Pacific Gas and
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PG&E Letter DCL-01-096

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 01-02,
Credit For Soluble Boron In The Spent Fuel Pool Criticality Analysis

Dear Commissioners and Staff:

Enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 pursuant to 10 CFR 50.90. This proposed License Amendment Request (LAR) would allow the use of credit for soluble boron in the spent fuel pool (SFP) criticality analysis. This criticality analysis was performed using methodology analogous to that developed by the Westinghouse Owners Group (WOG) and described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology." However, due to subsequent analyses which indicate nonconservatisms in the axial burnup biases reported in the WOG methodology, a criticality analysis was performed for the Diablo Canyon Units 1 and 2 SFPs that addresses the axial burnup biases and includes technically supported margins. The analysis is submitted to update the licensing bases which support the proposed Technical Specification (TS) changes.

Enclosure 1 provides a description of the proposed changes, the reasons for requesting the changes, the supporting safety evaluations, and significant hazards determinations. Enclosures 2 and 3 provide the marked-up and revised TS pages, respectively. Enclosure 4 provides the revised TS Bases pages for information. The Bases changes will be implemented under PG&E's TS 5.5.14 TS Bases Control Program after NRC approval of this LAR. Enclosure 5 provides a copy of the Diablo Canyon SFP criticality analysis (A-DP1-FE-0001, Revision 0) with credit for soluble boron. Enclosure 6 provides a copy of the Diablo Canyon SFP boron dilution analysis. Please note that the SFP criticality analysis assumed an IFBA Boron-10 density of 1.5 milligrams (mg) per inch but was incorrectly reported throughout Enclosure 5 as 1.5 grams (g) per inch. Westinghouse has confirmed that this is only a typographical error and does not affect the results of the criticality analysis.

A001

In response to NRC Generic Letter 96-04, "Boraflex Degradation In Spent Fuel Pool Storage Racks," issued on June 26, 1996, PG&E is conservatively storing fuel assemblies in a checkerboard configuration within the region containing Boraflex panels. This LAR would eliminate the need to credit Boraflex for maintaining a subcritical condition within that region, eliminating the conservative measure of storing the fuel assemblies in a checkerboard configuration. Therefore, to maintain adequate storage capacity in the SFP to accommodate a full core offload, PG&E requests that the review and approval of the LAR be completed by January 1, 2003, and that the LAR be made effective upon issuance, to be implemented within 30 days from the date of issuance.

Sincerely,

David H. Oatley
Vice President Diablo Canyon Operations

cc: Edgar Bailey, DHS
Ellis W. Merschoff
David L. Proulx
Girija S. Shukla
Diablo Distribution

Enclosures
MKF

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	)	Docket No. 50-275 Facility Operating License No. DPR-80
Diablo Canyon Power Plant Units 1 and 2	)	Docket No. 50-323 Facility Operating License No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath says that he is Vice President Diablo Canyon Operations of Pacific Gas and Electric Company; that he has executed LAR 01-02 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

D H Oatley

David H. Oatley
Vice President Diablo Canyon Operations

Subscribed and sworn to before me this 13th day of September, 2001

Sandra Eatherly

Notary Public
County of San Luis Obispo
State of California

SANDRA EATHERLY
Commission # 1225422
Notary Public - California
San Luis Obispo County
My Comm. Expires Jul 18, 2003

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

This proposed License Amendment Request (LAR) is a request pursuant to 10 CFR 50.90 to revise Technical Specification (TS) 3.7.16, "Spent Fuel Pool Boron Concentration," TS 3.7.17, "Spent Fuel Assembly Storage – Region 1/Region 2," and TS 4.3, "Fuel Storage" for Diablo Canyon Power Plant (DCPP) Units 1 & 2.

2.0 DESCRIPTION

This LAR proposes revisions to the DCPP TS associated with controlling the storage of assemblies with differing initial enrichments, initial Boron-10 content, fuel pellet diameter and burnup. The proposed TS changes also include a revised Surveillance Requirement (SR) to ensure proper boron concentration in the spent fuel pool (SFP).

The proposed changes to the DCPP TS are described below, and the specific wording changes are shown in Enclosure 2:

1. Table of Contents: The Table of Contents would be revised to remove TS 3.7.17, "Spent Fuel Assembly Storage – Region 1," and rename TS 3.7.17, "Spent Fuel Assembly Storage – Region 2" to "Spent Fuel Assembly Storage."

2. TS 3.7.16: The frequency of SR 3.7.16.1 would be changed from "31 days" to "7 days" to provide adequate assurance that boron concentration reduction is not taking place.

3. TS 3.7.17: LCO 3.7.17.1, "Spent Fuel Assembly Storage – Region 1," would be deleted and LCO 3.7.17.2, "Spent Fuel Assembly Storage – Region 2," would be renamed to LCO 3.7.17, "Spent Fuel Assembly Storage." This TS would be revised to provide storage guidelines for the entire SFP. The storage configurations consider the initial enrichment, initial Boron-10 content, fuel pellet diameter, and burnup for the stored fuel assemblies. This specification provides the requirements for the two new configurations (2-by-2 array and checkerboard) and the one existing configuration (all cell) in the SFP.

4. Figure 3.7.17-1: This figure would be deleted and replaced by a new figure to provide graphical representation of allowable storage configurations (all cell, 2-by-2 array, and checkerboard) in the SFP.

5. Figure 3.7.17-2: To clarify that this figure is a function of fuel pellet diameter and that it provides the acceptable/unacceptable enrichment-burnup requirements for an all cell configuration, the title would be revised to, "Figure 3.7.17-2 Minimum Required Assembly Discharge Burnup as a Function of Initial Enrichment and Fuel Pellet Diameter for an All Cell Storage Configuration."

6. Figure 3.7.17-3: This new figure would be added to provide the acceptable/unacceptable enrichment-burnup requirements for the burned fuel assemblies within a 2-by-2 array configuration.

7. TS 4.3.1.1(b): This section would be deleted and replaced with a new requirement for k_{eff} less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the Final Safety Analysis Report (FSAR).

8. TS 4.3.1.1(c): This section would add a new requirement for k_{eff} less than or equal to 0.95 if fully flooded with water borated to 806 ppm, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the FSAR.

9. TS 4.3.1.1(d): This section would be relabeled from "c" to "d".

10. TS 4.3.1.1 (e): This new section would allow the storage of fuel assemblies with a discharge burnup in the "acceptable" region of Figure 3.7.17-2 for the all cell configuration as shown in Figure 3.7.17-1.

11. TS 4.3.1.1 (f): This new section would allow the storage of fuel assemblies with a discharge burnup in the "acceptable" region of Figure 3.7.17-3 for the 2-by-2 array configuration as shown in Figure 3.7.17-1.

3.0 BACKGROUND

The DCPP Units 1 and 2 SFPs each consist of 16 stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells (nominal 11-inch center-to-center spacing). Three modules, currently designated as Region 1, contain 290 fuel assembly storage locations qualified for storage of new or irradiated fuel. Each Region 1 storage location is surrounded on all four sides by Boraflex panels. These Boraflex panels are neutron absorbers and decrease the k_{eff} of the stored fuel. The remaining 13 modules, currently designated as Region 2, consist of 1,034 fuel assembly storage locations with no Boraflex, qualified for storage of assemblies based on the initial enrichment and burnup of the fuel.

In response to NRC Generic Letter (GL) 96-04, "Boraflex Degradation In Spent Fuel Pool Storage Racks," issued on June 26, 1996, DCPP has monitored the Boraflex in Units 1 and 2 SFPs and concluded that the degradation has remained lower than other plants due to plant specific cooling flow patterns and rates. However, because the exact amount of degradation is difficult to determine, PG&E has conservatively implemented storage of fuel assemblies in a checkerboard configuration within Region 1 to ensure that the 5 percent subcriticality margin in unborated water will continue to be maintained. This action results in the loss of approximately 145 fuel assembly storage cells that would otherwise be available. In order to eliminate this measure and regain full storage capacity of the spent fuel racks, PG&E proposes to incorporate new limitations within the DCPP TS which utilize credit for soluble boron for the control of reactivity in the SFP while retaining the necessary margin of safety. The proposed changes will also eliminate the need to take credit for the spent fuel rack Boraflex neutron absorber panels in Region 1 of the SFP criticality analysis.

