ATTACHMENT "E"

EVALUATION OF THE HANDLING AND STORAGE OF 4.8 W/O ENRICHED FUEL

SOUTHERN CALIFORNIA EDISON
SAN ONOFRE NUCLEAR
GENERATING STATION
UNITS 2 AND 3

REVISION 0

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EXECUTIVE SUMMARY

This report supports Proposed Change Number PCN-449 to San Onofre Nuclear Generating Station Units 2 and 3 (SONGS 2 and 3) Facilities Operating Licenses, NPF-10 and NPF-15, respectively. This proposed change increases the licensed maximum fuel pin enrichment from 4.1 weight percent U-235 (w/o) to 4.8 w/o for SONGS 2 and 3. Increasing the maximum fuel pin enrichment from 4.1 w/o to 4.8 w/o will allow an increase of the current cycle length from about 520 effective full power days (EFPD) to about 600 EFPD resulting in economic benefit. Also, to increase the allowance for Boraflex degradation and to develop separate burnup criteria for the peripheral pool locations of the Region II spent fuel storage racks, the minimum discharge burnup vs initial enrichment tables and curves have been revised.

The results of criticality, radiological, and decay heat analyses show that the existing new and spent fuel storage racks, and supporting systems and components have been adequately designed to accommodate the storage and handling of SONGS 2 and 3 fuel with a maximum fuel pin enrichment of 4.8 w/o. For postulated accident conditions in the spent fuel pool, a minimum concentration of 1850 PPM (1800 PPM + 50 PPM uncertainty) soluble boron is required. The use of the higher enriched fuel in the reactor core will be analyzed each cycle in the reload safety analysis.

The criticality analyses also show that San Onofre Nuclear Generating Station Unit 1 (SONGS 1) fuel assemblies can be safely stored in the SONGS 2 and 3 spent fuel storage racks. The maximum initial enrichment of the SONGS 1 assemblies was 4.0 w/o. Due to the permanent shutdown of SONGS 1, the maximum initial enrichment limit is not being changed.

The burnup criteria for unrestricted placement of SONGS 1, 2, and 3 fuel in the Region II spent fuel storage racks have been re-calculated. Due to conservative assumptions in the new calculational methodology and the inclusion of larger Boraflex gaps, slightly higher burnups are calculated than the current values. The SONGS 1 burnup curve has been replaced with a single value.

Finally, for the peripheral pool locations of the Region II spent fuel storage racks, substantially lower burnup criteria have been calculated. The large neutron leakage from the peripheral pool locations permits a lower discharge burnup than required for the interior locations. A table and curve are provided for SONGS 2 and 3 fuel assemblies. A single value is provided for SONGS 1 fuel assemblies.

1. INTRODUCTION

1.1 PURPOSE

Southern California Edison Company (Edison) plans to increase the allowable maximum fuel pin enrichment from 4.1 weight percent U-235 (w/o) to 4.8 w/o for the San Onofre Nuclear Generating Station Units 2 and 3 (SONGS 2 and 3). This report supports Proposed Change Number PCN 449 to the SONGS 2 and 3 Facilities Operating Licenses, NPF-10 and NPF-15,[1,2] respectively. Edison also plans to revise the burnup requirements for storing San Onofre Nuclear Generating Station Unit 1 (SONGS 1) and SONGS 2 and 3 fuel in the Region II spent fuel storage racks of SONGS 2 and 3.

Increasing the maximum fuel pin enrichment from 4.1 w/o to 4.8 w/o for SONGS 2 and 3 will allow an increase of the current cycle length from about 520 effective full power days (EFPD) to about 600 EFPD.

This evaluation addresses fuel handling and storage criticality, decay heat loads, and radiological consequences due to fuel handling and storage. Although the requested maximum fuel pin enrichment for SONGS 2 and 3 is 4.8 w/o, the criticality analyses were performed for up to 5.1 w/o enrichment and the results bound the requested 4.8 w/o for SONGS 2 and 3 fuel. The use of the higher enriched fuel (up to 4.8 w/o) in the reactor core will be analyzed each cycle in the reload safety analysis.

Due to the permanent shutdown of SONGS 1, the maximum initial fuel enrichment for SONGS 1 fuel is not being changed from the current limit of 4.0 w/o.

1.2 PRESENT DESIGN

The current limits on fuel enrichment and discharge burnup requirements include:

(1) The maximum permitted SONGS 2 and 3 fuel pin enrichment is 4.1 w/o for both the new and spent fuel storage racks

(2) The same initial enrichment vs discharge burnup criteria apply to all - interior and peripheral pool - Region II spent fuel storage rack locations for SONGS 2 and 3 assemblies
(Technical Specification [TS] Figure 3.7.18-2)
(3) The same initial enrichment vs discharge burnup criteria apply to all - interior and peripheral pool - Region II spent fuel storage rack locations for SONGS 1 assemblies (TS Figure 3.7.18-1)
(The current criteria are a curve, not a single value.)
(4) Spent fuel pool soluble boron concentration of 1850 PPM (1800 PPM + 50 PPM uncertainty)
(5) The maximum initial enrichment for SONGS 1 fuel was 4.0 w/o.

1.3 PROPOSED CHANGES

Edison proposes the following changes regarding fuel enrichment and discharge burnup requirements:

(1) Increase the maximum fuel pin enrichment to 4.8 w/o for SONGS 2 and 3 fuel assemblies in the new and spent fuel storage racks
(2) Revise the initial enrichment vs discharge burnup criteria for SONGS 2 and 3 fuel assemblies in the Region II spent fuel storage racks
(3) Provide initial enrichment vs (lower) discharge burnup criteria for SONGS 2 and 3 fuel assemblies in the peripheral pool locations of the Region II spent fuel storage racks
(4) Revise the initial enrichment vs discharge burnup criteria for SONGS 1 fuel assemblies in the Region II spent fuel storage racks to a single value
(5) Provide a single (lower) discharge burnup value for SONGS 1 fuel assemblies in the peripheral pool locations of the Region II spent fuel storage racks

1.4 REPORT FORMAT

This report generally follows the guidance of the NRC Position Paper entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling

Applications," dated April 14, 1978, as amended by the NRC letter dated January 18, 1979.[3] Structural and seismic re-analysis of the new and spent fuel storage racks is not required since the rack and fuel assembly designs and weights are bounded by previous SFP rerack analyses submitted to and approved by the NRC.

Section 2 of this report is a description of the new and spent fuel storage racks. Design data for the SONGS 1, 2, and 3 fuel assemblies currently stored in the spent fuel storage racks are also provided.

Section 3 of this report provides the criticality analyses of the new and spent fuel storage racks. The analyses include:

(1) Un-irradiated 4.8 w/o, unshimmed (No burnable poison rods - including IFBA, Gd, or Er) SONGS 2 and 3 fuel assemblies in the new fuel storage racks and the Region I spent fuel storage racks
(These analyses were done for 5.1 w/o and bound 4.8 w/o.)

(2) Minimum discharge burnup requirements versus initial enrichment for both the interior and peripheral pool locations of the Region II spent fuel storage racks for SONGS 2 and 3 fuel assemblies

(3) Minimum discharge burnup requirements for both the interior and peripheral pool locations of the Region II spent fuel storage racks for SONGS 1 fuel assemblies

(5) Fuel handling activities

(6) Postulated accidents

(7) Boraflex Erosion or Dissolution

Section 4 provides a decay heat analysis of the spent fuel pools.

Section 5 describes the impact of increased enrichment on waste generation, effluents, fuel handling building shielding, personnel exposure during fuel handling operations, and the radiological consequences of fuel handling accidents and pool boiling.

1.5 CONCLUSIONS

On the basis of the information and evaluations presented in this report, Edison concludes that the proposed increase in enrichment and changes in fuel storage for the SONGS 2 and 3 new and spent fuel storage facilities will provide safe fuel storage and are in conformance with NRC requirements. The changes will have no significant impact on the health and safety of the general public.

Technical Specification (TS) Changes

To implement the proposed increase in enrichment and revised burnup requirements for the Region II spent fuel storage racks, the following Technical Specifications will have to be changed:

3.7.18 Spent Fuel Assembly Storage

The current Figure 3.7.18-1 (Unit 1 Minimum Burnup vs Initial Enrichment For Region II Racks) will be replaced with single values as follows: 18.0 GigaWatt-Days per metric ton of Uranium (GWD/T) for interior locations

5.5 GWD/T for peripheral pool locations (Peripheral pool locations have one or two faces towards the spent fuel pool sides.)

Figure 3.7.18-2 (Units 2 and 3 Minimum Burnup vs Initial Enrichment For Region II) will be renumbered to Figure 3.7.18-1.
Figure 3.7.18-1 will become the SONGS 2 and 3 burnup curve for the interior locations of Region II. Also the data in this curve have been recalculated.

A new Figure 3.7.18-2 will be provided. The new figure provides lower burnup criteria for the Region II peripheral pool locations for SONGS 2 and 3 fuel.

Thus Figures 3.7.18-1 and 3.7.18-2 are both for Units 2 and 3 fuel and included revised data:

Figure 3.7.18-1 - "Minimum Burnup vs Initial Enrichment For Unrestricted Placement Of SONGS 2 And 3 Fuel In Region II Racks"

Figure 3.7.18-2 - "Minimum Burnup vs Initial Enrichment For Placement Of SONGS 2 And 3 Fuel In Region II Peripheral Pool Locations"

The bases for Technical Specification 3.7.18 will be revised accordingly.

4.3.1.1 Criticality - Spent Fuel Storage Racks
The current enrichment limit of 4.1 w/o for SONGS 2 and 3 fuel will be increased to 4.8 w/o.

4.3.1.2 Criticality - New Fuel Storage Racks
The current enrichment limit of 4.1 w/o for SONGS 2 and 3 fuel will be increased to 4.8 w/o.

Technical Specification 3.7.17, Spent Fuel Pool Boron Concentration, does not need to be changed. The current boron concentration of the spent fuel pool - 1850 PPM (1800 PPM + 50 PPM uncertainty) - is acceptable. However, the bases need to be changed. Previously, the misloading analyses assumed that Region II was completely filled with un-irradiated 4.1 w/o fuel assemblies. The new analyses assume a worst case misloading of nine (9) un-irradiated fuel assemblies of 5.1 w/o (bounds 4.8 w/o) in a 3x3 array in the Region II spent fuel storage racks.

1.6 REFERENCES

1. San Onofre Nuclear Generating Station Unit 2 Facility Operating License NPF-10, Docket No. 50-361.

2. San Onofre Nuclear Generating Station Unit 3 Facility Operating License NPF-15, Docket No. 50-362.

3. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees, from B. K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," as amended by the NRC letter dated January 18, 1979.

2. FUEL STORAGE DESCRIPTION

2.1 INTRODUCTION

This section presents a description of the new and spent fuel storage racks. Design data for the SONGS 1, 2, and 3 fuel assemblies currently stored in the spent fuel storage racks are also provided.

2.2 FUEL ASSEMBLY DESCRIPTIONS

Two fuel assembly designs are currently stored in the SONGS 2 and 3 fuel storage racks:

(1) ABB Combustion Engineering (ABB/CE), Zircaloy-clad, 16x16 Fuel Assemblies, 4.1 w/o maximum enrichment

(2) Westinghouse, Stainless-steel-clad, 14x14 Fuel Assemblies transhipped from Unit 1, 4.0 w/o maximum enrichment

Edison plans to increase the maximum fuel pin enrichment of the ABB/CE SONGS 2 and 3 fuel assemblies to 4.8 w/o.

The characteristics of the ABB/CE SONGS 2 and 3 and Westinghouse SONGS 1 fuel assembly designs are given in Table 2-1.

2.3 NEW FUEL STORAGE RACK DESCRIPTION

The new fuel storage racks[1] provide for safe storage of un-irradiated fuel assemblies in a geometry which prevents criticality under all normal and accident conditions. The new fuel storage racks are designed to protect the stored assemblies against possible impact loading due to handling of neighbor assemblies, and to guide the assemblies into their locations.

The new fuel storage racks provide dry storage for 80 fuel assemblies at a nominal centerline spacing of 29 inches and 38 inches (Figure 2-1). The racks are fabricated from stainless steel.

2.4 SPENT FUEL STORAGE RACK DESCRIPTION

The spent fuel storage racks[1,2] provide for storage of new and spent fuel assemblies in appropriate regions of the spent fuel pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loadings. SONGS 1, 2, and 3 fuel may be stored in the SONGS 2 and SONGS 3 racks, as well as miscellaneous storage items (e.g., trash baskets, dummy fuel assemblies, neutron sources), and the failed rod storage baskets.

Fuel is stored in two regions within each pool (Table 2-2, Figure 2-2):

(1) Region I (312 locations)

(2) Region II (1230 locations)

Both regions use Boraflex, a neutron absorbing material. Boraflex consists of fine boron carbide particles distributed in a polymeric silicone encapsulant. Its length and width are designed to allow for both shrinkage and edge deterioration and still meet criticality requirements.

Cells located in the interior of a Region I rack have Boraflex on all four sides. Periphery cells facing the pool walls have Boraflex on the three sides not facing the wall. Cells facing adjacent racks have Boraflex on all four sides. Rack corner cells which face two pool walls have Boraflex only on the two remaining sides. Those corner cells adjacent to another rack and the pool wall require Boraflex on three sides. Corner cells adjacent to other racks in both directions have Boraflex on all four sides.

Cells located in the interior of a Region II rack have Boraflex on all four sides. Periphery side cells have Boraflex on the three rack interior sides. Rack corner cells have Boraflex only on the two rack interior sides.

The Region I and Region II racks are constructed from Type 304LN stainless steel except the leveling screws which are SA-564 Type 630 stainless steel and some leveling pads which are either SA-182 Type F-304 stainless steel or SA-240 (or SA-479) Type 304 stainless steel. The floor plates under the rack support pads are made from SA-240 Type 304 stainless steel, which has the same corrosion resistance characteristics as the rack materials.

The Region I and Region II racks are neither anchored to the floor nor braced to the pool walls or each other. Also, the pool floor plates are not attached to the pool floor.

2.4.1 Region I Spent Fuel Storage Rack Description

Region I consists of two high density fuel racks, each with 12x13 cells. The nominal dimensions of each rack are 125.5 inches by 135.9 inches. The cells within a rack are interconnected by grid assemblies and stiffener clips to form an integral structure as shown in Figure 2-3.

Region I is typically used to store un-irradiated fuel, and fuel which has not achieved the minimum required burnup for unrestricted storage in Region II. Region I can hold a full core off load (217 fuel assemblies), plus 95 locations.

2.4.2 Region II Spent Fuel Storage Rack Description

Region II (1230 locations) has six high density fuel racks, four with 14x15 cells and two with 13x15 cells and provides normal storage for spent fuel assemblies. The nominal dimensions of the 14x15 rack are 124.82 inches by 133.67 inches; the nominal dimensions of the 13x15 rack are 115.97 inches by 133.67 inches.

Region II is designed to accommodate irradiated fuel which meets a predetermined burnup. Placement of fuel in Region II racks is restricted by burnup and enrichment limits, or by prescribed storage patterns.

The six Region II storage racks consist of stainless steel cells assembled in a checkerboard pattern, producing a honeycomb type structure as shown in

Figure 2-4. Cells are located in every other location and are welded together at the cell corners. This results in "non-cell" storage locations, each one formed by one outside wall of four checkerboard cells.

2.5 REFERENCES

1. San Onofre Nuclear Generating Station Units 2 and 3 Updated Final Safety Analysis Report, Revision 10, Chapter 9, Docket Nos. 50-361 and 50-362.

2. Spent Fuel Pool Reracking Licensing Report, Revision 6, Southern California Edison San Onofre Nuclear Generating Station Units 2 and 3, February 16, 1990.

