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Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Supplement to License Amendment Request (LAR) to Revise the Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage" (TAC Nos. MC5811 and MC5812)

By letter dated February 1, 2005, Nuclear Management Company (NMC) submitted an LAR to revise the spent fuel pool criticality analyses and Technical Specifications (TS) 3.7.17, "Spent Fuel Pool Storage" and 4.3, "Fuel Storage". By letter dated September 16, 2005, NMC provided responses to the NRC request for additional information regarding this LAR. This letter supplements the subject LAR to address October 6 and November 15, 2005 telephone discussions with the Nuclear Regulatory Commission (NRC) Staff. NMC is submitting this supplement in accordance with the provisions of 10 CFR 50.90.

Enclosure 1 provides supporting information on the regulatory basis for NRC approval of this LAR.

Enclosure 2 to this letter provides Westinghouse WCAP-16517-NP, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis", dated November 2005. This WCAP replaces in its entirety the Westinghouse calculation provided as Exhibit D in the NMC February 1, 2005 LAR. This WCAP differs technically from the calculation in that the analysis and results are based on demonstration that k_{eff} is less than 0.995 using no soluble boron, including biases and uncertainties, for the fresh and spent fuel storage configurations. The WCAP also contains administrative differences that include removal of the Appendices and references to them, and removal of references to Westinghouse internal procedures. This WCAP is marked as Westinghouse Non-Proprietary Class 3 on each page, and therefore, NMC is not requesting that the document be withheld from public disclosure under Title 10 Code of Federal Regulations Section 2.390.

Enclosure 3 provides revised TS Figure 3.7.17-1, Figure 4.3.1-3 and Figure 4.3.1-4 which have been revised to incorporate the results from WCAP-16517-NP. These figures replace the figures previously submitted in Exhibit C with the NMC February 1, 2005 LAR.

The proposed changes in this supplement do not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the February 1, 2005 submittal as supplemented February 22, 2005 and September 16, 2005.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR supplement by transmitting a copy of this letter and enclosures to the designated State Official.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on DEC 0 2 2005

Thomas J. Palmisano
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Nuclear Management Company, LLC

Enclosures (3)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

Enclosure 1

Regulatory Basis for Prairie Island Nuclear Generating Plant Fuel Storage Criticality Prevention

The Prairie Island Nuclear Generating Plant (PINGP) regulatory basis for fuel storage criticality prevention is Atomic Energy Commission (AEC) General Design Criterion (GDC) 66, "Prevention of Fuel Storage Criticality" as proposed on July 10, 1967. AEC GDC 66 specifies, "Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configuration shall be emphasized over procedural controls". Physical systems require analyses, including criticality analyses, to demonstrate that they provide a safe configuration.

For the original plant license, the specific neutron multiplication factor, k_{eff}, criteria which the criticality analyses were required to meet to demonstrate compliance with AEC GDC 66 were included in TS Section 5.6.A. The original plant Technical Specifications (TS) required $k_{eff} \leq 0.90$ with unborated water. This limit became the licensing basis and thus, the regulatory basis for future analyses, through issuance of the PINGP TS by the AEC. Nuclear Management Company (NMC) is not aware of any regulations in effect at that time that specified this neutron multiplication factor limit.

To support reracking of the fuel storage pools, in 1976 Northern States Power Company (NSP[1]) submitted a License Amendment Request (LAR) to the NRC with new criticality analyses and revised TS 5.6.A to limit $k_{eff} \leq 0.95$ with unborated water. The NRC reviewed and approved the proposed criticality analyses and issued the revised TS in License Amendments (LA) -22 (Unit 1)/16 (Unit 2), dated August 16, 1977. With this license amendment the NRC revised the PINGP licensing basis fuel pool analysis multiplication factor acceptance criteria and the regulatory basis for future analyses to $k_{eff} \leq 0.95$. NMC is not aware of any regulations in effect at that time that specified this neutron multiplication factor limit.

In the mid-1990s, PINGP was the pilot plant for a Westinghouse Owners Group initiative to license criticality analyses crediting soluble boron. NSP submitted an LAR with new criticality analyses and proposed to revise TS 5.6.A.1 requiring the pool to be maintained with:

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Reference 3;
c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 750 ppm, which includes an allowance for uncertainties as described in Reference 3;

[1] NSP was the plant operating entity prior to NMC.

Reference 3 above was "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", Westinghouse Commercial Nuclear Fuel Division, dated February 1997. This analysis utilized the methodology in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology".

The NRC reviewed and approved the SFP analyses (identified as Reference 3 above) which credited soluble boron and issued TS 5.6.A.1 on June 12, 1997 with the paragraph b. and c. requirements quoted above in LA-129/121. This LA approval was based on NRC approval of WCAP-14416-NP-A. NMC is not aware of any regulations in effect at that time that specified these neutron multiplication factor limits. The multiplication factor analysis limits became the regulatory limit and licensing basis through NRC issuance of the TS. This LA established new PINGP licensing basis multiplication factors for future fuel storage analyses.

On February 1, 2005, NMC submitted an LAR to the NRC with new SFP criticality analyses based on a different methodology and proposed to revise TS curves relating to fuel storage. This LAR proposes to replace the reference in TS 4.3[2], "Fuel Storage", with a reference to the new analyses. The new analyses continue to meet the current PINGP licensing basis multiplication factor limits, that is, $k_{eff} < 1.0$ if fully flooded with unborated water and $k_{eff} \leq 0.95$ if fully flooded with borated[3] water. The February 1, 2005 submittal did not propose to change the multiplication factor limits, thus, the current licensing basis continues to be met and these limits form the regulatory basis for issuance of the LA. NMC only proposes to replace the methodology and analyses which demonstrate compliance with these limits.

In conclusion, the regulatory basis, for approving the LAR dated February 1, 2005 as supplemented on February 22, 2005, September 16, 2005 and by this letter, is AEC GDC 66 as it has been throughout the plant operating history. The current NRC approved licensing basis for demonstrating compliance with AEC GDC 66 is the neutron multiplication factor limits in the current PINGP TS 4.3. The methodology and analyses presented in the February 1, 2005 LAR and its supplements demonstrate that these current licensing basis limits are met.

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[2] LA-158/149 converted the PINGP TS to the format and content guidance of NUREG-1431, "Standard Technical Specifications, Westinghouse Plants". The new TS 4.3 contains the same requirements as TS 5.6.A previously contained.

[3] Based on the new analyses, the boron limit is proposed to be changed to 730 ppm.

Enclosure 2

WCAP-16517-NP
Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis
November 2005

74 pages follow

Westinghouse Non-Proprietary Class 3

WCAP-16517-NP
Revision 0

November 2005

Prairie Island Units 1 and 2
Spent Fuel Pool Criticality Analysis

WCAP-16517-NP

Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis

November, 2005

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1.0 Objective

This report presents the results of criticality analyses for the Prairie Island Units 1 & 2 spent fuel storage racks with credit for assembly burnup, Fuel Burnable Absorber (Gd_2O_3), ^{241}Pu decay and soluble boron. The primary objectives of this calculation are as follows:

1. To determine the design basis fuel assembly for all of the fuel assembly storage configurations. They include the "All-Cell" and "3x3 array" fuel assembly storage configurations.

2. To determine the assembly burnup versus initial enrichment limits required for safe storage of fuel assemblies in the "All-Cell" storage configuration

3. To determine the assembly burnup versus initial enrichment limits required for safe storage of peripheral fuel assemblies in the "3x3 array" with the center fuel assembly initially enriched to 4.95 w/o ^{235}U. This will be accomplished with credit for 5, 10, 15, and 20 years of ^{241}Pu decay.

4. To determine the assembly burnup versus initial enrichment limits required for safe storage of peripheral fuel assemblies in the "3x3 array" with the center fuel assembly initially enriched to 4.95 w/o ^{235}U and shimmed with 4 Gd_2O_3 rods. These limits will be derived based upon a Gd_2O_3 concentration of 4.0 w/o. This will be accomplished with credit for 5, 10, 15, and 20 years of ^{241}Pu decay.

5. To determine if the current interface between storage configurations is still valid

6. To determine the amount of soluble boron required to maintain k_{eff} less than or equal to 0.95, including all biases and uncertainties, assuming the most limiting plausible reactivity accident.

The methodology employed in this analysis for soluble boron credit is analogous to that of Reference 2 and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 3. Reference 2 was reviewed and approved by the US NRC. The methodology employed in this analysis and in Reference 2 employs axially distributed burnups to represent discharged fuel assemblies. This analysis was prepared in accordance with the Westinghouse Quality Assurance Program.

1.1 Design Criteria

The design criteria are consistent with GDC 62, Reference 4, and NRC guidance given in Reference 5. Section 1.4 describes the analysis methods including a description of the computer codes used to perform the criticality safety analysis. A brief summary of the analysis approach and criteria follows.

1. Determine the fresh and spent fuel storage configurations using no soluble boron conditions such that the 95/95 upper tolerance limit value of k_{eff}, including applicable

biases and uncertainties, is less than 0.995. This is accomplished with infinite arrays of either fresh or spent fuel assembly configurations. Note that the actual NRC k_{eff} limit for this condition is unity. Therefore, an additional safety margin equal to 0.005 Δk_{eff} units is included in the infinite array analysis results. Additional margin to the k_{eff} limit will be identified based upon the KENO results for the entire spent fuel pool #2.

2. Determine the amount (ppm) of soluble boron necessary to reduce the k_{eff} value of all storage configurations by at least 0.05 Δk_{eff} units. This is accomplished by constructing a KENO model for spent fuel pool #2 which includes the storage configurations which are least sensitive to changes in soluble boron concentration. As an example, storage configurations which contain depleted fuel assemblies (and represented by depleted isotopics) are less reactivity-sensitive to changes in soluble boron concentration than an assembly represented by zero burnup and a relatively low initial fuel enrichment. Note that spent fuel pool #2 is much larger than spent fuel pool #1 and therefore the results will be bounding for both spent fuel pools.

3. Determine the amount of soluble boron necessary to compensate for 5% of the maximum burnup credited in any storage configuration. In addition, determine the amount of soluble boron necessary to account for a reactivity depletion uncertainty equal to 1.0% Δk_{eff} per 30,000 MWD/MTU of credited assembly burnup. This is accomplished by multiplying this derivative by the maximum burnup credited in any storage configuration and converting to soluble boron using the data generated in Step 2.

4. Determine the largest increase in reactivity caused by postulated accidents and the corresponding amount of soluble boron needed to offset this reactivity increase.

An alternative form of expressing the soluble boron requirements is given in Reference 3. The final soluble boron requirement is determined from the following summation.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

Where:

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

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

SBC_{RE} = soluble boron required to account for burnup and reactivity depletion uncertainties (ppm).

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

For purposes of the analyses contained herein, minimum burnup limits established for fuel assemblies to be stored in the storage racks do include burnup credit established in a manner which takes into account approximations to the operating history of the fuel

assemblies. Variables such as the axial burnup profile as well as the axial profile of moderator and fuel temperatures have been factored into the analyses. Also, the axial reactivity effect associated with the absence of Gd_2O_3 at both ends of the fuel assembly was directly included in this analysis

1.2 Design Approach

The Soluble Boron Credit Methodology provides additional reactivity margin in the spent fuel storage analyses which may then be used to implement added flexibility in storage criteria and, for example, to eliminate the need to credit any of the degraded Boraflex. Boraflex in the spent fuel racks is not credited in this analysis.

All of the storage cells modeled in this analysis employ a realistic representation of the pitch between storage locations. The square storage cell pitch for the "All-Cell" and "3x3" fuel assembly storage configurations employed for this analysis is equal to 9.5 inches.

The selection of the design basis fuel assembly type was based on an evaluation of the variety of fuel assemblies employed in the reactor to date and selecting the most reactive type for a given fuel assembly storage configuration. The candidate fuel assembly types include the Westinghouse and Exxon 14x14 Standard (STD), the Westinghouse 14x14 Optimized (OFA), and the Exxon TOPROD fuel assembly designs. The Westinghouse 14x14 OFA fuel assembly has been evaluated to be the design basis fuel assembly to represent fresh fuel assemblies in the center location of the "3x3" fuel assembly storage configurations. The Westinghouse 14x14 Standard fuel assembly has been evaluated to be the design basis fuel assembly to represent discharged fuel assemblies in the "All-Cell", and peripheral locations of the "3x3" fuel assembly storage configurations. The most reactive moderator conditions (water density equal to 1.0 g/cc) will be employed for each fuel assembly storage configuration such that the analysis results are valid over the nominal spent fuel temperature range (50 to 150 degrees Fahrenheit).

The reactivity characteristics of the storage racks were evaluated using infinite lattice analyses; this environment was employed in the evaluation of the burnup limits versus initial enrichment as well as the evaluation of physical tolerances and uncertainties. A full spent fuel pool model was also employed to evaluate soluble boron worth, the reactivity worth of postulated accidents, and the multiplication factor for the zero soluble boron condition.

1.3 Methodology

This section describes the methodology employed to assure the criticality safety of the spent fuel pools and to define limits placed on fresh and depleted fuel assembly storage configurations. The analysis methodology employs: (1) SCALE-PC, a personal computer version of the SCALE-4.3 code system, as documented in Reference 6, with the updated SCALE-4.3 version of the 44 group ENDF/B-V neutron cross section library, and (2) the two-dimensional integral transport code DIT, Reference 7, with an ENDF/B-VI neutron cross section library.

SCALE-PC was used for calculations involving infinite arrays for the "All-Cell" and "3x3" fuel assembly storage configurations. In addition, it was employed in a full representation of spent fuel pool #2 to evaluate soluble boron worth and postulated accidents.

SCALE-PC, used in both the benchmarking and the fuel assembly storage configurations, includes the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. All references to KENO in the text to follow should be interpreted as referring to the KENO V.a module.

The DIT code is used for simulation of in-reactor fuel assembly depletion. The following sections describe the application of these codes in more detail.

1.3.1 SCALE-PC

The SCALE system was developed for the Nuclear Regulatory Commission to satisfy the need for a standardized method of analysis for evaluation of nuclear fuel facilities and shipping package designs. SCALE-PC is a version of the SCALE code system that runs on specific classes of personal computers.

1.3.2 Validation of SCALE-PC

Validation of SCALE-PC for purposes of fuel storage rack analyses is based on the analysis of selected critical experiments from two experimental programs. The first program is the Babcock & Wilcox (B&W) experiments carried out in support of Close Proximity Storage of Power Reactor Fuel, Reference 8. The second program is the Pacific Northwest Laboratory (PNL) Program carried out in support of the design of Fuel Shipping and Storage Configurations; the experiments of current interest to this effort are documented in Reference 9. Reference 10, as well as several of the relevant thermal experiment evaluations in Reference 11, were found to be useful in updating pertinent experimental data for the PNL experiments.

Nineteen experimental configurations were selected from the B&W experimental program; these consisted of the following experimental cores: Core X, the seven measured configurations of Core XI, Cores XII through XXI, and Core XIIIA. These analyses employed measured critical data, rather than the extrapolated configurations to a fixed critical water height reported in Reference 8, so as to avoid introducing possible biases or added uncertainties associated with the extrapolation techniques. In addition to the active fuel region of the core, the full environment of the latter region, including the dry fuel above the critical water height, was represented explicitly in the analyses.

The B&W group of experimental configurations employed variable spacing between individual rod clusters in the nominal 3 x 3 array. In addition, the effects of placing either SS-304 or Borated Aluminum plates of different boron contents in the water channels between rod clusters were measured. Table 1-1 summarizes the results of these analyses.

Eleven experimental configurations were selected from the PNL experimental program. These experiments included unpoisoned uniform arrays of fuel pins and 2 x 2 arrays of rod clusters with and without interposed SS-304 or B/Al plates of different blackness. As

in the case of the B&W experiments, the full environment of the active fuel region was represented explicitly. Table 1-2 summarizes the results of these analyses.

The approach employed for the determination of the mean calculational bias and the mean calculational variance is based on Criterion 2 of Reference 12. For a given KENO calculated value of k_{eff} and associated one sigma uncertainty, the magnitude of $k_{95/95}$ is computed by the following equation; by this definition, there is a 95 percent confidence level that in 95 percent of similar analyses the validated calculational model will yield a multiplication factor less than $k_{95/95}$.

$$k_{95/95} = k_{KENO} + \Delta k_{bias} + M_{95/95}\left(\sigma_m^2 + \sigma_{KENO}^2\right)^{1/2}$$

Where,

k_{keno} is the KENO calculated multiplication factor,

Δk_{bias} is the mean calculational method bias,

$M_{95/95}$ is the 95/95 multiplier appropriate to the degrees of freedom for the number of validation analyses, and is obtained from the Tables of Reference 13.

σ_m^2 is the mean calculational method variance deduced from the validation analyses,

σ_{KENO}^2 is the square of the KENO standard deviation.

$M_{95/95}(\sigma_m^2+\sigma_{KENO}^2)^{1/2}$ is equal to the methodology uncertainty

The equation for the mean calculational methods bias is as follows.

$$\Delta k_{bias} = \frac{1}{n}\sum_{i=1}^{n}(1-k_i)$$

Where

k_i is the i^{th} value of the multiplication factor for the validation lattices of interest.

The equation for the mean calculational variance of the relevant validating multiplication factors is as follows.

$$\sigma_m^2 = \frac{n\sum_{1}^{n}\left(k_i - k_{ave}\right)^2 \sigma_i^2}{(n-1)\sum_{1}^{n}\frac{1}{\sigma_i^2}} - \sigma_{ave}^2$$

where k_{ave} is given by the following equation.

$$k_{ave} = \frac{\sum_{1}^{n} \frac{k_i}{\sigma_i^2}}{\sum_{1}^{n} \frac{1}{\sigma_i^2}},$$

σ_{ave}^2 is given by the following equation.

$$\sigma_{ave}^2 = \frac{\sum_{1}^{n} \sigma_i^2 G_i}{\sum_{1}^{n} G_i},$$

G_i is the number of generations.

For purposes of this bias evaluation, the data points of Table 1-1 and Table 1-2 are pooled into a single group. With this approach, the mean calculational methods bias, Δk_{bias}, and the mean calculational variance, (σ_m^{2}), calculated by equations given above, are determined to be 0.00259 and $(0.002882)^2$, respectively. The magnitude of $M_{95/95}$ is obtained from Reference 13 for the total number of pooled data points, 30.

The magnitude of $k_{95/95}$ is given by the following equation for SCALE 4.3 KENO analyses employing the 44-group ENDF/B-V neutron cross section library and for analyses where these experiments are a suitable basis for assessing the methods bias and calculational variance.

$$k_{95/95} = k_{KENO} + 0.00259 + 2.22\left[(0.00288)^2 + \sigma_{KENO}^2\right]^{\frac{1}{2}}$$

Based on the above analyses, the mean calculational bias, the mean calculational variance, and the 95/95 confidence level multiplier are deduced as 0.00259, $(0.00288)^2$, and 2.22, respectively

1.3.3 Application to Fuel Storage Pool Calculations

As noted above, the CSAS25 control module was employed to execute the functional modules within SCALE-PC. The CSAS25 control module was used to analyze either infinite arrays of single or multiple storage cells or the full spent storage pool.

Standard material compositions were employed in the SCALE-PC analyses consistent with the design input given in Section 2.0; these data are listed in Table 1-3. For fresh fuel conditions, the fuel nuclide number densities were derived within the CSAS25 module using input consistent with the data of Table 1-3. For burned fuel representations, the fuel isotopics were derived from the DIT code as described below.