The precedent of using soluble boron in water to provide criticality control, aside from normal reactor operations, has already been established. Credit for soluble boron in the SFP has been previously permitted when considering abnormal or accident conditions. Also, during refueling, soluble boron in the reactor vessel is the only direct control utilized to ensure that the reactor remains subcritical. This proposed LAR would allow the use of credit for soluble boron in the SFP criticality analysis. This criticality analysis was performed using methodology analogous to that developed by the Westinghouse Owners Group (WOG) and described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology." However, due to subsequent analyses which indicate nonconservatisms in the axial burnup biases reported in the WOG methodology, a criticality analysis (Enclosure 5) was performed for the DCPP Units 1 and 2 SFPs that addresses the axial burnup biases and includes technically supported margins.

4.0 TECHNICAL ANALYSIS

General Design Criterion (GDC) 62 states that "criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by the use of geometrically safe configurations." The NRC established a 5 percent subcriticality margin (k_{eff} no greater than 0.95) to comply with GDC 62 in Section 9.1.2 of NRC Standard Review Plan NUREG-0800. For the current criticality analysis, all of the applicable biases and uncertainties were combined with k_{eff} to provide a one-sided, upper tolerance limit on k_{eff} such that the true value was less

than the calculated value with a 95 percent probability at a 95 percent confidence level (95/95). This LAR permits the use of SFP soluble boron to offset these uncertainties to maintain k_{eff} less than or equal to 0.95. Additionally, the criticality analysis shows that the spent fuel rack k_{eff} would remain less than 1.0 (subcritical) on a 95/95 basis if flooded with unborated water.

The new criticality analysis (Enclosure 5) performed for the DCPP SFPs shows that the acceptance criteria for criticality is met for the storage of 17-by-17 fuel assemblies under both normal and accident conditions with soluble boron credit, no credit for the spent fuel rack Boraflex neutron absorber panels, and the storage configurations and enrichment limits described below. In addition to reactivity credit for soluble boron, credit is also taken for the burnup of each assembly, as well as the presence of Integral Fuel Burnable Absorbers (IFBA) for fuel assemblies with an enrichment between 4.9 and 5.0 weight percent U-235. An IFBA consists of a neutron absorbing material applied as a thin Zirconium Diboride coating on the outside of a fuel pellet and is used as an integral part of some fuel rods. These parameters are controlled by the proposed TS.

This LAR also eliminates the distinction of storage cell types. Because the storage requirements associated with Region 2 cells are more restrictive and yield more conservative reactivity results than the Region 1 cells, the criticality analysis was performed for Region 2 racks only and the results were conservatively applied to the less reactive Region 1 racks. Likewise, PG&E proposes to apply the current Region 2 criticality analysis (Reference 1), which provides the all cell configuration evaluation approved for use in License Amendment Nos. 104/103, to the less reactive Region 1 racks.

While this LAR proposes the use of credit for soluble boron in the SFP criticality analysis, storage configurations have been defined to ensure that the spent fuel rack k_{eff} will be less than 1.0, including uncertainties on a 95/95 basis, without the presence of any soluble boron or Boraflex. Soluble boron credit provides significant negative reactivity in the criticality analysis to provide subcritical margin such that the SFP k_{eff} is maintained less than or equal to 0.95. The following storage configurations and enrichment limits resulted from the criticality analysis:

2-by-2 Array A repeating array of fuel assemblies arranged in a 2-by-2 storage configuration with: one fuel assembly with an initial enrichment less than or equal to 4.9 weight percent U-235, or with an initial enrichment less than or equal to 5.0 weight percent U-235 and an IFBA loading equivalent to 16 rods each with 1.5 milligrams

of Boron-10 per inch over 120 inches; and three fuel assemblies with a discharge burnup in the "acceptable" region of the burnup-enrichment curve provided within the criticality analysis.

Checkerboard A repeating checkerboard array of: one fuel assembly with an initial enrichment less than or equal to 5.0 weight percent U-235; and one empty storage location defined as a cell without a fuel assembly or a cell loaded with nonfissile components.

The revised TS continue to specify the requirements for the SFP storage configurations. The only significant changes relate to the criteria for determining the storage configuration. In addition to the current criteria based on initial enrichment, burnup, and fuel pellet diameter, IFBA loading will also be used to determine the SFP storage configuration. Since the proposed SFP storage configuration limitations will be similar to those currently in the DCPP TS, the new limitations will have minimal effect on normal pool operations and maintenance. The proposed changes will ensure that fuel is stored in a configuration consistent with the criticality analysis and will ensure that the design requirements are met.

The analysis of the reactivity effects of fuel storage in the DCPP spent fuel racks was performed with SCALE-PC, a personal computer version of the SCALE-4.3 code package (which includes BON-AMI, NITAWL-II, CSAS-25, and KENO-V.a), with the 44-group ENDF/B-V neutron cross section library. Since the KENO-V.a code package does not have burnup capability, depletion analyses were made with the two-dimensional integral transport theory code (DIT), which uses an 89-group structure collapsed from the ENDF/B-VI library. The SCALE-PC models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the DCPP racks are designed and have been found to adequately reproduce the critical values. The selected critical experiments included the Babcock & Wilcox experiments carried out in support of Close Proximity Storage of Power Reactor Fuel and the Pacific Northwest Laboratory Program carried out in support of the design of Fuel Shipping and Storage Configurations. This experimental data is sufficiently diverse to establish that the method bias and uncertainty are applicable to the DCPP storage rack conditions. The DIT code and its cross section set have been used in the design of reload cores and extensively benchmarked against operating reactor history and test data.

The spent fuel criticality analysis in Enclosure 5 addresses all the fuel types currently stored in the SFP and in use in the reactor at DCPP. For

purposes of this analysis, the different types of fuel assemblies were surveyed so as to define a reference design fuel assembly that would assure conservative results. The design basis fuel assembly for the fresh fuel was taken to be a conservative representation of the Westinghouse OFA 17-by-17 fuel assembly having a nominal enrichment of 4.9 weight percent U-235 and no IFBAs. The design basis fuel assembly for the burned fuel was taken to be a conservative approximation to the Westinghouse Standard 17-by-17 fuel assembly. This conservative approximation to the burned fuel assembly envelopes the characteristics of all burned fuel assemblies currently stored in the DCPP SFPs. This design basis burned fuel assembly was represented by a four-node axial representation of the assembly burnup and applicable fuel and moderator temperatures.

A methodology bias (determined from benchmark calculations), as well as a reactivity bias to account for the effect of the normal range of SFP water temperature (50°F to 212°F), was included. Uncertainties due to fuel assembly manufacturing tolerances, rack fabrication tolerances, and the KENO methodology were appropriately determined with a 95/95 basis.

The concept of reactivity equivalencing due to fuel burnup was used in the criticality analysis to equate the reactivity of a fuel assembly that has a particular initial enrichment and burnup combination to the reactivity of a fuel assembly that has a lower initial enrichment and zero burnup. The NRC has accepted the use of reactivity equivalencing for this purpose provided the geometric configuration and the conditions under which the equivalency was determined remain unchanged.

The DCPP criticality analysis also addresses postulated accidents in the SFP. The soluble boron concentration required to maintain k_{eff} less than or equal to 0.95 under accident conditions was determined by first surveying all possible events which increase the k_{eff} value of the SFP. Examples of such accidents are the misplacement of a fresh fuel assembly in place of a burned fuel assembly within a rack module, the accidental drop of a fresh fuel assembly outside the rack module, and the T-Bone drop. The event that would produce the largest increase in the SFP k_{eff} value was evaluated to be the misloading of a 4.9 weight percent U-235 fuel assembly with zero burnup and no IFBAs just outside of the fuel rack and immediately adjacent to a fresh fuel assembly inside the fuel rack. However, the minimum SFP boron concentration value of 2,000 ppm required by the TS is more than sufficient to maintain k_{eff} less than or equal to 0.95 for this reactivity increase. By virtue of the double contingency principle, which has been endorsed by the staff, two unlikely independent and concurrent events are beyond the scope of the required analysis. Therefore, credit for the presence of the entire 2,000 ppm of

soluble boron may be assumed in evaluating accident conditions such as a fuel misplacement.

Based on the results of the new criticality analysis (Enclosure 5), a SFP boron concentration of 798 ppm would be adequate to maintain the spent fuel storage rack k_{eff} less than or equal to 0.95, on a 95/95 basis, assuming the most limiting single fuel mishandling accident. This soluble boron concentration assumes an atomic fraction for Boron-10 of 0.199. For a Boron-10 isotopic fraction equal to 0.197, as recently measured at DCPP, the total soluble boron concentration required to maintain the same concentration of Boron-10 atoms would be equal to 806 ppm. PG&E proposes to retain the current minimum SFP soluble boron concentration limit of 2,000 ppm within the DCPP TS as it will conservatively maintain the spent fuel storage rack k_{eff} less than or equal to 0.95 when fuel is stored in accordance to the configurations evaluated in the criticality analysis.