Table 2-1

FUEL ASSEMBLY DATA FOR SONGS 1, 2, AND 3

	SONGS 1	SONGS 2&3
Maximum Fuel Pin Enrichment (w/o)	4.0	4.8*
Cladding Type	SS	Zr
Rod Array	14x14	16x16
Fuel Rod Pitch (in.)**	0.556	0.506
Number of Rods Per Assembly	180	236
Fuel Rod Outer Diameter (in.)	0.422	0.382
Fuel Pellet Diameter (in.)	0.3835	0.325***
Active Fuel Length (in.)	120.0	150.0
Cladding Thickness (in.)	0.0165	0.025
Number of Guide Tubes	16	5
Guide Tube Outer Diameter (in.)	0.535	0.980
Guide Tube Inner Diameter (in.)	0.511	0.900
Guide Tube Material	SS	Zr

* The current maximum enrichment is 4.1 w/o. It is proposed to increase the maximum enrichment to 4.8 w/o.

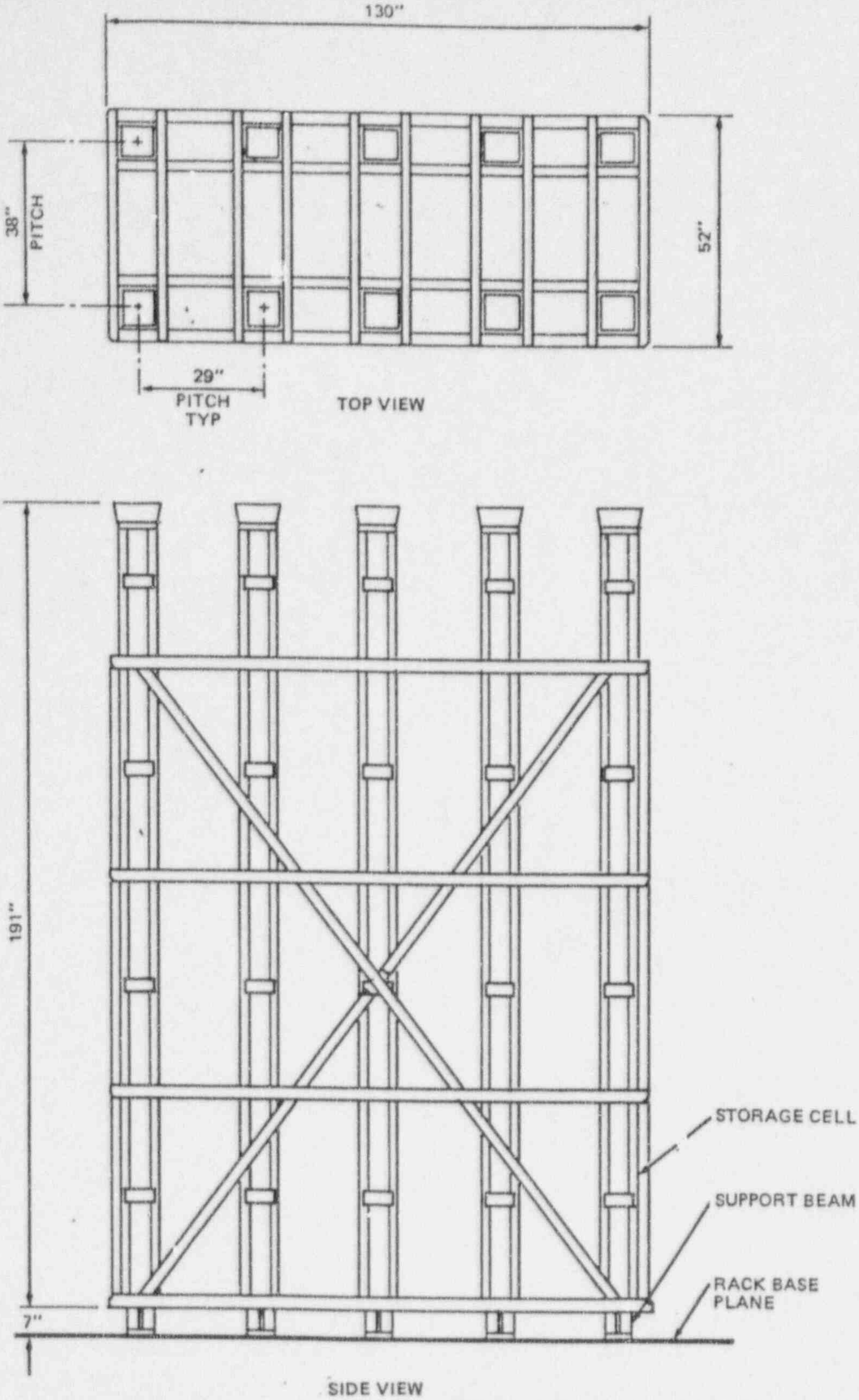
** Fuel rod pitch is the spacing between fuel rods measured as the distance from centerline to centerline of the rod. Both assembly types are square pitch arrays.

*** In the future, the fuel pellet diameter may increase to 0.3255 inches. There will be no impact on criticality because the present analyses assume a fuel stack height density which bounds the small amount of additional fuel which would result from the increase in fuel pellet diameter.

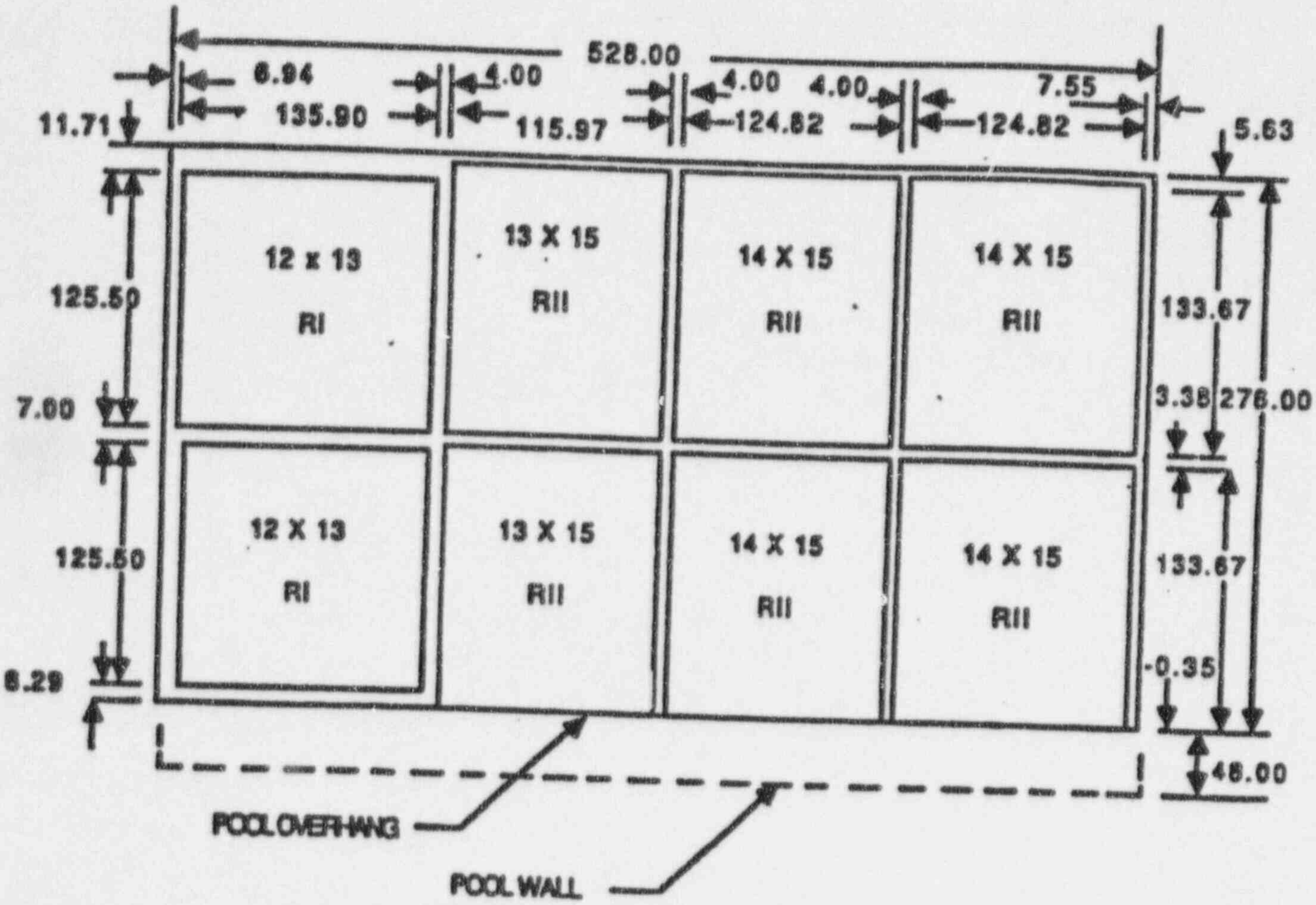
Table 2-2

SPENT FUEL RACK DATA
(Each Unit)

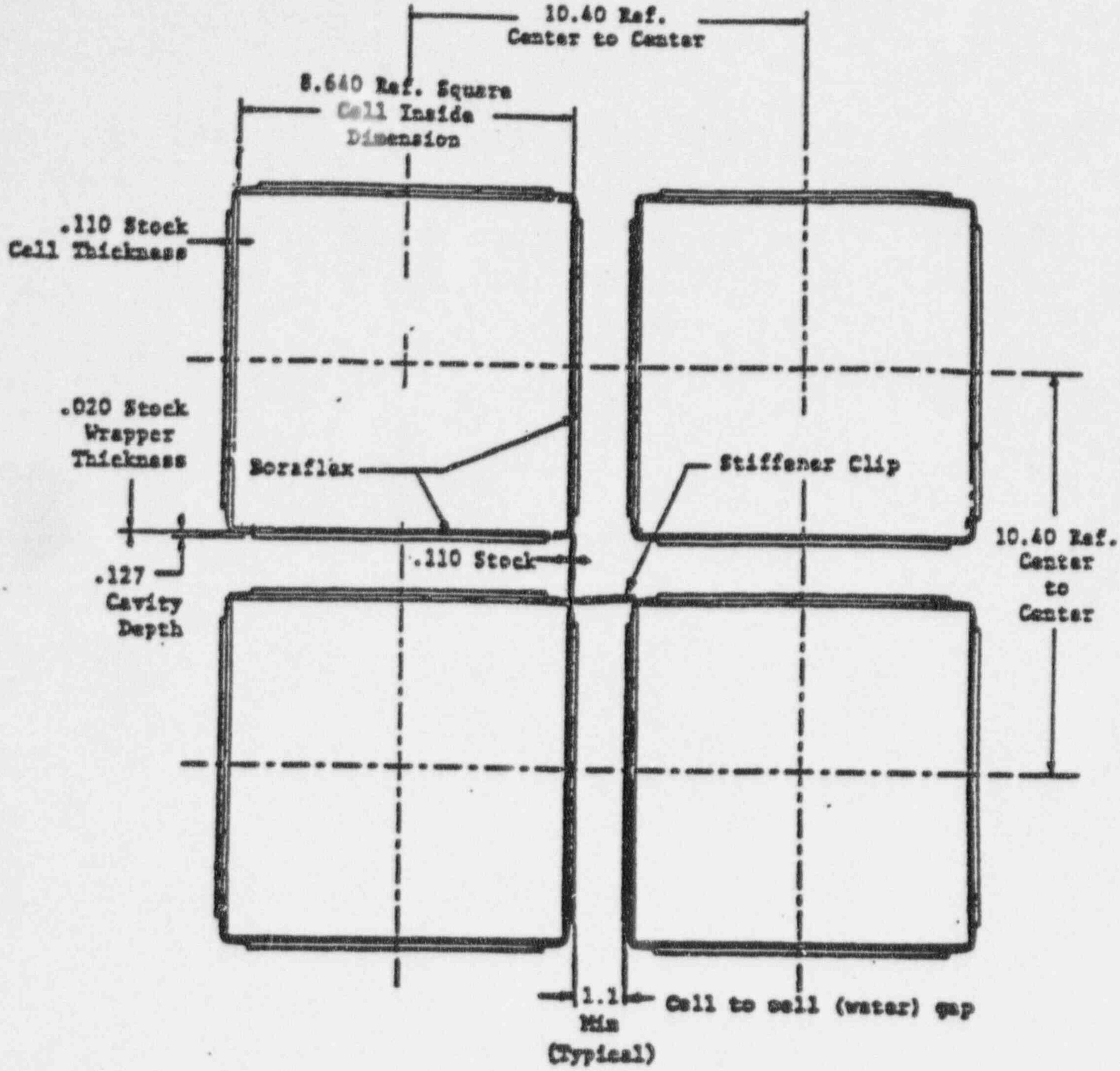
	Region I	Region II
Number of Storage Locations	312	1230
Number of Rack Arrays	Two 12x13	Four 14x15 Two 13x15
Center-to-Center Spacing (inches)	10.40	8.85
Cell Inside Width (inches)	8.64	8.63
Type of Fuel	SONGS 2 and 3 16x16 and/or SONGS 1 14x14	SONGS 2 and 3 16x16 and/or SONGS 1 14x14
Rack Assembly Outline Dimensions (inches)	126 x 136 x 198.5	125 x 134 x 198.5 (14 x 15) 116 x 134 x 198.5 (13 x 15)



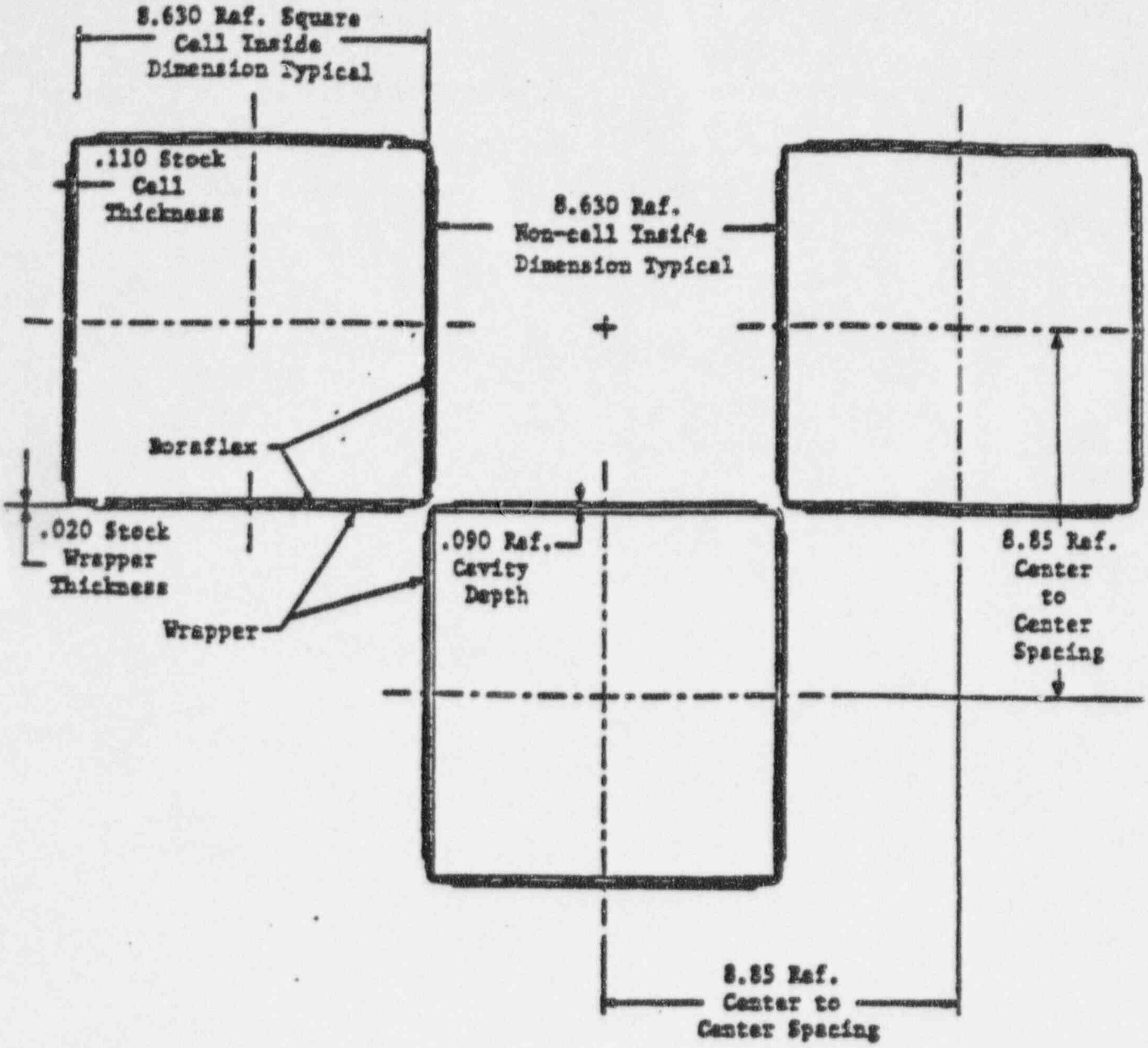
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
NEW FUEL STORAGE RACK ARRANGEMENT
FIGURE 2-1



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
SPENT FUEL STORAGE RACK ARRANGEMENT
FIGURE 2-2



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
REGION I FUEL STORAGE RACK
FIGURE 2-3



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
REGION II FUEL STORAGE RACK
FIGURE 2-4

3. FUEL STORAGE AND HANDLING CRITICALITY EVALUATION

3.1 INTRODUCTION

This section presents the criticality analyses performed to increase the SONGS 2 and 3 maximum enrichment from 4.1 w/o to 4.8 w/o.

