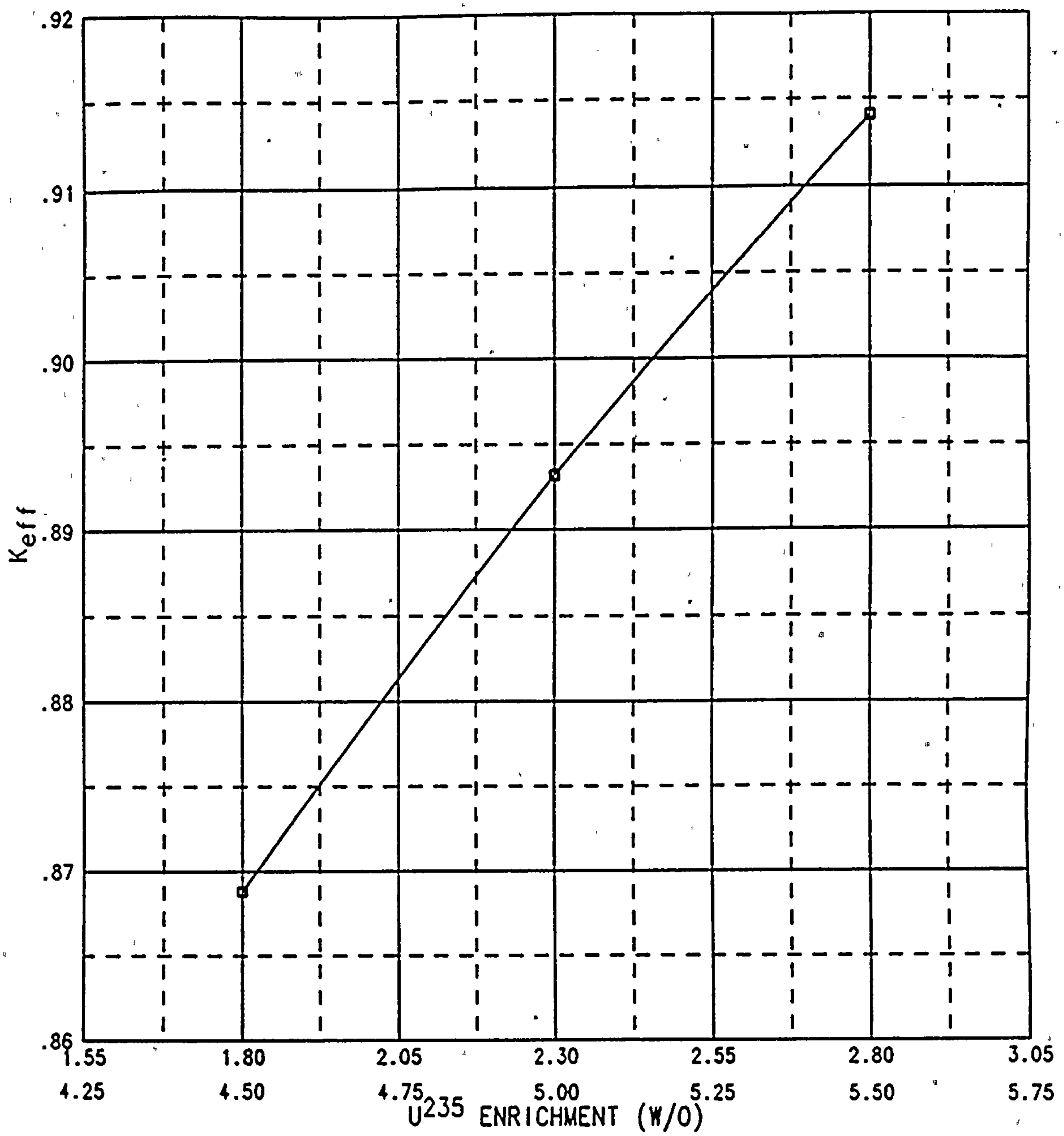
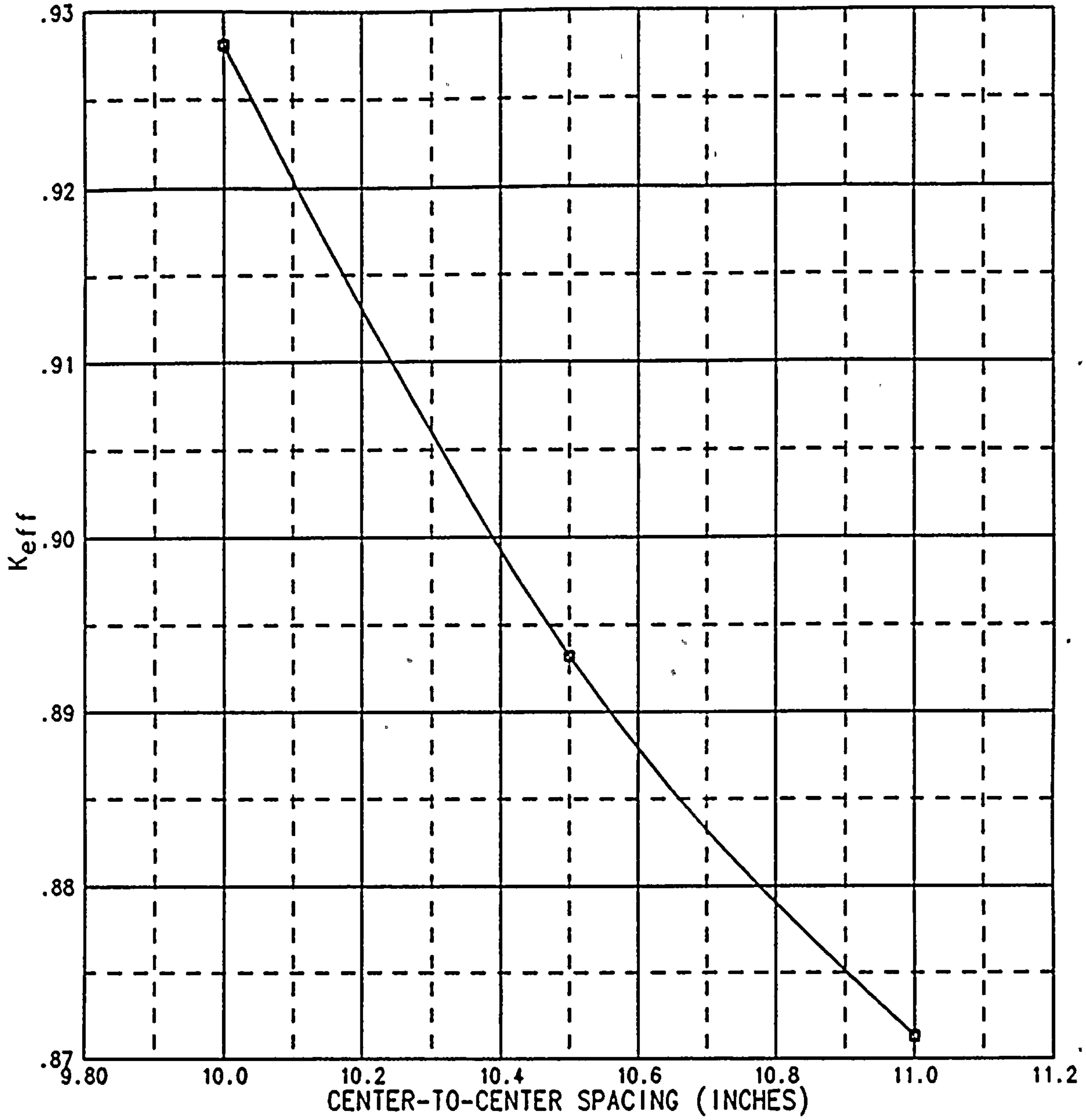



