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Westinghouse Spent Fuel Rack Criticality Analysis Methodology

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ABSTRACT

The purpose of this topical report is to document the Westinghouse Spent Fuel Rack Criticality Methodology that ensures the spent fuel rack multiplication factor, K_{eff}, is less than 0.95 as recommended by ANSI 57.2-1983 and NRC guidance. Explained within the document are the codes, methods and techniques used to satisfy this criterion on K_{eff}. Also presented in this document is the procedure for calculating K_{eff} with credit for spent fuel pool soluble boron.

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1.0 Introduction

The purpose of this report is to document the Westinghouse Spent Fuel Rack Criticality Methodology that ensures the spent fuel rack multiplication factor, K_{eff}, is less than 0.95 as recommended by ANSI 57.2-1983[1] and NRC guidance[2]. The individual sections of this report demonstrate the codes, methods and techniques used to satisfy this criterion on K_{eff}.

Section 2.0, Computer Code Methods and Benchmarking, explains the computer codes used in the evaluation of the spent fuel rack K_{eff} calculations. The methodology of the NITAWL-II, XSDRNPM-S, and KENO-Va codes is discussed and benchmark results are presented to establish a methodology bias and bias uncertainty. The PHOENIX-P computer code is also discussed in this section. PHOENIX-P is a nuclear design code used primarily for core reactor physics calculations but maintains the capability to simulate spent fuel storage rack geometries. The benchmarking of PHOENIX-P is discussed here.

In Section 3.0, Spent Fuel Rack Criticality Calculations, the maximum fresh fuel assembly enrichments that can be stored in the spent fuel racks are determined. The details on the assumptions made to model the spent fuel storage racks and the use of the results are also presented. Specific details are presented on KENO-Va calculations, PHOENIX-P tolerance calculations and the final 95/95 K_{eff} determination.

To allow higher enrichments than those determined in the previous section, Section 4.0, Reactivity Equivalencing Methodology, discusses the techniques used to allow higher fuel assembly enrichments to be stored in the spent fuel storage racks by taking credit for fuel assembly burnup and Integral Fuel Burnable Absorbers (IFBA). This section defines the concept of reactivity equivalencing and discusses the assumptions and uncertainties associated with each reactivity equivalencing technique. The use of PHOENIX-P in each reactivity equivalencing methodology is discussed.

To completely cover possible off-normal conditions in the spent fuel storage racks, Section 5.0, Postulated Accident Methodology, defines the postulated spent fuel rack accidents which are considered in the spent fuel rack criticality analysis. The methodology used to determine the reactivity impact of these accidents is discussed. Finally, the application of the double contingency principle to these spent fuel rack postulated accidents is presented which allows credit for spent fuel pool soluble boron to offset the potential reactivity increase caused by these off-normal conditions.

Finally, Section 6.0, Soluble Boron Credit Methodology, defines how the three previous sections are applied when credit for spent fuel pool soluble boron is used under normal storage configuration conditions. The normal storage configuration is defined using the maximum feasible K_{eff} calculation to ensure that the spent fuel rack K_{eff} will be less than 1.0 with no soluble boron under normal storage conditions. Soluble boron credit is then used to offset the uncertainties and tolerances and maintain K_{eff} less than or equal to 0.95 as explained. The use of

soluble boron credit for reactivity equivalencing uncertainties is discussed. The calculation of postulated accidents crediting soluble boron is discussed. Finally, a summary of all the soluble boron credit requirements is presented.

2.0 Computer Code Methods and Benchmarking

2.1 NITAWL, XSDRN, KENO-Va Benchmarking

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 spent fuel storage rack conditions which include strong neutron absorbers, large water gaps and soluble boron in the moderator.

The design method which insures the criticality safety of the spent fuel storage racks uses NITAWL-II[3] and XSDRNPM-S[4] for cross-section generation and KENO-Va[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 of KENO-Va and the KENO-Va storage rack calculations is generated from ENDF/B-V[6] data. The NITAWL-II 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-S program which is a one-dimensional S_n transport theory code. These multigroup cross-section sets are then used as input to KENO-Va which is a three dimensional Monte Carlo theory program designed for reactivity calculations.

KENO-Va Monte Carlo calculations are always performed with sufficient neutron histories to assure convergence. A typical KENO-Va Monte Carlo calculation involves more than 100,000 neutron histories which is significantly more than the default of 30,000. To assure adequate convergence, the KENO-Va edits which show Average K_{eff} per Generation Run and Average K_{eff} by Generation Skipped are examined. These edits provide a visual inspection on the overall convergence of the KENO-Va Monte Carlo results.

A set of 44 critical experiments[7,8,9,10,11] has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty. The benchmark experiments cover a wide range of geometries, materials, and enrichments, ranging from relatively low enriched (2.35, 2.46, and 4.31 w/o), water moderated, oxide fuel arrays separated by various materials (B_4C, aluminum, steel, water, etc.) that simulate LWR fuel shipping and storage conditions to dry, harder spectrum, uranium metal cylinder arrays at high enrichments (93.2 w/o) with various interspersed materials (Plexiglass and air). Comparison with these experiments demonstrates the wide range of applicability of the method. Table 1 on page 21 summarizes these experiments.

The highly enriched benchmarks show that the criticality code sequence can correctly predict the reactivity of a hard spectrum environment, such as the optimum moderation condition often considered in fresh rack and shipping cask analyses. However, the results of the 12 highly enriched benchmarks are not incorporated into the criticality method bias because the

enrichments are well above any encountered in commercial nuclear power applications. Basing the method bias solely on the 32 low enriched benchmarks results in a more appropriate and more conservative bias.

The 32 low enriched, water moderated experiments result in an average KENO-Va K_{eff} of [ ]$^{(a,c)}$. Comparison with the average measured experimental K_{eff} of 1.0007 results in a method bias of [ ]$^{(a,c)}$. The standard deviation of the bias value is [ ]$^{(a,c)}$. The 95/95 one-sided tolerance limit factor for 32 values is 2.20[12]. 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 [ ]$^{(a,c)}$.

2.2 PHOENIX-P Benchmarking

The transport theory computer code, PHOENIX-P[13] is used to determine the reactivity changes caused by changes in the fuel assembly and spent fuel racks (tolerances) and the spent fuel pool conditions (temperature and soluble boron). PHOENIX-P is a depletable, two-dimensional, multigroup, discrete ordinates, transport theory code which utilizes a 42 energy group nuclear data library.

PHOENIX-P has been used to demonstrate its predictive capability in a series of comparisons against direct experimental data from critical experiments and against isotopic measurements. These comparisons provide a good assessment of the code's ability to predict key physics parameters over a wide range of lattice variations. Reactivity comparisons are accomplished by comparing appropriate predictions to Strawbridge and Barry's 101 criticals[14] and the Babcock and Wilcox (B&W) cores XI-1,2,7,8,9 and cores XIV-1,6 spatial criticals[15,16,17].