The results show that the SONGS 2 and 3 new fuel storage racks, spent fuel storage racks, and fuel handling equipment can safely accommodate unshimmed (No burnable poison rods - including IFBA, Gd, or Er), 4.8 w/o enriched SONGS 2 and 3 fuel. The neutron multiplication factor (K-eff) is less than 0.95 for normal conditions and all postulated accidents. In addition, a minimum boron concentration of 1850 PPM (1800 PPM with 50 PPM uncertainty) is sufficient to maintain the k-eff below 0.95 for SONGS 2 and 3 fuel with up to 4.8 w/o enrichment under postulated accident conditions in the spent fuel pool.

Although the requested maximum fuel pin enrichment for SONGS 2 and 3 is 4.8 w/o, the analyses were performed for up to 5.1 w/o enrichment and the results bound the requested 4.8 w/o for SONGS 2 and 3 fuel.

The minimum burnup criteria for unrestricted storage of SONGS 1, 2, and 3 fuel in Region II have been re-calculated. Also, for peripheral pool locations, substantially lower burnups than required for interior locations have been calculated. The large neutron leakage from the peripheral locations permits a lower discharge burnup than required for the interior locations.

3.2 ACCEPTANCE CRITERIA FOR CRITICALITY

The acceptance criteria for criticality for the new and spent fuel storage racks can be found in NUREG-0800, 'Standard Review Plan', and the NRC's 'OT Position For Review And Acceptance Of Spent Fuel Storage And Handling Applications'.[1]

(1) For new fuel storage racks, the neutron multiplication factor (k-eff) shall be less than about 0.95 when fully loaded and flooded with potential

moderators such as nonborated water fire extinguishant aerosols. K-eff will not exceed 0.98 with fuel of the highest anticipated reactivity in place assuming optimum moderation.

(2) For spent fuel storage racks, the neutron multiplication factor (k-eff) shall be less than or equal to 0.95, including all uncertainties, under all conditions.

3.3 CRITICALITY ANALYTICAL METHODS

3.3.1 Compliance With Regulatory Standards

The analytical methods employed herein conform with:

- o ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System

- o ANSI 57.2-1983, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 6.4.2

- o ANSI/ANS-8.1-1983 (formerly ANSI N16.1-1975), "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"

- o ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety"

- o NUREG-0800, Rev 2, NRC Standard Review Plan, Section 9.1.1, "New Fuel Storage"

- o NUREG-0800, Rev 3, NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage"

- o NRC guidance, "NRC OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 14, 1978, as amended by NRC letter dated January 18, 1979

3.3.2 Computer Programs

CELLDAN, NITAWL-II, KENO V.a, and CASMO-3 are the computer programs used in the analyses.[2,3]

CELLDAN calculates the atoms/barn-cm of U^{235}, U^{238}, and Oxygen in the UO_2 fuel. CELLDAN also calculates the atoms/barn-cm of Hydrogen, Oxygen, B^{10}, and B^{11} in the water. Finally, CELLDAN calculates the Dancoff factor, and U^{235} and Oxygen scattering cross-sections per U^{238} atom for NITAWL-II.

NITAWL-II generates a binary cross-section library for KENO V.a. The library contains 27 group cross-section data for every nuclide in the KENO V.a problem. Using the U^{238} number density, Dancoff factor, and U^{235}/Oxygen scattering cross-sections per U^{238} atom from CELLDAN, NITAWL-II uses the Nordheim Method to do resonance shielding of the U^{238} cross-section.

KENO V.a is the nuclear industry standard program for criticality analyses. KENO V.a is a three-dimensional, multi-group, Monte Carlo program.

CASMO-3 is a multi-group two-dimensional transport theory program for calculations on BWR and PWR fuel assemblies. It is extensively used by utilities in the U.S. In these analyses, CASMO-3 is used for two purposes. First, CASMO-3 is used to evaluate the reactivity variations (Δk) due to the rack and Boraflex manufacturing tolerances. Second, CASMO-3 is used to generate the initial enrichment versus discharge burnup criteria for Region II storage.

3.4 NEW FUEL STORAGE RACKS

This section presents the criticality analyses of the new fuel storage racks. Although the maximum requested enrichment of the fuel for SONGS 2 and 3 is 4.8 w/o, the analyses show that the new fuel storage racks can accommodate up to 5.1 w/o. The results clearly bound 4.8 w/o.

3.4.1 Calculational Methodology

The final k-eff for the new fuel storage racks is calculated as:

$$
\begin{aligned}
\text{k-eff}_{\text{Final}} = \text{k-eff}_{\text{Reference}} &+ \text{Methodology Bias} \\
&+ \text{SQRT}\Big[(\text{95/95 Methodology Bias Uncertainty})^2 \\
&+ (\text{95/95 KENO V.a Uncertainty in k-eff}_{\text{Reference}})^2 \\
&+ \sum (\Delta k_{\text{Tolerance}})^2 \Big]
\end{aligned}
$$

3.4.1.1 Reference Model

The reference KENO V.a model for the new fuel storage racks is:

(1) 5.10 w/o -- SONGS 2 and 3 **UN-IRRADIATED** Fuel
No U-234 or U-236 in the fuel pellet. These naturally occurring isotopes act as a neutron absorber in the pellet. Thus it is conservative to remove them.

(2) UO_2 theoretical density = 96%

(3) Unshimmed (No burnable poison rods - including IFBA, Gd, or Er)

(4) All materials at 20 degrees C (68 F)

(5) Nominal dimensions

(6) The 2.0 inch wide SS-304 angle pieces which form the storage locations are modelled.
No other structural materials are considered.

(7) 1/4 of the new fuel rack storage array and neighboring concrete walls are modelled. Reflective boundary conditions have the effect of modelling the full new fuel storage rack array.

(8) Water density of 0.02 gms/cc
(Conservatively bounds all normal humidity variations)

(9) Axially, the active fuel region is reflected by concrete at the bottom and water at the top.

KENO V.a is executed with 503 neutron generations and 2000 neutrons per generation. KENO V.a results are used after skipping three generations.

Two KENO V.a models are used to analyze the flooding:

(1) Water density from 0.02 gms/cc to 0.7 gms/cc
(2) Water density from 0.7 gms/cc to 1.0 gms/cc

Although the spacing is 29 inches center-to-center, the fuel assemblies still interact neutronically with each other at water densities from 0.02 gms/cc to about 0.7 gms/cc. The reference 1/4 storage array model described above is used for these water densities.

For water densities greater than 0.7 gms/cc, the fuel assemblies are essentially neutronically isolated from each other. Therefore, an infinite array of fuel assemblies separated by 29 inches center-to-center is used for water densities greater than 0.7 gms/cc.

3.4.1.2 Methodology Bias

The bias and 95/95 uncertainty in the bias for CELLDAN, NITAWL-II, KENO V.a, and the 27 group cross-section library are 0.00928 and 0.00148, respectively. The bias and uncertainty were determined by analyses of 16 B&W critical experiments[4] for standard fuel storage (Table 3-1).

3.4.1.3 Exclusion Of Manufacturing Tolerances And Calculational Uncertainties

Since (1) the maximum final k-eff under all conditions is expected to be less than 0.91, and (2) delta k-eff from the statistical combination of manufacturing tolerances and calculational uncertainties is typically less than 0.01 (See sections 3.5.2.1 and 3.6.2.1), there is sufficient margin to 0.95 that these small contributions to the final k-eff can be neglected.

3.4.2 Results

Under all normal and postulated accident conditions, k-eff of the new fuel storage racks is less than 0.95 when fully loaded with un-irradiated, unshimmed 5.1 w/o ABB/CE fuel assemblies.

3.4.2.1 Normal Conditions

Under normal conditions, k-eff is less than 0.72 for dry storage of unshimmed, un-irradiated 5.1 w/o SONGS 2 and 3 fuel assemblies in the new fuel storage racks.

$$\text{k-eff}_{Final} = \text{k-eff}_{Reference} + \text{Methodology Bias}$$
$$= 0.70809 + 0.00928 = 0.71737$$

3.4.2.2 Postulated Accidents

The only significant postulated accident for the new fuel storage racks is flooding at a water density which maximizes k-eff. This accident is analyzed by calculating k-eff for the full range of water density from 0.02 gms/cc to 1.0 gms/cc.

At the optimum water density of 0.045 gms/cc, k-eff = 0.856 for storage of un-irradiated 5.1 w/o ABB-CE fuel assemblies in the new fuel storage racks.

$$\text{k-eff}_{Final} = \text{k-eff}_{Reference} + \text{Methodology Bias}$$
$$= 0.84683 + 0.00928 = 0.85611$$

When the new fuel racks are completely flooded with unborated water at 1.0 gm/cc, k-eff = 0.904.

$$\text{k-eff}_{Final} = \text{k-eff}_{Reference} + \text{Methodology Bias}$$
$$= 0.89516 + 0.00928 = 0.90444$$

The variation of k-eff with water density is given in Table 3-4 and Figure 3-3.

3.5 SPENT FUEL STORAGE RACKS -- REGION I

This section presents the criticality analyses of the Region I spent fuel storage racks. Although the maximum requested enrichment of the fuel for SONGS 2 and 3 is 4.8 w/o, the analyses show that the Region I spent fuel storage racks can accommodate up to 5.1 w/o. The results clearly bound 4.8 w/o SONGS 2 and 3 fuel assemblies and SONGS 1 fuel assemblies.

3.5.1 Calculational Methodology

The final k-eff for the Region I spent fuel storage racks is calculated as:

$$
\begin{aligned}
\text{k-eff}_{\text{Final}} = \text{k-eff}_{\text{Reference}} &+ \text{Methodology Bias} + \Delta k_{\text{Pool Water Temp}} \\
&+ \text{SQRT}\Big[\,(\text{95/95 Methodology Bias Uncertainty})^2 \\
&+ (\text{95/95 KENO V.a Uncertainty in k-eff}_{\text{Reference}})^2 \\
&+ \textstyle\sum (\Delta k_{\text{Tolerance}})^2 + (\Delta k_{\text{Eccentric}})^2\,\Big] \\
&+ \Delta k_{\text{Boraflex Gaps}}
\end{aligned}
$$

3.5.1.1 Reference Model

The reference KENO V.a model for the Region I spent fuel racks is:

(1) 5.10 w/o -- SONGS 2 and 3 **UN-IRRADIATED** Fuel
No U-234 or U-236 in the fuel pellet. These naturally occurring isotopes act as a neutron absorber in the pellet. Thus it is conservative to remove them.

(2) UO_2 theoretical density = 96%

(3) Unshimmed (No burnable poison rods - including IFBA, Gd, or Er)

(4) All materials at 20 degrees C (68°F)

(5) Nominal dimensions

(6) 0 PPM soluble boron
For analysis purposes, 0 PPM is "normal".

(7) No assembly grids or end fittings

(8) Nominal Boraflex thickness, length, and density

(9) 4% Boraflex width shrinkage

(10) Minimum B_4C content (w/o) in Boraflex

(11) No Boraflex gaps

The reference model has no Boraflex gaps. The reactivity effect of Boraflex gaps is added later.

(12) Infinite in lateral (X-Y) extent, finite in axial (Z) extent

(13) Water reflector (1 foot) at top and bottom of the active fuel length

KENO V.a is executed with 503 neutron generations and 2000 neutrons per generation. KENO V.a results are used after skipping three generations.

3.5.1.2 Methodology Bias

The bias and 95/95 uncertainty in the bias for CELLDAN, NITAWL-II, KENO V.a, and the 27 group cross-section library are 0.00928 and 0.00148, respectively. The bias and uncertainty were determined by analyses of 16 B&W critical experiments[4] for standard fuel storage (Table 3-1).

The bias (0.00928) is added directly to the reference k-eff. The 95/95 uncertainty in the bias (0.00148) is combined statistically with other random contributors to the final k-eff.

3.5.1.3 Pool Water Temperature Variation

The reference analysis temperature is 20 degrees C. CASMO-3 cases were run at 20 degrees C, 40 degrees C, 80 degrees C, 120 degrees C, and 120 degrees C + 10% void. At the bottom of the racks where pressure is greater than atmospheric, 120 degrees C is the approximate boiling temperature. K-effective decreases with increasing temperature and void at 0 PPM. Thus delta-k for pool water temperature variation is 0.0.

3.5.1.4 95/95 Methodology Bias Uncertainty

The 95/95 uncertainty in the bias (0.00148, section 3.5.1.2 above) is combined statistically with other random contributors to the final k-eff.

3.5.1.5 95/95 KENO V.a Uncertainty

KENO V.a results are reported as k-eff +/- sigma. The 95/95 KENO V.a uncertainty is $K_{95/95}$ * KENO V.a sigma. For 500 neutron generations, $K_{95/95}$ is 1.763 (Reference 5).

3.5.1.6 Manufacturing Tolerances

The contributions to the $\Delta k_{Tolerance}$ for manufacturing tolerances include:

Boraflex density	(+/- 0.121 grams)
Boraflex thickness	(+/- 0.007 inches)
Boraflex width	(+/- 0.063 inches)
Cell wall thickness	(+/- 0.004 inches)
Wrapper thickness	(+/- 0.004 inches)
Minimum cell inner dimension	(+/- 0.025 inches)
Center-to center spacing	(+/- 0.060 inches)

All tolerances - except Boraflex density - are rack manufacturer values. The Boraflex density tolerance is calculated from the minimum B-10 loading value provided by the Boraflex manufacturer.

Rather than include a tolerance for fuel density and enrichment, the fuel is analyzed at 96% of theoretical density, and maximum anticipated enrichment. The effect of pellet manufacturing tolerances is negligible compared to the other tolerances which are included. No tolerance is included for the Boraflex length since the effect of 6 inch Boraflex gaps (Section 3.5.1.8 below) is so much larger.

No statistical tolerance for B-10 loading in Boraflex is used. The minimum loading is used for all cases.

The delta k-eff's due to storage rack and Boraflex sheet dimensional tolerances, and Boraflex density are calculated with CASMO-3 because the delta k-eff's are small and can be lost in the statistical uncertainty in KENO V.a results (KENO V.a results are k-eff +/- sigma). The CASMO-3 delta k results are combined statistically (Square root of the sum of the squares) with uncertainties in methodology bias, reference KENO V.a k-eff, and eccentric positioning of assemblies in the storage locations.

The total statistical uncertainty is 0.00811.

3.5.1.7 Eccentric Loading

The effect of asymmetric locations of fuel assemblies in the storage cells was evaluated. The results are a higher k-eff for assemblies centered (k-eff = 0.91599) in the storage locations than for assemblies off-center (k-eff = 0.91378 for an infinite pattern of four assemblies moved as close together as possible in the corners of their storage locations). Therefore, $\Delta k_{Eccentric}$ is 0.0.