1.3.4 The DIT Code

The DIT (Discrete Integral Transport) code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The neutron transport equations are solved in integral form within each pin cell. The cells retain full heterogeneity throughout the discrete integral transport calculations. The multigroup spectra are coupled between cells through the use of multigroup interface currents. The angular dependence of the neutron flux is approximated at cell boundaries by a pair of second order Legendre polynomials. Anisotropic scattering within the cells, together with the anisotropic current coupling between cells, provide an accurate representation of the flux gradients between dissimilar cells.

The multigroup cross sections are based on the Evaluated Nuclear Data File Version 6 (ENDF/B-VI). Cross sections have been collapsed into an 89-group structure which is used in the assembly spectrum calculation. Following the multigroup spectrum calculation, the region-wise cross sections within each heterogeneous cell are collapsed to a few groups (usually 4 broad groups), for use in the assembly flux calculation. A B1 assembly leakage correction is performed to modify the spectrum according to the assembly in- or out-leakage. Following the flux calculation, a depletion step is performed to generate a set of region-wise isotopic concentrations at the end of a burnup interval. An extensive set of depletion chains are available, containing 33 actinide nuclides in the thorium, uranium and plutonium chains, 171 fission products, the gadolinium, erbium and boron depletable absorbers, and all structural nuclides. The spectrum-depletion sequence of calculations is repeated over the life of the fuel assembly. Several restart capabilities provide the temperature, density, and boron concentration dependencies needed for three-dimensional calculations with full thermal-hydraulic feedback effects.

The DIT code and its cross Section library are employed in the design of initial and reload cores and have been extensively benchmarked against operating reactor history and test data.

For the purpose of spent fuel pool criticality analysis calculations, the DIT code is used to generate the detailed fuel isotopic concentrations as a function of fuel burnup and initial feed enrichment. Each complete set of fuel isotopics is reduced to a smaller set of burned fuel isotopics at specified time points after discharge. The latter burned fuel representation includes the following nuclides: ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{149}Sm, ^{16}O, and ^{10}B. The DIT code lists the Samarium-149 isotopics for ^{149}Sm and ^{149D}Sm (a metastable isomer). Since ^{149}Sm is a stable isotope, the concentration of this Samarium isotope is the sum of the individual concentration of these two isomers.

The isotopic number densities from the DIT calculation are based upon pin cell averaged values. The input to KENO calculations requires that the number densities be specified for the fuel pellet. Therefore, the number densities from the DIT calculations are scaled by the ratio of area of the cell to the area of the fuel pellet for use in the KENO calculations. The concentration of ^{10}B supplied to KENO is such that the KENO and DIT assembly k_∞ values (at room temperature and unborated conditions) agree to within one sigma of the KENO calculation.

1.4 Assumptions

- The Westinghouse OFA was modeled as the design basis fuel assembly to conservatively represent all fuel assemblies residing in the center locations of the "3x3" fuel assembly storage configurations.

- The Westinghouse Standard fuel assembly was modeled as the design basis fuel assembly to conservatively represent all fuel assemblies residing in the "All-Cell" and peripheral locations of the "3x3" fuel assembly storage configurations.

- Fresh Standard and OFA fuel assemblies were conservatively modeled with a UO_2 density equal to 10.576 g/cc (96.5% of theoretical density). This translates into a pellet density equal 97.6% of theoretical density with a 1.1% dishing (void) fraction.

- All fuel assemblies, fresh and depleted, were conservatively modeled as containing solid right cylindrical pellets and uniformly enriched over the entire length of the fuel stack height. This conservative assumption bounds fuel assembly designs which incorporate lower enrichment blanket or annular pellets.

- All of the Boraflex poison material residing in the storage racks is conservatively omitted for this analysis .

- The intra module water gaps were conservatively modeled as 1.0 inches.

Table 1-1 Calculational Results for Cores X Through XXI of the B&W Close Proximity Experiments

Core	Run No.	KENO k_{eff}	Plate Type[1]	Spacing[2]
X	2348	0.99610 ± 0.00084	none	3
XI	2355	1.00049 ± 0.00080	SS-304	1
XI	2359	0.99884 ± 0.00077	SS-304	1
XI	2360	1.00315 ± 0.00081	SS-304	1
XI	2361	0.99831 ± 0.00080	SS-304	1
XI	2362	1.00060 ± 0.00078	SS-304	1
XI	2363	0.99957 ± 0.00078	SS-304	1
XI	2364	1.00246 ± 0.00080	SS-304	1
XII	2370	0.99990 ± 0.00082	SS-304	2
XIII	2378	0.99754 ± 0.00089	B/Al	1
XIIIA	2423	0.99575 ± 0.00087	B/Al	1
XIV	2384	0.99465 ± 0.00086	B/Al	1
XV	2388	0.99158 ± 0.00084	B/Al	1
XVI	2396	0.99230 ± 0.00088	B/Al	2
XVII	2402	0.99478 ± 0.00079	B/Al	1
XVIII	2407	0.99440 ± 0.00083	B/Al	2
XIX	2411	0.99821 ± 0.00081	B/Al	1
XX	2414	0.99498 ± 0.00082	B/Al	2
XXI	2420	0.99318 ± 0.00094	B/Al	3

[1] Entry indicates metal separating unit assemblies.

[2] Entry indicates spacing between unit assemblies in units of fuel rod pitch.

Table 1-2. Calculational Results for Selected Experimental PNL Lattices, Fuel Shipping and Storage Configurations

Experiment	k_{eff}	Comments
043	0.99787 ± 0.00106	Uniform rectangular array, no poison
044	1.00104 ± 0.00102	"
045	0.99955 ± 0.00101	"
046	0.99960 ± 0.00103	"
061	0.99792 ± 0.00099	2 x 2 array of rod clusters, no poison
062	0.99628 ± 0.00096	"
064	0.99696 ± 0.00103	2 x 2 array of rod clusters, 0.302 cm thick SS-304 cross
071	0.99970 ± 0.00101	2 x 2 array of rod clusters, 0.485 cm thick SS-304 cross
079	0.99463 ± 0.00102	2 x 2 array of rod clusters, cross of 0.3666 g boron/cm^2
087	0.99423 ± 0.00099	2 x 2 array of rod clusters, cross of 0.1639 g boron/cm^2
093	0.99787 ± 0.00098	2 x 2 array of rod clusters, cross of 0.1425 g boron/cm^2

Table 1-3. Standard Material Compositions Employed in Criticality Analysis for Prairie Island Units 1 & 2 Spent Fuel Storage Racks

Material	Element	Weight Fraction
Zircaloy-4, Density = 6.56 g/cm^3 @ 293.15 K	Zr	0.9829
	Sn	0.0140
	Fe	0.0021
	Cr	0.0010
4.0 w/o Gd_2O_3 @ 293.15 K	^{154}Gd	0.0007304
	^{155}Gd	0.0050360
	^{156}Gd	0.0070440
	^{157}Gd	0.0054304
	^{158}Gd	0.0086676
	^{160}Gd	0.0077296
	^{16}O	0.0053620
Water	SCALE Standard Composition Library Density = 1.0 g/cm^3 @ 293.15 K	
Stainless Steel	SCALE Standard Composition Library Density = 7.94 g/cm^3 @ 293.15 K	
Fresh UO_2	Fraction of Theoretical Density = 0.965 Enrichment = 4.95 w/o ^{235}U @ 293.15 K	
Regular Concrete	SCALE Standard Composition Library Density = 2.3 g/cm^3 @ 293.15 K	

2.0 Design Input

This section provides a brief description of the Prairie Island Units 1 & 2 spent fuel storage racks with the objective of establishing a basis for the analytical models employed in the criticality analyses described in Section 3.0.

2.1 Design Input from NMC

NMC provided Westinghouse a comprehensive package of design data related to the Prairie Island Units 1 & 2 spent fuel pools. This design input package includes the necessary data, drawings, or references required to develop the KENO models discussed herein. Specifically, it includes drawing NF-90044 which was employed to develop the KENO model for spent fuel pool #2. The nominal storage cell dimensions were obtained from drawing NF-90046 provided by NMC. Note that drawing NF-90044 contains a typographical error. Module 90047-11 is labeled as a 7x8 module in drawing NF-90044. It is actually a 7x7 module.

2.2 Spent Fuel Pool Storage Configuration Description

There are two spent fuel pools which provide storage for Prairie Island Units 1 & 2. Spent fuel pool #1 is the small pool, and spent fuel pool #2 is the large pool. Spent fuel pool #1 contains 9 spent fuel storage modules; there are six 7x7 modules and three 7x8 modules. Spent fuel pool #2 contains 21 spent fuel storage modules; there are seven 7x7 modules, ten 7x8 modules, and four 8x7 modules. The modules are separated by a minimum water gap of 1 inch. Spent fuel pool #2 has a liner inside dimension equal to 227 inches in the north to south direction and 521 inches in the west to east direction. The modules in spent fuel pool #2 are located 2 inches from the southwest corner. Figure 2-2 displays the arrangement of the spent fuel pool storage modules and was produced by scanning drawing NF-90044. Table 2-2 summarizes the overall geometry data for the Prairie Island Units 1 & 2 spent fuel pool #2.

2.3 Individual Storage Cell Description

The nominal storage cell is centered on a pitch equal to 9.5 inches. Each storage cell consists of an inner stainless steel canister and outer stainless steel sheathing. The original Boraflex material (not modeled in this analysis) was located in the cavity between the inner canister and outer sheathing. The inner stainless steel canister has a nominal inside dimension equal to 8.27 inches and is 0.09 inches thick. The outer stainless steel sheathing has an inside dimension equal to 8.70 inches and is 0.024 inches thick.

The nominal storage rack dimensions are summarized in Table 2-1. The nominal rack dimensions are reported with manufacturing tolerances, where available. Figure 2-1 displays the Prairie Island Units 1 & 2 storage cell geometry. Figure 2-3 displays the dimensions of the individual storage cell and was produced by scanning drawing NF-90046.

Table 2-1. Prairie Island Units 1 & 2 Storage Cell Dimensions (All dimensions given in inches)

Parameter	Value
Nominal Cell Pitch	9.50 ± 0.06
Box Wall Thickness	0.09 ± 0.01
Box ID	8.27 ± 0.10
Boraflex Cavity Width	8.20
Boraflex Cavity Thickness	0.125
Sheathing Thickness	0.024

Table 2-2. Prairie Island Units 1 & 2 Spent Fuel Pool #2 Dimensions (All dimensions given in inches)

Parameter	Value
Pool Length	227
Pool Width	521
Intra Module Gap[3]	1.0
Wall / Module Gap in SW Corner	2.0
Wall Thickness	24

[3] The intra module water gap is conservatively modeled as 1 inch.

Figure 2-1. Prairie Island Units 1 & 2 Storage Cell

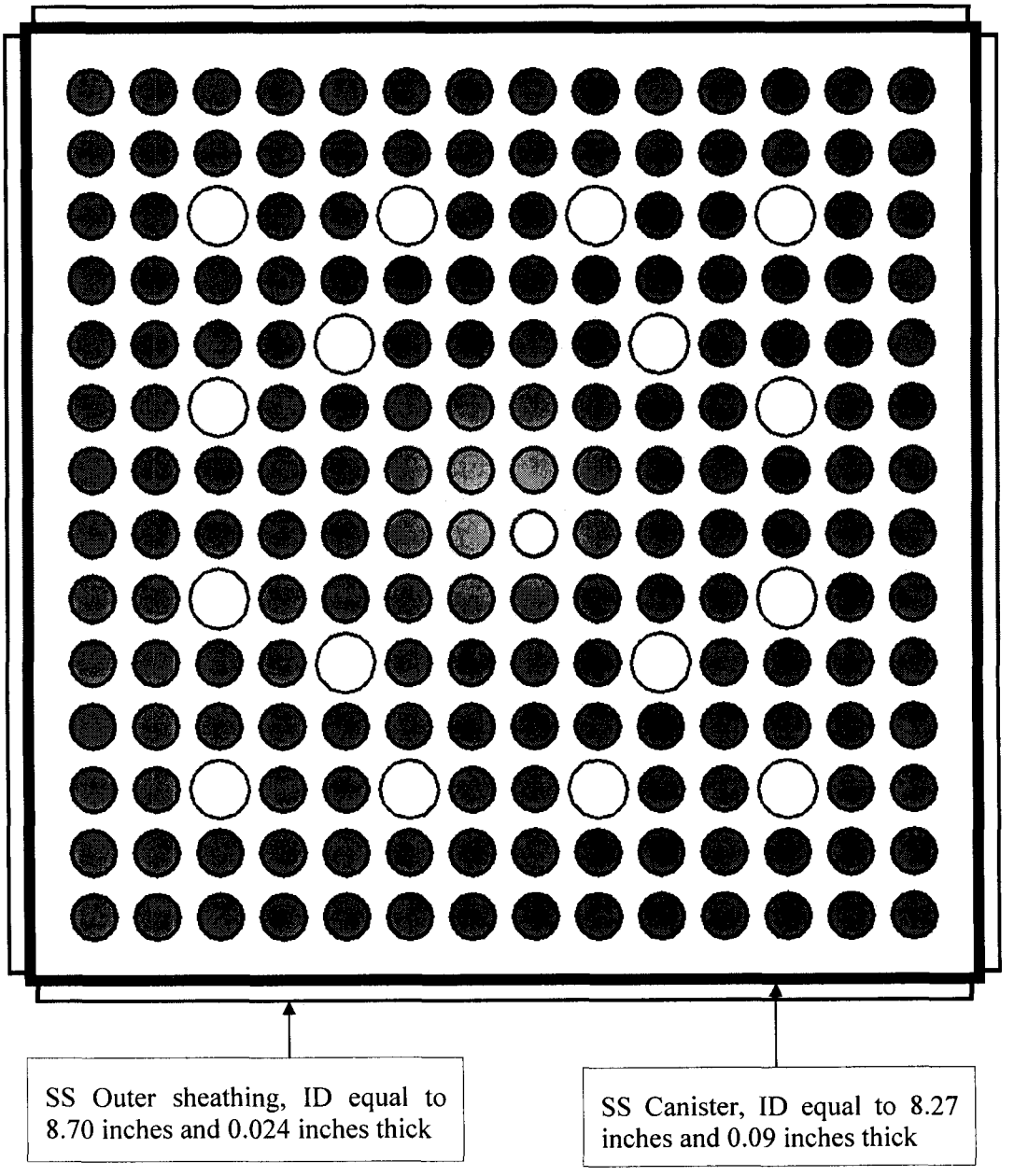


Figure 2-2. Prairie Island Units 1 & 2 Spent Fuel Pool

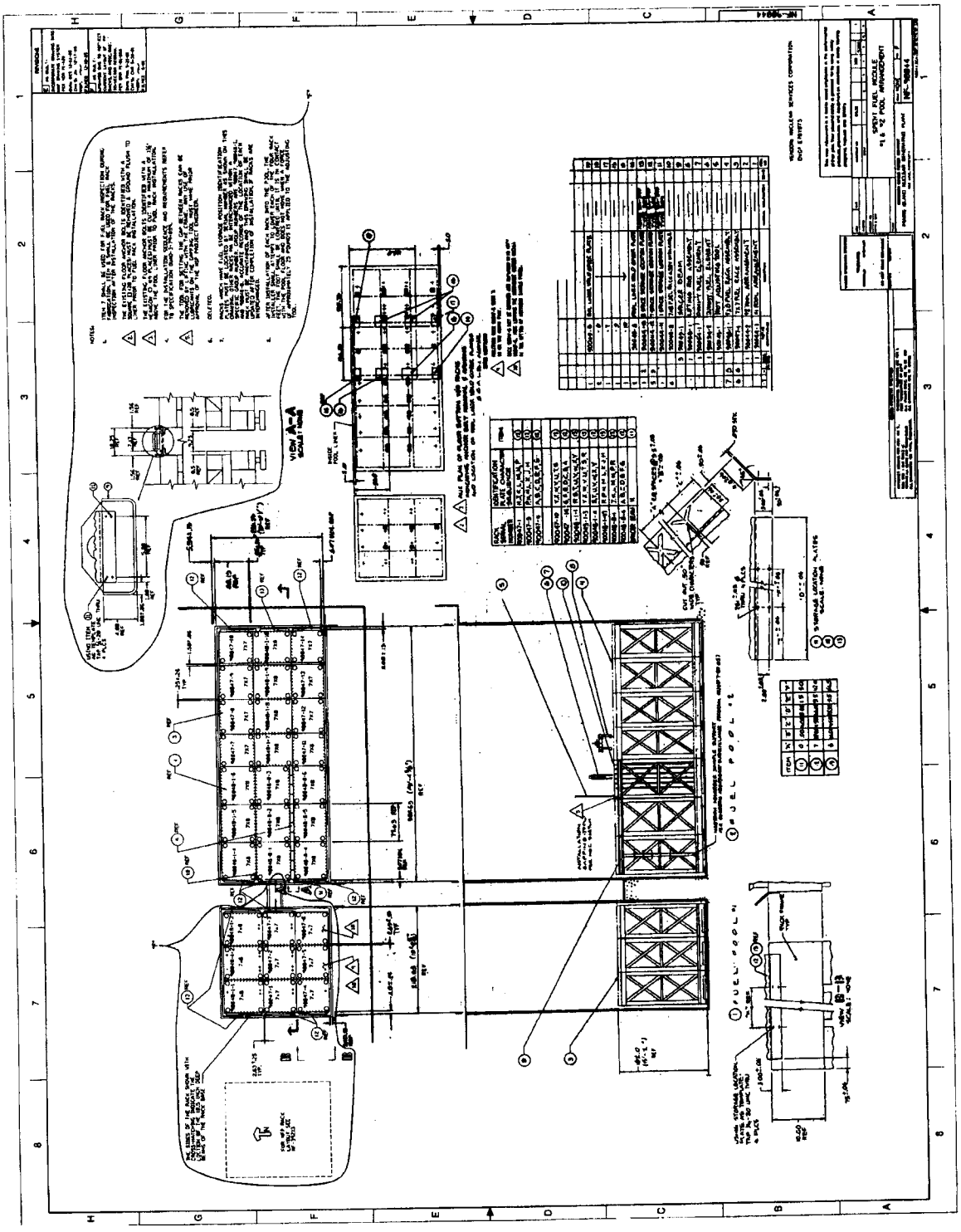
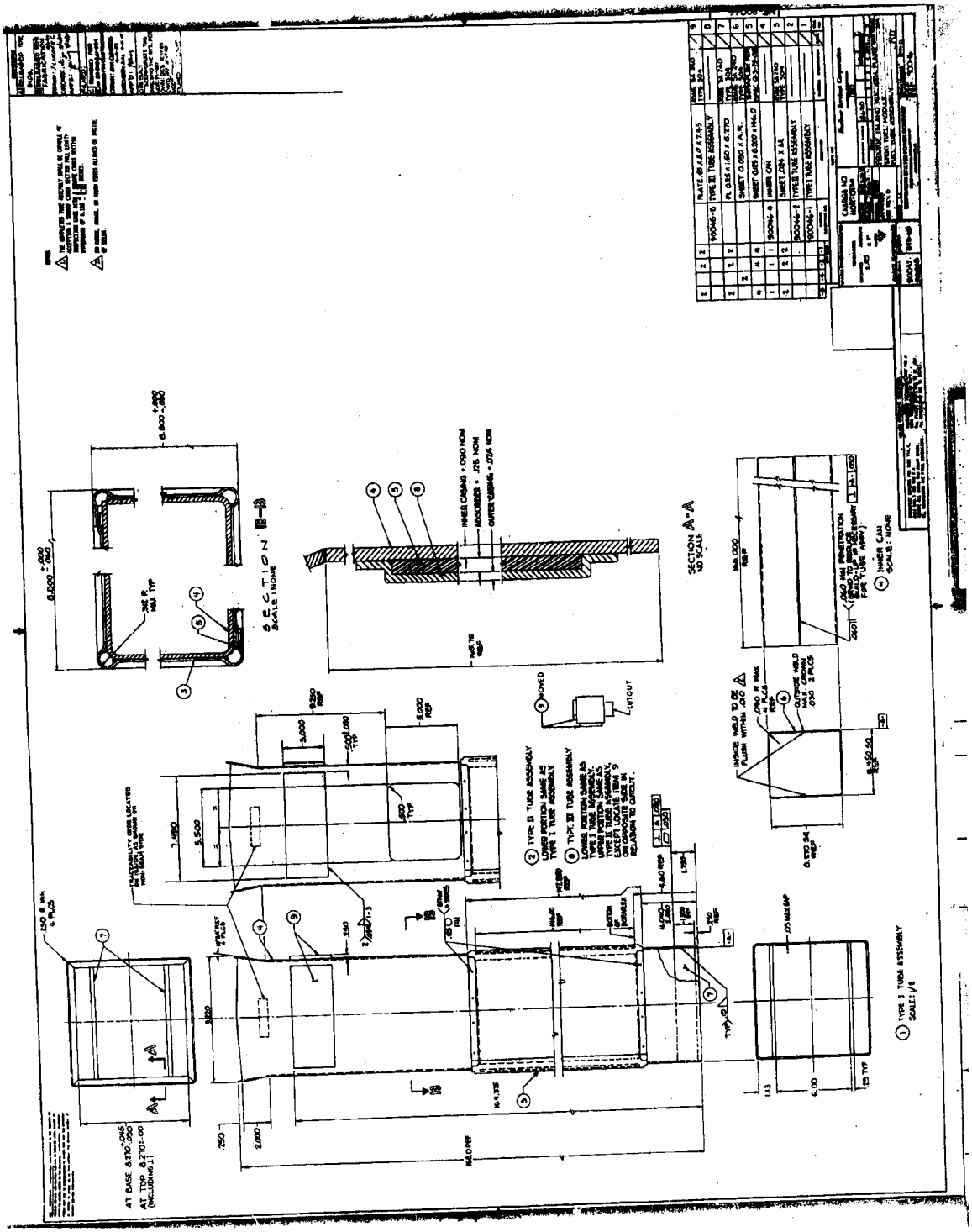