The range of lattice parameters of the 101 criticals are given in Table 2 on page 22. The resulting mean PHOENIX-P k_{eff} for all 101 criticals is [ ]$^{(a,c)}$ with a standard deviation of [ ]$^{(a,c)}$. This shows PHOENIX-P results to be in excellent agreement with experimental data for all dependent parameters, with no significant bias or trends as a function of lattice parameters.

The core loadings and compositions studied in the seven B&W core spatial criticals are given in Table 3 on page 23. The resulting mean K_{eff} of the seven critical experiments was [ ]$^{(a,c)}$ with a standard deviation of [ ]$^{(a,c)}$. The overall mean core K_{eff} for this set of critical experiments with diverse lattice configurations is in very good agreement with expected experimental values. The overall standard deviation indicates excellent stability of the PHOENIX-P library and methodology.

The data points calculated using the reactivity equivalencing methodology of Section 4.0 are generated with the transport theory computer code, PHOENIX-P. The validation of PHOENIX-P for use in generating fuel assembly burnup credit data points is discussed below.

The PHOENIX-P code has been validated by comparisons with experiments where the isotopic fuel composition has been examined following reactor shutdown. In addition, an extensive set of benchmark critical experiments has been analyzed with PHOENIX-P. Comparisons between

measured and predicted uranium and plutonium isotopic fuel compositions are shown in Table 4 on page 24. The measurements were made on fuel discharged from Yankee Core 5[18]. The PHOENIX-P predictions agree quite well with measurements for all measured isotopes throughout the burnup range.

3.0 Spent Fuel Rack Criticality Calculations

This section describes the analytical techniques and models employed to perform the criticality analysis calculations for spent fuel storage racks.

Section 3.1 describes the fresh fuel assembly reactivity calculations performed for the spent fuel storage racks using defined nominal enrichments, storage configuration and rack conditions. Section 3.2 describes the tolerance calculations used to determine the reactivity uncertainty associated with fuel assembly and storage rack tolerances. Finally, Section 3.3 discusses the final 95 percent probability with a 95 percent confidence interval (95/95) K_{eff} calculation performed to ensure K_{eff} is less than or equal to 0.95.

3.1 Reactivity Calculations using KENO-Va

To show that storage of fuel assemblies in the spent fuel storage racks satisfies the 0.95 K_{eff} criticality acceptance criteria, KENO-Va is used to establish a nominal reference reactivity using fresh fuel assemblies.

The following are the basic assumptions which are used to develop the nominal case KENO-Va model for the spent fuel storage rack calculation:

Spent Fuel Rack Storage Cell: The nominal spent fuel rack storage cell dimensions are used.

Fuel Assembly Types: The fuel assembly parameters relevant to the criticality analysis are listed in Table 5 on page 25. All fuel assembly types considered for storage in the spent fuel storage racks must be evaluated.

Fuel Rod Enrichment: The nominal fresh fuel enrichment modeled for each fuel pin is modeled. The pin locations within a fuel assembly with multiple enrichments will be considered, if applicable.

Fuel Pellet Density and Dishing Fraction: The nominal values for theoretical density and dishing fraction of the fuel pellets are modeled.

Axial Blankets: If axial blankets are modeled, the length and enrichment of the blanket fuel pellets are considered.

^{234}U and ^{236}U: No amount of ^{234}U or ^{236}U is modeled in the fuel pellet.

Spacer Grids or Sleeves: No amount of material from spacer grids or spacer sleeves is modeled in the fuel assembly.

Burnable Absorbers: No amount of burnable absorber poison material is modeled in the fuel assembly.

Fission Product Poisons: No amount of fission product poison material is modeled in the fuel assembly.

Moderator Temperature and Density: The moderator is pure water (no boron) at a temperature of 68°F and a density of 1.0 gm/cm^3.

Neutron Absorbing Poison Panels: If credit is taken for any fixed neutron absorbing poison panels present, they are modeled using the as-built or manufacturer-specified poison material loadings and dimensions.

Fuel Rack Storage Cell Configuration: If all storage cells are not loaded with the same fuel assembly type and enrichment, the specific storage configuration will be modeled. Different types of configurations includes checkerboard patterns, empty cell locations, specific pool configurations and other layouts as defined.

Using the above listed assumptions on the fuel assembly, spent fuel pool water conditions, and fuel rack storage cell loading configuration, a KENO-Va model will be developed using the nominal dimensions of the spent fuel rack and the K_{eff} of the storage racks loaded with fresh fuel assemblies will be calculated. The resulting nominal K_{eff} will be combined with the tolerance in the next section to develop the 95/95 K_{eff}.

3.2 Tolerance Calculations using PHOENIX-P

To evaluate the reactivity effects of possible variations in material characteristics and mechanical/construction dimensions, PHOENIX-P perturbation calculations are performed.

The reactivity effects of the following tolerances are considered in the spent fuel storage rack tolerance calculations using PHOENIX-P:

^{235}U Enrichment: The standard DOE enrichment tolerance of ±0.05 w/o ^{235}U about the nominal fresh reference enrichments is considered.

UO_2 Density: A ±2.0% variation about the nominal reference theoretical density is considered.

Fuel Pellet Dishing: A variation in fuel pellet dishing fraction from 0% to twice the nominal dishing is considered.

Storage Cell I.D.: The tolerance about the nominal reference cell inner diameter is considered.

Storage Cell Pitch: The tolerance about the nominal reference cell pitch is considered.

Cell Material Thicknesses: The tolerance about the nominal reference cell material thicknesses for all modeled rack structures is considered.

Neutron Absorber Panels Dimensions: The tolerances about the nominal width, length, and thickness of neutron absorber panels is considered.

Neutron Absorber Panels Poison Loading: The tolerances about the nominal poison loading of the neutron absorbing poison panels is considered if the nominal poison loading assumed in the KENO-Va model is not the minimum manufacturer specified loading.

Assembly Position: The KENO-Va reference reactivity calculation assumes fuel assemblies are symmetrically positioned within the storage cells. Calculations are performed to determine any reactivity increase caused by positioning fuel assemblies asymmetrically within the storage cells.

The results of the tolerance calculation are used in the next section to develop the 95/95 K_{eff}.

3.3 95/95 K_{eff} Calculations

To develop the 95 percent probability at a 95 percent confidence level K_{eff} for the spent fuel storage racks, the following reactivity biases must be included:

Methodology: The benchmarking bias as determined for the Westinghouse KENO-Va methodology in Section 2.1 is considered.

Water Temperature: A reactivity bias is applied to account for the effect of the normal range of spent fuel pool water temperatures.

^{10}B Self Shielding: If applicable, a reactivity bias is added to the results to correct for the modeling assumption that individual ^{10}B atoms are homogeneously distributed within the absorber material (versus clustered about each B_4C particle). A commonly used neutron poison material which requires this bias is Boraflex.