**Figure 4. Donald C. Cook Nuclear Plant SFP Region 1 "Burned" Fuel Assembly Minimum Burnup vs. Initial $U^{235}$ Enrichment Curve**



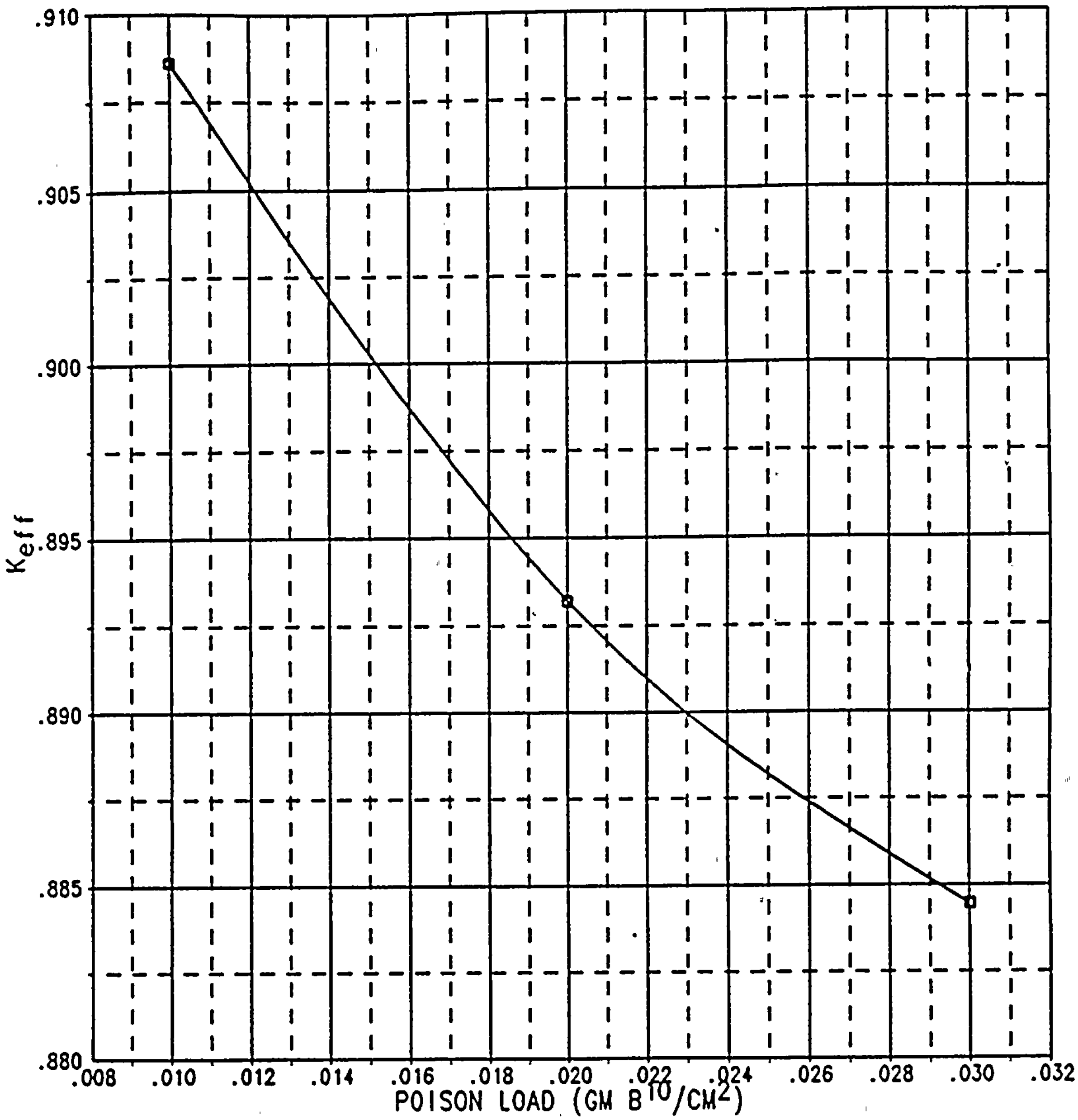

BORAL HELD AT .02 GM $B^{10}/CM^2$
CENTER TO CENTER HELD AT 10.50"

**Figure 5. Sensitivity of $k_{eff}$ to Enrichment in the Donald C. Cook Nuclear Plant SFP Region 1 Storage Area with Checkerboard Loading**