Figure 2-3. Prairie Island Units 1 & 2 Assembly Storage Cell



3.0 Analysis

3.1 KENO Models

The Prairie Island Units 1 & 2 spent fuel storage racks employ two different fuel assembly storage configurations; namely the "All-Cell" and "3x3" fuel assembly storage configurations. The "3x3" fuel assembly storage configuration is analyzed with and without credit for Gd_2O_3 burnable absorbers. The purpose of this section is to describe the models employed in KENO to represent these assembly storage configurations and spent fuel pool #2.

3.1.1 KENO Model for the "All-Cell" Fuel Assembly Storage Configuration

An "All-Cell" fuel assembly storage configuration is modeled in KENO as an infinitely repeating storage cell that contains either a fresh or depleted fuel assembly. An inner stainless steel canister controls the fuel assembly position.

Each cell location is modeled in KENO as a square cell with a pitch equal to 9.50 inches. The inner stainless steel canister is modeled with an inside dimension equal to 8.27 inches and is 0.09 inches thick. The outer stainless steel sheathing is modeled with an inside dimension equal to 8.70 inches and is 0.024 inches thick. The cavity between the canister and outer sheathing is modeled with water. All of these dimensions employed to model the Prairie Island Units 1 & 2 storage cell are consistent with the values given in Table 2-1.

The fuel assembly, inner stainless steel canister, and outer stainless steel sheathing are modeled in KENO as 144 inches tall. Reflective boundary conditions are applied to the X and Y surfaces of the assembly, thus simulating an infinitely repeating array. A two-foot water reflector is modeled above and below the storage cell geometry. The pool water is simulated to be full density (1 g/cm^3) at room temperature (68 °F). The top and bottom surfaces of the water reflector have reflected boundary conditions.

The fuel assembly modeled in KENO represents the Westinghouse 14x14 Standard design. Note that the fuel pellets in a fuel rod are modeled as a fully enriched right solid cylinder that is 144 inches tall. This assumption conservatively bounds fuel rod designs that incorporate annular and/or lower enrichment fuel pellets such as those employed for axial blankets. A top down image of a KENO produced plot of a single "All-Cell" fuel assembly storage configuration is shown in Figure 3-4.

3.1.2 KENO Model for the "3x3" Fuel Assembly Storage Configuration

The "3x3" fuel assembly storage configuration is modeled in KENO as a repeating 3x3 array with a fresh fuel assembly occupying the center location of the array and the remaining locations are occupied by discharged fuel assemblies. This storage configuration is analyzed with and without credit for Gd_2O_3 burnable absorbers. An inner stainless steel canister controls each fuel assembly position within the array.

Each of the nine storage cell locations is modeled in KENO as a square cell with a pitch equal to 9.50 inches. The inner stainless steel canister is modeled with an inside dimension equal to 8.27 inches and is 0.09 inches thick. The outer stainless steel sheathing is modeled with an inside dimension equal to 8.70 inches and is 0.024 inches thick. The cavity between the canister and outer sheathing is modeled with water. All of these dimensions employed to model the Prairie Island Units 1 & 2 storage cell are consistent with the values given in Table 2-1.

The fuel assembly, inner stainless steel canister, and outer stainless steel sheathing are modeled in KENO as 144 inches tall. Reflective boundary conditions are applied to the X and Y surfaces of the 3x3 array, thus simulating an infinitely repeating "3x3" fuel assembly storage configuration. A two-foot water reflector is modeled above and below the storage cell geometry. The pool water is simulated to be full density (1 g/cm^3) at room temperature (68 °F). The top and bottom surfaces of the water reflector have reflected boundary conditions.

The center fuel assembly that is modeled in KENO represents the Westinghouse 14x14 OFA design. The enrichment of all fuel pellets is equal to 4.95 w/o ^{235}U (with no Gd_2O_3 credit) and the pellet density is equal to 96.5% of theoretical density. The remaining fuel assemblies that are modeled by KENO represent the Westinghouse 14x14 Standard design. Note that the fuel pellets in a fuel rod are modeled as fully enriched right solid cylinders that are 144 inches tall. This assumption conservatively bounds fuel rod designs which incorporate annular and/or lower enrichment fuel pellets such as those employed for axial blankets. A top down image of a KENO produced plot of a single "3x3" fuel assembly storage configuration is shown in Figure 3-5.

Storage of fresh fuel assemblies with Gd_2O_3 burnable absorbers in the center location of the 3x3 array allow for storage of more reactive fuel assemblies on the periphery than is allowed by the configuration described above. Gd_2O_3 credit accounts for the reactivity decrease associated with the addition of a neutron poison material. The following assumptions are used to represent the Gd_2O_3 pellets in the KENO model of the 3x3 storage region.

- A 6 inch burnable absorber cutback (top and bottom) is used. This produces a 132 inch shimmed length that is centered about the active fuel height.
- The Gd_2O_3 amount is limited to four fuel pins at a concentration of 4.0 w/o Gd_2O_3 in Gd_2O_3-UO_2. The pin placement is shown in Figure 3-2.
- The ^{235}U enrichment is reduced to 4.0 w/o in the shimmed portion of the fuel pin for fuel temperature considerations.
- The ^{235}U enrichment in the blanket region of the shimmed fuel pins is also reduced to 4.0 w/o.
- The density of the UO_2 and Gd_2O_3 mixture is found with the following empirical expression (Reference 18),

$$\rho_{U+Gd} = \rho_{U_{th}} \cdot (1 - 0.00301 \cdot \mathrm{X})$$

where,

$$\rho_{U+Gd} \equiv density\ of\ UO_2\ and\ Gd_2O_3\ mixture$$
$$\rho_{U_{th}} \equiv theoretical\ UO_2\ density$$
$$\mathrm{X} \equiv Gd_2O_3\ concentration\ in\ weight\ percentage$$

The calculation performed for this analysis is as follows,

$$\rho_{U_{th}} = 10.5764\ g \cdot cm^{-3}\ (96.5\%\ T.D.) \quad and \quad \mathrm{X} = 4.0\ w/o,$$
$$\therefore \quad \rho_{U+Gd} = 10.5764 \cdot (1 - 0.00301 \cdot 4.0) = 10.4491\ g \cdot cm^{-3}$$

This value is utilized in the "3x3" storage configuration KENO models with Gd_2O_3 credit.

3.1.3 KENO Model for Entire Spent Fuel Pool

There is a relatively large amount of leakage in the Prairie Island spent fuel pool #1 (the small pool), therefore only spent fuel pool #2 (the large pool) need be modeled for conservatism. Spent fuel pool #2 is modeled in KENO as a rectangular water cell that is 521 inches in the west to east direction and 227 inches in the north to south direction. The floor and sides of the spent fuel pool are modeled by surrounding the rectangular water cell with two feet of concrete on the bottom and sides.

Twenty one (21) fuel storage modules are inside the spent fuel pool #2 rectangular water cell. The fuel storage modules vary in size from a 7x7 to a 7x8/8x7 array of storage cells. All of the individual assembly storage cells were modeled exactly the same and as described in Sections 3.1.1 through 3.1.2. The minimum intra module water gap of 1.0 inch was modeled conservatively. The fuel storage rack modules are placed within 2.0 inches from the southwest corner of the spent fuel pool liner. Note that a 2 inch gap of water between the modules and pool wall is maintained on the south and west faces and the intra module water gap shown by section A-A in Figure 2-2 is modeled as 9.75 inches wide. These pool dimensions are shown in Table 2-2. The pool water was modeled at room temperature conditions, 68 °F, and as full density (1.0 g/cc).

The storage modules are modeled with both the "All-Cell" and "3x3" storage configurations. Figures 3-6 and 3-7 show KENO produced plots of the spent fuel pool loaded with these storage configurations. These arrangements conform to the restrictions outlined in sections 3.1.1 and 3.1.2. No Gd_2O_3 burnable absorber credit is modeled in this portion of the analysis.

3.2 Design Basis Fuel Assembly

Prairie Island Units 1 & 2 have been in operation for many years and during that time interval a variety of reload batches containing different fuel assembly designs have been cycled through the reactors. Thus, the criticality safety analysis of their spent fuel pool

must take into account possible differences in the reactivity characteristics of the different assembly types. For purposes of this analysis, the different types of fuel assemblies were surveyed so as to define a reference design fuel assembly that would assure conservative results for the analysis.

Table 3-1 provides the relevant dimensions employed to model the Westinghouse 14x14 Standard and Westinghouse 14x14 OFA fuel assemblies in the spent fuel pool environment. Figure 3-1 displays the Westinghouse 14x14 fuel assembly with both the OFA and STD parameters. Based on the results of scoping calculations for the ^{235}U loading and storage configuration considered here, the most reactive fresh fuel assembly design is the Westinghouse 14x14 OFA fuel assembly for the center location of the "3x3" fuel assembly storage configuration. The Westinghouse Standard fuel assembly design was modeled as the design basis fuel assembly to conservatively represent discharged fuel assemblies residing in the "All-Cell" and peripheral locations of the "3x3" fuel assembly storage configurations. Other fuel assembly designs are found to be less reactive in these fuel assembly storage configurations than the design basis fuel assemblies.

The unshimmed design basis fuel assemblies are modeled with the maximum enrichment over the active fuel length. The fresh fuel pellets in a fuel rod are modeled as a solid right cylinder with a UO_2 density equal to 10.576 g/cc (96.5% of theoretical density). No credit is taken for the nominal 1.1 to 1.2 void fraction percentages that are associated with dishing or chamfering. In addition, no credit is taken for any natural or reduced enrichment pellets. These assumptions result in equivalent or conservative calculations of reactivity for all fuel assemblies used at Prairie Island Units 1 & 2, including those with annular pellets or lower enrichment pellets at the ends of the fuel rods. No credit is taken for any spacer grids or sleeves.

The shimmed fuel assemblies are of the Westinghouse OFA design and incorporate the design features outlined above in section 3.1.2.

3.3 Modeling of Axial Burnup Distributions

A key aspect of the burnup credit methodology employed in this analysis is the inclusion of an axial burnup profile correlated with feed enrichment and discharge burnup of the burned fuel assemblies. This effect is important in the analysis of the spent fuel pool characteristics since the majority of spent fuel assemblies stored in the pool have a discharge burnup well beyond the limit for which the assumption of a uniform axial burnup shape is conservative. Therefore, it is necessary to represent the burnt fuel assembly with a representative axial burnup profile.

For any given spent fuel assembly, the fuel burnup is a continuous function of axial position. However, from a computational point of view, this function can be discretized in such a manner that the axial "end-effect" is adequately captured. It is often common practice to divide the fuel assembly into several axial zones with each zone assumed to be uniform in burnup. Moreover, it is required that the size of the top and bottom axial zones be small (typically less than 8 inches) so as to capture the steep burnup gradient with axial position while that of the central zone may be larger. In spent fuel pool calculations,

an eight-zone axial model is found to be adequate (Reference 19) to represent the spent fuel assembly. Such an eight-zone model would have seven zones with fine mesh spacing (four at the top of the fuel assembly and three at the bottom) and the remaining zone represents the center portion of the fuel assembly. Figure 3-3 provides a pictorial view of the axial zones employed in the eight-zone axial model.

The individual power fractions of each zone are so modeled that they give the same volume averaged burnup when compared to a uniform burnup model. This model is validated due to the fact that the relative contribution of the bottom zones of the fuel assembly to the k_{eff} value is negligible.

Input to this analysis is based on the limiting axial burnup profile data provided in the DOE Topical Report, as documented in Reference 14. The burnup profile in the DOE Topical Report is based on a database of 3169 axial-burnup profiles for PWR fuel assemblies compiled by Yankee Atomic. This profile is derived from the burnups calculated by utilities or vendors based on core-follow calculations and in-core measurement data. The axial burnup profile in the DOE report is based on the most limiting axial burnup shape found in the database. The eight-zone model is constructed based on this limiting axial burnup profile.

DIT was used to generate the isotopic concentrations for each segment of the axial profile. Table 3-2 lists the fuel and moderator temperatures employed in the spectral calculations for each node of the eight-zone axial burnup model. These values are based on mid-cycle temperature profiles for Prairie Island Units 1 and 2. The fuel temperatures for each axial zone are calculated based on a representative fuel temperature correlation while the moderator temperatures are based on a linear relationship with axial position. These node dependent moderator and fuel temperature data and power profile data were employed in DIT to deplete the fuel to the desired burnup for each initial enrichment and axial zone. The values of assembly average burnups versus feed enrichment for which burned fuel assemblies were simulated are tabulated in Table 3-3.

A constant soluble boron concentration of 800 ppm was employed in all the burnup steps. This value is representative of a cycle average soluble boron concentration in the core. For the purpose of extracting the number densities, the DIT computer code was executed in two modes. First, a normal depletion was continued in steps of 1000 MWD/MTU (with respect to the assembly average case) until the desired burnup was reached. Then a restart is performed at cold, spent fuel pool conditions and the fuel assembly is allowed to decay for 100 hours. At this point of time, the reactivity of the burned fuel assembly is at its highest. The k_∞ and the isotopic number densities are then extracted for the KENO model development at these assembly conditions.

The DIT computed isotopic concentrations were transferred into the KENO models of the storage cells using a limited set of isotopes. That is, the ^{235}U, ^{238}U, ^{236}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{16}O, and equilibrium ^{149}Sm at shutdown are represented explicitly in the KENO models. All other fission product isotopic number densities are represented by an equivalent ^{10}B concentration; the magnitude of this concentration is determined by matching the DIT k_{eff} value with the KENO k_{eff} value to a one sigma tolerance level.

Reference 19 contains a listing of the isotopic number densities employed in the KENO calculations. The format of the listing is compatible with the KENO input description and can directly be used as part of KENO input for material specification. The isotopic number densities are listed for the combination of initial enrichment and burnup listed in Table 3-3. The listing is for the Westinghouse 14x14 Standard fuel assembly design.

Reference 19 also contains a listing of the ^{10}B number densities determined by matching the DIT k_{eff} and KENO calculated k_{eff} values. The ^{10}B number density, the DIT calculated k_{eff} and the KENO calculated k_{eff}, for the eight-zone axial model (and the average fuel assembly model) are listed in each table. The first four tables contain these values for 3.0 w/o, the next four tables contain the data for 4.0 w/o, and the final four tables contain data for 5.0 w/o.

3.4 Tolerance / Uncertainty Calculations

Previous sections described the storage racks and fuel assembly storage configurations within the spent fuel pool and the KENO models employed to represent repeating arrays of these fuel assembly storage configurations. In addition, the method of modeling the axial profiles of fuel assembly burnup, moderator temperature, and fuel temperature were discussed.

Using the above input, analytic models were developed to perform the quantitative evaluations necessary to demonstrate that the effective multiplication factor for the spent fuel pool is less than 0.995 with zero soluble boron present in the pool water. Applicable biases to be factored into this evaluation are: (1) the methodology bias deduced from the validation analyses of pertinent critical experiments, and (2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range of 50 °F to 150 °F (from Reference 20).

A second allowance is based on a 95/95 confidence level assessment of tolerances and uncertainties; included in the summation of variances are the following.

(a) the 95/95 confidence level methods variance,

(b) the 95/95 confidence level calculational uncertainty,

(c) fuel rod manufacturing tolerances,

(d) storage rack fabrication tolerances,

(e) tolerances due to positioning the fuel assembly in the storage cell.

(f) burnup uncertainty

(g) burnable absorber concentration (if applicable)

Items a) and b) are based on the calculational methods validation analyses. For Item c), the fuel rod manufacturing tolerance for the reference design fuel assembly is assumed to consist of two components; an increase in fuel enrichment equal to 0.05 w/o ^{235}U and an

increase in pellet density from 96.5 to 98.5% of theoretical density; the individual contributions of each change are combined by taking the square root of the sum of the squares of each component. There is no allowance for dishing and chamfer and therefore the pellet density conservatively represents the stack density of the UO_2 pellets in the fuel rod.

For item d), the following uncertainty components were evaluated. The inner stainless steel canister ID was increased from 8.27 inches to 8.37 inches and the thickness of the canister was decreased from 0.09 inches to 0.08 inches. The storage cell pitch for the "All-Cell" and "3x3" fuel assembly storage configurations was decreased from 9.50 inches to 9.44 inches.

In the case of the tolerance due to positioning of the fuel assembly in the storage cells (item e), all nominal calculations are carried out with fuel assemblies conservatively centered in the storage cells. One case was run to investigate the effect of off-center position of the fuel assemblies for each of the fuel assembly storage configurations. These cases positioned the assemblies as close as possible in four adjacent storage cells. Eccentric positioning has a slightly positive reactivity effect for all of the fuel assembly storage configurations.

For item f), a 5% burnup uncertainty is included. The 5% burnup uncertainty is applied to the fuel assembly storage configurations that contain depleted fuel assemblies.

For item g), the nominal gadolinia concentration is equal to 4.0 wt %. The tolerance analyzed for the gadolinia concentration is equal to -0.2 wt %.