The tolerances calculated in the previous section are combined with the following uncertainties:

Calculation Uncertainty: The 95 percent probability/95 percent confidence level uncertainty on the KENO-Va nominal reference K_{eff} is considered.

Methodology Uncertainty: The 95 percent probability/95 percent confidence uncertainty in the benchmarking bias as determined for the Westinghouse KENO-Va methodology in Section 2.1 is considered.

The following formula is used to determine the 95/95 K_{eff} for the spent fuel storage racks:

$$K_{eff} = K_{nominal} + B_{method} + B_{temp} + B_{self} + B_{uncert} \quad (3.1)$$

where:

$K_{nominal}$ = nominal conditions KENO-Va K_{eff}.

B_{method} = method bias determined from benchmark critical comparisons.

B_{temp} = temperature bias.

B_{self} = ^{10}B self shielding bias, if applicable.

B_{uncert} = statistical summation of uncertainty components =

$$\sqrt{\sum_{i=1}^{n} ((tolerance_i \ldots or \ldots uncertainty_i)^2)}$$ for n tolerances/uncertainties.

The final 95/95 K_{eff} calculated above will satisfy ANSI 57.2[1] and NRC guidance[2] which require K_{eff} to be less than or equal to 0.95.

4.0 Reactivity Equivalencing Methodology

Increased flexibility for the storage of higher enrichment fuel assemblies is achievable using reactivity equivalencing. Reactivity equivalencing is predicated upon the reactivity decrease associated with fuel depletion or the addition of Integral Fuel Burnable Absorbers (IFBA). A series of reactivity calculations is performed to generate a set of enrichment-burnup or enrichment-IFBA ordered pairs which all yield an equivalent K_{eff} when the fuel is stored in the spent fuel storage racks. The data points on the reactivity equivalence curve are generated with the transport theory computer code, PHOENIX-P. The next two sections detail the assumptions, methodology, and uncertainties used when calculating fuel assembly burnup credit or IFBA credit.

4.1 Fuel Assembly Burnup Credit

Storage of burned fuel assemblies in the spent fuel storage racks with higher initial enrichments than those allowed by the methodology in Section 3.0 is achievable using reactivity equivalencing. The concept of reactivity equivalencing is predicated upon the reactivity decrease associated with fuel depletion. For burnup credit, a series of reactivity calculations are performed to generate a set of enrichment-fuel assembly discharge burnup ordered pairs which all yield an equivalent K_{eff} when stored in the spent fuel storage racks.

The first step in calculating burnup credit is fuel depletion. The fuel assembly is depleted under hot full power core operating conditions. A conservatively high soluble boron letdown curve is chosen to enhance the buildup of plutonium thus making the fuel assembly more reactive when stored in the spent fuel storage racks.

To determine the most reactive time after shutdown of a burned fuel assembly, a study was done which examined fission product decay after shutdown using the computer code CINDER[19]. CINDER is a point-depletion computer code used to determine fission product activities. The fission products were permitted to decay for 30 years after shutdown. The fuel reactivity was found to reach a maximum at approximately 100 hours after shutdown. At this time, the major fission product poison, ^{135}Xe, has nearly completely decayed away. Furthermore, the fuel reactivity was found to decrease continuously from 100 hours to 30 years following shutdown. Therefore, the most reactive time for a fuel assembly after shutdown of the reactor can be conservatively approximated by removing the ^{135}Xe.

Uncertainties associated with the depletion of the fuel assembly and reactivities computed with PHOENIX-P are accounted for in the development of the individual reactivity equivalence limits. An uncertainty is applied to the PHOENIX-P calculational results which starts at [ ]$^{(a,c)}$ and increases linearly with burnup, [ ]$^{(a,c)}$. The bias as a function of enrichment is provided in Figure 1 on page 27. This bias is considered to be very conservative and is based on consideration of the good agreement between PHOENIX-P predictions and measurements and on conservative estimates of fuel assembly reactivity variances with depletion history.

Based on the most reactive time after shutdown study and the uncertainty in the PHOENIX-P calculations, the burned fuel assembly is restarted in PHOENIX-P at various burnup steps with no fission product decay at cold (68°F) spent fuel storage rack conditions with all ^{135}Xe removed. The K_∞ results from these calculations are compared to the nominal rack condition K_∞ at cold spent fuel storage rack conditions with the zero burnup enrichment. An equivalent burnup at each higher enrichment is determined by finding the burnup which yields a K_∞ (including uncertainty) equal to the zero burnup enrichment nominal rack condition K_∞. Multiple sets of burnup-enrichment pairs are used to establish a burnup credit curve which covers the enrichment range of the zero burnup enrichment to the highest enrichment stored in the spent fuel racks.

It is important to recognize that the curve is based on calculations of constant rack reactivity. In this way, the environment of the storage rack and its influence on assembly reactivity is implicitly considered.

To better illustrate this methodology, a sample burnup credit curve is provided in Figure 2 on page 28. Note in Figure 2, the endpoints are (0 MWD/MTU, 2.0 w/o ^{235}U) and (33,000 MWD/MTU, 5.0 w/o ^{235}U). The interpretation of the endpoint data is as follows: the reactivity of the spent fuel rack containing 5.0 w/o ^{235}U fuel at 33,000 MWD/MTU is equivalent to the reactivity of the rack containing 2.0 w/o ^{235}U fresh fuel. The endpoint data at 5.0 w/o includes a reactivity uncertainty of [ ]$^{(a,c)}$ consistent with the minimum burnup requirement of 33,000 MWD/MTU. Reactivity uncertainty is also applied linearly to all points on Figure 2 consistent with Figure 1.

As part of the burnup credit calculation, no specific uncertainty is added for measured burnup predictions. Uncertainty associated with measured burnups is dependent on the code or method used to predict the measured burnup. Additional burnup necessary to offset any measured burnup uncertainty must be added to the burnup credit requirement determined by the criticality analysis to determine the final acceptance curve for burnup credit.

The effect of axial burnup distribution on assembly reactivity has been considered in the development of the burnup credit methodology. Westinghouse evaluations[20] have been performed to quantify axial burnup reactivity effects and to confirm that the reactivity equivalencing methodology described above results in calculations of conservative burnup credit limits. The evaluations show that axial burnup effects can cause assembly reactivity to increase only at burnup-enrichment combinations (ex. 4.0 w/o ^{235}U @ 40,000 MWD/MTU) which are well beyond those typically calculated for burnup credit limits. Additional accounting for axial burnup distribution effects is not necessary provided the burnup credit required does not exceed the previously determined limits. These limits are presented in Table 6 on page 26.