3.5.1.8 Boraflex Gap methodology

The $\Delta k_{Boraflex\ gaps}$ term is based on randomly placing a 6 inch gap (4% shrinkage) in every Boraflex panel. The gaps are in addition to the 4% width shrinkage in the reference model. The analysis is done with KENO V.a, and assumes an infinite array of storage locations.

Although the gaps are randomly located in the Boraflex panels, the effect of a gap is to increase the final k-eff. A gap never decreases k-eff. Therefore, delta-k due to gaps can not be combined statistically with the manufacturing tolerances and calculational uncertainties; delta-k due to gaps is an additive term to the reference k-eff.

The delta-k contribution ($\Delta k_{Boraflex\ gaps}$) from a randomly placed 6 inch Boraflex gap in every Boraflex panel is 0.00792.

3.5.2 Region I Results

The neutron multiplication factor (k-eff) for the Region I spent fuel storage racks completely loaded with un-irradiated, unshimmed 5.1 w/o fuel is less than 0.95, including all uncertainties, under all conditions.

3.5.2.1 Normal Conditions

Under non-accident conditions, k-eff is 0.941 for storage of unshimmed, un-irradiated 5.1 w/o SONGS 2 and 3 fuel assemblies in the Region I spent fuel storage racks at a soluble boron concentration of 0 PPM .

	Delta-k	k-eff
Nominal KENO Reference Reactivity:		0.91599
Methodology Bias:	+ 0.00928	
Pool Water Temperature Variation:	0.00000	
TOTAL Bias + Temp Variation	+ 0.00928	
Best Estimate Nominal k-eff		0.92527
Tolerances & Uncertainties		
Methodology Bias Uncertainty (95/95)	0.00148	
KENO Calculational Uncertainty (95/95)	0.00120	
Boraflex Density	0.00242	
Boraflex Thickness	0.00352	
Boraflex Width	0.00050	
Cell Wall Thickness	0.00083	
Wrapper Thickness	0.00298	
Minimum Cell Inner Dimension	0.00137	
Center-to-center Spacing	0.00568	
Eccentric Positioning	0.00000	
TOTAL Uncertainty (statistical)	+ 0.00811	
Final k-eff Including Tolerances/Uncertainties		0.93338
Boraflex Gaps	+ 0.00792	
Final k-eff With Boraflex Gaps		0.94130

3.5.2.2 Postulated Accidents

Under postulated accident conditions, k-eff remains below 0.95 when credit is taken for 1800 PPM (no uncertainty) soluble boron.

The analyses were performed for un-irradiated, unshimmed 5.1 w/o SONGS 2 and 3 fuel. The results bound 4.8 w/o SONGS 2 and 3 fuel assemblies, and SONGS 1 fuel assemblies.

The accidents considered for the Region I spent fuel storage racks include:

(1) Fuel Assembly Dropped Horizontally On Top Of The Racks
(2) Fuel Assembly Dropped Vertically Into A Storage Location Already Containing A Fuel Assembly
(3) Fuel Assembly Dropped to The SFP Floor
(4) Loss Of Cooling Systems
(5) Fuel Misloading Accidents
(6) Heavy Load Drops
(7) Seismic Event
(8) Boron Dilution

The proposed design of the 4.8 w/o enriched fuel will result in a slight weight increase. However, the seismic event is bounded by the analyses performed for the rerack project and does not need to be considered further.

A boron dilution accident is not analyzed since the spent fuel storage racks have k-eff of 0.941 at a soluble boron concentration of 0 PPM.

For accident conditions, the double contingency principle of ANSI/ANS-8.1-1983 (formerly ANSI N16.1-1975) is applied. This principle states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Therefore, for those accidents during which k-eff increases, the presence of soluble boron may be credited, since the absence of boron would be a second unlikely event.

3.5.2.2.1 Fuel Assembly Dropped Horizontally On Top Of The Racks

Analysis has shown that more than 12 inches of water separates the active fuel region of the dropped assembly lying on top of the racks from the active fuel region of assemblies in the storage racks. Thus the fuel regions are neutronically isolated and reactivity does not increase.

A single un-irradiated, unshimmed 5.1 w/o fuel assembly in water at 68 degrees F and 0 PPM has k-eff = 0.92.

3.5.2.2.2 Fuel Assembly Dropped Vertically Into A Storage Location Already Containing A Fuel Assembly

Analysis has shown that more than 12 inches of water separates the active fuel region of the dropped assembly from the active fuel region of assemblies in the storage racks. Thus the fuel regions are neutronically isolated and reactivity does not increase.

3.5.2.2.3 Fuel Assembly Dropped To The SFP Floor

A dropped fuel assembly can not fit between rack modules. However, a fuel assembly can fit between a Region I module and the pool wall. A soluble boron concentration of 1800 PPM (no uncertainty) will keep k-eff less than 0.95.

3.5.2.2.4 Loss Of Cooling Systems

From the reference temperature of 20 degrees C (68 F), k-effective decreases with increasing temperature and void at 0 PPM (Section 3.5.1.3). No credit is taken for soluble boron in this accident scenario.

3.5.2.2.5 Fuel Misloading Accidents

Since the Region I racks can accommodate un-irradiated, unshimmed 5.1 w/o fuel in every storage location at 0 PPM, the fuel misloading accident is not credible for the Region I racks.

3.5.2.2.6 Heavy Load Drops

Two potential heavy load drops are considered:

(1) Spent Fuel Pool Gate Drop

(2) Test Equipment Skid Drops

These heavy loads may fall onto the Region I spent fuel storage racks containing:

(1) Fuel assemblies stored without control rods

(2) Fuel assemblies stored with inserted control rods

(1) Fuel Assemblies Stored Without Control Rods

For unshimmed, un-irradiated 5.1 w/o fuel assemblies stored without control rods, k-eff remains below 0.95 at a boron concentration of 0 PPM following a heavy load drop on to the Region I racks provided the following lift height and weight limits are met:

Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool except for the following two cases:

a. Spent fuel pool gates shall not be carried at a height greater than 30 inches (elevation 36′ 4") over the fuel racks.

b. Test equipment skid (4500 pounds) shall not be carried at a height greater than 72 inches (elevation 39′ 10") over rack cells which contain Unit 2/3 fuel assemblies or greater than 30 feet 8 inches (elevation 64′ 6") over rack cells which contain Unit 1 fuel assemblies.

Structural analyses have been performed which demonstrate that there is no significant damage to the spent fuel racks in the active fuel and Boraflex region if the above weight and height restrictions are observed. The structural analyses and resulting restrictions were developed for postulated drops of heavy loads and subsequent penetrations into the Region II racks where the storage cells share walls. The penetration distances provided from these analyses are conservative for the Region I racks because each Region I storage cell is separately enclosed by a cell wall. Increasing the number of cell walls reduces the penetration distances of dropped loads, since the impact is shared by more supporting surfaces.

Test Equipment Skid

The top of a SONGS 2/3 fuel assembly is 13.2" below the top of the racks. The top of a SONGS 1 fuel assembly is 51.5" below the top of the racks. If the test equipment skid is dropped from 72" above the racks containing SONGS 2 and 3 assemblies, it is calculated to penetrate only 13.0" and does not contact any assembly upper end fittings. If the test equipment skid is dropped from 30′ 8" above the racks containing SONGS 1 assemblies, it is calculated to penetrate only 16.0" and does not contact any assembly upper end fittings. There are no significant deformations in the body of the racks which would alter the center to center spacing of the fuel assemblies or degrade the performance of the Boraflex. Therefore, without further analyses, k-eff is less than 0.95 at 0 PPM and credit for 1800 PPM is not needed.

Spent Fuel Pool Gate

Previous evaluations of the fuel pool gate drop event determined that up to six SONGS 2 and 3 fuel assemblies could have been impacted and damaged, as discussed in UFSAR Section 15.7.3.6. Recently, the structural and radiological consequences of the gate drop event have been reevaluated[6] to revise conservatisms which were used in the previous analyses. The new analyses concluded that only one fuel assembly would be impacted and potentially damaged.

The dropped fuel pool gate is calculated to penetrate 9.5" for the primary impact. The gate then rotates to a secondary impact. During rotation, one gate corner penetrates one storage cell 13.9" and makes contact with a single assembly. No other assemblies are damaged during the secondary impact. The active fuel and Boraflex region are about 20" below the top of the upper end fitting. There are no significant deformations in the body of the racks which would alter the center to center spacing of the fuel assemblies or degrade the performance of the Boraflex. Therefore, without further analyses, k-eff is less than 0.95 at 0 PPM and credit for 1800 PPM is not needed.

(2) <u>Fuel Assemblies Stored With Control Rods</u>

Control rods stored integrally with fuel assemblies extend above the top of the fuel assembly upper end fitting about 1.4" for a SONGS 1 fuel assembly and 11.1" for a SONGS 2 and 3 fuel assembly. Therefore, postulated drops of heavy loads represent potentially greater physical damage to SONGS 2 and 3 fuel assemblies with inserted control rods. SONGS 1 inserted control rods do not pose any damage to the fuel assembly since dropped heavy loads can not penetrate the racks far enough to impact the top of the control rods.

Therefore, to prevent any damage to fuel assemblies and the Boraflex region of the storage cells, the following administrative controls are imposed on SONGS 2 and 3 fuel:

(a) Prior to lifting or lowering of the test equipment skid over the spent fuel racks, all control rods shall be removed from the potential impact zone. A minimum area of 10 by 12 cells shall be designated as the potential drop impact zone beneath test equipment while lifting or lowering over the Region I racks.

(b) When moving the test equipment skid above the fuel racks after being lowered, the skid height shall not exceed 11" above the top of the racks.

(c) Prior to and during rigging for removal and reinstallation of the transfer pool bulkhead gates, control rods shall be relocated outside of the potential gate primary impact zone. The primary impact zone for the Transfer Pool Gate is located within storage racks nos. 1 and 2, which are Region I type racks. Cells adjacent to the gate in rows F through P and 1 through 3 are included (30 cells total).

Therefore, for un-irradiated, unshimmed 5.1 w/o fuel assemblies stored with control rods, k-eff remains below 0.95 at a boron concentration of 0 PPM following a heavy load drop on to the Region I racks.

Increasing the enrichment to 4.8 w/o requires no changes to the administrative controls governing heavy loads.

3.5.2.3 Boraflex Erosion or Dissolution

Recently, elevated silica concentrations have been observed in spent fuel pools of numerous plants. SONGS has also experienced elevated silica concentrations in the SFP. This elevated concentration originates from the Boraflex panels.

Calculations have been performed to investigate the criticality consequences due to the loss of Boraflex thickness in the SONGS 2 and 3 Region I spent fuel storage racks. Using the reference KENO V.a models described above, up to 50% decrease in Boraflex thickness has been evaluated. The results are listed in Table 3-5.

Assuming un-irradiated 5.1 w/o fuel, and a 6 inch random gap in every Boraflex panel, about 20% of the Boraflex thickness can be lost uniformly before k-eff reaches 0.95 at a soluble boron concentration of 0 PPM (Table 3-5). The current spent fuel pool water silica level indicates that the loss of Boraflex has been negligible (less than 3 PPM in five years). Based on this experience, the loss of Boraflex and its reactivity effect for the remaining lifetime at SONGS 2 and 3 is expected to be insignificant. Edison will continue to monitor the Boraflex integrity through the Boraflex coupon surveillance program; silica levels in the pool will be monitored; and, industry (EPRI) experience with Boraflex erosion will be closely followed.

To date, four Boraflex surveillance coupons from each unit have been tested. The first coupon was removed during the cycle 5-6 refueling outage; the second coupon was removed during the cycle 6-7 refueling outage; the third and fourth coupons were removed during the cycle 7-8 refueling outage. The results of the coupon tests and inspections show that the Boraflex is performing within the EPRI acceptance criteria.

3.6 SPENT FUEL STORAGE RACKS -- REGION II

This section documents the criticality analyses of the Region II spent fuel storage racks. Although the maximum requested enrichment of the fuel for SONGS 2 and 3 is 4.8 w/o, the analyses were performed for an enrichment up to 5.1 w/o. The results clearly bound 4.8 w/o SONGS 2 and 3 fuel, and SONGS 1 fuel assemblies.

3.6.1 Calculational Methodology

The final k-eff for the Region II spent fuel storage racks is calculated as:

$$\begin{aligned} \text{k-eff}_{Final} = {} & \text{k-eff}_{Reference} + \text{Methodology Bias} + \Delta k_{Pool\ Water\ Temp} \\ & + \text{SQRT}\Big[(\text{95/95 Methodology Bias Uncertainty})^2 \\ & + (\text{95/95 KENO V.a Uncertainty in k-eff}_{Reference})^2 \\ & + \textstyle\sum (\Delta k_{Tolerance})^2 + (\Delta k_{Eccentric})^2 \Big] \\ & + \Delta k_{Boraflex\ Gaps} \\ & + \Delta k_{Axial\ Burnup\ Effect} \end{aligned}$$

Spent fuel storage in the Region II racks takes credit for fuel assembly burnup. This methodology - called 'Reactivity Equivalencing' - is described in section 3.6.1.10 below.

3.6.1.1 Reference Model

The reference KENO V.a model for the Region II spent fuel racks is:

(1) 1.85 w/o -- SONGS 2 and 3 **UN-IRRADIATED** Fuel
2.56 w/o -- SONGS 1 **UN-IRRADIATED** Fuel
No U-234 or U-236 in the fuel pellet. These naturally occurring isotopes act as a neutron absorber in the pellet. Thus it is conservative to remove them.

(2) UO_2 theoretical density = 96%

(3) Unshimmed (No burnable poison rods - including IFBA, Gd, or Er)

(4) All materials at 20 degrees C (68°F)

(5) Nominal dimensions

(6) 0 PPM soluble boron
For analysis purposes, 0 PPM is "normal".
(7) No assembly grids or end fittings
(8) Nominal Boraflex thickness, length, and density
(9) 4% Boraflex width shrinkage
(10) Minimum B_4C content (w/o) in Boraflex
(11) No Boraflex gaps
The reference model has no Boraflex gaps. The reactivity effect of Boraflex gaps is added later.
(12) Infinite in lateral (X-Y) extent, finite in axial (Z) extent
(13) Water reflector (1 foot) at top and bottom of the active fuel length

KENO V.a is executed with 503 neutron generations and 2000 neutrons per generation. KENO V.a results are used after skipping three generations.

3.6.1.2 Methodology Bias

The bias and 95/95 uncertainty in the bias for CELLDAN, NITAWL-II, KENO V.a, and the 27 group cross-section library are 0.00928 and 0.00148, respectively. The bias and uncertainty were determined by analyses of 16 B&W critical experiments[4] for standard fuel storage (Table 3-1).

The bias (0.00928) is added directly to the reference k-eff. The 95/95 uncertainty in the bias (0.00148) is combined statistically with other random contributors to the final k-eff.

3.6.1.3 Pool Water Temperature Variation

The reference analysis temperature is 20 degrees C. CASMO-3 cases were run at 20 degrees C, 40 degrees C, 80 degrees C, 120 degrees C, and 120 degrees C + 10% void. At the bottom of the racks where pressure is greater than atmospheric, 120 degrees C is the approximate boiling temperature.
K-effective decreases with increasing temperature and void at 0 PPM. Thus delta-k for pool water temperature variation is 0.0.

3.6.1.4 95/95 Methodology Bias Uncertainty

The 95/95 uncertainty in the bias (0.00148, section 3.6.1.2 above) is combined statistically with other random contributors to the final k-eff.

3.6.1.5 95/95 KENO V.a Uncertainty

KENO V.a results are reported as k-eff +/- sigma. The 95/95 KENO V.a uncertainty is $K_{95/95}$ * KENO V.a sigma. For 500 neutron generations, $K_{95/95}$ is 1.763 (Reference 5).