Table 3-4, Table 3-5, and Table 3-6 provide a summary of the KENO results used in the calculation of biases and uncertainties for the "All-Cell", "Unshimmed 3x3", and "Shimmed 3x3" fuel assembly storage configurations, respectively. The total biases and uncertainties for these fuel assembly storage configurations are 0.02678, 0.02403, and 0.02816 Δk_{eff} units respectively.

3.5 No Soluble Boron 95/95 k_{eff} Calculational Results

The purpose of the following five subsections is to present the KENO calculated multiplication factors for the "All-Cell" and "3x3" fuel assembly storage configurations along with the result for the entire spent fuel pool at the zero soluble boron condition.

Due to the burnup requirements for storage in these configurations, ^{241}Pu decay and ^{241}Am production burnup credit is included. The concentrations for ^{241}Pu are decayed using the equation below and a half life, $t_{1/2}$, value of 14.4 years.

$$n = n_0 \cdot e^{-\frac{(\ln 2) \cdot t}{t_{1/2}}}$$

The production rate for ^{241}Am is equal to the rate of ^{241}Pu decay, using an initial ^{241}Am concentration, n_o, of zero. The decay time, t, extends 20 years in intervals of 5 years.

3.5.1 "All-Cell" Fuel Assembly Storage Configuration

As described in Section 3.1.1, the "All-Cell" fuel assembly storage configuration consists of an infinitely repeating storage cell that contains either fresh or depleted fuel assemblies. The fuel assembly modeled in this analysis is the Westinghouse Standard fuel assembly design.

k_{eff} was evaluated for an infinite array of "All-Cell" storage locations over a range of initial enrichment values up to 5.0 w/o ^{235}U and assembly average burnups up to 45.0 GWD/T. These calculations were performed at 68 °F, with maximum water density equal to 1.0 g/cc, to maximize the array reactivity. KENO calculations were performed for this fuel assembly storage configuration with an axially distributed burnup profile. The relative axial burnup profile employed for these calculations is discussed in Section 3.3. These resulting KENO calculated k_{eff} data are then employed to determine the burnup versus initial enrichment limits for a target k_{eff} value at zero soluble boron. The target value of k_{eff} is selected to be less than 0.995 by an amount sufficient to cover the magnitude of the analytical biases and uncertainties in these analyses. From Table 3-4, the sum of the biases and uncertainties is equal to 0.02678. Therefore, the target k_{eff} value for the "All-Cell" fuel assembly storage configuration is equal to 0.96822 (0.995-0.02678).

Table 3-7 lists the KENO calculated k_{eff} values for the "All-Cell" fuel assembly storage configuration versus initial enrichment and fuel assembly average burnup for an axially distributed burnup profile. The first entry in each of these tables lists the initial enrichment for no assembly burnup. Based upon the target k_{eff} value, the interpolated enrichment for no assembly burnup is equal to 1.79 w/o ^{235}U. The derived burnup limits, for enrichments greater than 1.79 w/o ^{235}U, are based upon the KENO calculated k_{eff} values for 3.0, 4.0, and 5.0 w/o ^{235}U. For each of these enrichments, KENO calculations were performed at three assembly average burnup values for an axially distributed burnup profile. A second degree fit of the burnup versus k_{eff} data was then employed to determine the burnup required to meet the target k_{eff} value of 0.96822. The resulting assembly burnup versus initial enrichment storage limits are provided in Table 3-10. The first entry in these tables lists the initial enrichment, 1.79 w/o ^{235}U, for fuel assemblies at zero burnup. The data in this table is plotted in Figure 4-5. The required assembly burnups as a function of initial enrichment were fitted to third degree polynomials. These polynomials are given in Table 4-1 and will be used to determine the burnup as a function of initial enrichment.

3.5.2 "3x3" Fuel Assembly Storage Configuration

As described in Section 3.1.2, the "3x3" fuel assembly storage configuration consists of a repeating 3x3 array with a fresh fuel assembly occupying the center location of the array and the remaining locations are occupied by discharged assemblies. The center assembly is the Westinghouse OFA design and the peripheral assemblies are the Westinghouse Standard design. The unshimmed case contains no Gd_2O_3 burnable absorbers, and the shimmed case contains four Gd_2O_3 burnable absorber pins at a concentration of 4.0 w/o.

k_{eff} was evaluated for an infinite array of "Unshimmed 3x3" storage locations over a range of initial enrichment values up to 5.0 w/o ^{235}U and assembly average burnups up to 55.0 GWD/T. These calculations were performed at 68 °F, with maximum water density equal to 1.0 g/cc, to maximize the array reactivity. KENO calculations were performed for this fuel assembly storage configuration with an axially distributed burnup profile. The relative axial burnup profile employed for these calculations is discussed in Section 3.3. These resulting KENO calculated k_{eff} data are then employed to determine the burnup versus initial enrichment limits for a target k_{eff} value at zero soluble boron. The target value of k_{eff} is selected to be less than 0.995 by an amount sufficient to cover the magnitude of the analytical biases and uncertainties in these analyses. From Table 3-5, the sum of the biases and uncertainties is equal to 0.02403. Therefore, the target k_{eff} value for the "Unshimmed 3x3" fuel assembly storage configuration is equal to 0.97097 (0.995-0.02403).

Table 3-8 lists the KENO calculated k_{eff} values for the "Unshimmed 3x3" fuel assembly storage configuration versus initial enrichment and fuel assembly average burnup for an axially distributed burnup profile. The first entry in these tables lists the initial enrichment for no assembly burnup. Based upon the target k_{eff} value, the interpolated enrichment for no assembly burnup is equal to 1.30 w/o ^{235}U. The derived burnup limits, for enrichments greater than 1.30 w/o ^{235}U, are based upon the KENO calculated k_{eff} values for 3.0, 4.0, and 5.0 w/o ^{235}U. For each of these enrichments, KENO calculations were performed at three assembly average burnup values with an axially distributed burnup profile. A second degree fit of the burnup versus k_{eff} data was then employed to determine the burnup required to meet the target k_{eff} value equal to 0.97097. The resulting assembly burnup versus initial enrichment storage limits are provided in Table 3-11. The first entry in these tables lists the initial enrichment, 1.30 w/o ^{235}U, for fuel assemblies at zero burnup. The data in this table is plotted in Figure 4-6. The required assembly burnups as a function of initial enrichment were fitted to third degree polynomials. These polynomials are given in Table 4-2 and will be used to determine the burnup as a function of initial enrichment.

k_{eff} was also evaluated for an infinite array of "Shimmed 3x3" storage locations over a range of initial enrichment values up to 5.0 w/o ^{235}U and assembly average burnups up to 45.0 GWD/T. These calculations were performed at 68 °F, with maximum water density equal to 1.0 g/cc, to maximize the array reactivity. KENO calculations were performed for this fuel assembly storage configuration with an axially distributed burnup profile. The relative axial burnup profile employed for these calculations is discussed in Section 3.3. These resulting KENO calculated k_{eff} data are then employed to determine the burnup versus initial enrichment limits for a target k_{eff} value at zero soluble boron. The target value of k_{eff} is selected to be less than 0.995 by an amount sufficient to cover the magnitude of the analytical biases and uncertainties in these analyses. From Table 3-6, the sum of the biases and uncertainties is equal to 0.02816. Therefore, the target k_{eff} value for the "shimmed 3x3" fuel assembly storage configuration is equal to 0.96684 (0.995-0.02816).

Table 3-9 lists the KENO calculated k_{eff} values for the "Shimmed 3x3" fuel assembly storage configuration versus initial enrichment and fuel assembly average burnup for both a uniform and axially distributed burnup profile. The first entry in these tables lists

the initial enrichment for no assembly burnup. Based upon the target k_{eff} value, the interpolated enrichment for no assembly burnup is equal to 1.39 w/o ^{235}U. The derived burnup limits, for enrichments greater than 1.39 w/o ^{235}U, are based upon the KENO calculated k_{eff} values for 3.0, 4.0, and 5.0 w/o ^{235}U. For each of these enrichments, KENO calculations were performed at three assembly average burnup values for an axially distributed burnup profile. A second degree fit of the burnup versus k_{eff} data was then employed to determine the burnup required to meet the target k_{eff} value equal to 0.96684. The resulting assembly burnup versus initial enrichment storage limits are provided in Table 3-12. The first entry in these tables lists the initial enrichment, 1.39 w/o ^{235}U, for fuel assemblies at zero burnup. The data in this table is plotted in Figure 4-7. The required assembly burnups as a function of initial enrichment were fitted to third degree polynomials. These polynomials are given in Table 4-3 and will be used to determine the burnup as a function of initial enrichment.

3.5.3 Entire Spent Fuel Pool

KENO models for the entire Prairie Island spent fuel pool #2 were constructed for this analysis and are shown in Figures 3-6 and 3-7. Figure 3-6 displays the KENO model for spent fuel #2, based upon the "All-Cell" storage configuration, with the 2 inch wall gap maintained on the south and west sides of spent fuel pool #2. Figures 3-7 illustrate the same KENO models based upon the "Unshimmed 3x3" storage configuration. These spent fuel pool KENO models are described in section 3.1.3. The largest KENO calculated multiplication factors for the spent fuel pool models and the respective infinite array models are shown in Table 3-13, and are based upon no soluble boron. The differences in the infinite array and spent fuel pool model's k_{eff} values are attributed to neutron leakage from the spent fuel #2 model. The biases and uncertainties, from Table 3-4 and Table 3-5, were added to the spent fuel pool multiplication factors and the results are shown in Table 3-13. As can be seen from Table 3-13, the final $k_{95/95}$ values at zero soluble for spent fuel pool #2 are all below the design basis limit equal to 0.995 at zero soluble boron.

The interface between the "All-Cell" and "3x3" storage configurations was directly simulated in a KENO model for spent fuel pool #2. The interface modeled is depicted in Figure 4-4. Note that the KENO calculated multiplication factor for this interface model is 0.96346 +/- 0.00038. This value is less than the value given in Table 3-13 for the "3x3" storage configuration. Therefore, the interface configuration (with biases and uncertainties) also meets the design basis limit equal to 0.995 at zero soluble boron.

3.6 Soluble Boron

The NRC Safety Evaluation Report (SER) for WCAP-14416-P is given in Reference 3; page 9 of the enclosure to Reference 3 defines the soluble boron requirement as follows. The total soluble boron credit requirement is defined as the sum of three quantities.

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where,

SBC_{TOTAL} is the total soluble boron credit requirement,(ppm),

$SBC_{95/95}$ is the soluble boron requirement for 95/95 k_{eff} less than or equal to 0.95, (ppm),

SBC_{RE} is the soluble boron required for burnup and reactivity uncertainties, (ppm),

SBC_{PA} is the soluble boron required for k_{eff} less than or equal to 0.95 under accident conditions, (ppm).

Each of these terms will be discussed in the following subsections.

3.6.1 Soluble Boron Requirement to Maintain k_{eff} Less Than or Equal to 0.95

Table 3-14 contains the KENO calculated k_{eff} values for the Prairie Island Units 1 & 2 spent fuel pool #2 from 0 to 600 ppm of soluble boron, in increments of 200 ppm. These KENO models assume that the pool is filled with the geometries and storage configurations outlined in section 3.1.3. The reactivity worth, Δk_{eff}, of the soluble boron was determined by subtracting the k_{eff} value, for a given soluble boron concentration, from the k_{eff} value for zero soluble boron. The soluble boron concentration and reactivity worth data was then fitted to a second degree polynomial, the limiting of which is shown on the bottom of Table 3-14. This polynomial was then employed to determine the amount of soluble boron required to reduce k_{eff} by 0.05 Δk_{eff} units, which is 276 ppm.

3.6.2 Soluble Boron Requirement for Burnup and Reactivity Uncertainties

The soluble boron credit, in units of ppm, required for reactivity uncertainties was determined by converting the uncertainty in fuel assembly reactivity and the uncertainty in absolute fuel assembly burnup values to a soluble boron concentration, in units of ppm, necessary to compensate for these two uncertainties. The first term, uncertainty in fuel assembly reactivity, is calculated by employing a depletion reactivity uncertainty equal to 0.010 Δk_{eff} units per 30,000 MWD/MTU of assembly burnup (obtained from Reference 3) and multiplying by the maximum amount of assembly burnup credited in a storage region analysis. For this analysis, the maximum amount of assembly burnup credited is 52,400 MWD/MTU (for the "Unshimmed 3x3" storage configuration). Therefore, the depletion reactivity uncertainty is 0.017467 Δk_{eff}.

The uncertainty in absolute fuel assembly burnup values is conservatively calculated as 5% of the maximum fuel assembly burnup credited in a storage region analysis. The maximum fuel assembly burnup credited in the storage configurations considered here, the uncertainty in these burnup values, and the corresponding reactivity values are given in Table 3-15. The reactivity associated with a change in burnup of 2,250 MWD/MTU at 45,000 MWD/MTU for the "All-Cell" storage region was conservatively calculated to be 0.01016 Δk_{eff} units.

The total of these two reactivity effects is equal to 0.027627 (0.017467 + 0.01016) Δk_{eff} units. The soluble boron concentration (ppm) necessary to compensate for this reactivity is calculated to be 145 ppm. A value of 183 ppm is conservatively chosen to be used.

3.6.3 Soluble Boron Required to Mitigate Accidents

The soluble boron concentration, in units of ppm, required to maintain k_{eff} less than or equal to 0.95 under accident conditions is determined by first surveying all possible events which increase the k_{eff} value of the spent fuel pool. The accident event which produced the largest increase in spent fuel pool k_{eff} value is employed to determine the required soluble boron concentration necessary to mitigate this and all less severe accident events. The list of accident cases considered include:

- Dropped fresh fuel assembly on top of the storage racks,
- Misloaded fresh fuel assembly into incorrect storage rack location,
- Misloaded fresh fuel assembly between storage racks (in gap between storage racks),
- Intramodule water gap reduction due to seismic event,
- Spent fuel pool temperature greater than 150 °F.

It is possible to drop a fresh fuel assembly on top of the spent fuel pool storage racks. In this case the physical separation between the fuel assemblies in the spent fuel pool storage racks and the assembly lying on top of the racks is sufficient to neutronically decouple the accident. In other words, dropping the fresh fuel assembly on top of the storage racks will only produce a very small positive reactivity increase. This very small positive reactivity increase will not be as limiting as the reactivity increase associated with fuel mishandling events.

Several fuel mishandling events were simulated with KENO to assess the possible increase in the k_{eff} value of the Prairie Island Units 1 & 2 spent fuel pool #2. The fuel mishandling events all assumed that a fresh Westinghouse OFA fuel assembly enriched to 4.95 w/o ^{235}U (and no burnable poisons) was misloaded into the described area of the spent fuel pool. These cases were simulated with the KENO model for spent fuel pool #2. These cases involved placing a fresh fuel assembly either inside a storage location intended for a burned fuel assembly or inside the gap of water between the storage modules in the southwest corner of spent fuel pool #2. The results of these KENO cases are contained in Table 3-16 which indicates that the highest increase in reactivity occurred when a fresh fuel assembly was placed in the gap of water between storage modules and next to another fresh fuel assembly. The reactivity increase associated with this accident was calculated to be 0.05914 Δk_{eff} units. The amount of soluble boron necessary to mitigate the consequence of this accident was determined to be 263 ppm by performing a KENO case for the same accident at 300 ppm and linear interpolation of the soluble boron for a reduction of 0.05914 Δk_{eff} units.

For the accident due to a seismic event the intramodule water gap is reduced to zero and each storage module makes contact. Based upon the comparison of the k_{eff} values for the

entire spent fuel pool and infinite arrays (see Table 3-13) the reactivity associated with this accident is approximately 0.011 Δk_{eff} units, and therefore not as limiting as the fuel mishandling events discussed above.

For the change in spent fuel pool water temperature accident, a temperature range of 150 F to 240 F was considered. From Reference 20, the maximum reactivity increase occurred for the "All-Cell" storage configuration and was calculated to be 0.01729 Δk_{eff} units. This reactivity increase is far less limiting than the reactivity increase associated with the fuel mishandling events discussed above.

3.6.4 Total Soluble Boron Requirement

Soluble boron in the spent fuel pool coolant is used in this criticality safety analysis to offset the reactivity allowances for calculational uncertainties in modeling, storage rack fabrication tolerances, fuel assembly design tolerances, and postulated accidents. The total soluble boron requirement is defined above.

The magnitude of each soluble boron requirement is shown below.

$SBC_{95/95}$ = 276 ppm

SBC_{RE} = 183ppm

SBC_{PA} = 263 ppm

SBC_{TOTAL} = 722 ppm

Therefore, a total of 722 ppm of soluble boron is required to maintain k_{eff} less than or equal to 0.95 (including all biases and uncertainties) assuming the most limiting single accident. Note that these soluble boron concentrations assumes an atomic fraction for ^{10}B equal to 0.199. For a ^{10}B isotopic fraction equal to 0.197, the soluble boron concentrations, required to maintain the same concentration of ^{10}B atoms, would be calculated as below.

$SBC_{95/95}$ = 279 ppm

SBC_{RE} = 185ppm

SBC_{PA} = 266 ppm

SBC_{TOTAL} = 730 ppm

Thus a recommended soluble boron level of 730 ppm is sufficient to accommodate all the design requirements.

Table 3-1. Summary of Fuel Assembly Characteristics (from Reference 20)

Characteristics	Westinghouse Standard	Westinghouse OFA
Cell Pitch (in)	0.556	0.556
Pellet OD (in)	.3659*	.3444
Fuel Rod Clad ID (in)	.3734	.3514
Fuel Rod Clad OD (in)	.422	.400
Fuel Rod Clad Material	Zirc-4	Zirc-4/ ZIRLO™ **
Guide Tube ID (in)	.505	.492
Guide Tube OD (in)	.539	.526
Instrument Tube ID	0.374	0.352
Instrument Tube OD	0.422	0.399
Enrichment w/o U_{235}	5.00	5.00

* Note that 0.3669 inches was conservatively employed to represent this pellet diameter

** Note that the clad material was conservatively modeled as Zr-4.

Table 3-2. Relative Power, Fuel, and Moderator Temperatures for Eight Zone Model

Zone No.	Height (in.)	Relative Power	Fuel Temperature (°F)	Moderator Temperature (°F)
1	6.15	0.488	991.022	544.190
2	6.15	0.813	1101.020	545.018
3	6.15	1.003	1211.018	545.360
4	107.1	1.092	1218.956	574.034
5	6.15	0.936	1138.010	603.860
6	3.075	0.841	1085.522	604.526
7	6.15	0.624	980.528	605.741
8	3.075	0.297	875.516	606.488

Table 3-3. Burnup and Initial Enrichment Combinations Used to Determine the Isotopic Number Densities

3 w/o ^{235}U	4 w/o ^{235}U	5 w/o ^{235}U
[MWD/MTU]	[MWD/MTU]	[MWD/MTU]
5,000	15,000	25,000
15,000	25,000	35,000
25,000	35,000	45,000
35,000	45,000	55,000

Table 3-4. k_{eff} for the Various Physical Tolerance Cases for the "All-Cell" Storage Configuration

Case Description	k_{eff}	Δk_{eff}
1.80 w/o Nominal Case[4]	0.97157 ± 0.00033	
4.95 w/o Nominal Case[5]	1.23265 ± 0.00038	
Increase in ^{235}U Enrichment	1.23368 ± 0.00038	0.00179
Increase in Stack Density	1.23299 ± 0.00038	0.00110
Decrease in Cell Pitch	0.97962 ± 0.00034	0.00872
Decrease in Rack Thickness	0.97769 ± 0.00033	0.00678
Decrease in Rack ID	0.97364 ± 0.00034	0.00274
Off-Center Assembly Positioning	0.97703 ± 0.00033	0.00612
Burnup Uncertainty		0.01016
Methodology Uncertainty[6]		0.00646
Statistical Sum of Uncertainties		0.01779
Methodology Bias[7]		0.00259
Pool Temperature Bias[8]		0.00640
Sum of Uncertainties and Biases		0.02678

[4] Note the 1.80 w/o nominal KENO case for the All Cell storage contains STD fuel at the fresh enrichment of 1.80 $^{w}/_{o}$ ^{235}U.
[5] Note the 4.95 w/o nominal KENO case for the All Cell storage contains STD fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ ^{235}U.
[6] See page 11 for definition of methodology uncertainty
[7] Methodology bias or the mean calculational methods bias is evaluated to be 0.00259.
[8] Pool temperature bias obtained from Reference 20.