A boron dilution analysis (Enclosure 6) has also been completed to support crediting boron in the DCPP Units 1 and 2 SFP criticality analysis. The boron dilution analysis includes an evaluation of the following plant specific features:

- Dilution Sources
- Boration Sources
- Instrumentation
- Administrative Procedures
- Piping
- Loss of Offsite Power Impact
- Boron Dilution Initiating Events
- Boron Dilution Times and Volumes

As part of that analysis, calculations were performed to define the dilution times and volumes for the SFP. The dilution sources available were compiled and evaluated against the calculated dilution volumes to determine the potential of a SFP dilution event. The analysis shows that a large volume of water (approximately 347,000 gallons) is necessary to dilute the SFP from the TS limit of 2,000 ppm to 800 ppm boron. The 800 ppm endpoint was utilized to ensure that k_{eff} for the spent fuel racks would remain less than or equal to 0.95. As part of the criticality analysis for the SFPs, a calculation has been performed on a 95/95 basis to show that the spent fuel rack k_{eff} remains less than 1.0 with unborated water in the pool. Thus, even if the SFP were diluted to zero ppm boron, which would take significantly more water than evaluated in the dilution analysis, the SFP would remain subcritical and the health and safety of the public would be protected.

The boron dilution analysis demonstrates that adequate time is available to identify and mitigate the dilution event before the spent fuel rack k_{eff} design basis of 0.95 is exceeded. A dilution event large enough to result in a significant reduction in the SFP boron concentration would involve the transfer of a large quantity of water from a dilution source and a significant increase in SFP level, which would ultimately overflow the pool. Such a large water volume turnover, and the overflow of the SFP, would be readily detected and terminated by plant personnel (Enclosure 6).

In addition, because of the large quantities of water required and the low dilution flow rates available at DCPP, any significant dilution of the SFP boron concentration would only occur over a long period of time (hours to days). Detection of a SFP boron dilution via level alarms and/or visual inspection would be expected before a dilution event sufficient to increase k_{eff} above 0.95 could occur.

The proposed changes to the SFP boron concentration sampling requirements are intended to provide more frequent verification that the SFP boron concentration limits are being met. Because significant reductions in the SFP boron concentration will be accompanied by an increase in pool volume or significant changes in the source of unborated water to the pool, any significant reductions in the pool boron concentration would be readily detected during normal operator rounds or by the pool level instrumentation. Sampling and verification of the SFP boron concentration on a 7-day frequency will provide adequate assurance that smaller and less readily identifiable boron concentration reductions are not taking place.

The results of the SFP dilution analysis, summarized in Enclosure 6, conclude that an unplanned or inadvertent event that would result in the dilution of the SFP boron concentration from 2,000 ppm to 800 ppm is not a credible event.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequence of an accident previously evaluated?

Response: No.

There is no increase in the probability of a fuel assembly drop accident in the spent fuel pool (SFP) when considering the presence of soluble boron in the SFP water for criticality control. The handling of the fuel assemblies does not change as a result of crediting soluble boron in the SFP.

There is no increase in the probability of the accidental misloading of a fuel assembly into the SFP racks when considering the presence of soluble boron in the SFP water for criticality control. Fuel assembly placement will continue to be controlled pursuant to approved fuel handling procedures and will be in accordance with the Technical Specification (TS) SFP storage configuration limitations.

There is no increase in the consequences of an accidental drop or accidental misloading of a fuel assembly into the SFP racks because the criticality analysis demonstrates that the pool will remain subcritical following either event even if the pool contains a boron concentration less than that currently specified in the TS. The current TS limitation will ensure that an adequate SFP boron concentration will be maintained.

There is no increase in the probability of the loss of normal cooling to the SFP water considering the presence of soluble boron in the pool water for subcriticality control since a high concentration of soluble boron has always been maintained in the SFP water.

There is no increase in the consequences of a loss of normal SFP cooling because the 2,000 ppm boron concentration required by TS provides significant negative reactivity to provide subcritical margin such that the SFP k_{eff} is maintained less than or equal to 0.95 up to boiling (212°F).

Therefore, the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

Spent fuel handling accidents are not new or different types of accidents; they have been analyzed in Section 15.5.22 of the Updated Final Safety Analysis Report (UFSAR).

Criticality accidents in the SFP are not new or different types of accidents; they have been analyzed in the UFSAR and in the Criticality Analysis reports associated with the specific license amendments for fuel enrichments up to 5.0 weight percent U-235.

Because soluble boron has always been required in the SFP water, and is currently required by TS, credit for soluble boron will have no effect on normal pool operation and maintenance. Crediting soluble boron in the SFP criticality analysis will only result in increased sampling to verify the boron concentration. This increased sampling frequency will not create the possibility of a new or different kind of accident.

The SFP dilution analysis demonstrates that a dilution which could increase the rack k_{eff} to greater than 0.95 is not a credible event. Therefore, crediting soluble boron in the SFP criticality analysis will not result in the possibility of a new kind of accident.

Revised specifications continue to specify the requirements for SFP storage configurations. The only significant changes relate to the criteria for determining the storage configuration. Because the proposed SFP storage configuration limitations will be similar to those currently contained in the TS, the new limitations will not have any significant effect on normal SFP operations and maintenance and will not create the possibility of a new or different kind of accident. A SFP loading verification will continue to be performed to ensure that the SFP loading configuration meets the specified requirements.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The TS changes proposed by this license amendment request and the resulting spent fuel storage limitations will provide an adequate safety margin to ensure that the stored fuel assembly array will always remain subcritical. Those limits are based on a plant specific criticality

analysis performed for the Diablo Canyon Units 1 and 2 SFPs that includes technically supported margins.

While the criticality analysis utilized credit for soluble boron, storage configurations have been defined to ensure that the spent fuel rack k_{eff} will be less than 1.0 with no soluble boron with a 95 percent probability at a 95 percent confidence level. Soluble boron credit is used to offset uncertainties, tolerances and off-normal conditions, and to provide subcritical margin such that the SFP k_{eff} is maintained less than or equal to 0.95. Since k_{eff} is less than or equal to 0.95, the current margin of safety is maintained.

A substantial reduction in the SFP soluble boron concentration that could lead to exceeding a k_{eff} of 0.95 has been evaluated and shown not to be credible.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PG&E concludes that the proposed amendment presents no significant hazards considerations under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

6.0 ENVIRONMENTAL EVALUATION

PG&E has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. PG&E has evaluated the proposed changes and has determined that the changes do not involve (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (3) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CRF 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

References

1. Holtec Report HI-931076, "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0% Enrichment," S.E.Turner, October 1993.

2. WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", Revision 1, November 1996.

AFFECTED TECHNICAL SPECIFICATION PAGES
(MARKED-UP PAGES)

Pages TS Table of Contents Page 3 of 4 (1 page)
TS 3.7-27, 3.7-28 thru -32, 4.0-1 (7 pages)

(continued)

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Boron Concentration

LCO 3.7.16 The spent fuel pool boron concentration shall be ≥ 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	------------------NOTE------------------ LCO 3.0.3 is not applicable. --	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the spent fuel pool boron concentration is within limit.	~~31~~ days 7

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage - Region 1

LCO 3.7.17.1 The combination of initial enrichment, initial B-10 content, burnup, and storage pattern of each spent fuel assembly stored in Region 1 shall be:

a. The initial enrichment is 4.5 weight percent U-235 or less; or

b. The initial enrichment is from 4.5 up to a maximum of 5.0 weight percent U-235, and any of the following conditions are met:

1) The combination of initial enrichment and cumulative burnup of the assemblies is within the acceptable area of Figure 3.7.17-1; or

2) The assemblies initially contained a minimum of a nominal 36 mg/in. per assembly of the isotope B-10 integrated in the fuel rods; or

3) The assemblies are put in a checkerboard pattern with any of the following:

a) water cells, or

b) assemblies that initially contained a minimum of a nominal 72 mg/in. per assembly of the isotope B-10 integrated in the fuel rods, or

c) partially irradiated fuel of at least 8000 MWD/MTU cumulative burnup: or

4) The assemblies are put into a pattern with alternate rows of fuel assemblies and water cells.

APPLICABILITY: Whenever any fuel assembly is stored in Region 1 of the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements for the LCO not met.	A.1	-----------NOTE-------------- LCO 3.0.3 is not applicable. ---------------------------------- Initiate action to move the noncomplying fuel assembly into an acceptable pattern that complies with this LCO or LCO 3.7.17.2.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1.1	Verify by administrative means that the fuel assembly characteristics and its expected storage location is in accordance with LCO 3.7.17.1.	Prior to each fuel assembly move, when the assembly will be stored in Region 1.