BORAL HELD AT .02 GM $B^{10}/CM^2$
CHECKERBOARD ENRICHMENT HELD AT 2.3/5.0 W/0

**Figure 6. Sensitivity of $k_{eff}$ to Center-to-Center Spacing in the Donald C. Cook Nuclear Plant SFP Region 1 Storage Area with Checkerboard Loading**




CENTER TO CENTER HELD AT 10.50"
CHECKERBOARD ENRICHMENT HELD AT 2.3/5.0 W/O

**Figure 7. Sensitivity of $k_{eff}$ to $B^{10}$ Loading in the Donald C. Cook Nuclear Plant SFP Region 1 Storage Area with Checkerboard Loading**

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ATTACHMENT 2 TO AEP:NRC:1071N

DESCRIPTION OF PROPOSED T/Ss CHANGES

AND 10 CFR 50.92 SIGNIFICANT

HAZARDS CONSIDERATION ANALYSIS

## Introduction

This letter requests T/Ss changes in the storage pattern in the spent fuel pool. The changes involve Section 5 (Design Section) and are based on criticality and thermal-hydraulic concerns. As discussed later in this attachment, analyses have been performed that demonstrate the acceptability of the proposed T/Ss changes with regard to the criticality concerns. We have also concluded that the current analysis of record for the thermal-hydraulic concerns remains bounding.

## Description of Changes

The proposed T/Ss changes are as follows:

1) In Units 1 and 2 T/S 5.6.1.1, subparagraph c.1 has been modified, the existing subparagraph c.2 has been slightly modified and renumbered as c.3, and a new subparagraph c.2 has been added.

2) Subparagraph c.1 identifies "Region 1" of the spent fuel storage racks for storage of Westinghouse fuel types with maximum nominal fuel assembly enrichments of 3.95 weight percent or greater. Fuel stored in Region 1 must be stored in a checkerboard configuration alternating Westinghouse Category 1 and Category 2 fuel, as shown in Figure 5.6-1 (see Attachment 1, Figure 3). Category 1 fuel is defined as Westinghouse fuel shown in Figure 5.6-2 as "acceptable for storage as Category 1 fuel." Category 2 fuel is defined as Westinghouse fuel with an enrichment greater than 3.95 weight percent U-235 and a burnup less than 5,550 MWD/MTU shown on the shaded area of Figure 5.6-2. These are the maximum enrichments and minimum burnups for Westinghouse fuel that can be stored in Region 2, as discussed in the next paragraph.

3) The new subparagraph c.2 identifies "Region 2" of the spent fuel storage racks. This Region 2 shall be for the storage of Westinghouse fuel with an enrichment less than or equal to 3.95 weight percent U-235 or with an enrichment greater than 3.95 weight percent U-235 but with a burnup greater than or equal to 5,550 MWD/MTU, and Exxon/ANF fuel of an enrichment less than or equal to 4.23 weight percent U-235 for a 17 x 17 assembly or less than or equal to 3.50 weight percent U-235 for a 15 x 15 assembly. The maximum enrichment and minimum burnup limits for Westinghouse fuel that can be stored in Region 2 were justified in our previous submittal AEP:NRC:1071F.

4) The existing subparagraph c.2 is renumbered as c.3 and is being modified slightly to address the boundary condition between Regions 1 and 2. This boundary condition is that the checkerboard pattern requirement for Region 1 must be carried into Region 2 by at least one row. This boundary condition is illustrated in the revised figure, Figure 5.6-1.

5) The second page of Table 5.7-1 in the Unit 2 T/Ss is incorrectly labeled as Table 5.9-1. This has been revised. Being purely editorial, this change is not discussed in the significant hazards consideration portion of this attachment.

Summary of Criticality Analyses

Attachment 1 contains criticality analyses for Region 1 of the spent fuel pool and for the boundary between Regions 1 and 2, prepared for us by Westinghouse. The analyses were performed in such a way that they bound all types of Westinghouse fuel currently in use or planned for use in the Cook Nuclear Plant.

In accordance with the acceptance criteria of Chapter 9.1.2 (Spent Fuel Storage) of the NRC Standard Review Plan, the Westinghouse analyses demonstrate that the center-to-center spacing between fuel assemblies and any strong, fixed neutron absorbers in the storage racks are sufficient to maintain the array, when loaded with fuel assemblies with a maximum nominal enrichment of 4.95 weight percent U-235 and flooded with pure water, in a subcritical condition with $k_{eff}$ less than 0.95. In order to achieve acceptable results, Westinghouse has determined that fresh fuel assemblies with nominal enrichments above 3.95 weight percent U-235 must be stored in Region 1 of the spent fuel storage racks in a checkerboard pattern configuration alternating Category 1 and Category 2 fuel. In addition, this checkerboard pattern of Region 1 must be carried into Region 2 by at least one row.

It should also be pointed out that one fuel assembly, which has an enrichment larger than 4.55 weight percent U-235, would have a $k_{eff}$ larger than 0.95 when out of the poisoned racks and with 0 ppm of boron in the spent fuel pool. However, it can be shown that with only 1000 ppm of boron, the $k_{eff}$ of a fuel assembly with an enrichment of 4.95 weight percent U-235 would be conservatively less than 0.95. The boron concentration in the spent fuel pool is required by T/S 3.9.15 to be greater than 2400 ppm by measurement when fuel assemblies with enrichment greater than 3.95 weight percent U-235 and with burnup less than 5,550 MWD/MTU are in the fuel storage pool.

Thermal-Hydraulic Considerations

The Cook Nuclear Plant Updated FSAR discusses analyses performed to demonstrate the adequacy of the spent fuel pool cooling system to remove decay heat, and also an assessment of the time it would take to reach bulk boiling in the event all spent fuel pool cooling is lost. The most recent analysis of this type was performed by ANF in support of using assemblies capable of 50,000 MWD/MTU exposure. The analyses were documented in ANF Report No. ANF-88-09. This analysis was submitted in our letter AEP:NRC:1071 dated August 19, 1988, and supported the Amendments 118 (Unit 1) and 104 (Unit 2) T/Ss changes. (The analyses were modified slightly as documented in Rev. 1 to ANF-88-09 to correct an error in determining the pool heat load. The revision resulted in a change in the time to reach bulk boiling from 8.6 hours to 5.5 hours.) The Unit 2 Vantage 5 assemblies are intended for use up to an average discharge burnup of only 48,000 MWD/MTU. Also, the Unit 2 power level remains the same at 3411 MWt. Therefore, spent fuel pool cooling analyses documented in ANF-88-09, Rev. 1, will bound the Vantage 5 assemblies and no new analyses are being submitted.