Table 3-5. k_{eff} for the Various Physical Tolerance Cases for the Unshimmed "3x3" Storage Configuration

Case Description	k_{eff}	Δk_{eff}
1.20 w/o Nominal Case[9]	0.96471 ± 0.00036	
4.95 w/o Nominal Case[10]	1.23181 ± 0.00037	
Increase in ^{235}U Enrichment	1.23350 ± 0.00038	0.00244
Increase in Stack Density	1.23243 ± 0.00040	0.00139
Decrease in Cell Pitch	0.96903 ± 0.00035	0.00503
Decrease in Rack Thickness	0.96752 ± 0.00036	0.00353
Increase in Rack ID	0.96498 ± 0.00037	0.00100
Off-Center Assembly Positioning	0.97372 ± 0.00036	0.00973
Burnup Uncertainty		0.00658
Methodology Uncertainty[11]		0.00646
Statistical Sum of Uncertainties		0.01504
Methodology Bias[12]		0.00259
Pool Temperature Bias[13]		0.00640
Sum of Uncertainties and Biases		0.02403

[9] Note the 1.20 w/o nominal KENO case for the 3x3 storage contains OFA fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ ^{235}U, which is surrounded by a ring of STD fuel at the fresh enrichment of 1.20 $^{w}/_{o}$ ^{235}U.

[10] Note the 4.95 w/o nominal KENO case for the 3x3 storage contains OFA fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ ^{235}U, which is surrounded by a ring of STD fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ ^{235}U.

[11] See page 11 for definition of methodology uncertainty.

[12] Methodology bias or the mean calculational methods bias is evaluated to be 0.00259.

[13] Pool temperature bias obtained from Reference 20.

Table 3-6. k_{eff} for the Various Physical Tolerance Cases for the Shimmed "3x3" Storage Configuration

Case Description	k_{eff}	Δk_{eff}
1.50 w/o Nominal Case[14]	0.97619 ± 0.00034	
4.95 w/o Nominal Case[15]	1.22695 ± 0.00039	
Increase in ^{235}U Enrichment	1.22923 ± 0.00039	0.00306
Increase in Stack Density	1.22880 ± 0.00039	0.00263
Decrease in Cell Pitch	0.98347 ± 0.00035	0.00797
Decrease in Rack Thickness	0.98016 ± 0.00035	0.00466
Increase in Rack ID	0.97704 ± 0.00035	0.00154
Off-Center Assembly Positioning	0.98483 ± 0.00035	0.00933
Decrease in Gd_2O_3 Concentration	0.97588 ± 0.00035	0.00038
Burnup Uncertainty		0.00752
Methodology Uncertainty[16]		0.00646
Statistical Sum of Uncertainties		0.01916*
Methodology Bias[17]		0.00259
Pool Temperature Bias[18]		0.00640
Sum of Uncertainties and Biases		0.02816

* Conservative, actual value is 0.01701 delta k-effective units.

[14] Note the 1.50 w/o nominal KENO case for the 3x3 storage contains OFA fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ ^{235}U, which is surrounded by a ring of STD fuel at the fresh enrichment of 1.50 $^{w}/_{o}$ ^{235}U.

[15] Note the 4.95 w/o nominal KENO case for the 3x3 storage contains OFA fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ ^{235}U, which is surrounded by a ring of STD fuel at the fresh enrichment of 4.95 $^{w}/_{o}$ ^{235}U.

[16] See page 11 for definition of methodology uncertainty.

[17] Methodology bias or the mean calculational methods bias is evaluated to be 0.00259.

[18] Pool temperature bias obtained from Reference 20.

Table 3-7. k_{eff} versus Initial Enrichment and Assembly Burnup for the "All-Cell" Storage Configuration with No Soluble Boron

Initial Enrichment (w/o ^{235}U)	Assembly Burnup (MWD/MTU)	k_{eff} Value				
		Decay Time (years)				
		0	5	10	15	20
1.79	0	N/A	N/A	N/A	N/A	N/A
3.0	5,000	1.05509	1.05296	1.05257	1.05203	1.05189
3.0	15,000	0.96342	0.95647	0.95173	0.94810	0.94511
3.0	25,000	0.89586	0.88528	0.87621	0.87085	0.86581
4.0	15,000	1.04167	1.03695	1.03345	1.03029	1.02818
4.0	25,000	0.97260	0.96512	0.95878	0.95428	0.95044
4.0	35,000	0.91680	0.90598	0.89802	0.89109	0.88668
5.0	25,000	1.03293	1.02597	1.02081	1.01745	1.01429
5.0	35,000	0.97891	0.97055	0.96311	0.95804	0.95357
5.0	45,000	0.93132	0.92004	0.91199	0.90535	0.90018

Table 3-8. k_{eff} versus Initial Enrichment and Assembly Burnup for the Unshimmed "3x3" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Assembly Burnup (MWD/MTU)	k_{eff} Value				
		Decay Time (years)				
		0	5	10	15	20
1.30	0	N/A	N/A	N/A	N/A	N/A
3.0	15,000	1.02206	1.01827	1.01439	1.01271	1.01091
3.0	25,000	0.98138	0.97613	0.97112	0.96693	0.96371
3.0	35,000	0.95284	0.94771	0.94381	0.93758	0.93601
4.0	25,000	1.02507	1.01832	1.01376	1.00931	1.00722
4.0	35,000	0.98691	0.97992	0.97508	0.97023	0.96734
4.0	45,000	0.95919	0.95277	0.94768	0.94336	0.94058
5.0	35,000	1.02408	1.01798	1.01123	1.00702	1.00468
5.0	45,000	0.99148	0.98192	0.97837	0.97205	0.96932
5.0	55,000	0.96393	0.95477	0.95042	0.94501	0.94286

Table 3-9. k_{eff} versus Initial Enrichment and Assembly Burnup for the Shimmed "3x3" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Assembly Burnup (MWD/MTU)	k_{eff} Value				
		Decay Time (years)				
		0	5	10	15	20
1.39	0	N/A	N/A	N/A	N/A	N/A
3.0	5,000	1.07394	1.07387	1.07322	1.07240	1.07205
3.0	15,000	1.00569	1.00044	0.99736	0.99285	0.99075
3.0	25,000	0.95830	0.95047	0.94594	0.94165	0.93888
4.0	15,000	1.06381	1.05964	1.05611	1.05356	1.05145
4.0	25,000	1.00643	0.99921	0.99372	0.98953	0.98634
4.0	35,000	0.96284	0.95564	0.94981	0.94696	0.94371
5.0	25,000	1.05250	1.04664	1.04146	1.03788	1.03478
5.0	35,000	1.00477	0.99767	0.99045	0.98597	0.98254
5.0	45,000	0.96744	0.96138	0.95526	0.95177	0.94908

Table 3-10. Fuel Assembly Burnup versus Initial Enrichment for the "All Cell" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/MTU)				
	Decay Time (years)				
	0	5	10	15	20
1.79	0.0	0.0	0.0	0.0	0.0
3.0	14,373.9	13,563.0	13,102.0	12,754.3	12,498.1
4.0	25,722.1	24,519.2	23,595.7	22,999.5	22,499.8
5.0	37,148.3	35,441.9	34,058.5	33,189.8	32,452.8

Table 3-11. Fuel Assembly Burnup versus Initial Enrichment for the Unshimmed "3x3" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/MTU)				
	Decay Time (years)				
	0	5	10	15	20
1.30	0.0	0.0	0.0	0.0	0.0
3.0	28,363.2	26,621.5	25,047.9	23,843.7	22,885.1
4.0	40,469.9	38,029.2	36,346.6	34,761.4	33,847.8
5.0	52,310.6	48,778.6	47,514.7	45,362.3	44,448.6

Table 3-12. Fuel Assembly Burnup versus Initial Enrichment for the Shimmed "3x3" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/MTU)				
	Decay Time (years)				
	0	5	10	15	20
1.39	0.0	0.0	0.0	0.0	0.0
3.0	23,011.8	21,438.1	20,620.5	19,730.4	19,258.4
4.0	33,994.7	32,205.1	30,830.0	29,995.8	29,234.1
5.0	45,175.0	43,353.2	41,428.4	40,258.2	39,341.2

Table 3-13. Entire Spent Fuel Pool #2 and Infinite Array k_{eff} Results for the Allowable Storage Configurations

Configuration Description	Without Biases & Uncertainties		With Biases & Uncertainties
	Entire Pool k_{eff}	Infinite Array k_{eff}	Entire Pool k_{eff}
All-Cell, 1.80 w/o ^{235}U and zero burnup	0.96045	0.97157	0.98723
Unshimmed 3x3, 1.30 w/o ^{235}U and zero burnup	0.96647	0.97477	0.99050

Table 3-14. k_{eff} as a Function of Soluble Boron Level

	"All-Cell" Storage Configuration		"3x3" Storage Configuration	
(ppm)	k_{eff}	Δk_{eff}	k_{eff}	Δk_{eff}
0	0.92117	0.00000	0.95695	0.00000
200	0.88289	0.03828	0.91989	0.03706
400	0.84953	0.07164	0.88657	0.07038
600	0.82076	0.10041	0.85623	0.10072

The most limiting case is the "3x3" storage configuration. The following second degree polynomial describes the soluble boron concentration as a function of Δk_{eff}:

$$\mathbf{ppm = 8894.9\ \Delta k_{eff}^2 + 5059.8\ \Delta k_{eff}}$$

Table 3-15. Reactivity Associated with 5 % Burnup Uncertainty for the Storage Configurations

Configuration	Maximum BU Considered (MWD/MTU)	5% BU Uncertainty	Δk_{eff}
All-Cell	45,000	2,250	0.01016
Unshimmed 3x3	52,400	2,620	0.0069
Shimmed 3x3	45,000	2,250	0.00752

Table 3-16. k_{eff} for Accident Events

	All Cell		3x3	
Description of Accident	**Δk_{eff}**	**Δppm**	**Δk_{eff}**	**Δppm**
Dropped fresh fuel assembly on top of the storage racks,	Not Limiting	Not Limiting	Not Limiting	Not Limiting
Misloaded fresh fuel assembly into burned storage rack location,	0.04158	179.5	0.05336	244.2
Misloaded fresh fuel assembly between storage racks,	0.03207	112.3	0.05914	262.5
Intramodule water gap reduction due to seismic event,	Not Limiting	Not Limiting	Not Limiting	Not Limiting
Spent fuel pool temperature greater than 185 °F	Not Limiting	Not Limiting	Not Limiting	Not Limiting