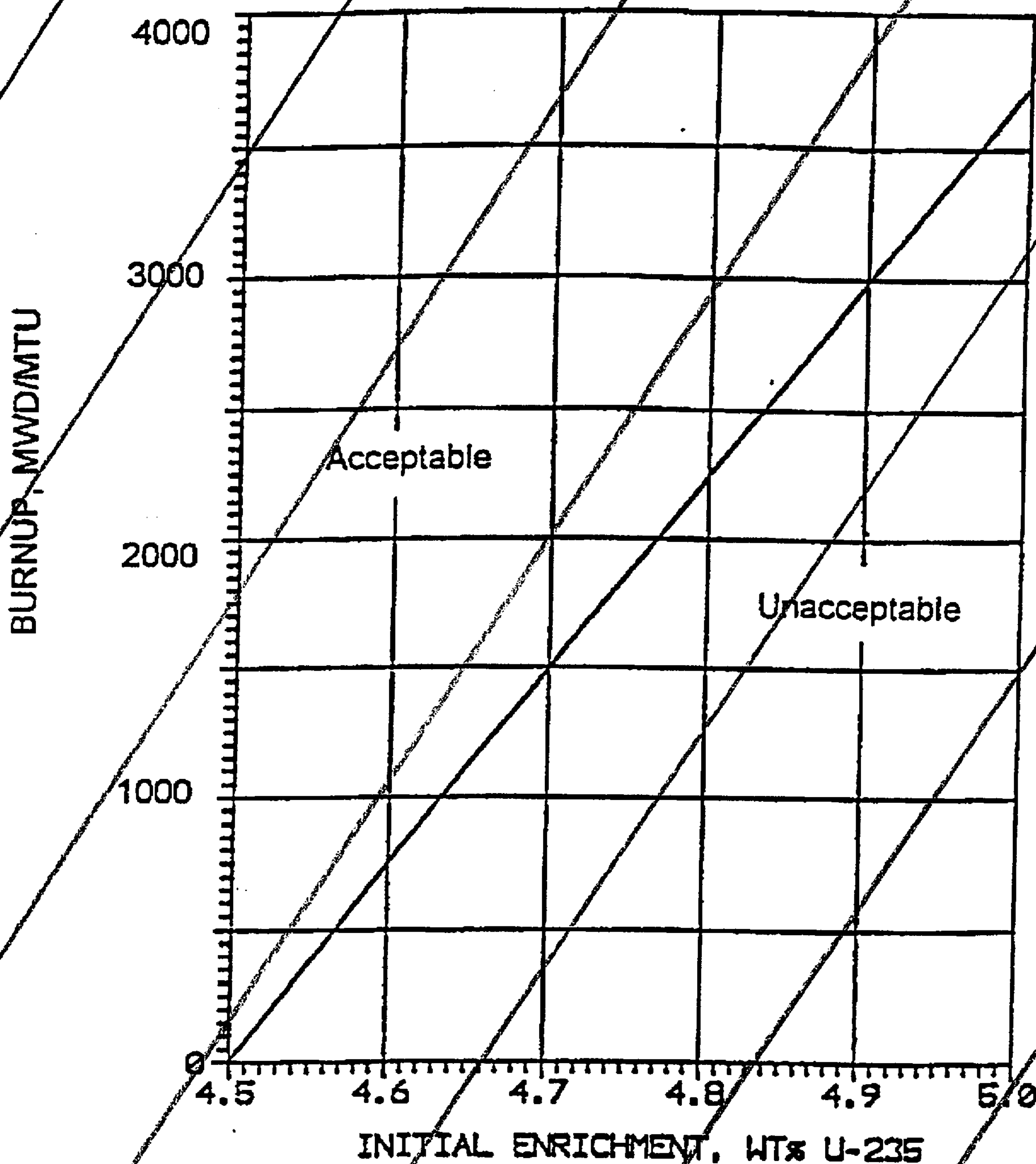

FIGURE 3.7.17-1
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT (NO IFBA) TO PERMIT
STORAGE IN REGION 1

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage ~~- Region 2~~

LCO 3.7.17~~.2~~ ~~The combination of initial enrichment, fuel pellet diameter and burnup of each spent fuel assembly stored in Region 2 shall be within the acceptable area of Figure 3.7.17-2, or the fuel assembly is stored in a checker board pattern with water cells or non-fissile material.~~

INSERT A

APPLICABILITY: Whenever any fuel assembly is stored in ~~Region 2 of~~ the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ------------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------ Initiate action to move the noncomplying fuel assembly into an acceptable ~~pattern that complies with this LCO or LCO 3.7.17.1~~ storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.~~2.~~1	Verify by administrative means that the fuel assembly characteristics and its expected storage location is in accordance with LCO 3.7.17.~~2~~	Prior to each fuel assembly move, when the assembly will be stored in ~~Region 2~~ the spent fuel pool.

Insert A – TS 3.7.17

Fuel assembly storage in the spent fuel pool shall be maintained such that any four cells shall be in a configuration as shown in Figure 3.7.17-1.

Insert B – TS 3.7.17

A	A
A	A

All Cell Configuration

F	B
B	B

2x2 Array Configuration

G	W
W	G

Checkerboard Configuration

<u>All Cell:</u>

A Fuel assembly with a discharge burnup in the "acceptable" region of Figure 3.7.17-2.

<u>2x2 Array:</u>

F (a) Fuel assembly with an initial enrichment ≤ 4.9 wt% U-235; or

(b) Fuel assembly with an initial enrichment ≤ 5.0 wt% U-235 and an IFBA loading equivalent to 16 rods each with 1.5 mg ^{10}B/in over 120 inches.

B Fuel assembly with a discharge burnup in the "acceptable" region of Figure 3.7.17-3.

<u>Checkerboard:</u>

G Fuel assembly with an initial enrichment ≤ 5.0 wt% U-235.

W Water cell – locations where fuel assemblies are not present, non-fissile components are permitted.