10 CFR 50.92 Significant Hazards Consideration Analysis

Per 10 CFR 50.92, a proposed amendment to an operating license will not involve a significant hazards consideration if the proposed amendment satisfies the following three criteria:

1) Does not involve a significant increase in the probability or consequences of an accident previously analyzed,

2) Does not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated, or

3) Does not involve a significant reduction in a margin of safety.

Criterion 1

Westinghouse has performed analyses that demonstrate the acceptability of the proposed changes with regard to criticality. The analyses demonstrate that fuel stored in the spent fuel pool will remain subcritical under design basis conditions. However, accidents or incidents can take place which would increase reactivity such as dropping a fuel assembly between the rack and pool wall or inadvertently placing a fuel assembly in the wrong location. For those conditions, the double contingency principle of ANSI N16.1-1975 can be applied. That principle states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against criticality. Thus, the presence of

greater than or equal to 2400 ppm of soluble boron in the spent fuel pool can be assumed as a realistic initial condition, since not assuming it would be a second unlikely event. The reactivity of the fuel stored in the spent fuel pool would be decreased by about 0.25 delta-k, with approximately 2000 ppm of boron; that is, for an accident or an incident resulting in an increase in reactivity, $k_{eff}$ would remain less than or equal to 0.95 due to the effect of the dissolved boron. In addition, paragraph 2.3 of the SER related to Amendments 118 and 104 for Cook Nuclear Plant Units 1 and 2, respectively, states that "the reactivity reduction due to the required pool boration of 2400 ppm of boron more than offsets the potential reactivity increases from postulated fuel mishandling accidents." It is concluded that the proposed T/Ss changes should not involve a significant increase in the probability or consequences of a previously analyzed accident.

Criterion 2

The Westinghouse analyses demonstrate continued acceptability of the spent fuel pool regarding criticality. The T/Ss changes will not result in physical changes to the plant (other than to the fuel assemblies, which were the subject of the Westinghouse analyses). Therefore, we believe the proposed T/Ss changes will not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3

Westinghouse has performed analyses that demonstrate the acceptability of the proposed changes with regard to criticality. The analyses demonstrate that the fuel stored in the spent fuel pool will remain subcritical under design basis conditions. However, accidents or incidents can take place which would increase reactivity such as dropping a fuel assembly between the rack and pool wall or inadvertently placing a fuel assembly in the wrong location. For those conditions, the double contingency principle of ANSI N16.1-1975 can be applied. That principle states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against criticality. Thus, the presence of greater than or equal to 2400 ppm of soluble boron in the spent fuel pool can be assumed as a realistic initial condition, since not assuming it would be a second unlikely event. The reactivity of the fuel stored in the spent fuel pool would be decreased by about 0.25 delta-k, with approximately 2000 ppm of boron; that is, for an accident or an incident resulting in an increase in reactivity, $k_{eff}$ would remain less than or equal to 0.95 due to the effect of the dissolved boron. In addition, paragraph 2.3 of the SER related to Amendment 118 and 104 for Cook Nuclear Plant Units 1 and 2, respectively, states that "the reactivity reduction due to the required pool boration of 2400 ppm of boron more than offsets the

potential reactivity increases from postulated fuel mishandling accidents." It is concluded that the proposed T/Ss changes should not involve a significant reduction in a margin of safety.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase in the probability of occurrence or in the consequences of a previously analyzed accident, but the results of which are within limits established as acceptable. The Westinghouse analyses demonstrate acceptable results from a criticality perspective using the acceptance criteria of the NRC Standard Review Plan. Therefore, we believe the example cited is applicable and that the proposed T/Ss changes do not involve a significant hazards consideration as defined in 10 CFR 50.92.

ATTACHMENT 3 TO AEP:NRC:1071N

PROPOSED, REVISED TECHNICAL SPECIFICATIONS PAGES

FOR DONALD C. COOK NUCLEAR PLANT

UNITS 1 AND 2

## DESIGN FEATURES

a. In accordance with the code requirements specified in Section 4.1.6 of FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

## VOLUME

5.4.2 The total contained volume of the reactor coolant system is 12,612 ± 100 cubic feet at a nominal $T_{avg}$ of 70°F.

## 5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

## 5.6 FUEL STORAGE

## CRITICALITY - SPENT FUEL

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

a. A $k_{eff}$ equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the storage racks.

c. 1. A separate region within the spent fuel storage racks (defined as Region 1) shall be established for storage of Westinghouse fuel types of enrichments greater than 3.95 weight percent U-235 and burnup less than 5,550 MWD/MTU in a checkerboard pattern configuration alternating Category 1 and Category 2 fuel as shown in Figure 5.6-1.

   Westinghouse Category 1 and Category 2 fuel definitions are given in Figure 5.6-2.

DESIGN FEATURES (cont'd)

## 5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL (cont'd)

2. A separate region within the spent fuel storage racks, defined as Region 2, shall be established for storage of Westinghouse fuel of an enrichment less than or equal to 3.95 weight percent U-235 or an enrichment greater than 3.95 weight percent U-235 but with a burnup greater than or equal to 5,550 MWD/MTU, and Exxon/ANF fuel of an enrichment less than or equal to 4.23 weight percent U-235 for a 17 x 17 assembly or less than or equal to 3.50 weight percent U-235 for a 15 x 15 assembly.

3. The boundary between the Regions 1 and 2 mentioned above shall be such that the checkerboard pattern storage requirement of Region 1 shall be carried into Region 2 by at least one row as shown in Figure 5.6-1.