Figure 3-1. Westinghouse 14x14 OFA & STD Fuel Assembly

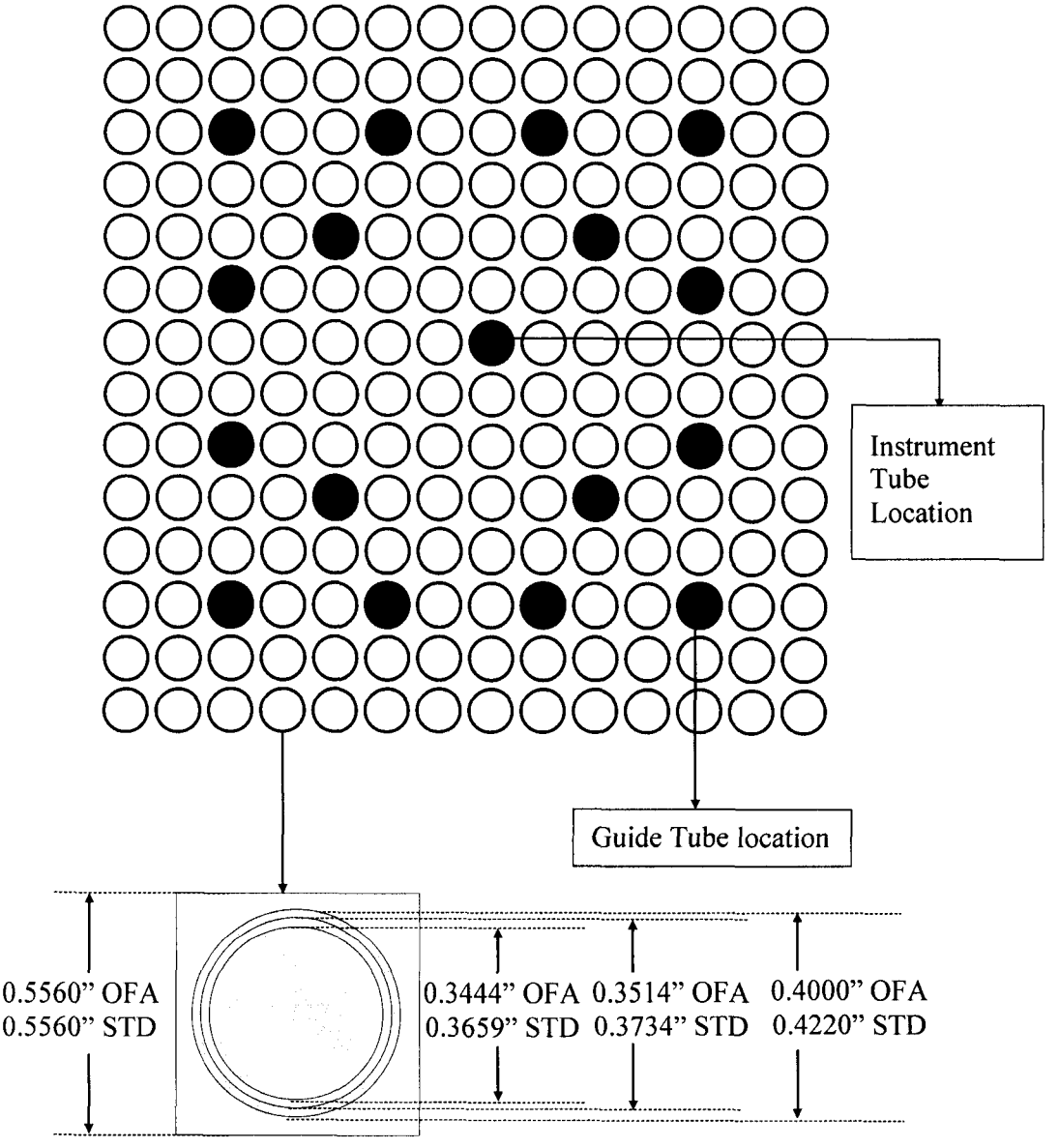


Figure 3-2. Gd_2O_3 Burnable Absorber Pin Pattern

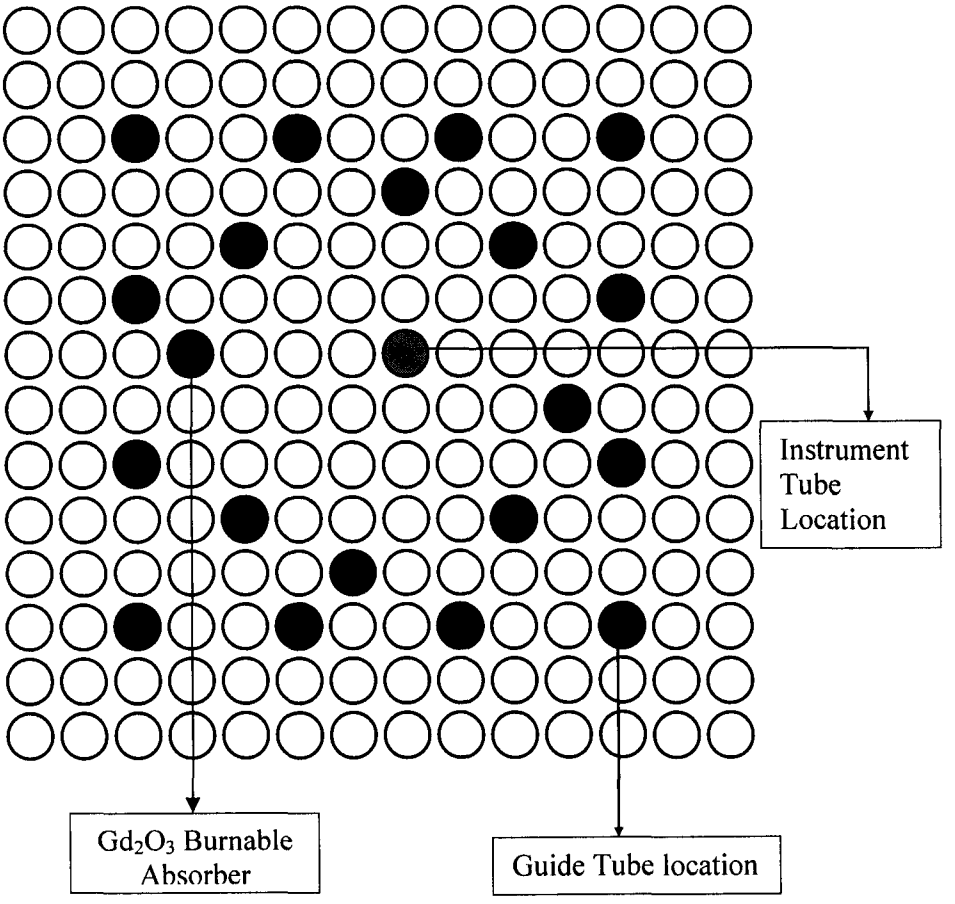


Figure 3-3. Sketch of Axial Zones Employed in Fuel Assembly

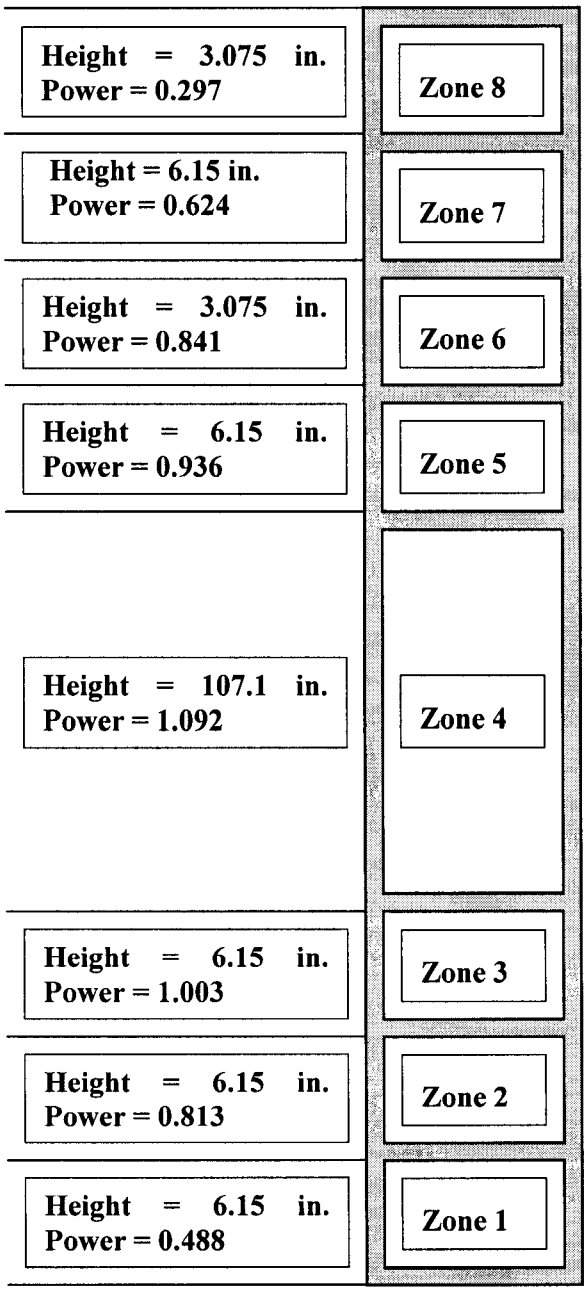


Figure 3-4. KENO Output Plot of the "All Cell" Model

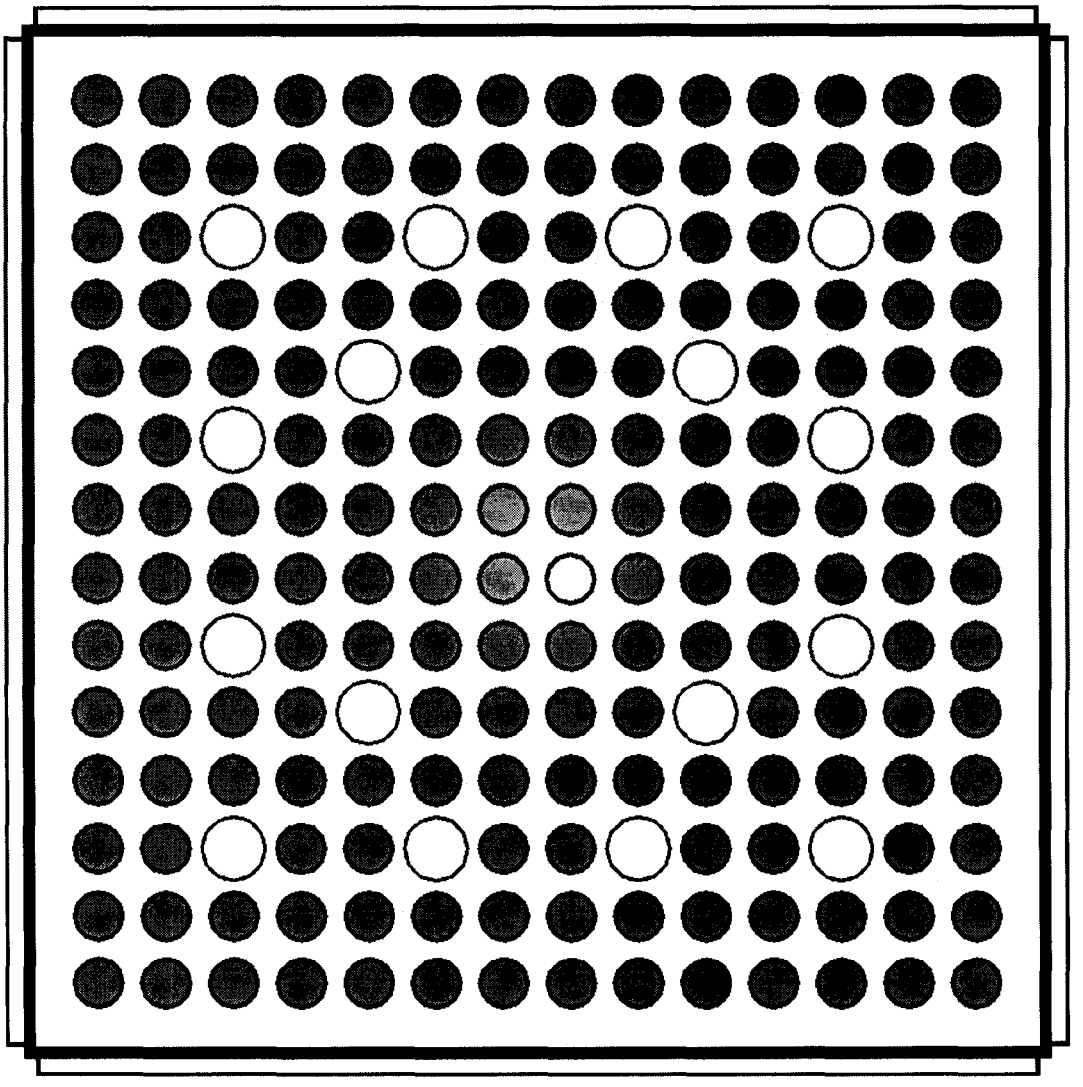


Figure 3-5. KENO Output Plot of the "3x3" Storage Model

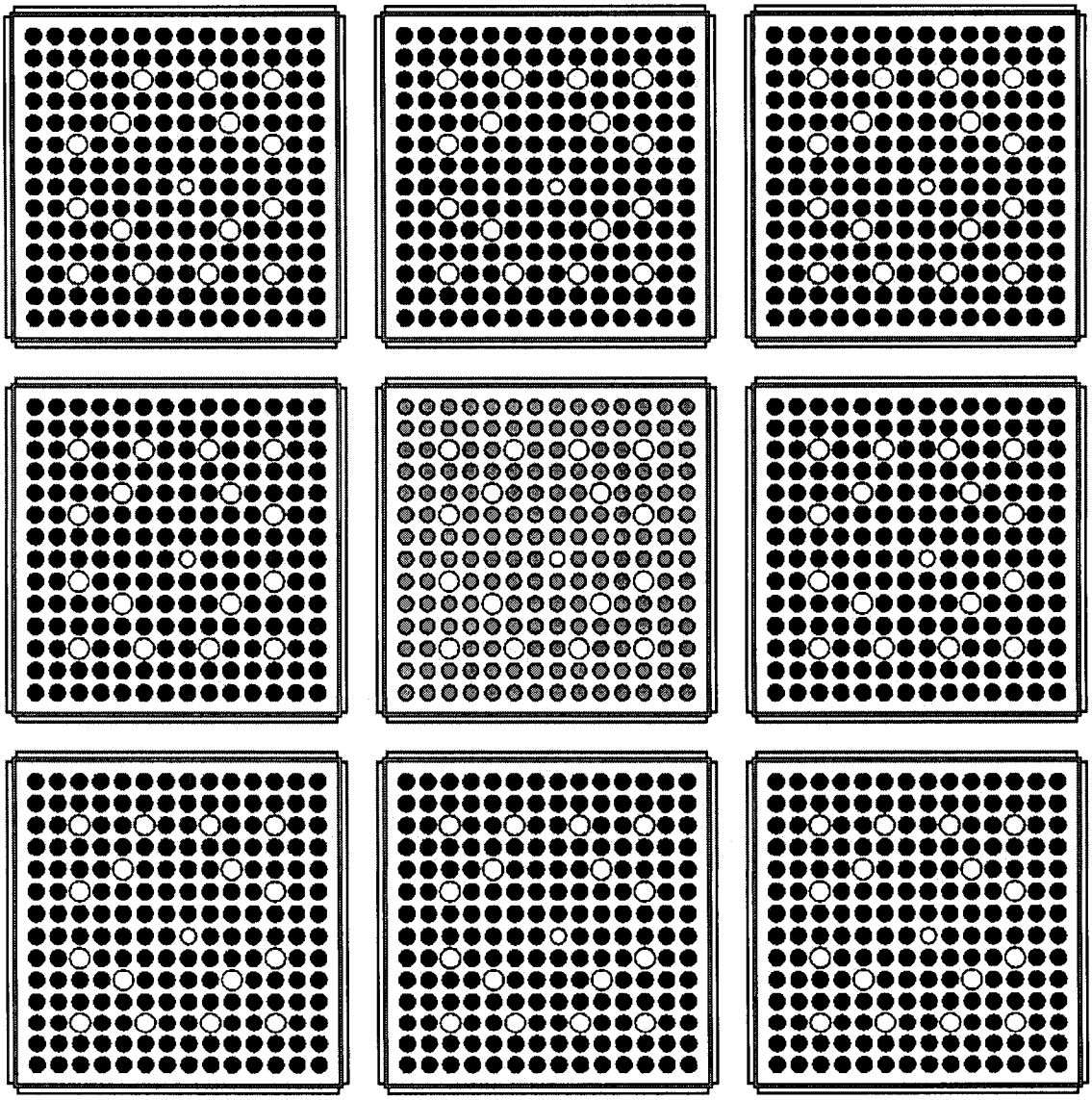
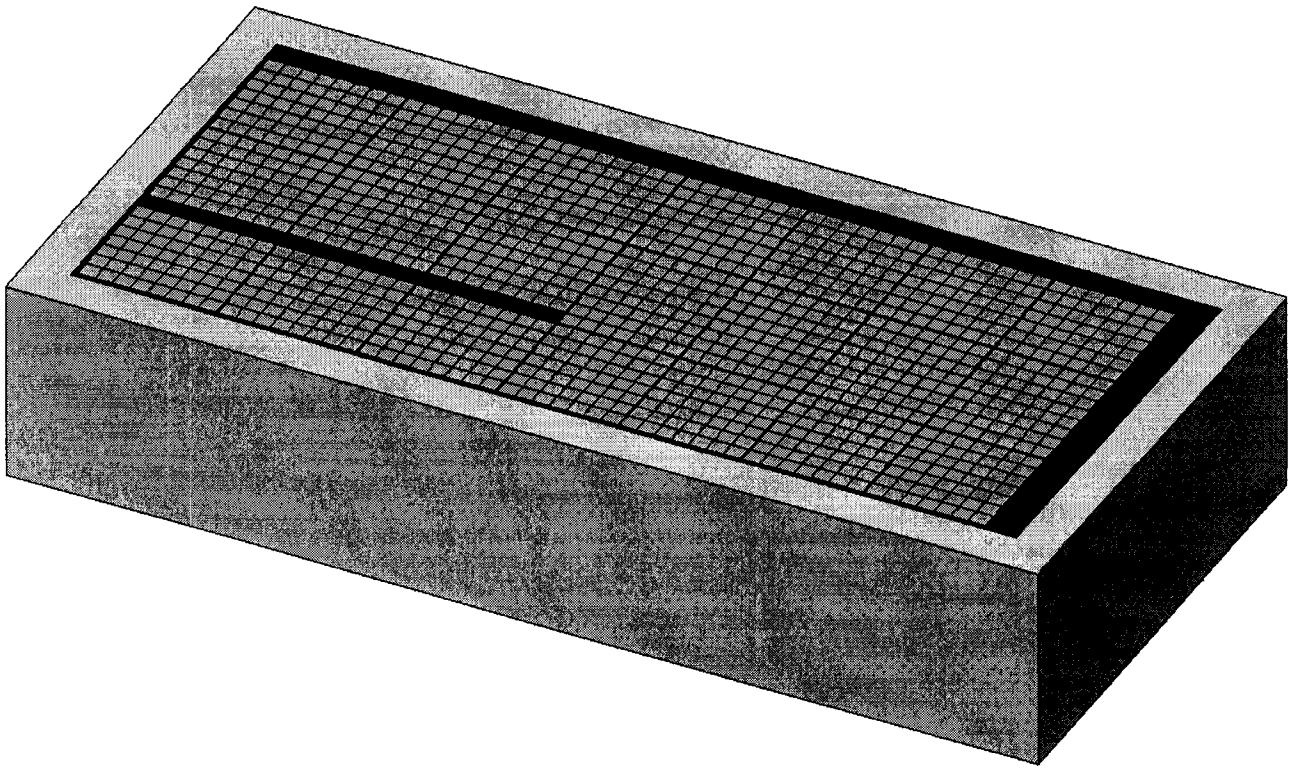
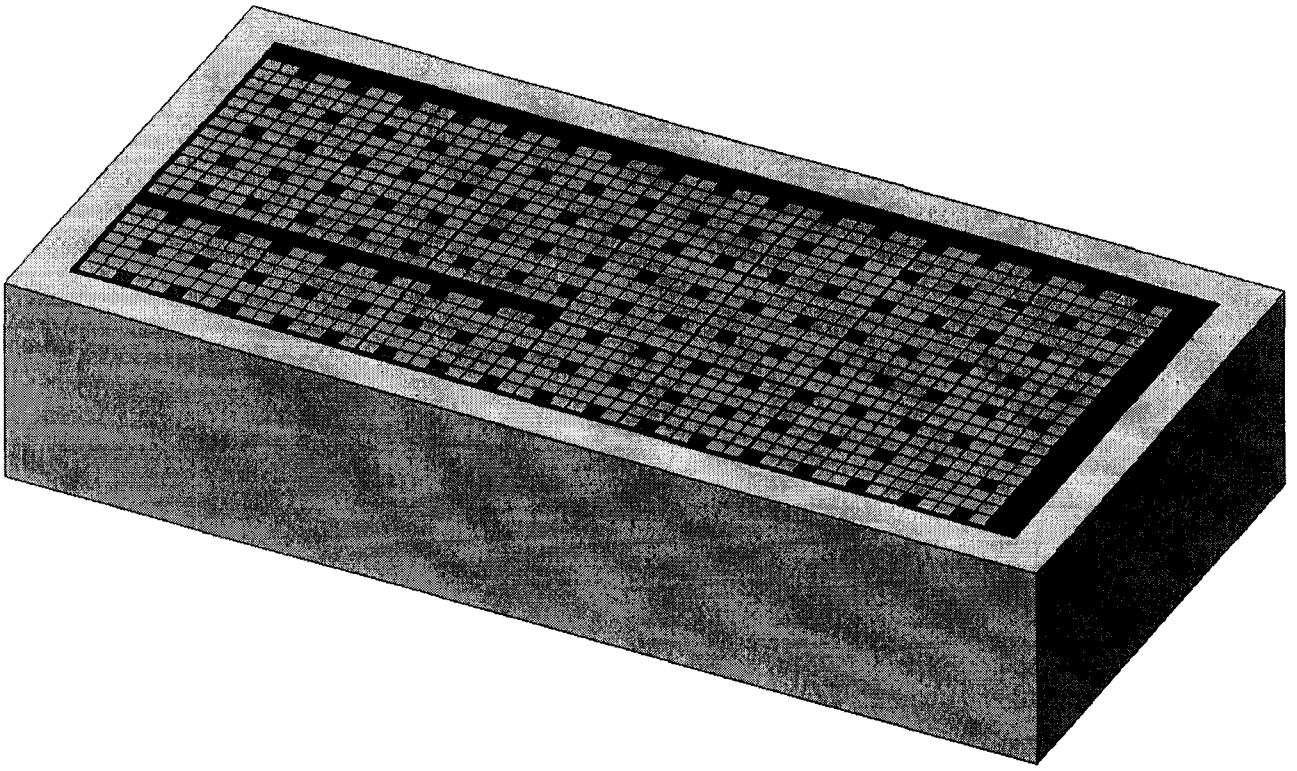


Figure 3-6. KENO Output Plot of the "All-Cell" Spent Fuel Pool Model



Note: Rack locations appear violet, pool water appears blue and pool wall appears gray in color.

Figure 3-7. KENO Output Plot of the "3x3" Spent Fuel Pool Model



Note: High enrichment rack locations appear green, low enrichment rack locations appear violet, pool water appears blue and pool wall appears gray in color.

4.0 Summary of Results

The following sections contain the criticality analysis results for the Prairie Island Units 1 & 2 spent fuel pool with soluble boron credit.

4.1 Allowable Storage Configurations and Interfaces

Figure 4-1 displays the allowable assembly arrangements for the All Cell storage configuration. The All Cell storage configuration will be employed to store depleted fuel assemblies that meet the requirements of Figure 4-5 or no fuel assembly. The assembly burnup versus initial enrichment limits should be calculated with the third degree polynomial given in Table 4-1 based upon the appropriate decay time.

Figure 4-2 displays the allowable assembly arrangements for the unshimmed "3x3" storage configuration. The unshimmed "3x3" storage configuration will be employed to store depleted fuel assemblies which meet the requirements of Figure 4-6 or no fuel assembly. The center storage cell will be employed to store fresh fuel assemblies of enrichment values of up to and including 4.95 w/o ^{235}U or no fuel assembly. The assembly burnup versus initial enrichment limits for the peripheral locations should be calculated with the third degree polynomial given in Table 4-2 based upon the appropriate decay time.

Figure 4-3 displays the allowable assembly arrangements for the Gd_2O_3 shimmed "3x3" storage configuration. The Gd_2O_3 shimmed "3x3" storage configuration will be employed to store depleted fuel assemblies which meet the requirements of Figure 4-7 or no fuel assembly. The center storage cell will be employed to store fresh fuel assemblies with a minimum of 4 Gd_2O_3 shimmed fuel rods. The minimum concentration of Gd_2O_3 is equal to 4.0 w/o. The maximum enrichment for unshimmed fresh fuel rods is equal to 4.95 w/o ^{235}U. The maximum enrichment for Gd_2O_3 shimmed fresh fuel rods is equal to 4.0 w/o ^{235}U. The assembly burnup versus initial enrichment limits for the peripheral locations should be calculated with the third degree polynomial given in Table 4-3 based upon the appropriate decay time.

The allowable interface between the storage configurations is displayed in Figure 4-4. Note that a row of empty storage cells at the interface may be used to separate the configurations. Also, it is acceptable to replace an assembly with an empty cell.

Note that a Failed Fuel Pin Basket (FFPB), a fully loaded Consolidation Rod Storage Basket (CRSB) with up to two (2) fuel rods missing, or a partially loaded CRSB with a maximum of 18 fuel rods may be substituted for any assembly in either the "All-Cell" or "3x3" fuel assembly storage configurations.

The FFPB is employed to store up to 16 fresh fuel rods (in a 4x4 array) with a maximum enrichment less than or equal to 4.95 w/o ^{235}U with no credit for burnup.

The fully loaded CRSC is a container designed to accommodate all of the fuel rods from two assemblies and fit into a single storage location. Note that up to two fuel rods may be missing from the container. The current burnup versus initial enrichment tech spec limits for the fully loaded CRSC are still valid.

A partially loaded CRSC may contain up to 18 fresh fuel rods with a maximum enrichment less than or equal to 4.95 w/o ^{235}U with no credit for burnup.

4.2 Burnup Credit

Figure 4-5 displays the assembly burnup versus initial enrichment storage curve for the All Cell storage configuration. The All Cell storage requirements are tabulated in Table 4-1 for 0, 5, 10, 15, and 20 years of decay time. The assembly burnup versus initial enrichment limits should be calculated with the third degree polynomial given in Table 4-1 based upon the appropriate decay time.

Figure 4-6 displays the burnup versus enrichment storage curve for the unshimmed "3x3" storage configuration. The unshimmed "3x3" storage requirements are tabulated in Table 4-2 for 0, 5, 10, 15, and 20 years of decay time. The assembly burnup versus initial enrichment limits for the peripheral locations should be calculated with the third degree polynomial given in Table 4-2 based upon the appropriate decay time.

Figure 4-7 displays the burnup versus enrichment storage curve for the shimmed "3x3" storage configuration. The shimmed "3x3" storage requirements are tabulated in Table 4-3 for 0, 5, 10, 15, and 20 years of decay time. The assembly burnup versus initial enrichment limits for the peripheral locations should be calculated with the third degree polynomial given in Table 4-3 based upon the appropriate decay time.