FIGURE 3.7.17-1
ALLOWABLE STORAGE CONFIGURATIONS
(ALL CELL, 2X2 ARRAY, CHECKERBOARD)
FOR THE SPENT FUEL POOL

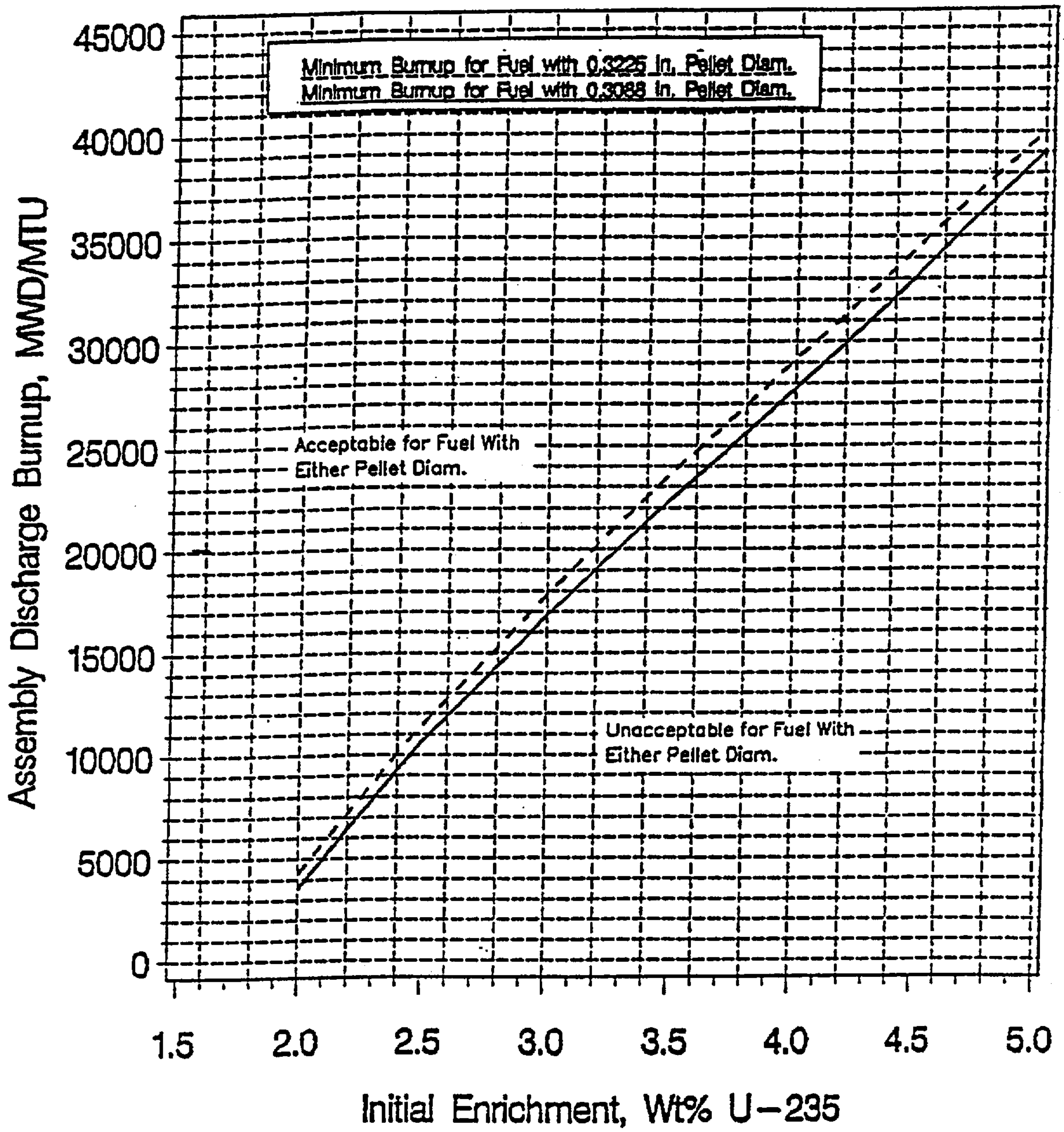

Figure 3.7.17-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT ~~TO PERMIT STORAGE IN REGION 2~~ AND FUEL PELLET DIAMETER FOR AN ALL CELL STORAGE CONFIGURATION

Insert C – TS 3.7.17

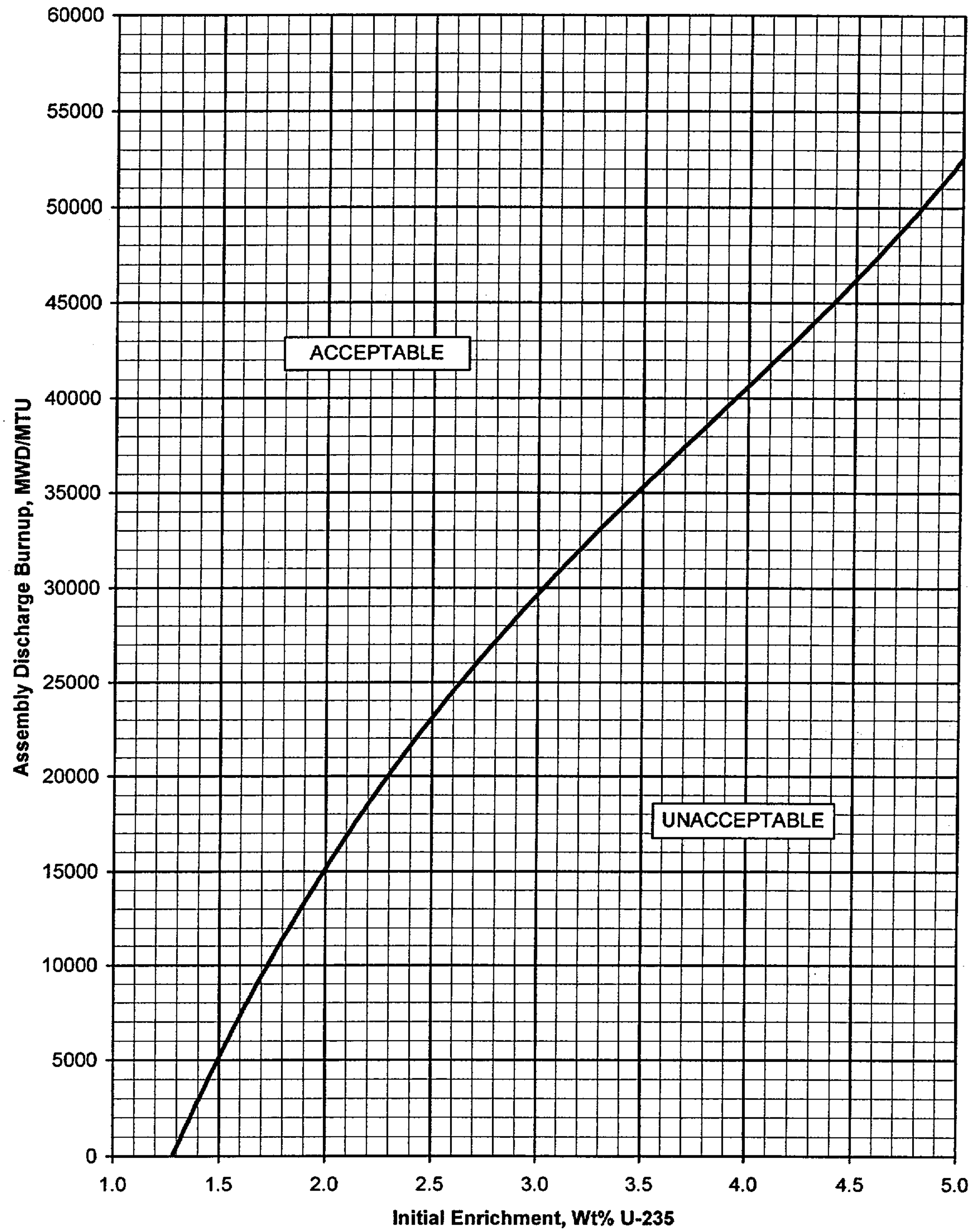

FIGURE 3.7.17-3
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR A 2X2 ARRAY STORAGE CONFIGURATION

4.0 DESIGN FEATURES

4.1 Site Location

The DCPP site consists of approximately 750 acres which are adjacent to the Pacific Ocean in San Luis Obispo County, California, and is approximately twelve (12) miles west-southwest of the city of San Luis Obispo.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control rod material shall be silver, indium, and cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent; < 1.0

b. k_{eff} ~~≤ 0.95~~ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the FSAR;

INSERT D

~~c.~~ d. A nominal 11 inch center to center distance between fuel assemblies placed in the fuel storage racks~~.~~ ;

INSERT E

(continued)

Insert D – TS 4.3.1

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 806 ppm, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the FSAR;

Insert E – TS 4.3.1

e. Fuel assemblies with a discharge burnup in the “acceptable” region of Figure 3.7.17-2 for the all cell configuration as shown in Figure 3.7.17-1;

f. Fuel assemblies with a discharge burnup in the “acceptable” region of Figure 3.7.17-3 for the 2x2 array configuration as shown in Figure 3.7.17-1.

REVISED TECHNICAL SPECIFICATION PAGE

Pages	TS Table of Contents Page 3 of 4 (1 page)
	TS 3.7-27, 3.7-28 thru 31, 4.0-1 thru -2 (7 pages)

(continued)

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Boron Concentration

LCO 3.7.16 The spent fuel pool boron concentration shall be $\geq$ 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	-------------------NOTE----------------- LCO 3.0.3 is not applicable. ---	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the spent fuel pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17 Fuel assembly storage in the spent fuel pool shall be maintained such that any four cells shall be in a configuration as shown in Figure 3.7.17-1.

APPLICABILITY: Whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ------------NOTE-------------- LCO 3.0.3 is not applicable. ---------------------------------- Initiate action to move the noncomplying fuel assembly into an acceptable storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.17.1	Verify by administrative means that the fuel assembly characteristics and its expected storage location is in accordance with LCO 3.7.17.	Prior to each fuel assembly move, when the assembly will be stored in the spent fuel pool.

A	A
A	A

All Cell Configuration

F	B
B	B

2x2 Array Configuration

G	W
W	G

Checkerboard Configuration

<u>All Cell:</u>

A Fuel assembly with a discharge burnup in the "acceptable" region of Figure 3.7.17-2.

<u>2x2 Array:</u>

F (a) Fuel assembly with an initial enrichment ≤ 4.9 wt% U-235; or

(b) Fuel assembly with an initial enrichment ≤ 5.0 wt% U-235 and an IFBA loading equivalent to 16 rods each with 1.5 mg ^{10}B/in over 120 inches.

B Fuel assembly with a discharge burnup in the "acceptable" region of Figure 3.7.17-3.

<u>Checkerboard:</u>

G Fuel assembly with an initial enrichment ≤ 5.0 wt% U-235.

W Water cell – locations where fuel assemblies are not present, non-fissile components are permitted.