4.2 Integral Fuel Burnable Absorber (IFBA) Credit

Storage of fresh fuel assemblies with nominal enrichments greater than those allowed by the methodology in Section 3.0 is achievable using reactivity equivalencing. The concept of reactivity equivalencing is predicated upon the reactivity decrease associated with the addition of

Integral Fuel Burnable Absorbers (IFBA). IFBAs consist of neutron absorbing material applied as a thin ZrB_2 coating on the outside of the UO_2 fuel pellet. As a result, the neutron absorbing material is a non-removable or integral part of the fuel assembly once it is manufactured.

Two analytical techniques are used to establish the criticality criteria for the storage of IFBA fuel assemblies in the fuel storage racks. The first method uses reactivity equivalencing to establish the poison material loading required to meet the criticality limits. The poison material considered in this analysis is a zirconium diboride (ZrB_2) coating manufactured by Westinghouse. The second method uses the fuel assembly infinite multiplication factor to establish a reference reactivity. The reference reactivity point is compared to the fuel assembly peak reactivity to determine its acceptability for storage in the fuel racks.

4.2.1 IFBA Requirement Determination

A series of reactivity calculations are performed to generate a set of IFBA rod number versus enrichment ordered pairs which all yield the equivalent K_{eff} when the fuel is stored in the spent fuel storage racks. The following assumptions are used for the IFBA rod assemblies in the PHOENIX-P models:

1. The fuel assembly is modeled at its most reactive point in life. This includes any depleted time in life where the IFBA has burned-out and the fuel assembly becomes more reactive.
2. Each IFBA rod has a poison material ^{10}B loading determined from Westinghouse IFBA design specifications for the given fuel assembly type, which is the minimum standard loading offered by Westinghouse for that fuel assembly type. Higher loadings (1.5X and 2.0X) are also considered.
3. The IFBA ^{10}B loading is reduced by [ ]$^{(a,c)}$ to conservatively account for manufacturing tolerances.
4. The IFBA ^{10}B loading is reduced by an amount which corresponds to the minimum poison length offered for the given fuel assembly type. For instance, a 144 inch fuel stack with a minimum poison length of 108 inches would result in a 25% IFBA ^{10}B loading reduction to conservatively model the minimum poison length for that fuel assembly type.

Uncertainties associated with IFBA dependent reactivities computed with PHOENIX-P are accounted for in the development of the IFBA credit limits. An uncertainty of approximately [ ]$^{(a,c)}$ of the total number of IFBA rods is accounted for in the development of the IFBA requirements.

Using the above assumptions, PHOENIX-P K_∞ results are calculated at higher fuel assembly enrichments (above the zero IFBA enrichment) which contain different number of IFBA fuel rods. The number of IFBA rods (at a given enrichment) which yields a K_∞ equal to the nominal rack condition K_∞ (with the zero IFBA enrichment) determines the number of IFBA for that enrichment. An additional number of IFBA rods are added to each data point to account for the uncertainty of [ ]$^{(a,c)}$ discussed above.

To better illustrate this methodology, a sample IFBA credit curve is provided in Figure 3 on page 29. Note in Figure 3, the endpoints are (0 IFBA rods, 4.2 w/o ^{235}U) and (64 IFBA rods, 5.0 w/o ^{235}U). The interpretation of the endpoint data is as follows: the reactivity of the spent fuel rack containing a 5.0 w/o ^{235}U fuel assembly with 64 IFBA rods is equivalent to the reactivity of the rack containing a 4.2 w/o ^{235}U fuel assembly with no IFBA rods. The endpoint data at 5.0 w/o includes an additional [ ]$^{(a,c)}$ IFBA rods consistent with the uncertainty on 64 IFBA rods.

4.2.2 Infinite Multiplication Factor

The infinite multiplication factor, K_∞, is used as a reference criticality reactivity point, and offers an alternative method for determining the acceptability of fuel assembly storage in the spent fuel racks. The fuel assembly K_∞ calculations are performed using PHOENIX-P. The following assumptions were used to develop the infinite multiplication factor model:

1. The fuel assembly is modeled at its most reactive point in life and no credit is taken for any burnable absorbers in the assembly. The zero IFBA enrichment limit is used for the fuel assembly.
2. The fuel array model is based on a unit assembly configuration (infinite in the lateral and axial extent) in reactor geometry (no rack).
3. The moderator is pure water (no boron) at a temperature of 68°F with a density of 1.0 g/cm^3.

Calculation of the infinite multiplication factor for the fuel assembly in core geometry results in a reference K_∞. A [ ]$^{(a,c)}$ reactivity bias is added to this reference K_∞ to conservatively account for calculational uncertainties. This bias is [ ]$^{(a,c)}$.

For IFBA credit, all fuel assemblies placed in the spent fuel racks must comply with the enrichment-IFBA requirement curve or have a reference K_∞ less than or equal to the value calculated above (including a [ ]$^{(a,c)}$ bias). By meeting either of these conditions, the maximum rack reactivity will be less than 0.95.

5.0 Postulated Accident Methodology

Accident conditions must be addressed in spent fuel storage rack criticality analysis to ensure that K_{eff} is maintained less than or equal to 0.95. Two types of accidents can occur in the spent fuel rack which can cause reactivity to increase. The first accident type, a fuel assembly misplacement, involves the placement of a fuel assembly into a position for which the restrictions on location, enrichment, burnup, or IFBA are not satisfied. The second accident type, a pool water temperature change, involves an increase or decrease in the spent fuel pool water temperature and density.

Two of the fuel assembly misplacement accidents have no impact on reactivity. These accidents include:

Fuel assembly drop on top of rack: The rack structure pertinent for criticality control is not excessively deformed and the dropped assembly which comes to rest horizontally or vertically on top of the rack has sufficient water separating it from the active fuel height of stored assemblies to preclude neutron interaction.

Fuel assembly drop between rack modules or between rack modules and spent fuel pool wall: In most of these cases, the design of the spent fuel rack is such that it precludes the insertion of a fuel assemblies in these locations. If this is not true, the same accident analysis procedure described next is used for this accident.

For the fuel assembly misplacement accidents, the amount of reactivity increase caused by each possible accident scenario is calculated using KENO-Va. A KENO-Va model of each misplaced fuel assembly condition is created and the reactivity increase caused by the misplaced assembly is determined.

For the pool water temperature accident, PHOENIX-P is used to determine the amount of reactivity increase associated with an increase or decrease in spent fuel pool water temperature. The water density is also adjusted accordingly. The temperature range considered is 32°F to 212°F.

For an occurrence of any of the above postulated accident condition, the double contingency principle of ANSI/ANS 8.1-1983 can be applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated 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.

To determine the reactivity decrease associated with spent fuel pool soluble boron, the reactivity change due to the presence of spent fuel pool soluble boron is calculated using PHOENIX-P.

Using the results of the PHOENIX-P soluble boron worth calculations, the amount of soluble boron required to offset the highest reactivity increase caused by all accident conditions is determined. This spent fuel pool soluble boron is thus the required amount of boron needed to maintain K_{eff} less than or equal to 0.95 under all postulated accident conditions.