4.3 Total Soluble Boron Requirement

The total soluble boron (sum of all three components) required to maintain the k_{eff} value (including all biases and uncertainties, without the adjustment for ^{10}B) less than or equal to 0.95 is determined to be 722 ppm for a ^{10}B atom percent equal to 19.9. The soluble boron concentration required for a ^{10}B atom percent equal to 19.7 is 730 ppm. The recommended minimum boron level is 730 ppm and is sufficient to accommodate all the design requirements.

Table 4-1. Fuel Assembly Burnup versus Initial Enrichment for the "All-Cell" Storage Configuration

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/MTU)				
	Decay Time (years)				
	0	5	10	15	20
1.79	0.0	0.0	0.0	0.0	0.0
3.0	14,373.9	13,563.0	13,102.0	12,754.3	12,498.1
4.0	25,722.1	24,519.2	23,595.7	22,999.5	22,499.8
5.0	37,148.3	35,441.9	34,058.5	33,189.8	32,452.8

The 3rd degree polynomial that describes the 0 years decay time curve is as follows:

$$\mathbf{BU = 94.80e^3 - 1098.60e^2 + 15530.98e - 24891.15}$$

The 3rd degree polynomial that describes the 5 years decay time curve is as follows:

$$\mathbf{BU = 37.66e^3 - 468.58e^2 + 12842.90e - 21765.17}$$

The 3rd degree polynomial that describes the 10 years decay time curve is as follows:

$$\mathbf{BU = 49.33e^3 - 607.38e^2 + 12920.33e - 21524.31}$$

The 3rd degree polynomial that describes the 15 years decay time curve is as follows:

$$\mathbf{BU = 39.90e^3 - 506.32e^2 + 12312.99e - 20705.14}$$

The 3rd degree polynomial that describes the 20 years decay time curve is as follows:

$$\mathbf{BU = 45.22\ e^3 - 567.00e^2 + 12297.63e - 20512.73}$$

Table 4-2. Fuel Assembly Burnup versus Initial Enrichment for the Unshimmed "3x3" Storage Configuration as a function of decay time

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/MTU)				
	Decay Time (years)				
	0	5	10	15	20
1.30	0.0	0.0	0.0	0.0	0.0
3.0	28,363.2	26,621.5	25,047.9	23,843.7	22,885.1
4.0	40,469.9	38,029.2	36,346.6	34,761.4	33,847.8
5.0	52,310.6	48,778.6	47,514.7	45,362.3	44,448.6

The 3rd degree polynomial that describes the 0 years decay time curve is as follows:

$$\mathbf{BU = 424.51e^3 - 5227.19e^2 + 32990.07e - 35024.12}$$

The 3rd degree polynomial that describes the 5 years decay time curve is as follows:

$$\mathbf{BU = 338.76e^3 - 4394.29e^2 + 29633.48e - 31876.86}$$

The 3rd degree polynomial that describes the 10 years decay time curve is as follows:

$$\mathbf{BU = 328.15e^3 - 4003.13e^2 + 27179.07e - 29321.24}$$

The 3rd degree polynomial that describes the 15 years decay time curve is as follows:

$$\mathbf{BU = 270.12e^3 - 3399.80e^2 + 24721.90e - 27016.94}$$

The 3rd degree polynomial that describes the 20 years decay time curve is as follows:

$$\mathbf{BU = 202.95e^3 - 2616.35e^2 + 21767.79e - 24350.83}$$

Table 4-3. Fuel Assembly Burnup versus Initial Enrichment for the Gd_2O_3 Shimmed "3x3" Storage Configuration as a function of decay time

Initial Enrichment (w/o ^{235}U)	Burnup (MWD/MTU)				
	Decay Time (years)				
	0	5	10	15	20
1.39	0.0	0.0	0.0	0.0	0.0
3.0	23,011.8	21,438.1	20,620.5	19,730.4	19,258.4
4.0	33,994.7	32,205.1	30,830.0	29,995.8	29,234.1
5.0	45,175.0	43,353.2	41,428.4	40,258.2	39,341.2

The 3rd degree polynomial that describes the 0 years decay time curve is as follows:

$$\mathbf{BU = 385.78e^3 - 4530.66e^2 + 28423.72e - 31899.43}$$

The 3rd degree polynomial that describes the 5 years decay time curve is as follows:

$$\mathbf{BU = 329.74e^3 - 3766.28e^2 + 24930.66e - 28360.31}$$

The 3rd degree polynomial that describes the 10 years decay time curve is as follows:

$$\mathbf{BU = 335.89e^3 - 3836.31e^2 + 24635.61e - 27828.70}$$

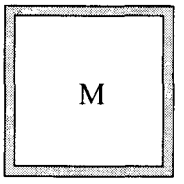
The 3rd degree polynomial that describes the 15 years decay time curve is as follows:

$$\mathbf{BU = 216.44e^3 - 2598.84e^2 + 20448.84e - 24070.61}$$

The 3rd degree polynomial that describes the 20 years decay time curve is as follows:

$$\mathbf{BU = 234.60e^3 - 2749.50e^2 + 20542.10e - 23956.57}$$

Figure 4-1. Allowable Fuel Assembly Combinations for the "All-Cell" Storage Configuration



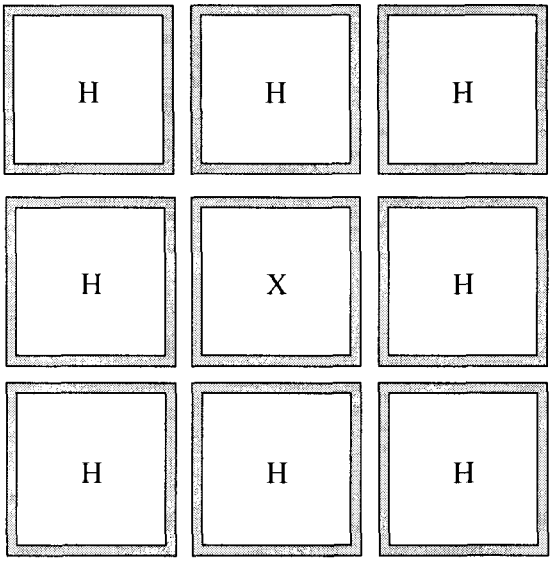
"M" represents

- Depleted fuel assembly with meets the requirements of Figure 4-5.

or

- An empty location.

Figure 4-2. Allowable Fuel Assembly Combinations for the Unshimmed "3x3" Storage Configuration



"H" represents

- Depleted fuel assembly which meets the requirements of Figure 4-6.

or

- An empty location.

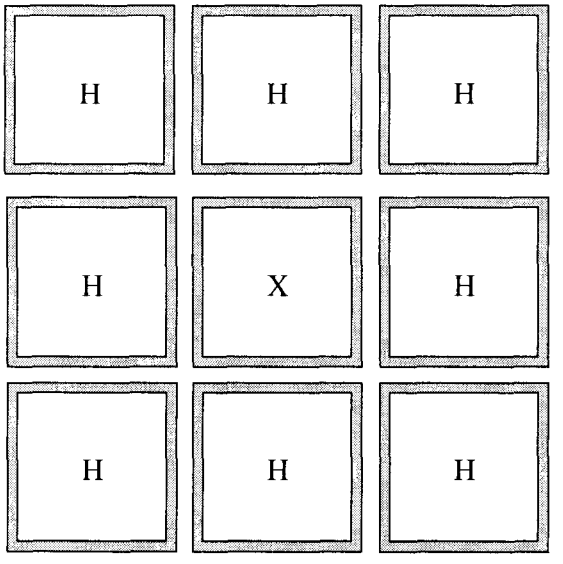
"X" represents

- Fresh unshimmed fuel assembly of enrichment values up to and including 4.95 $^{w}/_{o}$ ^{235}U.

or

- An empty location.

Figure 4-3. Allowable Fuel Assembly Combinations for the Gd_2O_3 Shimmed "3x3" Storage Configuration



"H" represents

- Depleted fuel assembly which meets the requirements of Figure 4-7.

or

- An empty location.

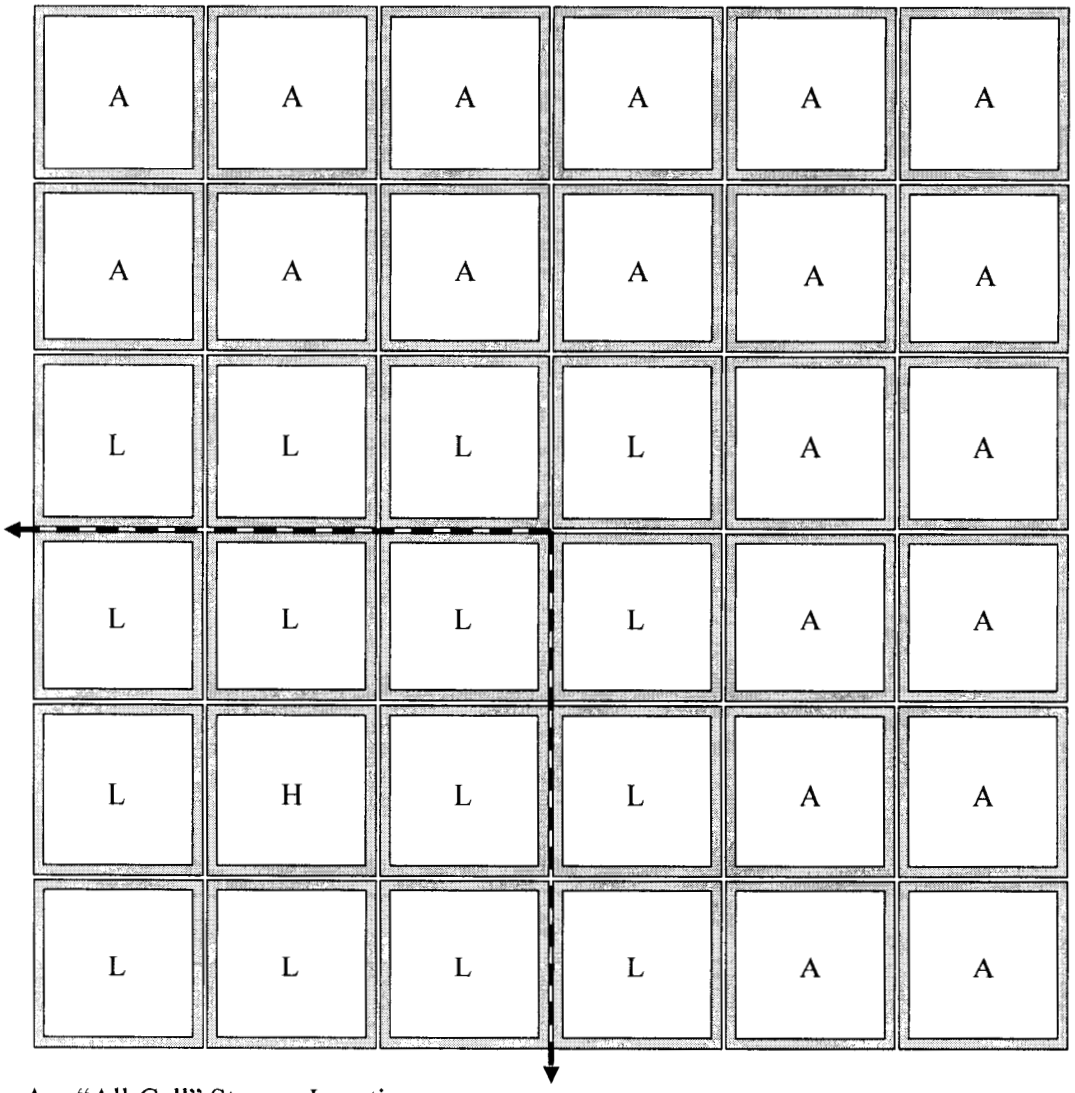
"X" represents

- Fresh Gd_2O_3 shimmed fuel assembly. Minimum of 4 shimmed fuel rods with a minimum of 4 w/o Gd_2O_3. Maximum enrichment for unshimmed rods is 4.95 $^{w}/_{o}$ ^{235}U. Maximum enrichment for shimmed rods is 4.0 $^{w}/_{o}$ ^{235}U.

or

- An empty location.

Figure 4-4. Boundary Between the "3x3" and "All-Cell" Storage Configurations



A = "All-Cell" Storage Location
L = "3x3" Low Enrichment Location
H = "3x3" High Enrichment Location

Notes:

1) A row of empty cells can be used at the interface to separate the configurations.

2) It is acceptable to remove an assembly from any storage location.

Figure 4-5. Prairie Island Units 1 & 2 Assembly Burnup Requirements for the "All-Cell" Storage Configuration

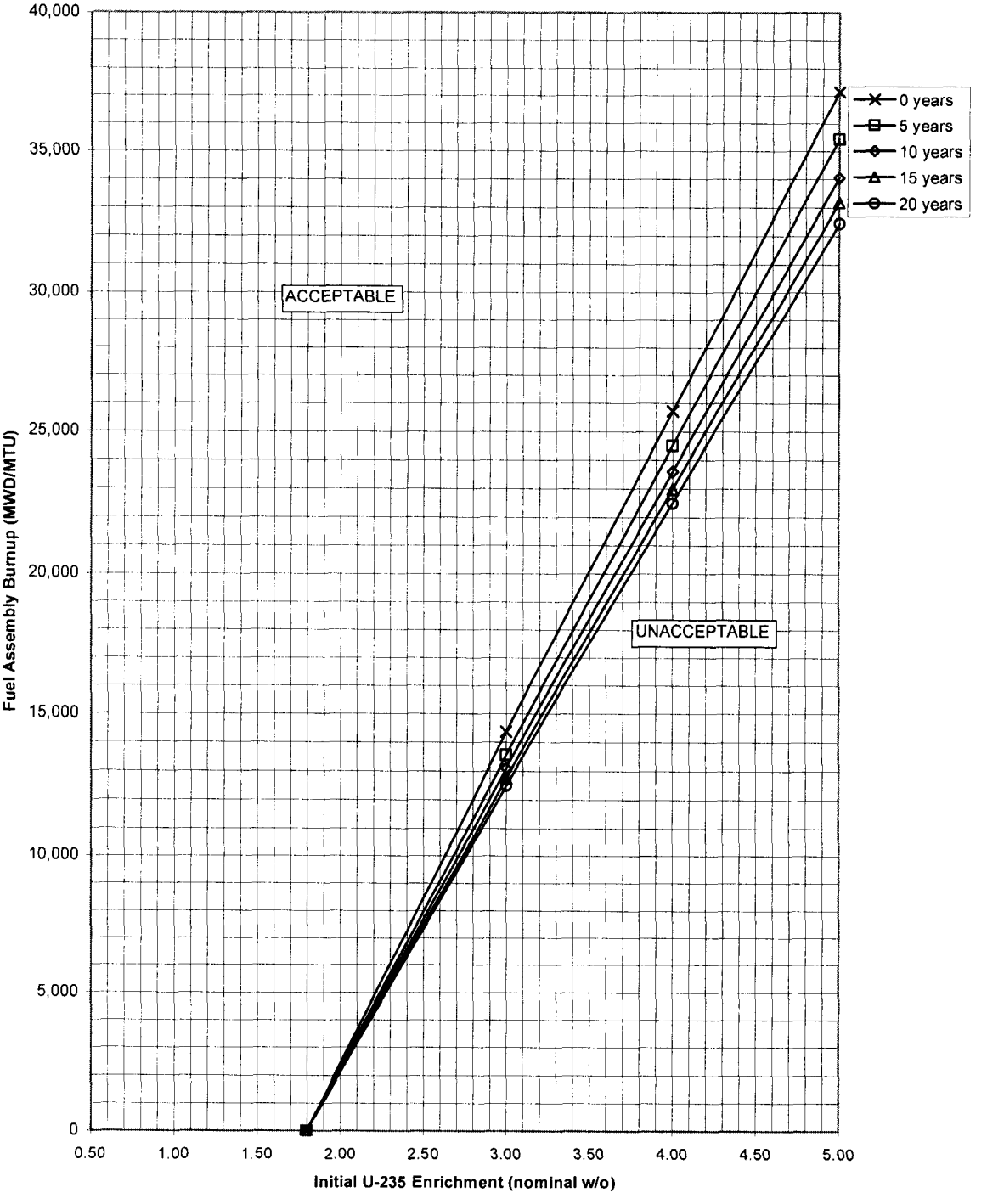


Figure 4-6. Assembly Burnup Requirements for the Peripheral Fuel Assemblies in the Unshimmed "3x3" Storage Configuration