5.6.1.2 Fuel stored in the spent fuel storage racks shall have a maximum nominal fuel assembly enrichment as follows:

| | Description | | Maximum Nominal Fuel Assembly Enrichment Wt. % $235_U$ |
|---|---|---|---|
| 1) | Westinghouse | 15 x 15 STD<br>15 x 15 OFA | 4.95 |
| 2) | Exxon/ANF | 15 x 15 | 3.50 |
| 3) | Westinghouse | 17 x 17 STD<br>17 x 17 OFA<br>17 x 17 V5 | 4.95 |
| 4) | Exxon/ANF | 17 x 17 | 4.23 |

CRITICALITY-NEW FUEL

5.6.2.1 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that $K_{eff}$ will not exceed 0.98 when fuel assemblies are placed in the pit and aqueous foam moderation is assumed.

5.6.2.2 Fuel stored in the new fuel storage racks shall have a maximum nominal fuel assembly enrichment as follows:

| Description | | Maximum Nominal Fuel Assembly Enrichment Wt. % $^{235}U$ |
|---|---|---|
| 1) Westinghouse | 15 x 15 STD<br>15 x 15 OFA | 4.55 |
| 2) Exxon/ANF | 15 x 15 | 3.50 |
| 3) Westinghouse | 17 x 17 STD<br>17 x 17 OFA<br>17 x 17 V5 | 4.55 |
| 4) Exxon/ANF | 17 x 17 | 4.23 |

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 629' 4".

CAPACITY

5.6.4 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2050 fuel assemblies.

## 5.7 SEISMIC CLASSIFICATION

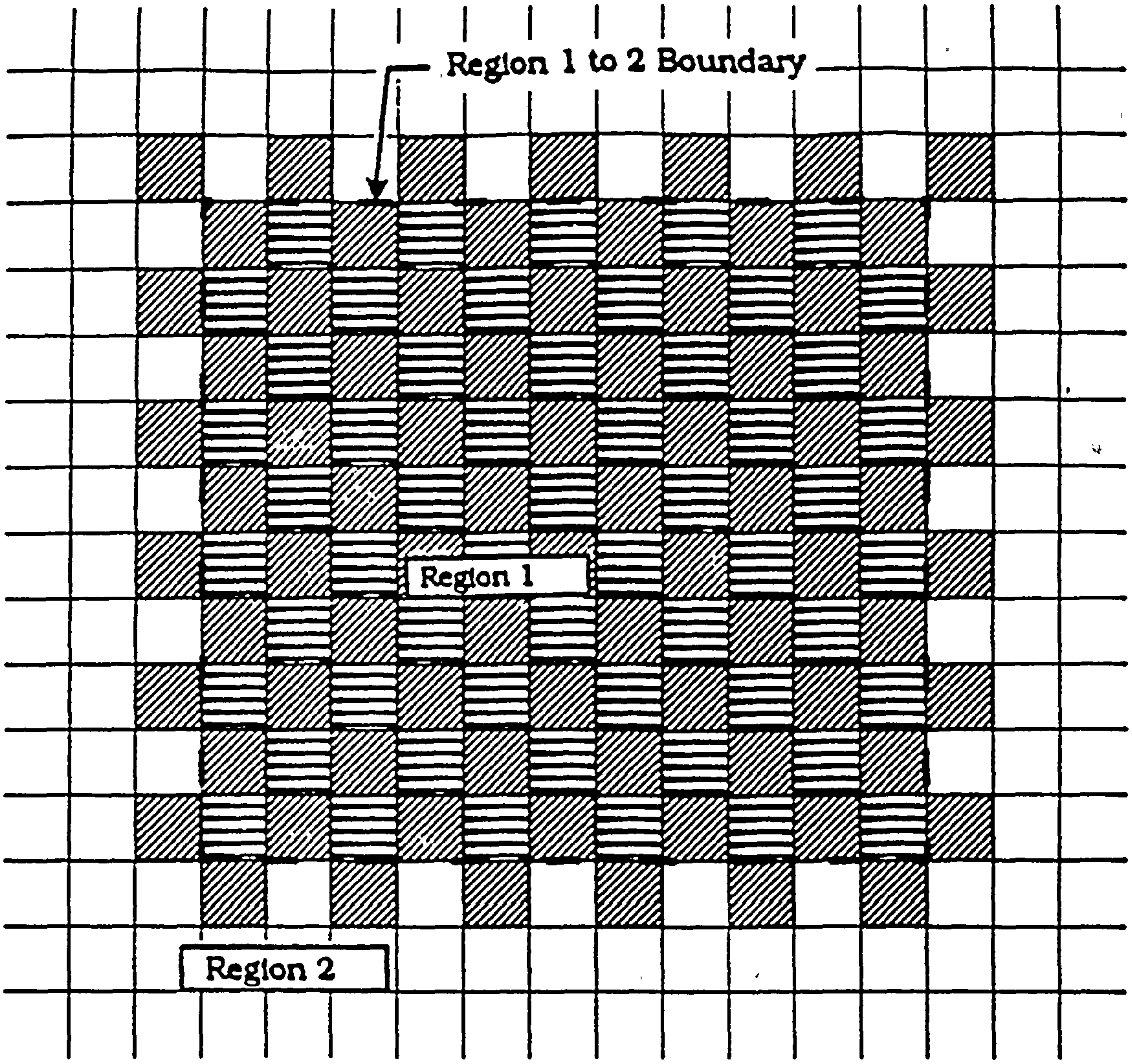
5.7.1 Those structures, systems and components identified as Category I items in the FSAR shall be designed and maintained to the original design provisions contained in the FSAR with allowance for normal degradation pursuant to the applicant Surveillance Requirements.