6.0 Soluble Boron Credit Methodology

This section describes the analytical techniques and models employed to perform the criticality analysis calculations for spent fuel storage racks with credit for spent fuel pool soluble boron.

Section 6.1 describes the maximum feasible K_{eff} calculation performed for the spent fuel storage racks to show that K_{eff} is less than 1.0 with no soluble boron credit. Section 6.2 describes the tolerance calculations used to determine the uncertainty presented by fuel assembly and storage rack tolerances and the use of spent fuel pool soluble boron to offset these reactivity uncertainties and maintain K_{eff} less than or equal to 0.95. Section 6.3 discusses the reactivity equivalencing calculations as performed with soluble boron credit. Section 6.4 discusses the calculation of postulated accident conditions with soluble boron credit. Finally, Section 6.5 summarizes the total soluble boron credit required by the spent fuel rack criticality analysis.

6.1 Maximum Feasible K_{eff}

The maximum feasible K_{eff} calculation defines the normal storage configuration for fuel assemblies in the spent fuel storage racks such that the maximum K_{eff} is less than 1.0. Normal storage rack configuration conditions are defined as nominal fuel assembly parameters and spent fuel rack dimensions. This calculation uses the same assumptions listed in Section 3.1. The calculation is performed at cold conditions with no soluble boron in the spent fuel pool water. A temperature bias is calculated to account for the normal operational temperature range of the spent fuel pool water, the method bias is determined from the benchmarking calculations performed in Section 2.1 and the ^{10}B self shielding bias is included, if applicable.

The final equation for determining the maximum feasible K_{eff} is shown below:

$$K_{max.\ feasible} = K_{normal} + B_{temp} + B_{method} + B_{self} \qquad (6.1)$$

where:

K_{normal} = normal condition KENO-Va K_{eff}

B_{temp} = temperature bias for normal operating range

B_{method} = method bias determined from benchmark critical comparisons

B_{self} = ^{10}B self shielding bias

The storage configuration used to calculate the maximum feasible K_{eff} using the above equation must be less than 1.0 with no soluble boron. This storage configuration is the basis for fuel assembly storage in the spent fuel pool with credit for soluble boron.

6.2 Soluble Boron Credit K_{eff}

To maintain adequate safety margin for criticality in the spent fuel storage racks, the K_{eff} of the spent fuel storage racks will be shown to be less than or equal to 0.95 with allowances for tolerances and uncertainties in the presence of spent fuel pool soluble boron. The same assumptions of Section 3.2 are applied here. The only difference between these assumptions and the calculations performed here is the presence of spent fuel pool soluble boron. A spent fuel pool soluble boron concentration is chosen which will provide a K_{eff} that is less than or equal to 0.95 when biases, tolerances and uncertainties are included. The tolerance calculations are performed assuming the presence of this spent fuel pool soluble boron concentration. The final 95/95 K_{eff} calculation is determined using equation 3.1 on page 8 The final 95/95 K_{eff} will be shown to be less than or equal to 0.95 with allowances for biases, tolerances, and uncertainties including the presence of the determined concentration of spent fuel pool soluble boron.

6.3 Reactivity Equivalencing with Soluble Boron Credit

The reactivity equivalencing methodology with soluble boron credit is similar to the methodology discussed in Section 4.0. The major differences are that the reactivity equivalencing calculations are performed based on the maximum feasible K_{eff} storage conditions and the uncertainty associated with the reactivity equivalencing methods are covered using credit for soluble boron. A detailed discussion on the specific calculations required for each type of reactivity equivalencing method follows.

6.3.1 Fuel Assembly Burnup Credit with Soluble Boron Credit

The reactivity equivalencing methodology for burnup credit with soluble boron credit is similar to the methodology discussed in Section 4.1. The first major difference is the basis for the PHOENIX-P reactivity calculations. The spent fuel rack restarts with burned fuel assemblies use spent fuel rack conditions which are established using the assumptions of the maximum feasible K_{eff} defined in Section 6.1. Using this set of conditions guarantees the spent fuel racks will not return critical under conditions of no soluble boron for the storage of burned fuel assemblies. As shown later in this section, the amount of soluble boron required to ensure K_{eff} remains less than or equal to 0.95 will also be determined.

The second major difference is the reactivity uncertainty associated with the burnup credit calculations and the uncertainty associated with measured burnup. A reactivity bias is typically applied to the reactivity calculations to account for uncertainties associated with the depletion of the fuel assembly and reactivities computed with PHOENIX-P. Also, if necessary, additional uncertainty is added to the burnup credit requirement to account for uncertainty in the measured burnup. Since the maximum feasible K_{eff} condition contains no soluble boron, calculations will be performed at the highest burnup requirement to determine the amount of soluble boron needed to maintain K_{eff} less than 0.95 including the appropriate uncertainty for depletion effects and

PHOENIX-P calculations (see Figure 1 on page 27) and appropriate amount of uncertainty on measured burnup. The increase in boron required to offset these uncertainties will be included in the final soluble boron credit requirement.

6.3.2 IFBA Credit with Soluble Boron Credit

The reactivity equivalencing methodology for IFBA credit with soluble boron credit is similar to the methodology discussed in Section 4.2. The number of IFBA rods required and the infinite multiplication factor calculations will be determined using the configuration assumed in the maximum feasible K_{eff} defined in Section 6.1. Using this configuration guarantees the spent fuel racks will not return critical under conditions of no soluble boron. As shown later in this section, the amount of soluble boron required to ensure K_{eff} remains less than or equal to 0.95 will also be determined.

The uncertainty associated with the determination of the IFBA rod requirement will be offset with credit for soluble boron. The uncertainties include the [ ]$^{(a,c)}$ decrease in IFBA ^{10}B loading for manufacturing uncertainty and [ ]$^{(a,c)}$ decrease in the number of IFBA rods for calculational uncertainty. To ensure that K_{eff} is maintained less than 0.95, calculations will be performed which include enough soluble boron to offset the reactivity increase cause by a [ ]$^{(a,c)}$ decrease in IFBA ^{10}B loading and [ ]$^{(a,c)}$ decrease in the number of IFBA rods for calculational uncertainty. The increase in boron required to offset these uncertainty values will be included in the final soluble boron credit requirement.

The calculation of the infinite multiplication factor for IFBA credit remains the same as Section 4.2.2. The fuel assembly enrichment is defined by the maximum feasible storage configuration of Section 6.1. The uncertainty of [ ]$^{(a,c)}$ associated with the infinite multiplication factor calculation is still applied.

6.4 Postulated Accidents with Soluble Boron Credit

The postulated accidents will be considered in the same manner as discussed in Section 5.0. The major differences are in the presence of soluble boron and the interpretation of the double contingency principle.