FIGURE 3.7.17-1
ALLOWABLE STORAGE CONFIGURATIONS
(ALL CELL, 2X2 ARRAY, CHECKERBOARD)
FOR THE SPENT FUEL POOL

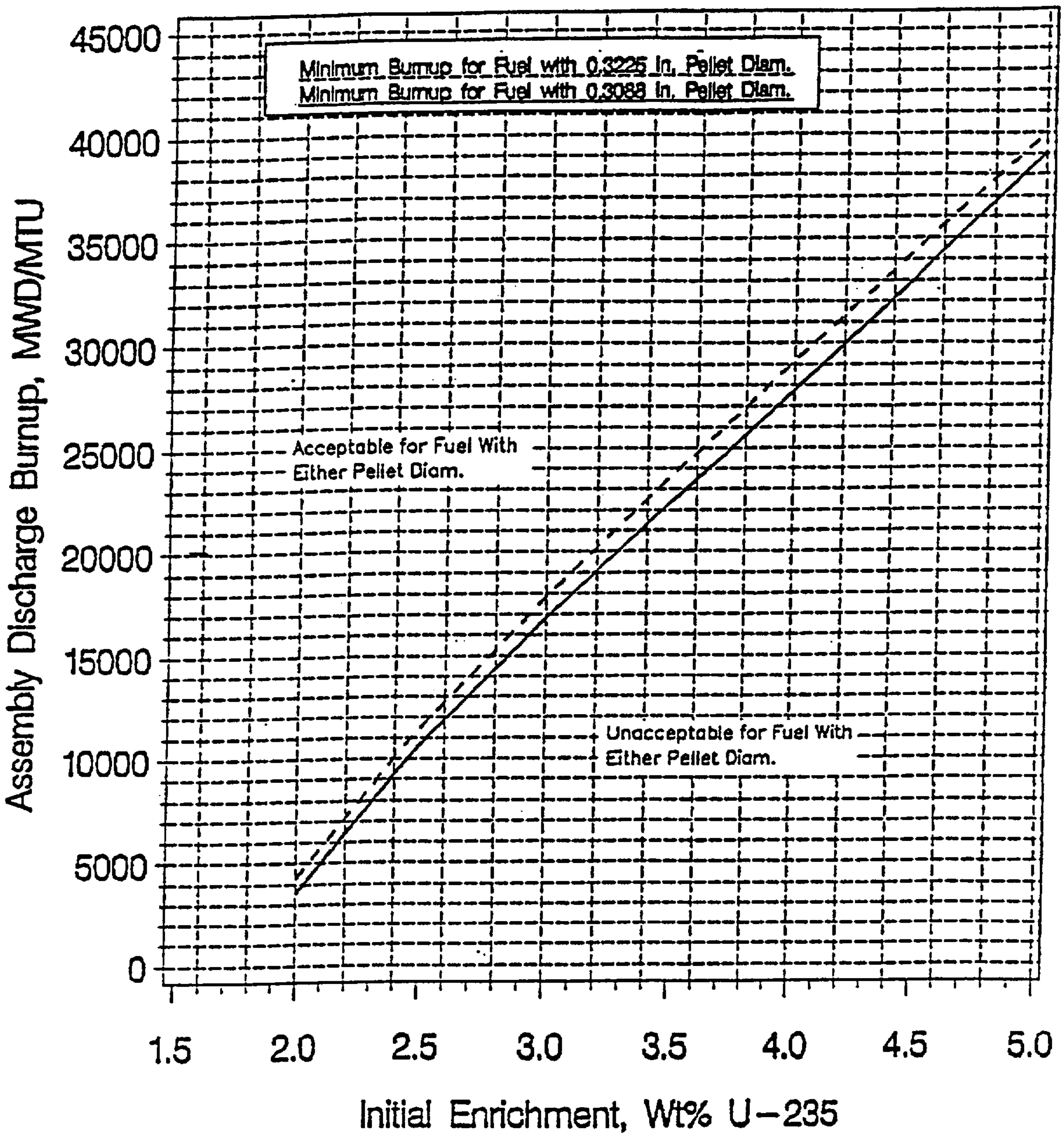

FIGURE 3.7.17-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT AND FUEL PELLET DIAMETER
FOR AN ALL CELL STORAGE CONFIGURATION

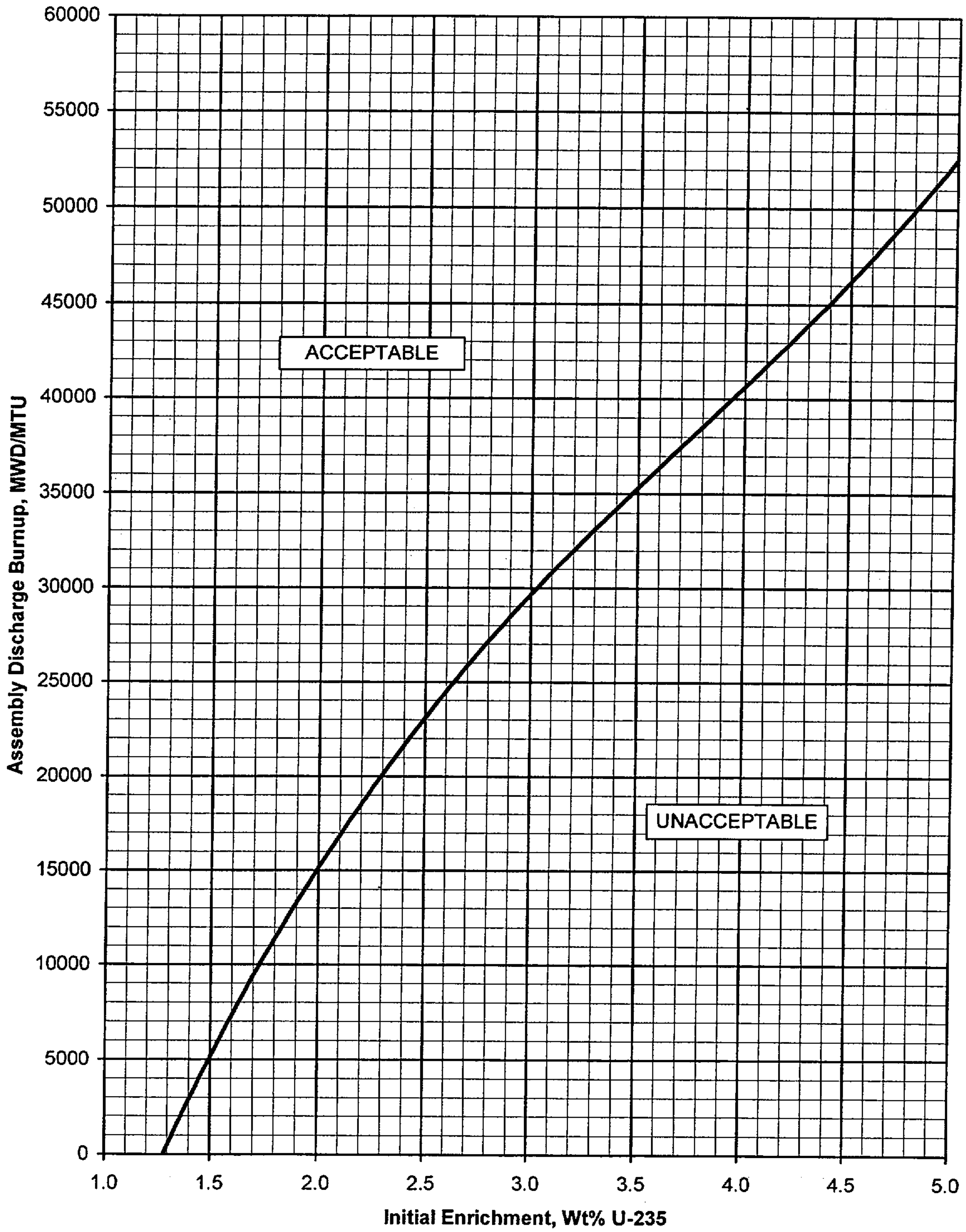

FIGURE 3.7.17-3
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR A 2X2 ARRAY STORAGE CONFIGURATION

4.0 DESIGN FEATURES

4.1 Site Location

The DCPP site consists of approximately 750 acres which are adjacent to the Pacific Ocean in San Luis Obispo County, California, and is approximately twelve (12) miles west-southwest of the city of San Luis Obispo.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control rod material shall be silver, indium, and cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the FSAR;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 806 ppm, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the FSAR;

d. A nominal 11 inch center to center distance between fuel assemblies placed in the fuel storage racks;

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

e. Fuel assemblies with a discharge burnup in the "acceptable" region of Figure 3.7.17-2 for the all cell configuration as shown in Figure 3.7.17-1;

f. Fuel assemblies with a discharge burnup in the "acceptable" region of Figure 3.7.17-3 for the 2x2 array configuration as shown in Figure 3.7.17-1.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR; and

d. A nominal 22 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies.