## 5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower shall be located as shown in Figure 5.1.1.

## 5.9 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.9.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.




REGION 1, CATEGORY 2 FUEL

REGION 1, CATEGORY 1 FUEL

REGION 2 FUEL

FIGURE 5.6-1

DONALD C. COOK NUCLEAR PLANT
SCHEMATIC FOR SFP INTERFACE BOUNDARY BETWEEN REGIONS 1 AND 2

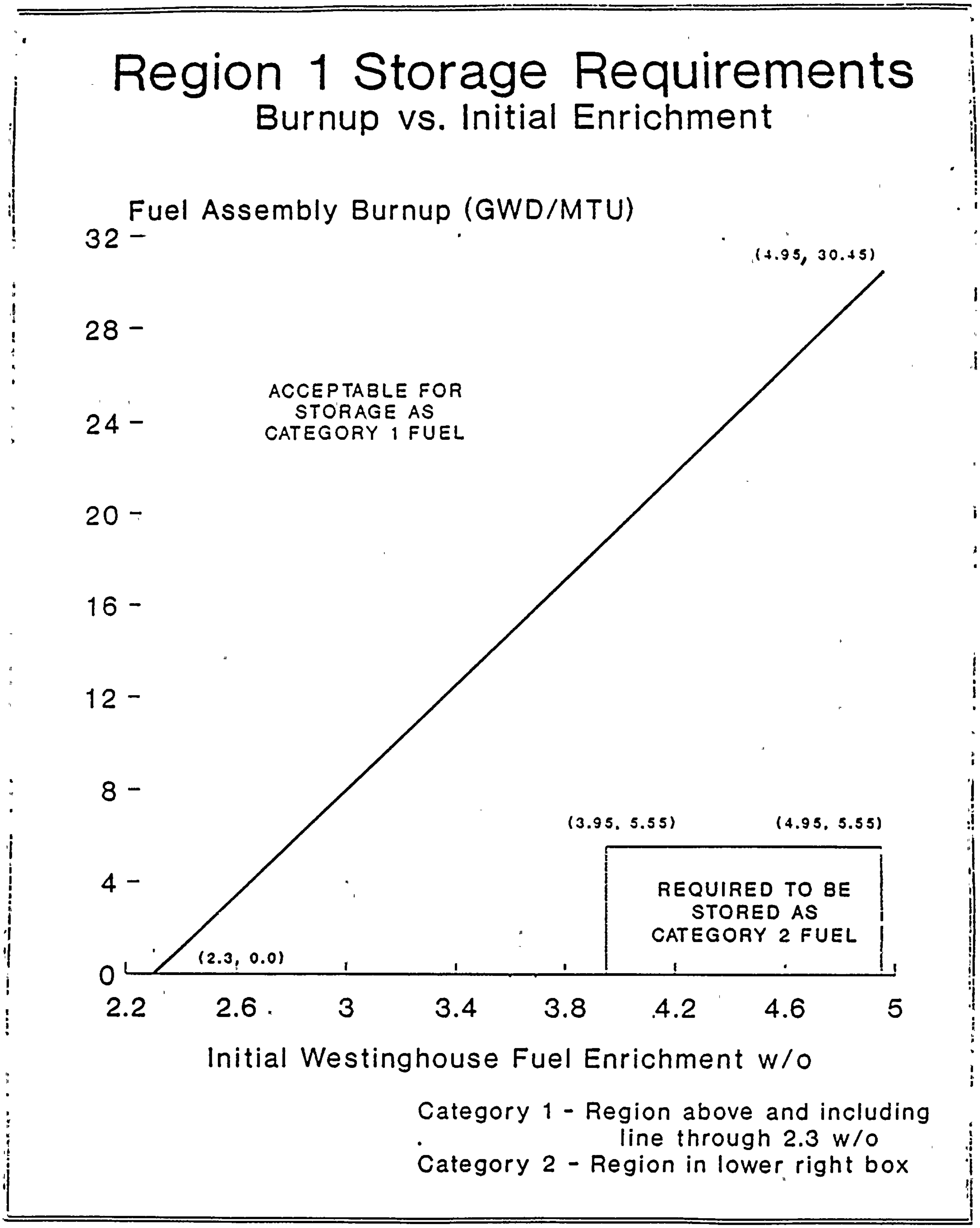



FIGURE 5.6-2

DONALD C. COOK NUCLEAR PLANT
SFP REGION 1 BURNED FUEL ASSEMBLY MINIMUM BURNUP VS. INITIAL U-235 ENRICHMENT CURVE

TABLE 5.9-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 200 heatup cycles at less than or equal to 100°F/hr and 200 cooldown cycles at less than or equal to 100°F/hr (pressurizer cooldown at less than or equal to 200°F/hr). | Heatup cycle - $T_{avg}$ from less than or equal to 200°F to greater than or equal to 547°F. Cooldown cycle - $T_{avg}$ from greater than or equal to 547°F to less than or equal to 200°F. |
| | 80 loss of load cycles. | Without immediate turbine or reactor trip. |
| | 40 cycles of loss of offsite A.C. electrical power. | Loss of offsite A.C. electrical power source supplying the onsite Class 1E distribution system. |
| | 80 cycles of loss of flow in one reactor coolant loop. | Loss of only one reactor coolant pump. |
| | 400 reactor trip cycles. | 100% to 0% of RATED THERMAL POWER. |
| | 200 large step decreases in load. | 100% to 5% of RATED THERMAL POWER with steam dump. |

## TABLE 5.9-1

## COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 1 main reactor coolant pipe break. | Break in a reactor coolant pipe greater than 6 inches equivalent diameter. |
| | Operating Basis Earthquakes | 400 cycles - 20 earthquakes of 20 cycles each. |
| | 50 leak tests. | Pressurized to 2500 psia. |
| | 5 hydrostatic pressure tests | Pressurized to 3107 psig. |
| Secondary System | 1 steam line break | Break in a steam line greater than 5.5 inches equivalent diameter. |
| | 5 hydrostatic pressure tests | Pressurized to 1356 psig. |

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,612 $\pm$ 100 cubic feet as a nominal Tavg of 70$^{o}$F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL

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

a. A $K_{eff}$ equivalent to less than 0.95 when flooded with unborated water,

b. A nominal 10.5-inch center-to-center distance between fuel assemblies, placed in the storage racks.

c. 1. A separate region within the spent fuel storage racks (defined as Region 1) shall be established for storage of Westinghouse fuel types of enrichment greater than 3.95 weight percent U-235 and burnup less than 5,550 MWD/MTU in a checkerboard pattern configuration alternating Category 1 fuel and Category 2 fuel as shown in Figure 5.6-1.