For the postulated accidents which cause a reactivity increase, the amount of reactivity increase will be calculated as before except the amount of soluble boron as determined in the calculations of Section 6.2 will be present. Based on the double contingency principle, one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Therefore, the presence of soluble boron in the storage pool water at normal concentrations (typically 2000 ppm) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

To determine the reactivity decrease associated with spent fuel pool soluble boron, the reactivity change due to the presence of spent fuel pool soluble boron is calculated using PHOENIX-P.

Using the results of the PHOENIX-P soluble boron worth calculations, the amount of soluble boron required to offset each reactivity increase caused by accident conditions is determined. The sum of the boron concentrations of Sections 6.2 and 6.3 is the starting point to determine the amount of additional soluble boron needed to offset the reactivity increase caused by the postulated accidents.

6.5 Soluble Boron Credit Summary

To summarize the soluble boron credit calculations, there are four calculations performed using the soluble boron credit methodology which determine three soluble boron credit concentrations. The four calculations are listed below.

1. Determine the storage configuration of the spent fuel racks using maximum feasible K_{eff} conditions such that K_{eff} is less than 1.0.
2. Using the resulting configuration from the previous step, calculate the spent fuel rack K_{eff} with soluble boron. Next determine the reactivity uncertainty associated with fuel assembly and storage rack tolerances and combine with the biases and other uncertainties to determine the final 95/95 K_{eff} at the concentration of spent fuel pool soluble boron which maintains K_{eff} less than or equal to 0.95.
3. Use reactivity equivalencing methodologies to determine burnup or IFBA credit for enrichments higher than allowed in step 1. Use soluble boron credit to offset uncertainties associated with each methodology, as appropriate.
4. Determine the increase in reactivity caused by postulated accidents and the corresponding additional amount of soluble boron needed to offset these reactivity increases.

The final soluble boron credit requirement is the summation of the boron credit requirements determined in steps 2, 3 and 4 above. The following equation relates these requirements.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA} \qquad (6.2)$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm).

$SBC_{95/95}$ = soluble boron credit required for 95/95 K_{eff} less than or equal to 0.95 (ppm).

SBC_{RE} = soluble boron credit required for reactivity equivalencing methodologies (ppm).

SBC_{PA} = soluble boron credit required for K_{eff} less than or equal to 0.95 under accident conditions (ppm).

The total soluble boron credit requirement along with the storage configuration specified in the maximum feasible K_{eff} calculations shows that the spent fuel rack K_{eff} will always be less than or equal to 0.95. Further the maximum feasible K_{eff} storage configuration will ensure the K_{eff} remains less than 1.0 with no soluble boron in the spent fuel pool.

Table 1. Benchmark Critical Experiments for KENO-Va

Critical Number	General Description	Enrichment	Reflector	Separating Material	Soluble Boron	Measured K_{eff}	KENO-Va Reactivity (K_{eff} +/- One Sigma)
1	UO_2 Rod Lattice	2.46	water	water	0	1.0002	[
2	UO_2 Rod Lattice	2.46	water	water	1037	1.0001	
3	UO_2 Rod Lattice	2.46	water	water	764	1.0000	
4	UO_2 Rod Lattice	2.46	water	B_4C pins	0	0.9999	
5	UO_2 Rod Lattice	2.46	water	B_4C pins	0	1.0000	
6	UO_2 Rod Lattice	2.46	water	B_4C pins	0	1.0097	
7	UO_2 Rod Lattice	2.46	water	B_4C pins	0	0.9998	
8	UO_2 Rod Lattice	2.46	water	B_4C pins	0	1.0083	
9	UO_2 Rod Lattice	2.46	water	water	0	1.0030	
10	UO_2 Rod Lattice	2.46	water	water	143	1.0001	
11	UO_2 Rod Lattice	2.46	water	stainless steel	514	1.0000	
12	UO_2 Rod Lattice	2.46	water	stainless steel	217	1.0000	
13	UO_2 Rod Lattice	2.46	water	borated aluminum	15	1.0000	
14	UO_2 Rod Lattice	2.46	water	borated aluminum	92	1.0001	
15	UO_2 Rod Lattice	2.46	water	borated aluminum	395	0.9998	
16	UO_2 Rod Lattice	2.46	water	borated aluminum	121	1.0001	
17	UO_2 Rod Lattice	2.46	water	borated aluminum	487	1.0000	
18	UO_2 Rod Lattice	2.46	water	borated aluminum	197	1.0002	
19	UO_2 Rod Lattice	2.46	water	borated aluminum	634	1.0002	
20	UO_2 Rod Lattice	2.46	water	borated aluminum	2 9	1.0003	
21	UO_2 Rod Lattice	2.46	water	borated aluminum	72	0.9997	
22	UO_2 Rod Lattice	2.35	water	borated aluminum	0	1.0000	
23	UO_2 Rod Lattice	2.35	water	stainless steel	0	1.0000	
24	UO_2 Rod Lattice	2.35	water	water	0	1.0000	
25	UO_2 Rod Lattice	2.35	water	stainless steel	0	1.0000	
26	UO_2 Rod Lattice	2.35	water	borated aluminum	0	1.0000	
27	UO_2 Rod Lattice	2.35	water	B_4C	0	1.0000	
28	UO_2 Rod Lattice	4.31	water	stainless steel	0	1.0000	
29	UO_2 Rod Lattice	4.31	water	water	0	1.0000	
30	UO_2 Rod Lattice	4.31	water	stainless steel	0	1.0000	
31	UO_2 Rod Lattice	4.31	water	borated aluminum	0	1.0000	
32	UO_2 Rod Lattice	4.31	water	borated aluminum	0	1.0000	
33	U-metal Cylinders	93.2	bare	air	0	1.0000	
34	U-metal Cylinders	93.2	bare	air	0	1.0000	
35	U-metal Cylinders	93.2	bare	air	0	1.0000	
36	U-metal Cylinders	93.2	bare	air	0	1.0000	
37	U-metal Cylinders	93.2	bare	air	0	1.0000	
38	U-metal Cylinders	93.2	bare	air	0	1.0000	
39	U-metal Cylinders	93.2	bare	plexiglass	0	1.0000	
40	U-metal Cylinders	93.2	paraffin	plexiglass	0	1.0000	
41	U-metal Cylinders	93.2	bare	plexiglass	0	1.0000	
42	U-metal Cylinders	93.2	paraffin	plexiglass	0	1.0000	
43	U-metal Cylinders	93.2	paraffin	plexiglass	0	1.0000	
44	U-metal Cylinders	93.2	paraffin	plexiglass	0	1.0000	](a,c)

Table 2. Summary of Lattice Parameters for Strawbridge and Barry 101 Criticals

Lattice Parameter	Range
Enrichment (a/o)	1.04 to 4.07
Boron Concentration (ppm)	0 to 3392
Water to Uranium Ratio	1.0 to 11.96
Pellet Diameter (cm)	0.44 to 2.35
Lattice Pitch (cm)	0.95 to 4.95
Clad Material	none, aluminum, and stainless steel
Lattice Type	square and hexagonal
Fuel Density (g/cm^3)	7.5 to 18.9