3.6.1.6 Manufacturing Tolerances

The contributions to the $\Delta k_{Tolerance}$ for manufacturing tolerances include:

Boraflex density	(+/- 0.132 grams)
Boraflex thickness	(+/- 0.007 inches)
Boraflex width	(+/- 0.063 inches)
Cell wall thickness	(+/- 0.004 inches)
Wrapper thickness	(+/- 0.004 inches)
Minimum cell inner dimension	(+/- 0.025 inches)
Center-to center spacing	(Not Applicable to Region II)

All tolerances - except Boraflex density - are rack manufacturer values. The Boraflex density tolerance is calculated from the minimum B-10 loading value provided by the Boraflex manufacturer.

Rather than include a tolerance for fuel density and enrichment, the fuel is analyzed at 96% of theoretical density, and maximum anticipated enrichment. The effect of pellet manufacturing tolerances is negligible compared to the other tolerances which are included. No tolerance is included for the Boraflex length since the effect of 6 inch Boraflex gaps (3.6.1.8 below) is so much larger.

No statistical tolerance for B-10 loading in Boraflex is used. The minimum loading is used for all cases.

The delta k-eff's due to storage rack and Boraflex sheet dimensional tolerances, and Boraflex density are calculated with CASMO-3 because the delta k-eff's are small and can be lost in the statistical uncertainty in KENO V.a results (KENO V.a results are k-eff +/- sigma). The CASMO-3 delta k results are combined statistically (Square root of the sum of the squares) with uncertainties in methodology bias, reference KENO V.a k-eff, and eccentric positioning of assemblies in the storage locations.

The total statistical uncertainty is 0.00726.

3.6.1.7 Eccentric Loading

The effect of asymmetric locations of fuel assemblies in the storage cells was evaluated. The results are a higher k-eff for assemblies centered (k-eff = 0.92356) in the storage locations than for assemblies off-center (k-eff = 0.91714 for an infinite pattern of four assemblies moved as close together as possible in the corners of their storage locations). Therefore, $\Delta k_{Eccentric}$ is 0.0.

3.6.1.8 Boraflex Gap methodology

The $\Delta k_{Boraflex\ gaps}$ term is based on randomly placing a 6 inch gap (4% shrinkage) in every Boraflex panel. The gaps are in addition to the 4% width shrinkage in the reference model. The analysis is done with KENO V.a, and assumes an infinite array of storage locations.

Although the gaps are randomly located in the Boraflex panels, the effect of a gap is to increase the final k-eff. A gap never decreases k-eff. Therefore, delta-k due to gaps can not be combined statistically with the manufacturing tolerances and calculational uncertainties; delta-k due to gaps is an additive term to the reference k-eff.

The delta-k contribution ($\Delta k_{Boraflex\ gaps}$) from a randomly placed 6 inch Boraflex gap in every Boraflex panel is 0.00779.

3.6.1.9 Axial Burnup Effects

The axial burnup effect ($\Delta k_{Axial\ Burnup\ Effect}$) is evaluated by converting a burnup distribution in terms of GigaWatt-Days per metric ton of Uranium fuel (GWD/T) into an equivalent enrichment (w/o U^{235}) distribution. Then KENO V.a cases are run for uniform axial burnup (single axial enrichment) and axial burnup distribution (varying axial enrichment). A higher k-eff results from the uniform burnup case compared to the axially varying burnup case. The main reason for this is that, although the burnup is less on the ends, higher neutron leakage compensates for this. Therefore, $\Delta k_{Axial\ Burnup\ Effect}$ is 0.0.

3.6.1.10 Reactivity Equivalencing For Burnup Credit

Spent fuel storage in the Region II spent fuel storage racks is achievable by means of 'reactivity equivalencing'. The concept of 'reactivity equivalencing' is based on the fact that reactivity decreases with fuel assembly burnup. A series of reactivity calculations are performed to generate a set of 'enrichment - fuel assembly discharge burnup' pairs which all give the equivalent k-eff when the fuel is stored in the Region II racks. This is the methodology approved by the NRC for the current burnup curves developed during the reracking project.

The enrichment - burnup pairs were generated with CASMO-3. CASMO-3 allows a fuel assembly to be depleted at hot full power reactor conditions, and then placed into fuel storage rack geometry at 20 degrees C, 0 PPM soluble boron concentration, and no Xenon. The most reactive point in time for a fuel assembly after discharge is conservatively approximated by removing the Xenon. Samarium buildup after shutdown is conservatively not modelled.

Because the burnup history is not known exactly for the discharged fuel assemblies, the fuel assembly isotopic content (U, Pu, etc) and distribution is not known exactly. Therefore, a 5% penalty is applied to the total reactivity decrement calculated by CASMO-3 from beginning of life to the burnup of interest.

3.6.2 Region II Results

The neutron multiplication factor (k-eff) for the Region II spent fuel storage racks is less than 0.95, including all uncertainties, under all conditions.

3.6.2.1 Normal Conditions

Under non-accident conditions and a soluble boron concentration of 0 PPM, k-eff is 0.948 for unrestricted storage in the Region II racks of unshimmed SONGS 1, 2, and 3 fuel assemblies which have the initial enrichment and discharge burnup combinations given in Sections 3.6.2.2.2 and 3.6.2.2.3.

	Delta-k	k-eff
Nominal KENO Reference Reactivity:		0.92356
Methodology Bias:	+ 0.00928	
Pool Water Temperature Variation	0.00000	
TOTAL Bias + Temp Variation	+ 0.00928	
Best Estimate Nominal k-eff		0.93284
Tolerances & Uncertainties		
Methodology Bias Uncertainty (95/95)	0.00148	
KENO Calculational Uncertainty (95/95)	0.00102	
Boraflex Density	0.00363	
Boraflex Thickness	0.00593	
Boraflex Width	0.00070	
Cell Wall Thickness	0.00020	
Wrapper Thickness	0.00020	
Minimum Cell Inner Dimension	0.00078	
Eccentric Positioning	0.00000	
TOTAL Uncertainty (statistical)	+ 0.00726	
Final k-eff Including Tolerances/Uncertainties		0.94010
Boraflex Gaps	+ 0.00779	
Final k-eff With Boraflex Gaps		0.94789

3.6.2.2 MINIMUM BURNUP CRITERIA FOR REGION II STORAGE

3.6.2.2.1 Fuel Assembly Burnup Determination

For the purpose of determining the eligibility for Region II storage of a fuel assembly, burnup of the fuel assembly will be estimated with the best available methodology. These best estimates of fuel assembly burnups will be decreased by their respective burnup calculational uncertainties, as defined by the following formula:

Fuel Assembly Burnup = (Calculated Assembly Burnup) *
(1.0 - 95/95 Calculational Uncertainty)

The 95/95 Calculational Uncertainty for SONGS 2 and 3 and SONGS 1 fuel assemblies have been determined based on the uncertainty components from each methodology. The values for SONGS 2 and 3 and SONGS 1 are 0.069 and 0.100, respectively.

3.6.2.2.2 SONGS 2 and 3 Fuel Assemblies

For interior rack locations, the assembly burnup vs. initial enrichment criteria for SONGS 2 and 3 fuel assemblies is given in Table 3-2 and Figure 3-1. This correlation can be applied to every rack location. Due to conservative assumptions and an increase in the assumed size and number of Boraflex gaps in the new methodology, the correlation results in a slightly higher acceptable burnup value than the current limitations for unrestricted placement of SONGS 2 and 3 fuel assemblies.

For rack locations with one or two faces towards the spent fuel pool sides (peripheral pool locations), analysis has been performed to determine a lower burnup. The resulting assembly vs. initial enrichment correlation is given in Table 3-3 and Figure 3-2.

Fuel assemblies, which do not meet the above burnup criteria for interior or peripheral pool storage, will be stored in accordance with the SONGS 2 and 3 Licensee Controlled Specifications.

3.6.2.2.3 SONGS 1 Fuel Assemblies

The minimum assembly burnup for unrestricted storage of SONGS 1 UO_2 fuel assemblies initially enriched to 4.0 w/o in SONGS 2 and 3 Region II racks is 18.0 GWD/T. This is equivalent to 2.56 w/o enrichment at 0.0 GWD/T.

For storage of 4.0 w/o initial enrichment SONGS 1 UO_2 fuel assemblies in Region II peripheral pool locations, the minimum burnup is 5.5 GWD/T.

All SONGS 1 UO_2 fuel assemblies remaining at San Onofre were initially enriched to 4.0 w/o except the following: A006 -- 3.40 w/o -- 26.593 GWD/T
A026 -- 3.40 w/o -- 31.499 GWD/T
A040 -- 3.40 w/o -- 26.220 GWD/T

The results for the initially enriched 4.0 w/o fuel bound these three assemblies. Therefore, no storage restrictions apply.

Fuel assemblies, which do not meet the above burnup criteria for interior or peripheral storage, will be stored in accordance with the SONGS 2 and 3 Licensee Controlled Specifications.

3.6.2.3 Postulated Accidents

Under postulated accident conditions, k-eff remains below 0.95 when credit is taken for 1800 PPM (no uncertainty) soluble boron.

The analyses were performed for un-irradiated, unshimmed 5.1 w/o SONGS 2 and 3 fuel. The results bound 4.8 w/o SONGS 2 and 3 fuel assemblies, and SONGS 1 fuel assemblies.

The accidents considered for the Region II spent fuel storage racks include:

(1) Fuel Assembly Dropped Horizontally On Top Of The Racks
(2) Fuel Assembly Dropped Vertically Into A Storage Location Already Containing A Fuel Assembly
(3) Fuel Assembly Dropped to The SFP Floor
(4) Loss Of Cooling Systems
(5) Fuel Misloading Accidents
(6) Heavy Load Drops

(7) Seismic Event

(8) Boron Dilution

The proposed design of the 4.8 w/o enriched fuel will result in a slight weight increase. However, the seismic event is bounded by the analyses performed for the rerack project and does not need to be considered further.

A boron dilution accident is not analyzed since the spent fuel storage racks have k-eff of 0.948 at a soluble boron concentration of 0 PPM.

For accident conditions, the double contingency principle of ANSI/ANS-8.1-1983 (formerly ANSI N16.1-1975) is applied. This principle states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Therefore, for those accidents during which k-eff increases, the presence of soluble boron may be credited, since the absence of boron would be a second unlikely event.

3.6.2.3.1 Fuel Assembly Dropped Horizontally On Top Of The Racks

Analysis has shown that more than 12 inches of water separates the active fuel region of the dropped assembly lying on top of the racks from the active fuel region of assemblies in the storage racks. Thus the fuel regions are neutronically isolated and reactivity does not increase.

A single un-irradiated, unshimmed 5.1 w/o fuel assembly in water at 68 degrees F and 0 PPM has k-eff = 0.92.

3.6.2.3.2 Fuel Assembly Dropped Vertically Into A Storage Location Already Containing A Fuel Assembly

Analysis has shown that more than 12 inches of water separates the active fuel region of the dropped assembly from the active fuel region of assemblies in the storage racks. Thus the fuel regions are neutronically isolated and reactivity does not increase.

3.6.2.3.3 Fuel Assembly Dropped To The SFP Floor

The separation between rack modules, and Region II rack modules and the pool walls/overhang does not permit this accident.

3.6.2.3.4 Loss Of Cooling Systems

From the reference temperature of 20 degrees C (68 F), k-effective decreases with increasing temperature and void at 0 PPM (Section 3.6.1.3). No credit is taken for soluble boron in this accident scenario.

3.6.2.3.5 Fuel Misloading Accidents

Taking credit for a soluble boron concentration of 1800 PPM (the current technical specification value without 50 PPM measurement uncertainty), k-eff is 0.932 for misloading nine (3x3) un-irradiated, unshimmed 5.1 w/o SONGS 2 and 3 fuel assemblies.

The misloaded assemblies are placed in empty rack locations surrounded by fuel assemblies which have the minimum permissible burnup for unrestricted storage.

Table 3-6 and Figure 3-4 provide the results of misloading 1, 2, 4, 6, 9, and 16 un-irradiated 5.1 w/o SONGS and 3 fuel assemblies into the Region II spent fuel storage racks.

3.6.2.3.6 Heavy Load Drops

[Note: The following discussion of heavy load drops is repeated from Section 3.5.2.2.6 for the Region I spent fuel storage racks, and has been modified to apply to the Region II racks.]

Two potential heavy load drops are considered:

(1) Spent Fuel Pool Gate Drop
(2) Test Equipment Skid Drops

These heavy loads may fall onto the Region II spent fuel storage racks containing:

(1) Fuel assemblies stored without control rods

(2) Fuel assemblies stored with inserted control rods

(1) <u>Fuel Assemblies Stored Without Control Rods</u>

For fuel assemblies stored without control rods, k-eff remains below 0.95 at a boron concentration of 0 PPM following a heavy load drop on to the Region II racks provided the following lift height and weight limits are met:

Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool except for the following two cases:

a. Spent fuel pool gates shall not be carried at a height greater than 30 inches (elevation 36′ 4") over the fuel racks.

b. Test equipment skid (4500 pounds) shall not be carried at a height greater than 72 inches (elevation 39′ 10") over rack cells which contain Unit 2/3 fuel assemblies or greater than 30 feet 8 inches (elevation 64′ 6") over rack cells which contain Unit 1 fuel assemblies.

Structural analyses have been performed which demonstrate that there is no significant damage to the spent fuel racks in the active fuel and Boraflex region if the above weight and height restrictions are observed. The structural analyses and resulting restrictions were developed for postulated drops of heavy loads and subsequent penetrations into the Region II racks.

<u>Test Equipment Skid</u>

The top of a SONGS 2/3 fuel assembly is 13.2" below the top of the racks. The top of a SONGS 1 fuel assembly is 51.5" below the top of the racks. If the test equipment skid is dropped from 72" above the racks containing SONGS 2 and 3 assemblies, it is calculated to penetrate only 13.0" and does not contact any assembly upper end fittings. If the test equipment skid is dropped from 30′ 8" above the racks containing SONGS 1 assemblies, it is calculated to penetrate only 16.0" and does not contact any assembly upper end fittings.

There are no significant deformations in the body of the racks which would alter the center to center spacing of the fuel assemblies or degrade the performance of the Boraflex. Therefore, without further analyses, k-eff is less than 0.95 at 0 PPM and credit for 1800 PPM is not needed.

Spent Fuel Pool Gate

Previous evaluations of the fuel pool gate drop event determined that up to six SONGS 2 and 3 fuel assemblies could have been impacted and damaged, as discussed in UFSAR Section 15.7.3.6. Recently, the structural and radiological consequences of the gate drop event have been reevaluated[6] to revise conservatisms which were used in the previous analyses. The new analyses concluded that only one fuel assembly would be impacted and potentially damaged.

The dropped fuel pool gate is calculated to penetrate 9.5" for the primary impact. The gate then rotates to a secondary impact. During rotation, one gate corner penetrates one storage cell 13.9" and makes contact with a single assembly. No other assemblies are damaged during the secondary impact. The active fuel and Boraflex region are about 20" below the top of the upper end fitting. There are no significant deformations in the body of the racks which would alter the center to center spacing of the fuel assemblies or degrade the performance of the Boraflex. Therefore, without further analyses, k-eff is less than 0.95 at 0 PPM and credit for 1800 PPM is not needed.

(2) Fuel Assemblies Stored With Control Rods

Control rods stored integrally with fuel assemblies extend above the top of the fuel assembly upper end fitting about 1.4" for a SONGS 1 fuel assembly and 11.1" for a SONGS 2 and 3 fuel assembly. Therefore, postulated drops of heavy loads represent potentially greater physical damage to SONGS 2 and 3 fuel assemblies with inserted control rods. SONGS 1 inserted control rods do not pose any damage to the fuel assembly since dropped heavy loads can not penetrate the racks far enough to impact the top of the control rods.

Therefore, to prevent any damage to fuel assemblies and the Boraflex region of the storage cells, the following administrative controls are imposed on SONGS 2 and 3 fuel:

(a) Prior to lifting or lowering of the test equipment skid over the spent fuel racks, all control rods shall be removed from the potential impact zone. A minimum area of 10 by 12 cells shall be designated as the potential drop impact zone beneath test equipment while lifting or lowering over the Region II racks.