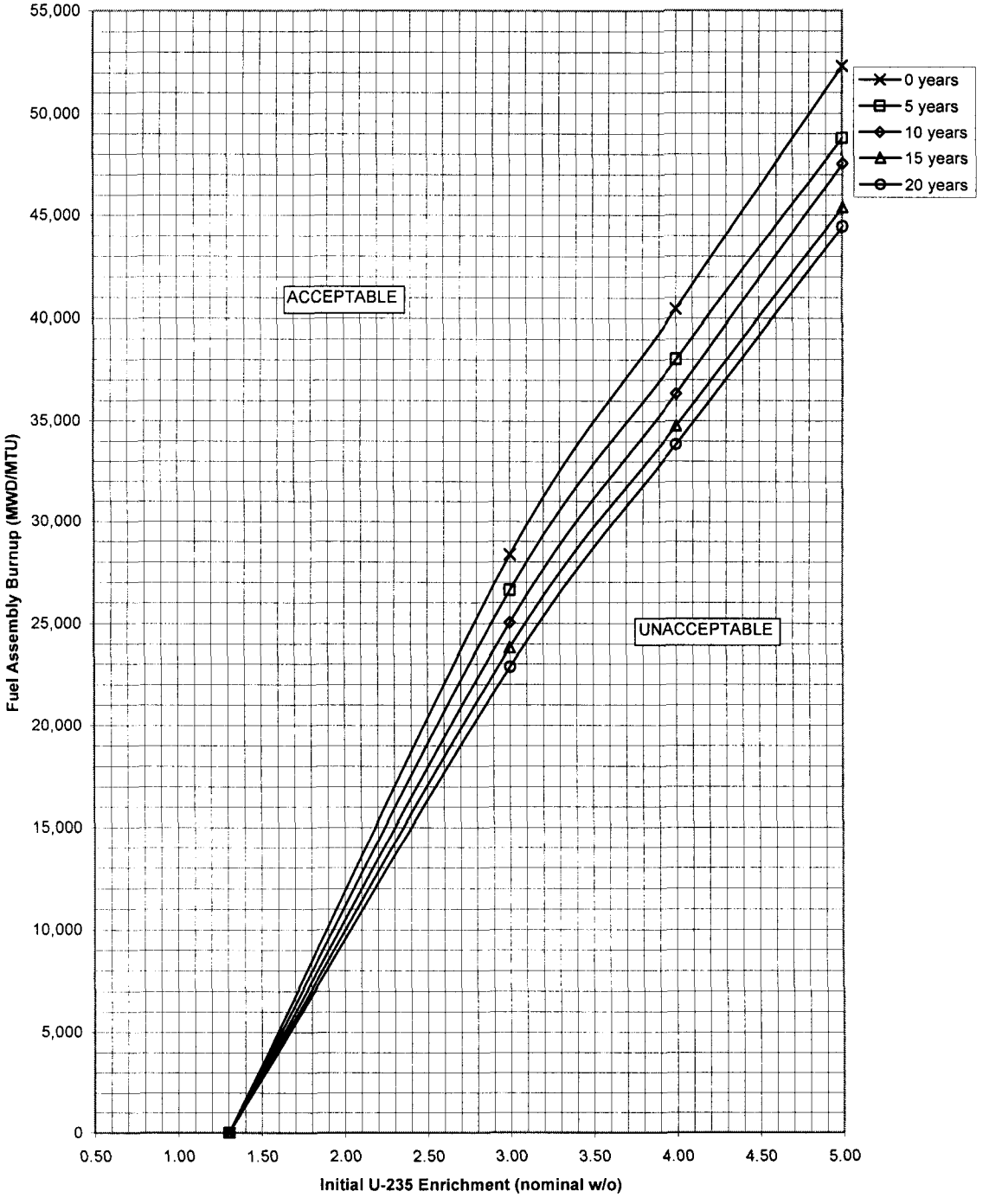
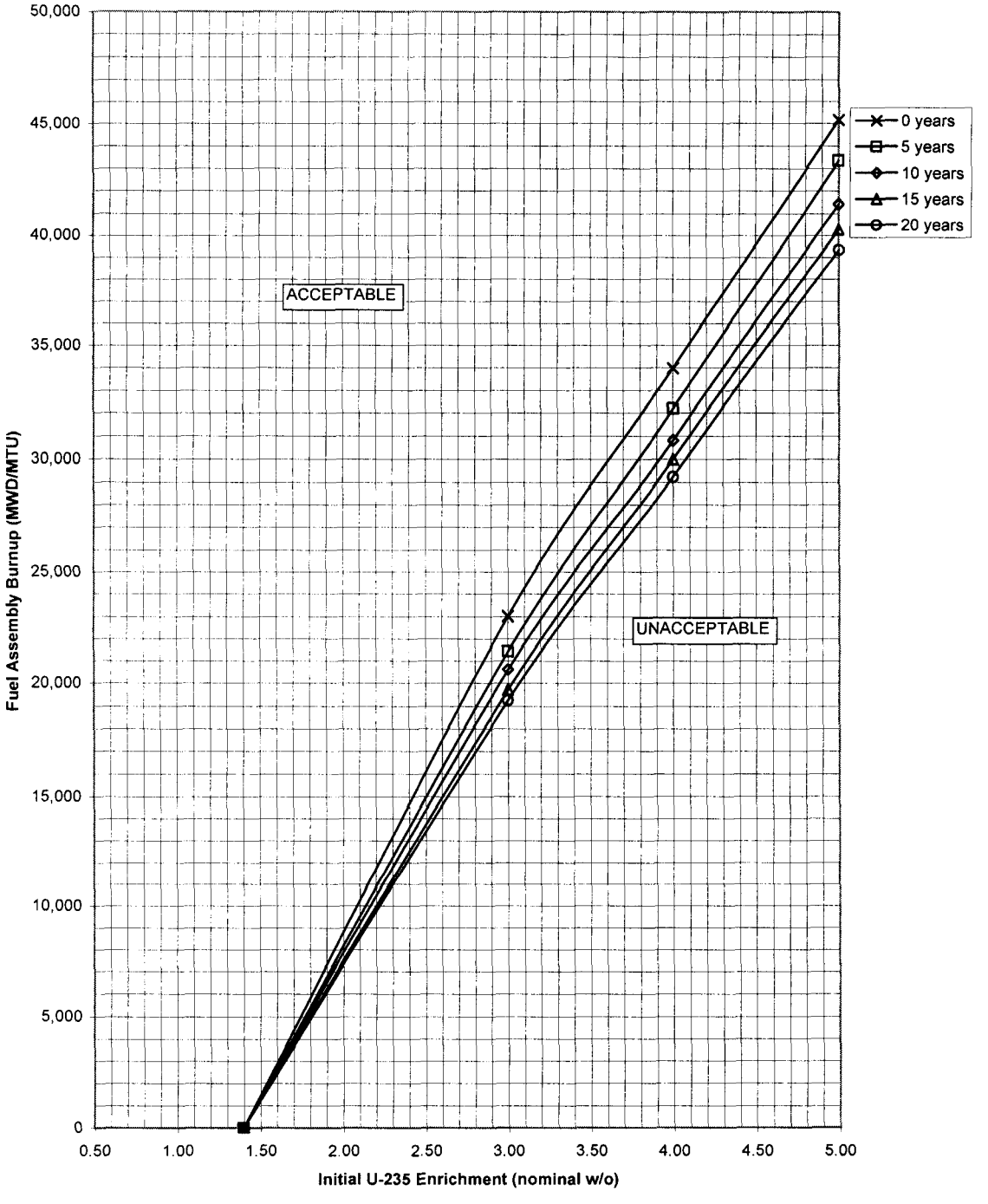


Figure 4-7. Assembly Burnup Requirements for the Peripheral Fuel Assemblies in the Shimmed "3x3" Storage Configuration



5.0 Computer Codes Used In Calculation

Table 5-1

Summary of Computer Codes Used in Calculation

Code No.	Code Name	Code Version	Verified and Configured (Yes/No)	Basis (or reference) that supports use of code in current calculation	Outstanding Issues (Yes/No).
1	SCALE-PC	4.3	Yes	QC-1	No, see Note

Note: There is a recent notification of an error in SCALE associated with the HOLE function. The error is documented in the SCALE notebook, titled "Error in KENO V.a for cylindrical holes with shared boundaries," and dated March 22, 2005. In the standard spent fuel pool analysis, none of the input files involve cylindrical holes with shared boundaries; therefore, the analysis is not affected from this code error.

6.0 References

1. Not Used.

2. Letter, G. S. Vissing (NRC) to R. C. Mecredy (RGE), "R. E. Ginna Nuclear Power Plant – Amendment Re-Revision to the Storage Configuration Requirements within the Existing Storage Racking and Taking Credit for a Limited Amount of Soluble Boron", December 7, 2000.

3. Letter, T. E. Collins (NRC) to T. Greene (WOG), "Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Spent Fuel Rack Methodology (TAC No. M93254)", October 25, 1996.

4. Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 62, "Prevention of Criticality in Fuel Storage and Handling".

5. L. Kopp (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants", February 1998.

6. "SCALE 4.3- Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers", NUREG/CR-200; distributed by the Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, July 1993.

7. "DIT: Discrete Integral Transport Assembly Design Code", CE-CES-11, Revision 4-P, April 1994.

8. M. N. Baldwin, et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel; Summary Report", BAW-1484-7, July 1979.

9. S. R. Bierman and E.D. Clayton, "Critical Experiments with Subcritical Clusters of 2.35 Wt% ^{235}U Enriched UO2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6", NUREG/CR-1547, PNL-3314, July 1980.

10. S.R. Bierman and E.D. Clayton, "Criticality Experiments with Subcritical Clusters of 2.35 and 4.31 Wt% U-Enriched UO2 Rods in Water with Steel Reflecting Walls", Nuclear Technology, Vol. 54, pg. 131, August 1981.

11. International Handbook of Evaluated Criticality Safety Benchmark Experiments, Nuclear Energy Agency and Organization for Economic Cooperation and Development.

12. W. Marshall, et al., "Criticality Safety Criteria", TANS Vol. 35, pg. 278, 1980.

13. D. B. Owen, "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans", SCR-607, Sandia Corporation Monograph, March 1963.

14. "Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages", DOE/RW-0472 Rev. 1, May 1997.

15. Not Used.

16. Parrington, Josef R., et al, "Nuclides and Isotopes: Chart of the Nuclides", Fifteenth Edition, General Electric Co. and KAPL, Inc., 1996.

17. "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs", CENPD-404-P, Rev. 0, January 2001.

18. "Methodology Manual for Cross Section Tableset Generation", CE-CES-124, Revision 2-P, June 1994.

19. P.F. O'Donnell, et al, "R.E. Ginna Nuclear Power Plant Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit", RGE-09-0009, Revision 0, November 1999.

20. J.R. Lesko, et al., "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit", CAA-97-042, Revision 0, February 1997.

Enclosure 3

Technical Specification revised Figures 3.7.17-1, 4.3.1-3 and 4.3.1-4

3 pages follow

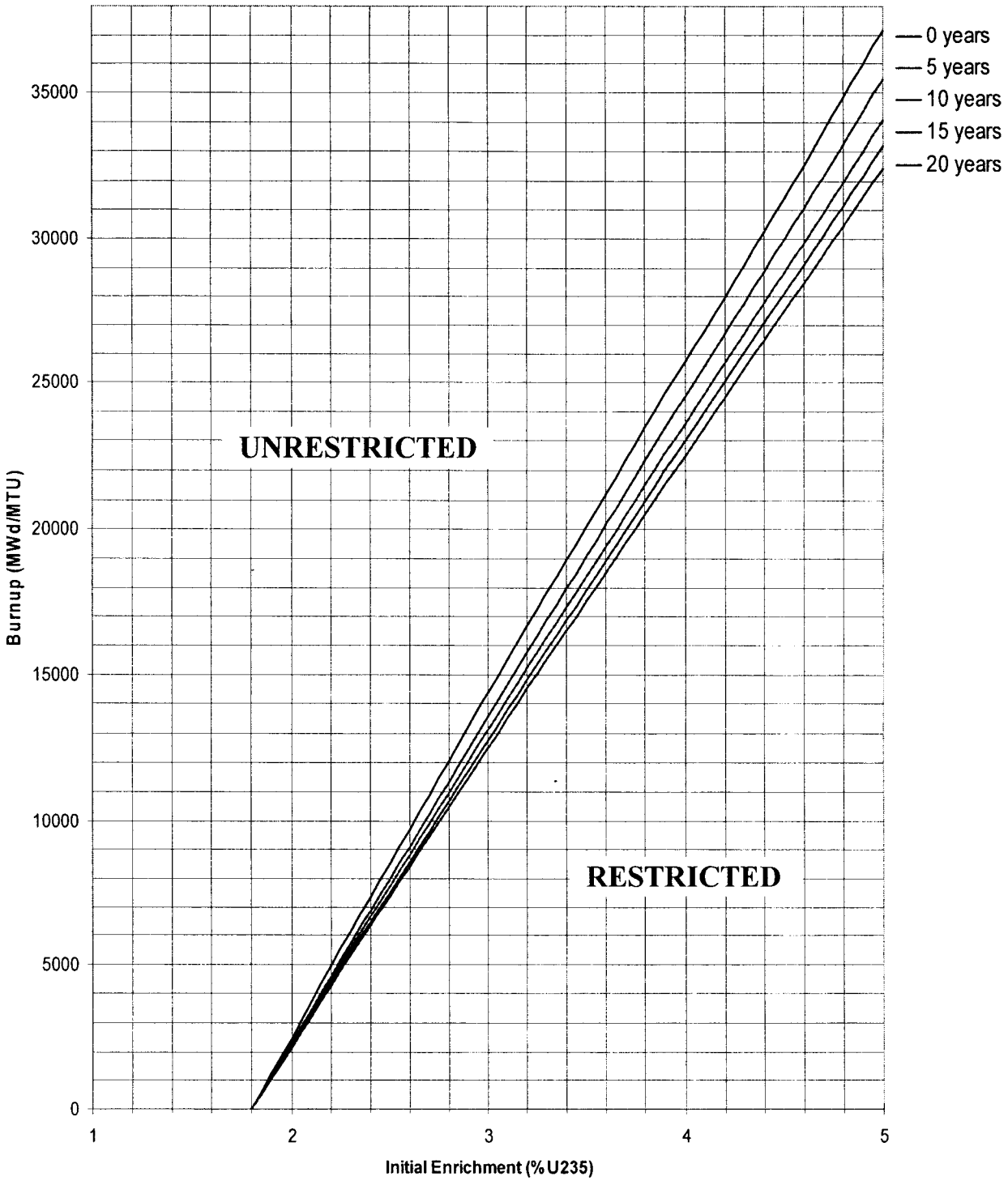


Figure 3.7.17-1
Spent Fuel Pool Unrestricted Region Burnup and Decay Time Requirements

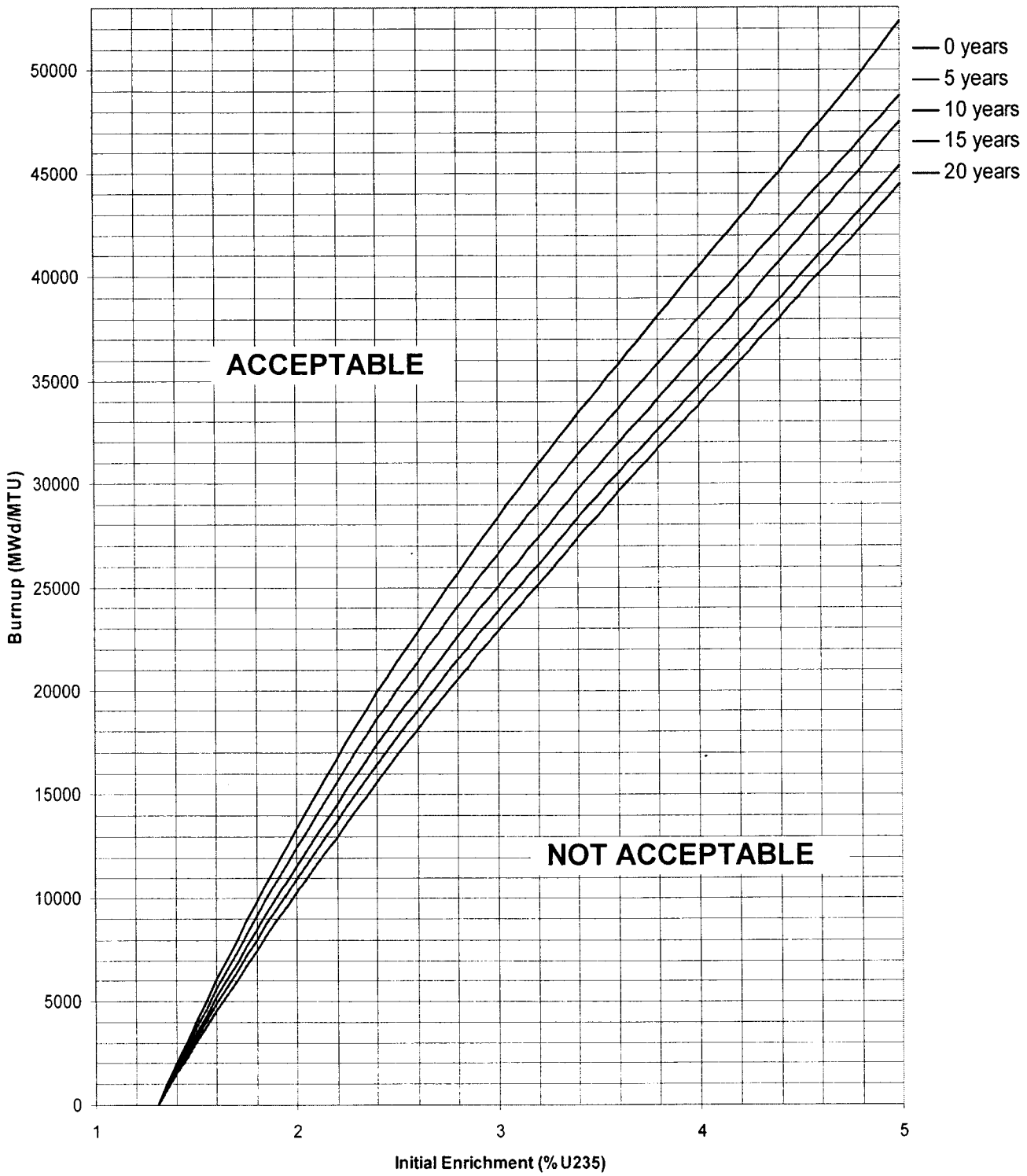


Figure 4.3.1-3
Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements
- No GAD

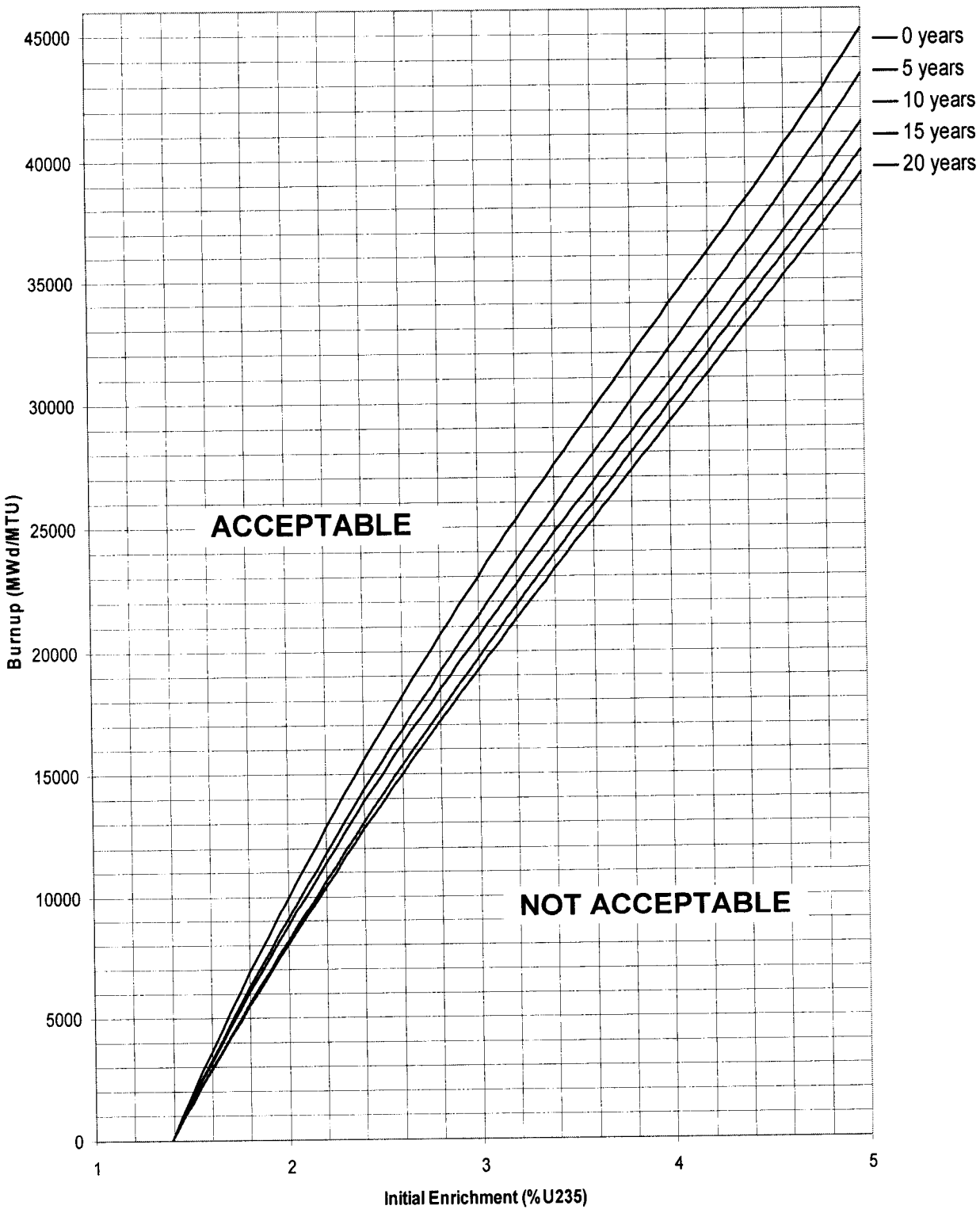


Figure 4.3.1-4
Spent Fuel Pool Checkerboard Region Burnup and Decay Time Requirements - Fuel with GAD