Table 3. B&W Core Loadings and Compositions Studied

Core	Loading	Number of Fuel Rods	Number of Water-Filled Positions	Number of Pyrex Rods	Soluble Boron (ppm)
XI	1	4961	0	0	1511
	2	4808	153	0	1334
	7	4808	81	72	1031
	8	4808	9	144	794
	9	4808	9	144	779
XIV	1	4736	225	0	1289
	6	4736	201	24	1179

Table 4. Comparison of PHOENIX-P Isotopics Predictions to Yankee Core 5 Measurements

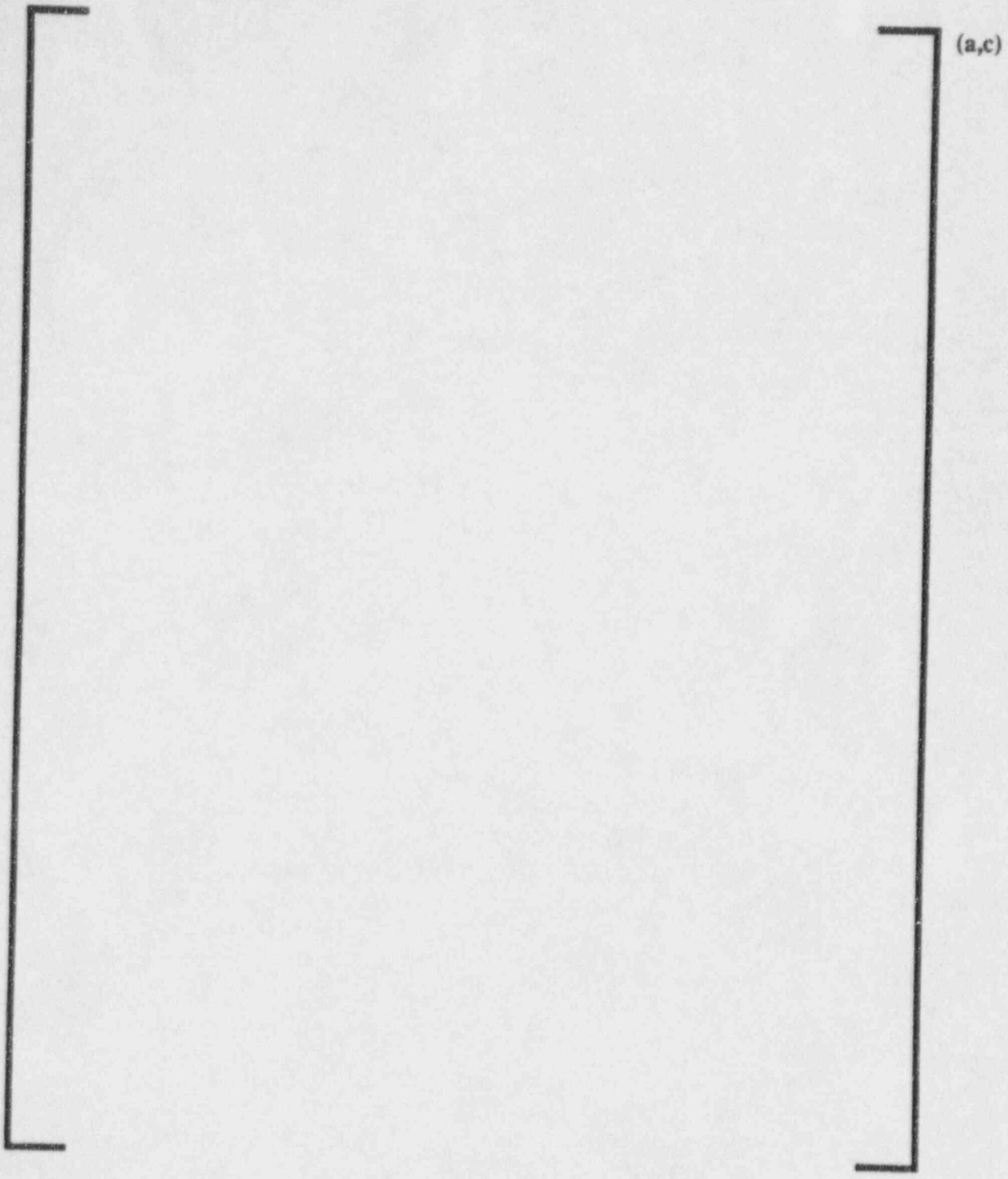
Quantity (Atom Ratio)	% Difference
$^{235}U/U$	[
$^{236}U/U$	
$^{238}U/U$	
$^{239}Pu/Pu$	
$^{240}Pu/Pu$	
$^{241}Pu/Pu$	
$^{242}Pu/Pu$	
$^{239}Pu/^{238}U$	
$^{235}U/^{238}U$	]$^{(a,c)}$

Table 5. Fuel Parameters Relevant to the Spent Fuel Storage Rack Criticality Analysis

Fuel Assembly Parameters
Number of Fuel Rods per Assembly
Fuel Rod Clad Material
Fuel Rod Clad Outer Diameter
Fuel Rod Clad Thickness
Fuel Pellet Outer Diameter
Fuel Pellet Density
Fuel Pellet Dishing Factor
Rod Pitch
Guide Tube Material
Number of Guide Tubes
Guide Tube Outer Diameter
Guide Tube Thickness
Instrument Tube Material
Number of Instrument Tubes
Instrument Tube Outer Diameter
Instrument Tube Thickness

Table 6. Axial Burnup Reactivity Bias for PHOENIX-P

Fuel Assembly Burnup (MWD/MTU)	3.0 w/o Enrichment Bias (ΔK)	4.0 w/o Enrichment Bias (ΔK)
[]$^{(a,c)}$	[]$^{(a,c)}$	[]$^{(a,c)}$



(a,c)