(b) When moving the test equipment skid above the fuel racks after being lowered, the skid height shall not exceed 11" above the top of the racks.

(c) Prior to and during rigging for removal and reinstallation of the Cask Handling Pool Gate, control rods shall be relocated outside of the potential gate primary impact zone. The primary impact zone for the Cask Handling Pool Gate is located within storage racks nos. 7 and 8, which are Region II type racks. Cells adjacent to the gate in rows HH through SS and 51 through 54 are included (44 cells total).

Therefore, for fuel assemblies stored with control rods, k-eff remains below 0.95 at a boron concentration of 0 PPM following a heavy load drop on to the Region II racks.

Increasing the enrichment to 4.8 w/o requires no changes to the administrative controls governing heavy loads.

3.6.2.4 Boraflex Erosion or Dissolution

[Note: The following discussion of Boraflex erosion or dissolution is repeated from Section 3.5.2.3 for the Region I spent fuel storage racks, and has been modified to apply to the Region II racks.]

Recently, elevated silica concentrations have been observed in spent fuel pools of numerous plants. SONGS has also experienced elevated silica concentrations in the SFP. This elevated concentration originates from the Boraflex panels.

Calculations have been performed to investigate the criticality consequences due to the loss of Boraflex thickness in the SONGS 2 and 3 Region II spent fuel storage racks. Using the reference KENO V.a models described above, up to 50% decrease in Boraflex thickness has been evaluated. The results are listed in Table 3-5.

Assuming the racks fully loaded with fuel assemblies which meet the burnup criteria for unrestricted storage, and a 6 inch random gap in every Boraflex panel, about 7% of the Boraflex thickness can be lost before k-eff reaches 0.95 at a soluble boron concentration of 0 PPM (Table 3-5). The current spent fuel pool water silica level indicates that the loss of Boraflex has been negligible (less than 3 PPM in five years). Based on this experience, the loss of Boraflex and its reactivity effect for the remaining lifetime at SONGS 2 and 3 is expected to be insignificant. Edison will continue to monitor the Boraflex integrity through the Boraflex coupon surveillance program; silica levels in the pool will be monitored; and, industry (EPRI) experience with Boraflex erosion will be closely followed.

To date, four Boraflex surveillance coupons from each unit have been tested. The first coupon was removed during the cycle 5-6 refueling outage; the second coupon was removed during the cycle 6-7 refueling outage; the third and fourth coupons were removed during the cycle 7-8 refueling outage. The results of the coupon tests and inspections show that the Boraflex is performing within the EPRI acceptance criteria.

3.7 CRITICALITY ANALYSES OF FUEL HANDLING ACTIVITIES

This section documents the criticality analyses of fuel movement activities in the spent fuel pool, but outside of the storage racks. Although the maximum requested enrichment of the fuel for SONGS 2 and 3 is 4.8 w/o, the analyses show that 5.1 w/o fuel can be safely handled. The results clearly bound 4.8 w/o.

The worst case scenarios (from a criticality perspective) for fuel handling activities can be bounded by two cases:

(1) Single Isolated Assembly In Unborated Water
(2) Fuel Transfer Carrier, in which two assemblies can be transported at once.

3.7.1 Calculational Methodology

The final k-eff for the fuel handling equipment is calculated as:

$$\begin{aligned} k\text{-eff}_{Final} = {} & k\text{-eff}_{Reference} + \text{Methodology Bias} \\ & + \text{SQRT}\Big[(95/95 \text{ Methodology Bias Uncertainty})^2 \\ & + (95/95 \text{ KENO V.a Uncertainty in } k\text{-eff}_{Reference})^2 \Big] \end{aligned}$$

3.7.1.1 Reference Model

The reference KENO V.a model for all fuel handling activities (except the fuel transfer carrier) is a single unshimmed, un-irradiated 5.1 w/o fuel assembly in unborated water at 68 degrees F. The model for the fuel transfer carrier is two such assemblies next to each other.

The fuel handling equipment itself is not modeled. No credit is taken for any neutron absorption in the fuel handling components.

KENO V.a is executed with 503 neutron generations and 2000 neutrons per generation. KENO V.a results are used after skipping three generations.

3.7.1.2 Methodology Bias And Uncertainty

The bias and 95/95 uncertainty in the bias for CELLDAN, NITAWL-II, KENO V.a, and the 27 group cross-section library are 0.00928 and 0.00148, respectively. The bias and uncertainty were determined by analyses of 16 B&W critical experiments[4] for standard fuel storage (Table 3-1).

The bias (0.00928) is added directly to the reference k-eff. The 95/95 uncertainty in the bias (0.00148) is combined statistically with the uncertainty in KENO V.a results.

3.7.1.3 95/95 KENO V.a Uncertainty

KENO V.a results are reported as k-eff +/- sigma. The 95/95 KENO V.a uncertainty is $K_{95/95}$ * KENO V.a sigma. For 500 neutron generations, $K_{95/95}$ is 1.763 (Reference 5).

3.7.2 Results

K-eff is less than 0.95, including all uncertainties, for all fuel handling activities involving 5.1 w/o fuel assemblies.

3.7.2.1 Single Isolated Fuel Assembly In Unborated Water

K-eff is 0.92 for a single isolated, un-irradiated, unshimmed 5.1 w/o SONGS 2 and 3 fuel assembly in water at 68 degrees F and 0 PPM.

$$\begin{aligned} \text{k-eff}_{\text{Final}} &= 0.90793 + 0.00928 \\ &\quad + \text{SQRT}\left[\ (0.00148)^2 + (1.763*0.00080)^2\ \right] \\ &= 0.91925 \end{aligned}$$

3.7.2.2 Fuel Transfer Carrier

Two un-irradiated, unshimmed 5.1 w/o SONGS 2 and 3 fuel assemblies may be moved in the fuel transfer carrier together.

Assuming 0 PPM and taking no credit for the carriage basket structural material, k-eff is 0.950.

$$k\text{-}eff_{Final} = 0.93884 + 0.00928 + \mathrm{SQRT}\left[(0.00148)^2 + (1.763*0.00063)^2 \right] \sim 0.94997$$

3.7.2.3 Postulated Accidents

A dropped fuel assembly is bounded by a single isolated un-irradiated, unshimmed 5.1 w/o assembly in unborated water. At 68 degrees F and a soluble boron concentration of 0 PPM, k-eff is 0.92.

Interaction of the dropped assembly with the spent fuel pool storage racks is included in the analyses of the racks. In the worst case, a dropped un-irradiated 5.1 w/o assembly enters an empty storage location in the Region II racks. K-eff is less than 0.95 assuming 1800 PPM soluble boron.

3.8 REFERENCES

1. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees, from B. K. Grimes, April 14, 1978, "OT Positions for Review and Acceptance of Spent Fuel Storage and Handling Applications," as amended by the NRC letter dated January 18, 1979

2. CCC-545, RSIC Computer Code Collection, "SCALE-4, a Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation," Oak Ridge National Laboratory

3. STUDSVIK/NFA-89/3, User's Manual, "CASMO-3, a Fuel Assembly Burnup Program," Version 4, Studsvik AB, 1989

4. BAW-1484-7, "Criticality Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," The Babcock & Wilcox Company, July, 1979

5. SCR-607, "Factors For One-Sided Tolerance Limits And For Variables Sampling Plans," Sandia Corporation, March 1963

6. UFSAR/UFHA Change Request SAR23-301, "FHA Inside Fuel Handling Building, SFP Gate Drop Accident", April 08, 1994

Table 3-1

KENO V.A ANALYSES OF CRITICAL EXPERIMENTS
FOR THE DETERMINATION
OF CALCULATIONAL BIAS AND UNCERTAINTY

B&W Core	Measured k-eff	KENO V.a Calculated k-eff
I	1.0002	0.99026
II	1.0001	0.99182
III	1.0000	0.99167
IX	1.0030	0.99087
X	1.0001	0.98896
XI	1.0000	0.99286
XII	1.0000	0.99134
XIII	1.0000	0.99649
XIV	1.0001	0.99350
XV	0.9998	0.98702
XVI	1.0001	0.98806
XVII	1.0000	0.98991
XVIII	1.0002	0.98962
XIX	1.0002	0.99171
XX	1.0003	0.99087
XXI	0.9997	0.99029

The mean and standard deviation of the (Meas k-eff - KENO k-eff) differences are: Mean = 0.00928

Standard Deviation = 0.00234

$$\text{Bias} +/- \text{95/95 Uncertainty} = \text{Mean} \quad +/- \quad \frac{K_{95/95} * \text{Std Dev}}{\text{SQRT(Number of Cases)}}$$

$$= 0.00928 +/- \ (2.524)(0.00234)/\text{SQRT}(16)$$

$$= 0.00928 +/- 0.00148$$

Table 3-2

MINIMUM BURNUP VS. INITIAL ENRICHMENT
FOR UNRESTRICTED PLACEMENT OF
SONGS 2 AND 3 FUEL IN REGION II RACKS

Enrichment	Burnup (GWD/T)
1.85	0.0
2.50	9.8
3.00	15.9
4.00	27.2
4.80	35.5 *
5.10	38.6

* Linearly Interpolated Value

Table 3-3

MINIMUM BURNUP VS. INITIAL ENRICHMENT
FOR PLACEMENT OF SONGS 2 AND 3 FUEL
IN REGION II PERIPHERAL POOL LOCATIONS

Enrichment	Burnup (GWD/T)
2.30	0.0
2.50	2.4
3.00	8.1
4.00	18.0
4.80	25.5 *
5.10	28.3

* Linearly Interpolated Value

Table 3-4

NEW FUEL STORAGE RACKS K-EFF VS. WATER DENSITY

Water Density (gms/cc)	KENO k-eff +/- sigma	KENO k-eff + 0.00928
0.02	0.70809 +/- 0.00132	0.717
0.03	0.80163 +/- 0.00145	0.811
0.04	0.84127 +/- 0.00135	0.851
0.045	0.84683 +/- 0.00130	0.856
0.05	0.84425 +/- 0.00133	0.854
0.06	0.82817 +/- 0.00135	0.837
0.07	0.80047 +/- 0.00135	0.810
0.08	0.76250 +/- 0.00135	0.772
0.09	0.72664 +/- 0.00135	0.736
0.10	0.69185 +/- 0.00134	0.701
0.125	0.61773 +/- 0.00127	0.627
0.15	0.56222 +/- 0.00126	0.572
0.20	0.50818 +/- 0.00128	0.517
0.30	0.51720 +/- 0.00131	0.526
0.50	0.63967 +/- 0.00141	0.649
0.70	0.75969 +/- 0.00144	0.769
0.80	0.80696 +/- 0.00153	0.816
0.90	0.85170 +/- 0.00151	0.861
1.00	0.89516 +/- 0.00151	0.904

Table 3-5

REACTIVITY EFFECT DUE TO BORAFLEX THINNING

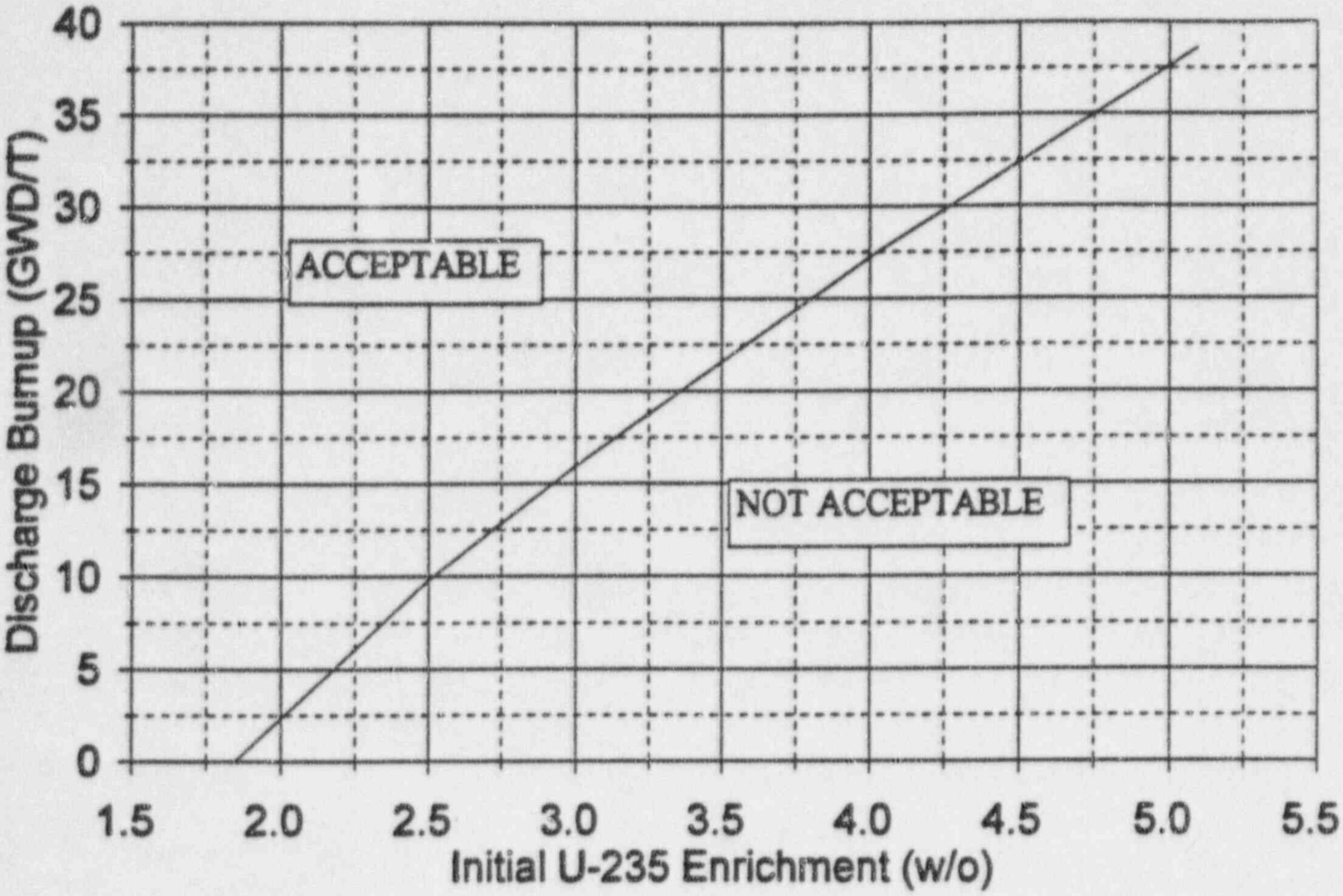
Percent Loss of Boraflex Thickness[†]	Region I k-eff Increase	Region II k-eff Increase
0%	0.000	0.000
1%	0.000	0.000
2%	0.001	0.000
5%	0.004	0.001
10%	0.004	0.004
20%	0.009	0.007
30%	0.012	0.013
40%	0.019	0.020
50%	0.027	0.030

[†]Based on nominal Boraflex thickness.