REVISED TECHNICAL SPECIFICATION BASES PAGES

Pages TS B3.7-69 thru B3.7-73 (5 pages)

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The DCPP Units 1 and 2 spent fuel pools (Ref. 1) each consist of 16 stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells. The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17.

Criticality analyses have been performed (Ref. 2 and 3) which demonstrate that the multiplication factor, k_{eff}, of the fuel assemblies in the spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of soluble boron is credited in the spent fuel pool criticality analysis. Reference 2 provides the analysis for the 2x2 array and checkerboard configurations, and Reference 3 provides the analysis for the all cell configuration. Both criticality analyses evaluate the region of the spent fuel pool that does not contain any Boraflex panels because the storage requirements are more restrictive and yield more conservative reactivity results than the region containing Boraflex. The results of the analyses may be conservatively applied to the less reactive region.

Storage configurations were defined in the criticality analyses (Ref. 2) to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95. A soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16.

Criticality analyses considering accident conditions have also been performed (Ref. 2 and 3). These analyses establish the amount of soluble boron necessary to ensure that k_{eff} will be maintained less than or equal to 0.95 should the most adverse postulated accident occur. For such an occurrence, the double contingency principle of ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 4) can be applied. The NRC letter states it is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for such a postulated accident condition, the presence of additional soluble boron in the spent fuel pool water (above the concentration required for normal conditions and reactivity equivalencing) can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

(continued)

BASES (continued)

BACKGROUND (continued)	In addition to consideration of spent fuel pool criticality, a boron dilution analysis (Ref. 5) was performed to evaluate the time and water volumes required to dilute the spent fuel pool from 2000 to 800 ppm. The 800 ppm endpoint was utilized to ensure that k_{eff} for the spent fuel racks would remain less than or equal to 0.95. However, analyses have been performed to demonstrate that even if the spent fuel pool boron concentration was diluted to zero ppm, which would take significantly more water than evaluated in the boron dilution analysis, the spent fuel would be expected to remain subcritical and the health and safety of the public would be assured.
APPLICABLE SAFETY ANALYSES	Most accident conditions result in a negligible reactivity effect in the spent fuel pool (Ref. 2 and 3). However, scenarios can be postulated that could have more than a negligible positive reactivity effect. Examples of such scenarios are the misplacement of a fuel assembly, a significant increase in spent fuel pool temperature above the design basis temperature of 150°F, or a cask drop accident (Ref. 1, 2, and 3). A soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 under accident conditions, which is well within the 2000 ppm requirement of LCO 3.7.16. The negative reactivity effect of the soluble boron more than compensates for the increased reactivity caused by the postulated accident scenarios. The accident analyses is provided in the FSAR (Ref. 1). The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The spent fuel pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 1, 2, and 3. The specified boron concentration of 2000 ppm ensures that the spent fuel pool k_{eff} will remain less than or equal to 0.95 due to a postulated fuel assembly misload accident or boron dilution event.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

(continued)

BASES (continued)

ACTIONS	A.1 and A.2 The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies and immediately taking actions to restore the spent fuel pool boron concentration to greater than or equal to 2000 ppm. This suspension of fuel movement does not preclude movement of fuel assemblies to a safe position. If the LCO is not met while moving fuel assemblies, LCO 3.0.3 would not be applicable since the inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.
SURVEILLANCE REQUIREMENTS	SR 3.7.16.1 This SR verifies by chemical analysis that the concentration of boron in the spent fuel pool is at or above the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place.
REFERENCES	1. FSAR, Section 9.1, 15.4.5, and 15.5.22. 2. "Diablo Canyon Units 1 and 2 Spent Fuel Pool Criticality Analysis," February 14, 2001, Paul F. O'Donnell, Westinghouse Doc. No. A-DP1-FE-0001. 3. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E.Turner, October 1993, Holtec Report HI-931077. 4. Double contingency principle of ANSI N16.1-1975, as specified in an April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A). 5. "Diablo Canyon Units 1 and 2 Spent Fuel Pool Boron Dilution Analysis," January, 2001, Gary J. Corpora.

B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The DCPP Units 1 and 2 spent fuel pools (Ref. 1) each consist of 16 stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells. The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17. The spent fuel storage racks are designed to accommodate three different storage configurations as shown in Figure 3.7.17-1.

Criticality analyses have been performed (Ref. 2 and 3) which demonstrate that the multiplication factor, k_{eff}, of the fuel assemblies in the spent fuel storage racks is less than or equal to 0.95. In order to maintain k_{eff} less than or equal to 0.95, the presence of soluble boron is credited in the spent fuel pool criticality analysis. Reference 2 provides the analysis for the 2x2 array and checkerboard configurations, and Reference 3 provides the analysis for the all cell configuration. Both criticality analyses evaluate the region of the spent fuel pool that does not contain any Boraflex panels because the storage requirements are more restrictive and yield more conservative reactivity results than the region containing Boraflex. The results of the analyses may be conservatively applied to the less reactive region. A discussion of how soluble boron is credited for the storage of fuel assemblies in the spent fuel pool is contained in the background for TS 3.7.16 Bases.

Storage configurations were defined in the criticality analyses (Ref. 2) to ensure that k_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95. A soluble boron concentration of 806 ppm is required to maintain k_{eff} less than or equal to 0.95 for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16.

Prior to movement of an assembly, it is necessary to verify that SR 3.7.16.1 is current.

APPLICABLE SAFETY ANALYSES

The analyzed accidents that could have significant reactivity effects are the misplacement of a fuel assembly, a significant increase in spent fuel pool temperature above the design basis temperature of 150°F, or a cask drop accident (Ref. 1, 2, and 3). For these accident conditions, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Spent Fuel Pool Boron Concentration") ensures that k_{eff} will remain at or below 0.95.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO	The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with LCO 3.7.17, ensure the k_{eff} of the spent fuel storage pool will always remain ≤ 0.95, assuming the pool to be flooded with borated water.
APPLICABILITY	This LCO applies whenever any fuel assembly is stored in the spent fuel pool.
ACTIONS	A.1 The Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since the inability to move fuel assemblies is not sufficient reason to require a reactor shutdown. When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with LCO 3.7.17, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with LCO 3.7.17 which will return the fuel pool to an analyzed condition.
SURVEILLANCE REQUIREMENTS	SR 3.7.17.1 This SR verifies by administrative means that the each fuel assembly and its expected storage location are in accordance with LCO 3.7.17 prior to each fuel assembly move when the assembly is to be stored in the spent fuel pool. A complete record of initial enrichment, initial integral boron content, fuel pellet diameter, and the cumulative burnup analysis shall be maintained for the time period that each fuel assembly remains in the spent fuel pool.
REFERENCES	1. FSAR, Section 9.1, 15.4.5, and 15.5.22. 2. "Diablo Canyon Units 1 and 2 Spent Fuel Pool Criticality Analysis," February 14, 2001, Paul F. O'Donnell, Westinghouse Doc. No. A-DP1-FE-0001. 3. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E.Turner, October 1993, Holtec Report HI-931077.

PACIFIC GAS & ELECTRIC COMPANY
DIABLO CANYON UNITS 1 AND 2
SPENT FUEL POOL CRITICALITY ANALYSIS

FEBRUARY 20, 2001

Title: **Diablo Canyon Units 1 & 2 Spent Fuel Pool Criticality Analysis**

Document Number: **A-DP1-FE-0001** **Revision Number:** **00**

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00	09/04/2001	Paul F.O'Donnell	Emilio Fuentes	Jeffrey A. Brown	n/a	n/a	n/a	62	App. A : 9 App. B : 14 App. C : 7	CDROM

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1.0 Introduction

1.1 Objective

This report presents the results of criticality analyses for the Diablo Canyon Nuclear Power Plant, Units 1 & 2, spent fuel storage pools assuming no credit for Boraflex. The physical description of the spent fuel storage racks is provided in Reference 1. The primary objectives of this analysis are as follows:

1. to employ Soluble Boron Credit in establishing the design/storage basis for the spent fuel pool,

2. to establish assembly burnup versus initial enrichment limits for the fuel assemblies in the spent fuel pool. The storage arrangement is a repeating 2X2 array containing one fresh fuel assembly checkerboarded with three burned fuel assemblies.