Figure 1. Burnup Credit Reactivity Bias

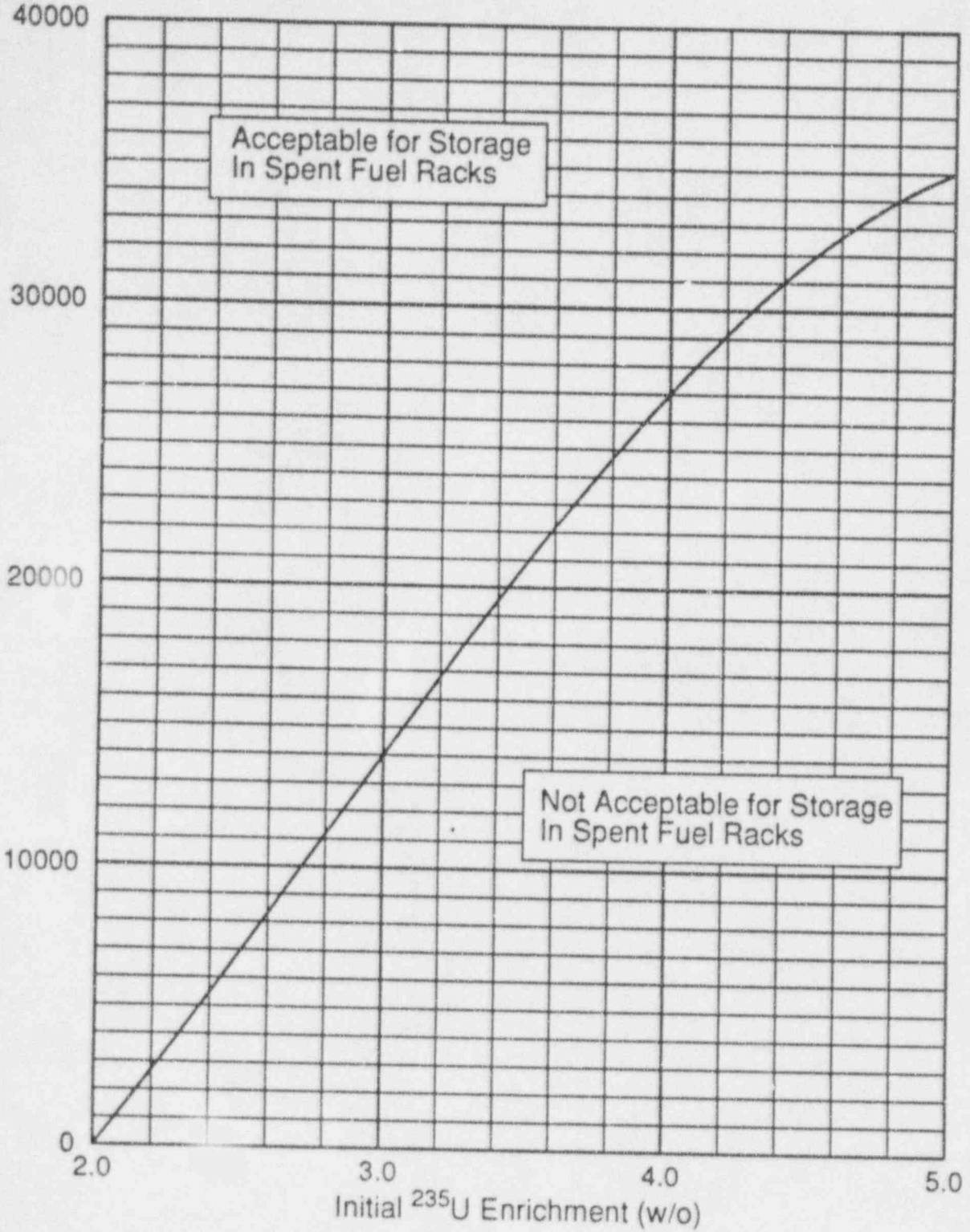


Figure 2. Example Burnup Credit Curve

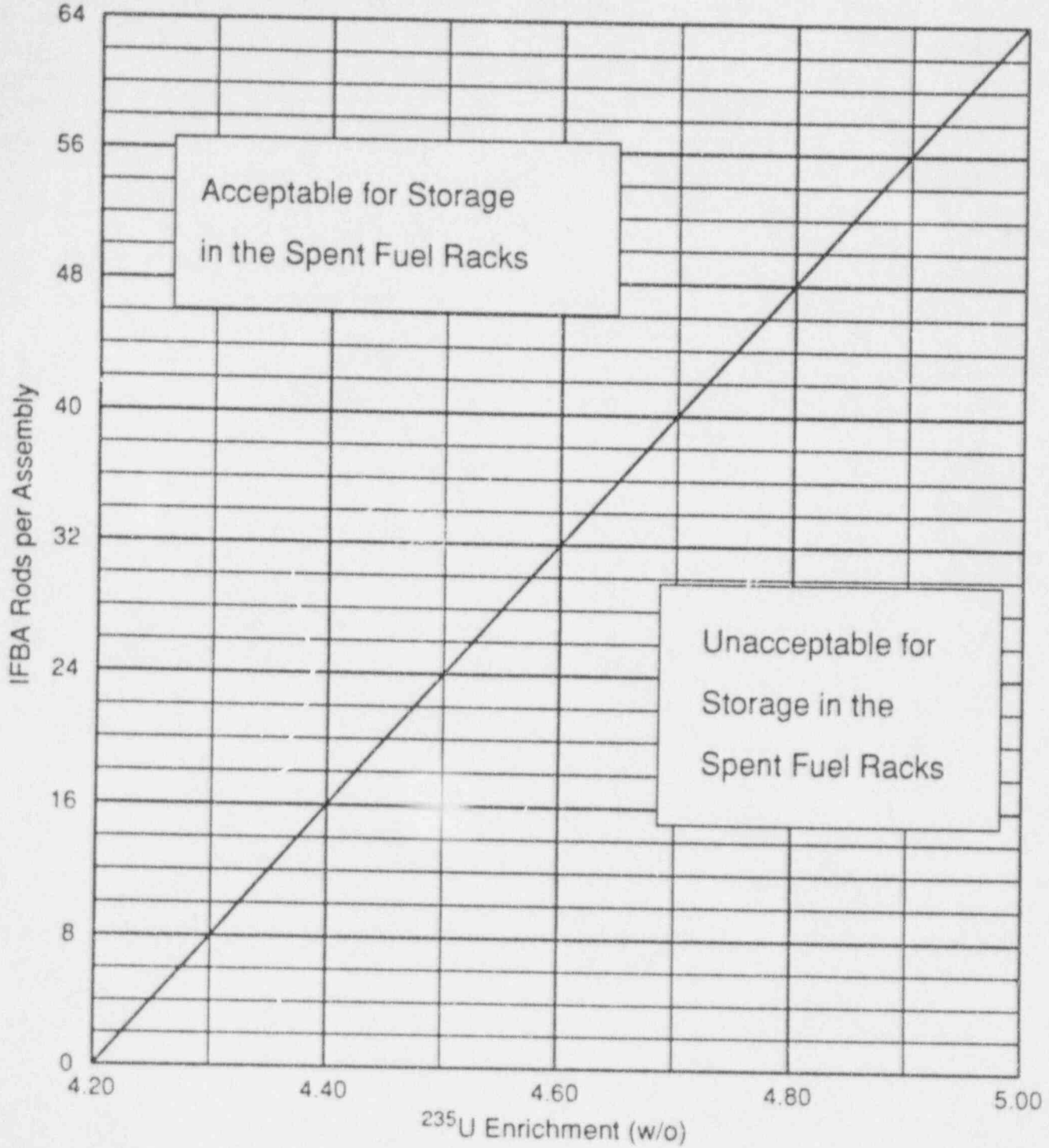


Figure 3. Example IFBA Credit Curve

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