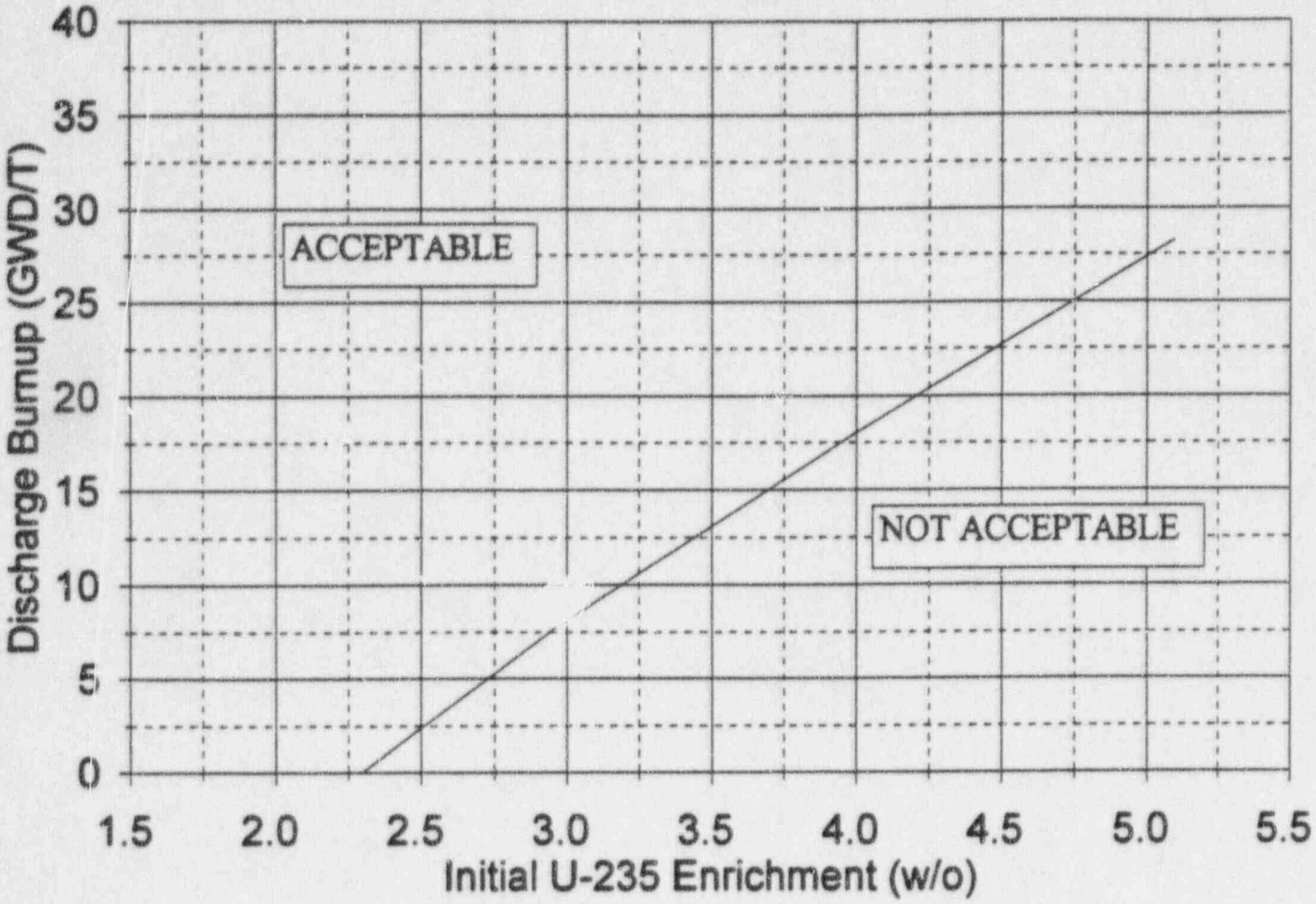
Table 3-6

REGION II K-EFF VS. NUMBER OF MISLOADED
5.1 W/O ASSEMBLIES -- 1800 PPM

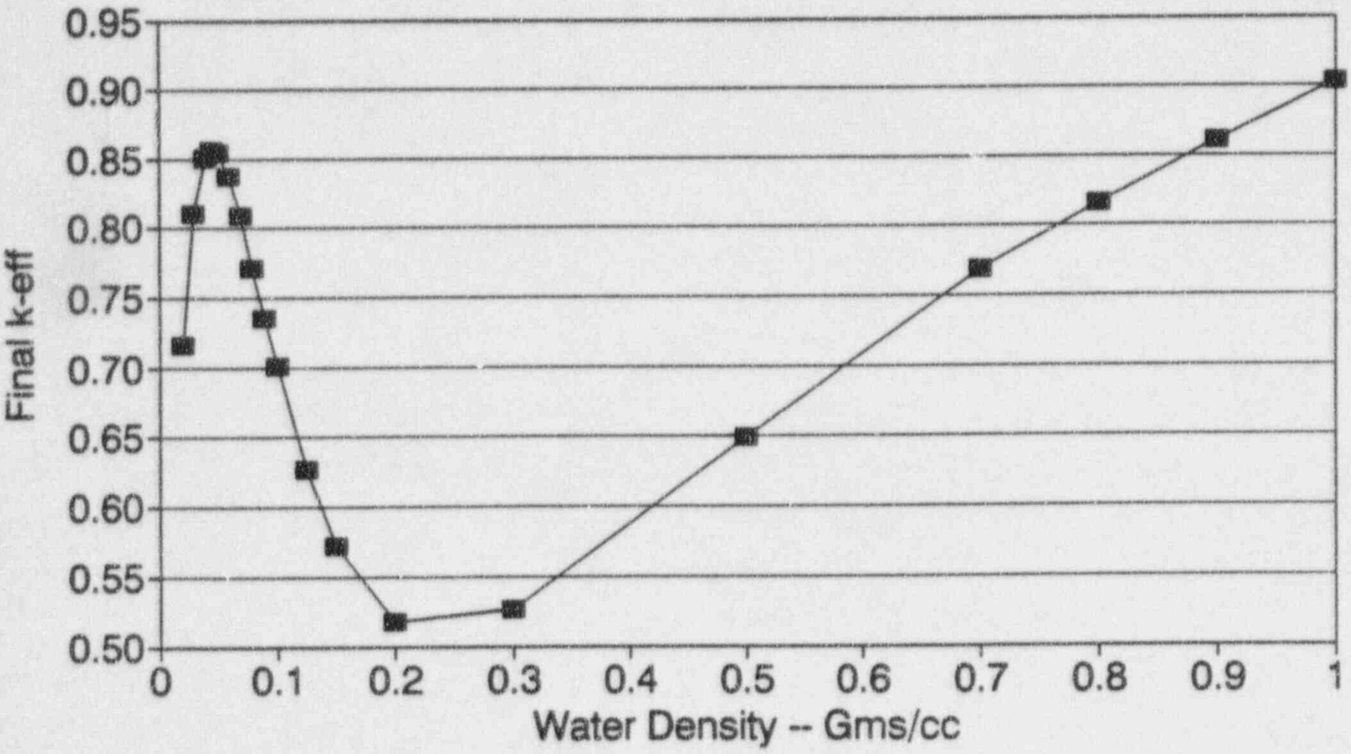
Number Of Misloaded Assemblies (5.1 w/o)	Region II K-eff
1	0.767
2	0.819
4	0.881
6	0.906
9	0.932
16	0.961



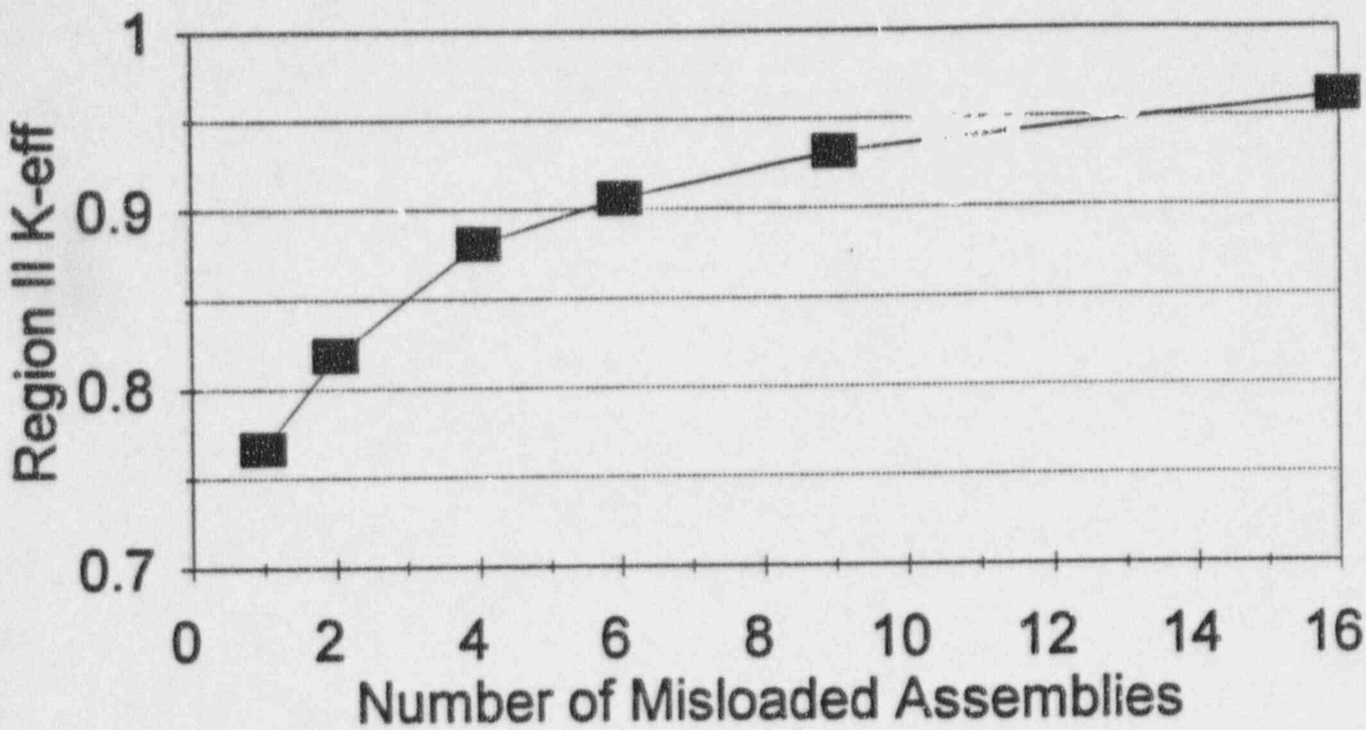
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
MINIMUM BURNUP VS. INITIAL ENRICHMENT FOR UNRESTRICTED PLACEMENT OF SONGS 2 AND 3 FUEL IN REGION II RACKS
FIGURE 3-1



SAN ONOFRE NUCLEAR GENERATING STATIO[illegible] Units 2 & 3
MINIMUM BURNUP VS. INITIAL ENRICHMENT FOR PLACEMENT OF SONGS 2 AND 3 FUEL IN REGION II PERIPHERAL POOL LOCATIONS
FIGURE 3-2



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
NEW FUEL STORAGE RACKS K-EFF VS. WATER DENSITY
FIGURE 3-3



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

REGION II K-EFF VS. NUMBER
OF MISLOADED 5.1 w/o ASSEMBLIES
-- 1800 PPM

FIGURE 3-4

I. DECAY HEAT EVALUATION

4.1 INTRODUCTION

This section presents the spent fuel pool decay heat analyses performed to support increasing the SONGS 2 and 3 maximum enrichment from 4.1 w/o to 4.8 w/o.

The results show that there is no impact on spent fuel pool decay heat loads from increasing the enrichment from 4.1 w/o to 4.8 w/o. The current UFSAR decay heat loads are conservative. The UFSAR analysis performed to calculate the maximum fuel cladding temperature and spent fuel pool cooling include assumptions which bound the use of more highly enriched fuel assemblies.

The decay heat load does not depend on enrichment. However, increasing the enrichment permits longer cycle lengths. Longer cycle lengths mean higher discharge assembly burnups, and higher discharge burnups increase the decay heat. However, the UFSAR decay heat loads remain conservative because the fuel batch size will be decreased from 108 assemblies to 104 or less assemblies.

All decay heat loads are calculated per NUREG-0800, Branch Technical Position ASB 9-2, 'Residual Decay Energy For Light-Water Reactors For long-Term Cooling'.[1] The decay heat from this methodology is a function of power level, irradiation time, and cooling time. The decay heat load does not depend on enrichment.

4.2 CURRENT LICENSING BASES

The current UFSAR decay heat loads are:

(1) Normal Maximum Heat Load* -- 24.7E+06 Btu/Hr
(2) Abnormal Maximum Heat Load* -- 51.3E+06 Btu/Hr

These heat loads are based on a cycle length of 570 EFPD, discharging 108 fuel assemblies, and the presence of transhipped SONGS 1 fuel assemblies.

4.3 HEAT LOADS FOR 4.8 W/O ENRICHMENT INCREASE

Increasing the enrichment permits longer cycle lengths and, therefore, increases the decay heat load. For these analyses a conservative cycle length of 635 EFPD is assumed. In addition to increasing the enrichment, the proposed fuel management plans will decrease the fuel batch size to 104 assemblies or less.

Therefore, the decay heat loads are:

(1) Normal Maximum Heat Load* -- 24.0E+06 Btu/Hr (Table 4-1)
(2) Abnormal Maximum Heat Load* -- 49.9E+06 Btu/Hr (Table 4-2)

for the following assumed conditions:

(1) 635 EFPD cycle length
(2) 104 assembly batch size
(3) 95% capacity factor
(4) 60 day refueling outages
(5) No SONGS 1 fuel assemblies
(Since SONGS 1 fuel assemblies have lower decay heat, it is conservative to omit the transhipped SONGS 1 fuel assemblies and completely fill the storage racks with SONGS 2 and 3 fuel assemblies.)
(6) The total number of assemblies shown in Tables 4-1 and 4-2 is greater than the actual number of available spent fuel storage locations. This is conservative for evaluating decay heat loads, and is not indicative of actual fuel discharge plans.

These heat loads are less than the current analyses of record in the UFSAR. The heat loads are smaller because the reduced discharge batch size more than offsets increasing the irradiation time from 570 EFPD to 635 EFPD.

* Per NUREG-0800/Standard Review Plan 9.1.3, Rev 1, July 1981, Section III.1.h, the heat loads are defined as:
(1) The normal maximum heat load is one refueling load at equilibrium conditions after 150 hours decay and one refueling load at equilibrium conditions after one year decay.
(2) The abnormal maximum heat load is one full core at equilibrium conditions after 150 hours decay and one refueling load at equilibrium conditions after 36 days decay.

4.4 REFERENCES

1. NUREG-0800, Standard Review Plan, Branch Technical Position ASB 9-2, "Residual Decay Energy For Light-Water Reactors For long-Term Cooling"

Table 4-1

SPENT FUEL POOL DECAY HEAT -- NORMAL MAXIMUM HEAT LOAD

Units 2/3 assemblies : Irradiated for 2 cycles of 635 EFPD (Except as noted)

Cycle	Unit 2/3 Assemblies Assemblies Discharged	 Total Assemblies	Cooling Time (Yrs)	Single Assembly Btu/Hr	Batch Btu/Hr	
1 – 5	484	484	15	3.817E+03	1.85E+06	(1)
6	108	592	13	4.004E+03	4.32E+05	
7	108	700	11	4.201E+03	4.54E+05	
8	108	808	9	4.412E+03	4.76E+05	
9	104	912	7	4.658E+03	4.84E+05	
10	104	1016	5	5.079E+03	5.28E+05	
11	104	1120	3	6.577E+03	6.84E+05	
12	104	1224	1	1.686E+04	1.75E+06	
13	104	1328	150 hrs	1.664E+05	1.73E+07	(2)
				TOTAL	2.40E+07	

TOTAL = 2.40E+07 Btu/Hr for 1,328 total assemblies (3)

(1) – 64(Cycle 1) + 89(Cycle 2) + 113(Cycle3) + 109(Cycle 4) + 109(Cycle 5)

(2) – For a 2 region core and cycle length of 635 EFPD,
the refuleing load which has cooled for 1 year was
irradiated for 2 cycles = 2 * 635 = 1,270 EFPD.
This leaves a region in the reactor irradiated for 635 EFPD.
One year later the burnup of this region is 635 + 365 = 1,000 EFPD.
This is the region which is then discharged with 150 hrs cooling.

(3) – Available storage locations = 1,542
- 217 (Full Core off-load)

1,325

(4) – Unit 1 assemblies are omitted to maximize the heat load.

(5) – Some assemblies will be irradiated for more than 2 cycles.
However, due to the low power in these assemblies, the decay heat
is bounded by two cycles at average power.