   Westinghouse Category 1 and Category 2 fuel definitions are given in Figure 5.6-2.

   2. A separate region within the spent fuel storage racks, defined as Region 2, shall be established for storage of Westinghouse fuel of an enrichment less than or equal to 3.95 weight percent U-235 or an enrichment greater than 3.95 weight percent U-235 but with a burnup greater than or equal to 5,550 MWD/MTU, and Exxon/ANF fuel of an enrichment less than or equal to 4.23 weight percent U-235 for a 17 x 17 assembly or less than or equal to 3.50 weight percent U-235 for a 15 x 15 assembly.

   3. The boundary between the Regions 1 and 2 mentioned above shall be such that the checkerboard pattern storage requirement of Region 1 shall be carried into Region 2 by at least one row as shown in Figure 5.6-1.

5.6 FUEL STORAGE

CRITICALITY - SPENT FUEL (cont'd)

5.6.1.2 Fuel stored in the spent fuel storage racks shall have a maximum nominal fuel assembly enrichment as follows:

| Description | | | Maximum Nominal Fuel Assembly Enrichment Wt. % $^{235}U$ |
|---|---|---|---|
| 1) | Westinghouse | 15 x 15 STD<br>15 x 15 OFA | 4.95 |
| 2) | Exxon/ANF | 15 x 15 | 3.50 |
| 3) | Westinghouse | 17 x 17 STD<br>17 x 17 OFA<br>17 x 17 V5 | 4.95 |
| 4) | Exxon/ANF | 17 x 17 | 4.23 |

CRITICALITY - NEW FUEL

5.6.2.1 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that $K_{eff}$ will not exceed 0.98 when fuel assemblies are placed in the pit and aqueous foam moderation is assumed.

5.6.2.2 Fuel stored in the new fuel storage racks shall have a maximum nominal fuel assembly enrichment as follows:

| Description | | | Maximum Nominal Fuel Assembly Enrichment Wt. % $^{235}U$ |
|---|---|---|---|
| 1) | Westinghouse | 15 x 15 STD<br>15 x 15 OFA | 4.95 |
| 2) | Exxon/ANF | 15 x 15 | 3.50 |
| 3) | Westinghouse | 17 x 17 STD<br>17 x 17 OFA<br>17 x 17 V5 | 4.95 |
| 4) | Exxon/ANF | 17 x 17 | 4.23 |

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 629'4".

CAPACITY

5.6.4 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2050 fuel assemblies.