Table 4-2

SPENT FUEL POOL DECAY HEAT -- ABNORMAL MAXIMUM HEAT LOAD

Units 2/3 assemblies : Irradiated for 2 cycles of 635 EFPD (Except as noted)

Cycle	Unit 2/3 Assemblies Assemblies Discharged	Total Assemblies	Cooling Time (Yrs)	Single Assembly Btu/Hr	Batch Btu/Hr
1 – 5	484	484	16	3.727E+03	1.80E+06 (1)
6	108	592	14	3.910E+03	4.22E+05
7	108	700	12	4.101E+03	4.43E+05
8	108	808	10	4.304E+03	4.65E+05
9	104	912	8	4.527E+03	4.71E+05
10	104	1016	6	4.825E+03	5.02E+05
11	104	1120	4	5.551E+03	5.77E+05
12	104	1224	2	9.066E+03	9.43E+05
13	104	1328	36 days	8.012E+04	8.33E+06 (2)
14	113	1441	150 hrs	1.678E+05	1.90E+07 (2 * 635 Days) (2)
14	104	1545	150 hrs	1.628E+05	1.69E+07 (635 Days)
				TOTAL	4.99E+07

TOTAL = 4.99E+07 Btu/Hr for 1,545 total assemblies (3)

(1) – 64(Cycle 1) + 89(Cycle 2) + 113(Cycle3) + 109(Cycle 4) + 109(Cycle 5)

(2) – For a 2 region core and cycle length of 635 EFPD:
104 assemblies – 2 * 635 EFPD – 36 days cooling
113 assemblies – 2 * 635 EFPD – 150 hrs cooling
104 assemblies – 635 EFPD – 150 hrs cooling

(3) – Available storage locations = 1,542

(4) – Unit 1 assemblies are omitted to maximize the heat load.

(5) – Some assemblies will be irradiated for more than 2 cycles. However, due to the low power in these assemblies, the decay heat is bounded by two cycles at average power.

5. RADIOLOGICAL EVALUATION

5.1 INTRODUCTION

This section presents the radiological analyses performed to support increasing the SONGS 2 and 3 maximum enrichment from 4.1 w/o to 4.8 w/o.

Increasing the SONGS 2 and 3 enrichment from 4.1 w/o to 4.8 w/o has insignificant impact on radwaste generation, gaseous effluent releases, spent fuel pool radiation shielding, personnel exposure during fuel handling operations, and the radiological consequences of fuel handling accidents including spent fuel pool boiling.

A number of SONGS 2 and 3 radiological analyses addressing fuel handling operations were revised when high density spent fuel racks were installed in the spent fuel pools.(1) To address the potential future use of higher enrichment and longer burnups, the revised analyses assumed a burnup of 60 GWD/T. A majority of the fuel handling operation radiological evaluations that are presented in UFSAR Section 15.7.3 reflect the results of these revised analyses. Based in part on the fact that many of the safety analyses of record are based on a burnup of 60 GWD/T, increasing the enrichment from 4.1 w/o to 4.8 w/o has no significant impact on the radiological consequences due to fuel handling.

The NRC has also reviewed the anticipated widespread use of extended burnup fuel in commercial LWR's. The NRC has concluded that there are no significant adverse radiological or non-radiological impacts associated with the use of extended fuel burnup and/or increased enrichment.(2) Moreover, the NRC has issued NUREG/CR-5009, "Assessment of the Use of Extended Burn-up Fuel in Light Water Reactors",(3) which concludes with a finding of no significant impact for fuel enrichment up to 5.0 w/o and burnup to 60 GWD/T.

5.2 RADWASTE GENERATION

SONGS 2 and 3 have separate and independent spent fuel pool cooling/ purification systems. Each fuel pool purification system, which is designed to remove soluble and insoluble foreign matter from the spent fuel pool water, is rated at 150 gal/min nominal flowrate. The purification flow path originates at the spent fuel pool, passes through a purification pump suction strainer (coarse) to protect the purification pump from solid material, a purification pump which discharges to a backflushable filter, followed by an ion exchanger, which discharges into the spent fuel pool cooling piping before it re-enters the spent fuel pool. The backflushable filter is designed to backflush solid material to the crud tank, after which the backflushed material is filtered and processed as radwaste. The backflushable filter can also be bypassed if the pressure drop across the filter cannot be reduced by backflushing due to clogging by Boraflex products. The ion exchanger resins are changed as necessary when the decontamination factor is low, and the spent resins are processed as solid waste.

The activity loading on the fuel pool filter and the fuel pool ion exchanger resin beads is addressed in UFSAR Section 11.4. An analysis has been performed to assess the isotopic inventory of the spent fuel pool purification system, assuming that the pool stores fuel with a burnup of either 33 GWD/T or 60 GWD/T. The analysis determined that the significant isotopes present in the radwaste are the crud isotopes of elements such as cesium, cobalt, iron, chromium and manganese. Many of these isotopes will be produced in greater quantity if the enrichment and burnup are increased. An increase in the burnup from 33 GWD/T to 60 GWD/T will result in the spent fuel pool activity concentration of a number of the collected isotopes increasing by no more than a factor of two, and typically much less.

The increase in the spent fuel pool activity concentrations will result in an increase in the quantity of solid and liquid radwaste produced, and a need for more frequent regeneration and/or changeout of the spent fuel pool cooling and purification system ion exchanger resin. The amount of additional liquid and solid radwaste that is produced by the cleanup of the spent fuel pool is a fraction of the total radwaste processed at the plant. Therefore, the overall impact of this increase in radwaste is not significant.

Of note is that the proposed enrichment is 4.8 w/o, and the proposed maximum burnup will remain below 60 GWD/T. These values are within the parameter limits evaluated in NUREG/CR-5009 which indicates a finding of no significant radiological impact.

5.3 GASEOUS EFFLUENT RELEASES

The dose rate to offsite individuals caused by gaseous effluent releases from the fuel handling building is the subject of UFSAR Sections 11.3.3 and 12.4.2. Per these sections, tritium is the only significant fuel handling building airborne effluent. During normal operations up to 320 curies/year of tritium can be released from the fuel handling building. During refueling operations 585 curies/year of tritium can be released from the fuel handling building. Based on these release quantities, an individual at the site boundary would receive an annual tritium inhalation dose of approximately 0.22 millirem.

An analysis has been performed to assess the dose to an individual at the site boundary who is exposed to airborne releases from the fuel handling building. The analysis determined that an increase in burnup from 33 GWD/T to 60 GWD/T would result in the fuel handling building airborne tritium activity concentration increasing by a factor of 1.7. Application of this change factor implies that an individual at the site boundary would receive an annual tritium inhalation dose of approximately 0.37 millirem. In addition, it is noted that the proposed increase in the fuel cycle length to approximately 24 months will reduce the exposure frequency associated with refueling operations. This reduction implies an average annual exposure of approximately 0.25 millirem for the proposed fuel cycle.

This 0.03 millirem dose increase does not alter the conclusion that the gaseous effluent releases would result in acceptable offsite radiological dose consequences if the proposed change to an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T is implemented.

5.4 FUEL HANDLING BUILDING SHIELDING EVALUATION

The predominant radioactivity sources in the spent fuel storage and transfer areas in the fuel handling building are the spent fuel assemblies stored in the spent fuel pool. Radiation Zone Maps presented in UFSAR Section 12.3

document maximum expected radiation levels in the fuel handling building assuming a spent fuel activity profile consistent with a burnup of 33 GWD/T and 3 days decay.

An analysis has been performed to assess the dose rates above and to the sides of the spent fuel pool, assuming that the pool stores fuel with a burnup of either 33 GWD/T or 60 GWD/T. The analysis determined that the highest dose rates are those associated with the storage of recently irradiated fuel assemblies (decayed for 3 days), and that these dose rates are virtually equivalent for the two burnups. As decay time increases the dose rates decrease, albeit the dose rates for the 60 GWD/T burnup fuel trend higher than the dose rates for the 33 GWD/T burnup fuel.

Since the fuel handling building shielding was designed assuming a spent fuel activity profile consistent with a burnup of 33 GWD/T and 3 days decay, and since the dose rates for this condition are equivalent to those of fuel with a burnup of 60 GWD/T and 3 days decay, it is concluded that an increase in fuel enrichment and burnup will have no significant impact on the fuel handling building radiation shielding evaluation.

5.5 PERSONNEL EXPOSURE DURING FUEL HANDLING OPERATIONS

Personnel exposure during fuel handling operations results from exposure to the contaminated fuel handling building air, exposure to contaminated spent fuel pool water, and exposure to direct gamma radiation shine from the stored spent fuel.

Tritium is the only significant fuel handling building airborne contaminant. An analysis has been performed to assess the dose to plant personnel exposed to airborne contamination in the fuel handling building. The analysis determined that an increase in burnup from 33 GWD/T to 60 GWD/T would result in the fuel handling building airborne tritium activity concentration increasing by a factor of 1.7. Application of this change factor implies that the annual tritium inhalation dose to an individual in the fuel handling building associated with normal and refueling operations would increase from approximately 1.07 to 1.82 millirem. In addition, it is noted that the proposed increase in the fuel cycle length to approximately 24 months will reduce the exposure frequency associated with refueling operations. This reduction implies an average annual exposure of approximately 1.64 millirem for the proposed fuel cycle. This dose increase does not alter the conclusion that the airborne contamination in the fuel handling building would result in acceptable radiological dose consequences if the proposed change to an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T is implemented.

The spent fuel pool water activity profile is dependent on the age and quantity of spent fuel stored in the pool. The maximum spent fuel pool water activity is present on the fourth day of the refueling period, when the Xenon-133 isotope has reached its peak activity. An analysis has been performed to assess the spent fuel pool water activity profile associated with an increase in burnup from 33 GWD/T to 60 GWD/T. The analysis determined that an increase in the burnup has a negligible effect on the maximum spent fuel pool activity. A second analysis was performed to assess the dose to plant personnel exposed to the spent fuel pool water. The analysis determined that an increase in burnup from 33 GWD/T to 60 GWD/T would result in a negligible change in the exposure dose.

With respect to personnel exposure to direct gamma radiation shine from the stored spent fuel, safety systems associated with fuel handling operations are not being changed. In addition, procedural controls associated with the handling or storage of fuel are not being changed. These safety systems and procedural controls ensure that sufficient water level will be maintained in the pool to provide adequate radiation shielding. As previously noted, an analysis has been performed to assess the dose rates above the spent fuel pool assuming that the pool stores fuel with burnups of either 33 GWD/T or 60 GWD/T. The analysis determined that the highest dose rates associated with the storage of recently irradiated fuel assemblies (decayed for a minimum of 3 days) are virtually equivalent for the two burnups. As decay time increases the dose rates decrease, albeit the dose rates for the 60 GWD/T burnup fuel trend higher than the dose rates for the 33 GWD/T burnup fuel.

The relationship between the 33 GWD/T and 60 GWD/T dose rates is indicative of the impact of the enrichment and burnup increase on personnel exposure during fuel handling operations. Since the maximum personnel exposure dose rates are equivalent for the two burnup cases, it is concluded that an increase in fuel enrichment and burnup will have no significant impact on individual or cumulative occupational exposure during fuel handling operations. In addition, it is noted that the proposed increase in the fuel cycle length to approximately 24 months will reduce the exposure frequency associated with refueling operations. This reduction will tend to actually decrease the individual and cumulative occupational exposure during fuel handling operations over the remaining life of the power plant.

5.6 DESIGN BASIS FUEL HANDLING ACCIDENTS

UFSAR Section 15.7.3 documents design basis analyses for the radiological consequences of a fuel handling accident in the containment building and several fuel handling accidents in the fuel handling building. The fuel handling accident inside the containment building addresses the dropping of a fuel assembly. The fuel handling accidents inside the fuel handling building address the dropping of a fuel assembly, the dropping of a spent fuel cask or other major load, the dropping of a spent fuel pool gate, and the dropping of a test equipment skid. For these accidents, no credit is taken for airborne activity removal by the fuel handling building charcoal or HEPA filtration. The control room emergency air cleanup system is credited to maintain control room doses below 10 CFR 50 Appendix A GDC 19 limits.

The dropped spent fuel assembly accident inside the containment building results in acceptable offsite and control room radiological dose consequences. The current design basis analysis assumes failure of all 236 fuel rods in the damaged fuel assembly. The failed fuel in the current design basis analysis is characterized by a burnup of 60 GWD/T and a 1.65 peaking factor. These characteristics conservatively envelop the case of an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T. As such, the results of this current analysis are applicable to the proposed change to an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T.

The dropped spent fuel assembly accident in the fuel handling building results in acceptable offsite and control room radiological dose consequences. The current design basis analysis assumes failure of 60 fuel rods in the damaged fuel assembly. The failed fuel in the current design basis analysis is characterized by a burnup of 60 GWD/T and a 1.65 peaking factor. These characteristics conservatively envelop the case of an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T. As such, the results of this current analysis are not altered by the proposed change to an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T.

The spent fuel cask drop accident in the fuel handling building does not result in any offsite or control room radiological dose consequences. The fuel handling building layout and design prevents the spent fuel cask from traveling over spent fuel stored in the spent fuel pool, or from damaging any spent fuel if dropped near the pool. Thus, no credible accident source exists from spent fuel external to the shipping cask. For spent fuel stored inside the shipping cask, both plant design and administrative controls limit the maximum drop impact energy to within the design capacity of the spent fuel cask. This prevents, by cask design, any release of radioactive material from the spent fuel inside the cask. For these reasons, there are no radiological consequences associated with a spent fuel cask drop accident. These conclusions as they relate to a spent fuel cask drop accident are not altered by the proposed change to an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T.

Previous evaluations of the fuel pool gate drop event determined that up to six SONGS 2 and 3 fuel assemblies could have been impacted and damaged, as discussed in UFSAR Section 15.7.3.6. Recently, the structural and radiological consequences of the gate drop event have been reevaluated to revise conservatisms which were used in the previous analyses. The new analyses concluded that only one fuel assembly would be impacted and potentially damaged.

The spent fuel pool gate drop accident in the fuel handling building results in acceptable offsite and control room radiological dose consequences. The current design basis analysis assumes failure of all 236 fuel rods in one fuel assembly impacted by the falling spent fuel pool gate. The failed fuel in the current design basis analysis is characterized by a burnup of 60 GWD/T and a 1.65 peaking factor. These characteristics conservatively envelop the case of an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T. As such, the results of this current analysis are not altered by the proposed change to an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T.

The test equipment skid drop accident in the fuel handling building does not result in any offsite or control room radiological dose consequences. The test equipment is procedurally prohibited from traveling over fuel assemblies unless certain height restrictions are met. These height restrictions ensure that in the event of a test equipment skid drop, no fuel assemblies would be damaged. If damage were to be postulated, the radiological consequences would be bounded by the radiological consequences for a spent fuel pool gate drop. And, as previously discussed, the acceptable results of the gate drop analysis are not altered by the proposed change to an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T.

5.7 SPENT FUEL POOL BOILING ACCIDENT

UFSAR Section 15.7.3.8 documents a design basis analysis for the radiological consequences of a spent fuel pool boiling accident due to the loss of spent fuel pool cooling. The spent fuel pool boiling accident results in acceptable offsite radiological dose consequences.

The current design basis analysis calculates offsite whole body immersion, thyroid inhalation, and tritium inhalation doses assuming a burnup of 33 GWD/T without taking credit for airborne activity removal by the fuel handling building charcoal or HEPA filtration. The results for this 33 GWD/T burnup are presented in the UFSAR. The current design basis analysis also calculates offsite doses for a burnup of 60 GWD/T. Per the analysis, an increase in the burnup to 60 GWD/T would not alter the reported results of the whole body immersion and thyroid inhalation doses, but would increase the tritium inhalation dose by a factor of 1.7. This increase, when applied to the less than 1 millirem dose calculated in the current design basis analysis, would not alter the conclusion that the spent fuel pool boiling accident would result in acceptable offsite radiological dose consequences if the proposed change to an enrichment of 4.8 w/o and a burnup of less than 60 GWD/T is implemented.

5.8 REFERENCES

1. Spent Fuel Pool Reracking Licensing Report, Revision 6, Southern California Edison San Onofre Nuclear Generating Station Units 2 and 3, February 16, 1990.

2. Federal Register 53 FR 6040, February 29, 1988

3. NUREG/CR-5009, "Assessment of the Use of Extended Burn-up Fuel in Light Water Reactors"