## 5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

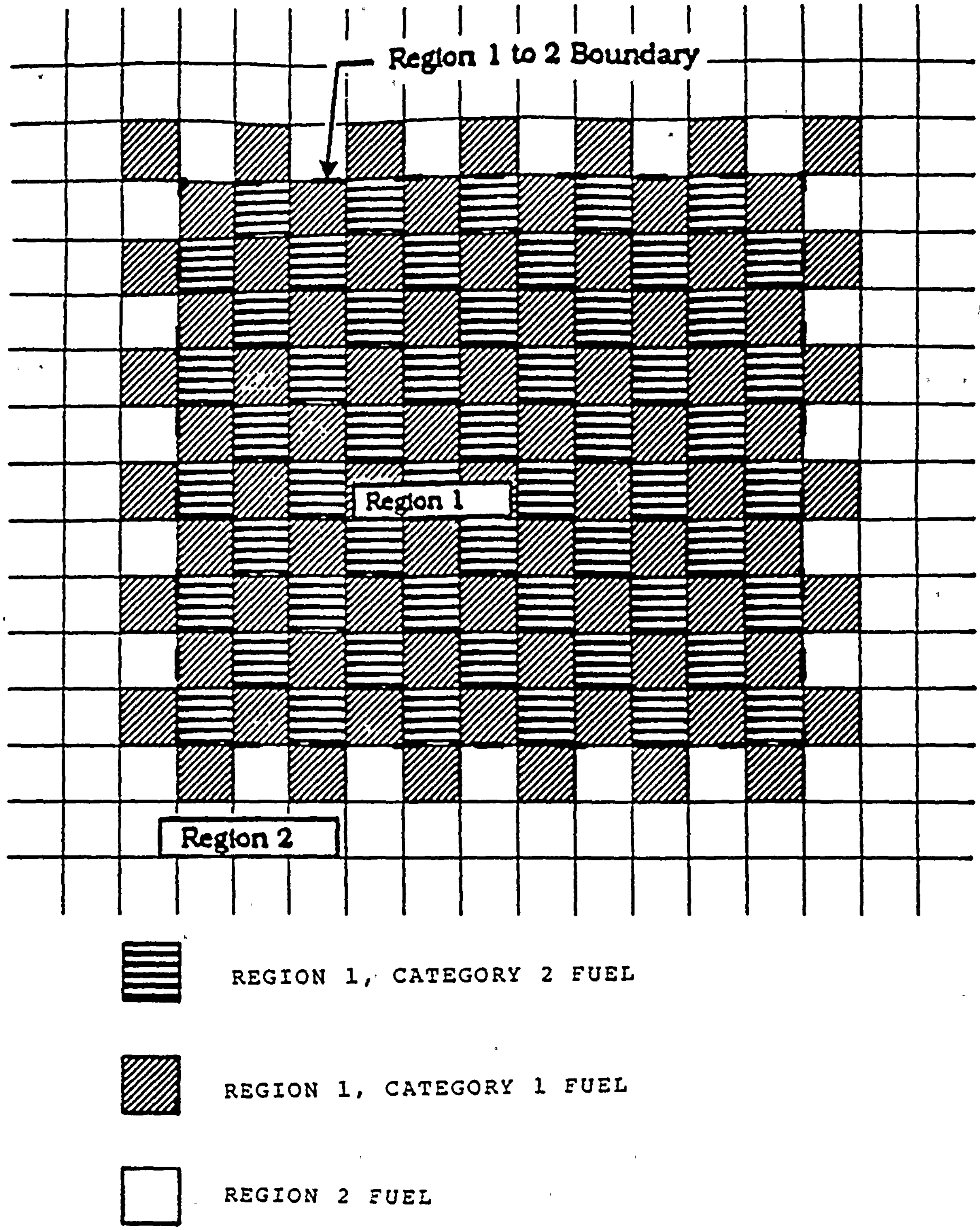



FIGURE 5.6-1
DONALD C. COOK NUCLEAR PLANT
SCHEMATIC FOR SFP INTERFACE BOUNDARY BETWEEN REGIONS 1 AND 2

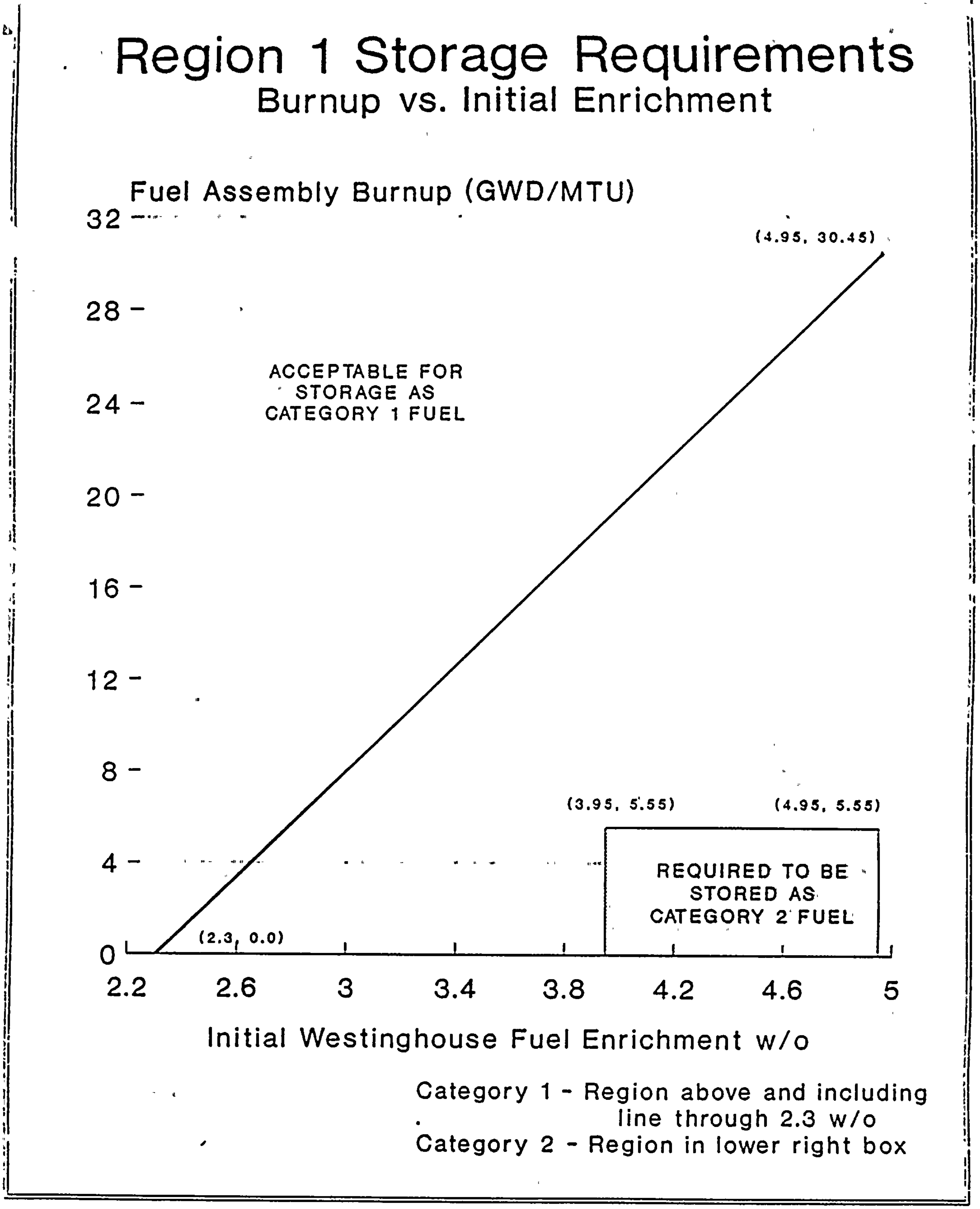



FIGURE 5.6-2

DONALD C. COOK NUCLEAR PLANT
SFP REGION 1 BURNED FUEL ASSEMBLY MINIMUM BURNUP VS. INITIAL U-235 ENRICHMENT CURVE

# TABLE 5.7-1

## COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 200 heatup cycles at less than or equal to 100°F/hr and 200 cooldown cycles at less than or equal to 100°F/hr (pressurizer cooldown at less than or equal to 200°F/hr). | Heatup cycle - $T_{avg}$ from less than or equal to 200°F to greater than or equal to 547°F. Cooldown cycle - $T_{avg}$ from greater than or equal to 547°F to less than or equal to 200°F. |
| | 80 loss of load cycles. | Without immediate turbine or reactor trip. |
| | 40 cycles of loss of offsite A.C. electrical power. | Loss of offsite A.C. electrical power source supplying the onsite Class 1E distribution system. |
| | 80 cycles of loss of flow in one reactor coolant loop. | Loss of only one reactor coolant pump. |
| | 400 reactor trip cycles. | 100% to 0% of RATED THERMAL POWER. |
| | 200 large step decreases in load. | 100% to 5% of RATED THERMAL POWER with steam dump. |

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

| COMPONENT | CYCLIC OR TRANSIENT LIMIT | DESIGN CYCLE OR TRANSIENT |
|---|---|---|
| Reactor Coolant System | 1 main reactor coolant pipe break. | Break in a reactor coolant pipe greater than 6 inches equivalent diameter. |
| | Operating Basis Earthquakes | 400 cycles - 20 earthquakes of 20 cycles each. |
| | 50 leak tests. | Pressurized to 2500 psia. |
| | 5 hydrostatic pressure tests | Pressurized to 3107 psig. |
| Secondary System | 1 steam line break | Break in a steam line greater than 5.5 inches equivalent diameter. |
| | 5 hydrostatic pressure tests | Pressurized to 1356 psig. |