3. eliminate the labeling of storage cell types. This analysis was performed for a 2X2 array in the present Region 2 cells. The Region 2 cells are known to yield more conservative reactivity results than the Region 1 cells. Therefore, the burnup versus initial enrichment storage curves produced in this analysis are valid throughout the spent fuel pool and the storage cells will not need different labels.

The methodology employed in this analysis for soluble boron credit is analogous to that of Reference 2 and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, Reference 3.

1.2 Design Criteria

The design criteria are consistent with GDC 62, Reference 4, and NRC guidance to all Power Reactor Licensees, Reference 5. Section 2.0 describes the analysis methods including a description of the computer codes used to perform the criticality safety analysis. A brief summary of the analysis approach and criteria follows.

1. Determine the fresh and spent fuel storage configuration of the spent fuel pool using no soluble boron conditions such that the 95/95 upper tolerance limit value of K_{eff} for the storage pool, including applicable biases and uncertainties, is less than unity.

2. Next, using the resulting storage configuration from the previous step, calculate the spent fuel rack effective multiplication factor with the chosen concentration of spent fuel pool soluble boron present. Then calculate the sum of: (a) the latter multiplication factor, (b) the reactivity uncertainty associated with fuel assembly and storage rack tolerances, and (c) the biases and other uncertainties required to determine the final 95/95 confidence level effective multiplication factor and show that at the chosen concentration of soluble boron, the system maintains the overall effective multiplication factor less than or equal to 0.95.

3. Use reactivity equivalencing methodologies to determine the minimum fuel assembly burnup for fuel assembly enrichments higher than allowed in Step 1, above.

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

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

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

where:

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

$SBC_{95/95}$ = soluble boron requirement for 95/95 $K_{eff} \leq 0.95$ (ppm),

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

SBC_{PA} = soluble boron required for $K_{eff} \leq 0.95$ under accident conditions (ppm).

For purposes of the analyses contained herein, minimum burnup limits established for fuel assemblies to be stored in the storage racks do include burnup credit established in a manner which takes into account conservative approximations to the operating history of the fuel assemblies. Variables such as the axial burnup profile as well as the axial profile of moderator and fuel temperatures have been factored into the analyses.

1.3 Design Approach

The design input employed in this analysis was directly obtained from Reference 1. The Soluble Boron Credit Methodology provides additional reactivity margin in the spent fuel storage analyses which may then be used to implement added flexibility in storage criteria and, for example, eliminate the need to implement the degraded boraflex

modeling. Boraflex in the spent fuel racks is not credited in this analysis. The Diablo Canyon spent fuel pool racks can be presently categorized into two regions – Region 1 and Region 2. The difference between the two types of racks is that Region 1 racks contain Boraflex in them while the Region 2 racks do not contain any Boraflex. All the calculations in this report are performed based on the Region 2 racks and the results can be conservatively applied to the Region 1 racks.

The selection of design basis fuel assembly types was based on an evaluation of the variety of fuel assemblies employed in the reactor to date and selecting the most reactive type for a given evaluation. The candidate fuel assembly types include the Westinghouse Standard and the Westinghouse OFA fuel assemblies of the 17X17 design. The Westinghouse Standard fuel assembly has been evaluated to be the design basis fuel assembly to represent burned fuel assemblies while the Westinghouse OFA has been chosen to represent the fresh fuel assemblies.

The reactivity characteristics of the Region 2 racks were evaluated using infinite lattice analyses; this environment was employed in the evaluation of the burnup limits versus initial enrichment as well as the evaluation of physical tolerances and uncertainties. A full spent fuel pool model was also employed to evaluate soluble boron worth and the reactivity worth of postulated accidents.

1.4 Methodology

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

SCALE-PC is used for calculations involving infinite arrays (one fresh fuel assembly checkerboarded with three burned fuel assemblies) of storage cells in Region 2 racks. In addition, it is employed in a full pool representation of the storage racks to evaluate soluble boron worth and postulated accidents.

SCALE-PC modules employed in both the benchmarking analyses and the spent fuel storage rack analyses include the control module CSAS25 and the following functional modules: BONAMI, NITAWL-II, and KENOV.a. All references to KENO in the text to follow should be interpreted as referring to the KENO V.a module.

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

1.4.1 SCALE-PC

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

1.4.2 Validation of SCALE-PC

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

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

The B&W group of experimental configurations employed variable spacing between individual rod clusters in the nominal 3 x 3 array. In addition, the effects of placing either SS-304 or Borated Aluminum plates of different boron contents in the water channels between rod clusters were measured. Table 1.4-1 summarizes the results of these analyses.

Eleven experimental configurations were selected from the PNL experimental program. These experiments included unpoisoned uniform arrays of fuel pins and 2 x 2 arrays of rod clusters with and without interposed SS-304 or B/Al plates of different blacknesses. As in the case of the B&W experiments, the full environment of the active fuel region was represented explicitly. Table 1.4-2 summarizes the results of these analyses.

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

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

where:

K_{KENO} is the KENO multiplication factor of interest,

ΔK_B is the mean calculational method bias,

$M_{95/95}$ is the 95/95 multiplier appropriate to the degrees of freedom for the number of validation analyses,

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

σ^2_{KENO} is the square of the KENO standard deviation

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

$$\Delta K_B = \frac{1}{n}\sum_{i=1}^{n}(1 - K_i),$$

where:

K_i is the ith value of the multiplication factor for the validation lattices of interest

$M_{95/95}$ is obtained from the tables of Reference 13.

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

$$(\sigma_m)^2 = \left[\frac{n\sum_1^n\left(K_i - K^{ave}\right)^2 \sigma_i^{-2}}{(n-1)\sum_1^n(\sigma_i)^{-2}}\right] - \sigma_{ave}^2$$

Where k^{ave} is given by the following equation.

$$K^{ave} = \frac{\sum_1^n K_i(\sigma_i)^{-2}}{\sum_1^n (\sigma_i)^{-2}}$$

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

$$\sigma^2_{ave} = \frac{\sum_1^n (\sigma_i)^2 G_i}{\sum_1^n G_i}$$

Where G_i is the number of generations.

For purposes of this bias evaluation, the data points of Tables 1.4-1 and 1.4-2 are pooled into a single group. With this approach, the mean calculational methods bias, ΔK_B, and the mean calculational variance, $(\sigma_m)^2$, calculated by equations given above, are determined to be 0.00259 and $(0.00288)^2$, respectively. The magnitude of $M_{95/95}$ is deduced from Reference 13 for the total number of pooled data points, 30.

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

$$K_{95/95} = K_{KENO} + 0.00259 + 2.22\,[0.00288^2 + (\sigma_{KENO}{}^2)]^{1/2}$$

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

1.4.3 Application to Fuel Storage Pool Calculations

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

Standard material compositions were employed in the SCALE-PC analyses consistent with those of Reference 1; these data are listed in Table 1.4-3. For fresh fuel conditions, the fuel nuclide number densities were derived within the CSAS25 module using input consistent with the data of Table 1.4-3. For burned fuel representations, the fuel isotopics were derived from the DIT code as described below.

1.4.4 The DIT Code

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

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

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

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

The isotopic number densities from the DIT calculation are based upon Cell average values. The input to KENO calculations require that the number densities be specified for the fuel pellet. Therefore, the number densities from the DIT calculations are scaled by the ratio of area of the cell to the area of the fuel pellet for use in the KENO calculations. The concentration of ^{10}B is determined by reactivity equivalencing a given DIT cell calculation with a corresponding KENO cell calculation to within the KENO one sigma uncertainty level.

1.5 Assumptions

1 The Westinghouse Standard fuel assembly was modeled with Zircaloy guide tubes in this analysis. In addition, this fuel assembly was modeled with no credit for grids and other structural materials.

2 The analysis conservatively assumed an infinite repeating array contained in the present Region 2 cells. The additional structural material contained in the current Region 1 cells was not modeled in this analysis.

1.6 Analysis Results

The primary objectives of this analysis were accomplished; a summary of the results is as follows.

1) Soluble boron credit methodology was employed to establish a target K_{eff} value of 0.9800 for the spent fuel pool at zero soluble boron. The allowance for applicable biases and uncertainties was deduced to be 0.01805; thus, the 95/95 upper tolerance limit value of K_{eff} was deduced to be 0.99805. The total soluble boron requirement for achieving a 95/95 value of $K_{eff} \leq 0.95$ was deduced to be the summation of the following three terms: $SBC_{95/95}$ = 233.2 ppm, SBC_{RE} = 51.6 ppm, and SBC_{PA} = 512.5 ppm for a total of 798 ppm. The soluble boron concentration was increased by 1% due to the difference in the ^{10}B atom percent used in the analysis (19.9 a/o) and that measured at Diablo Canyon (19.7 a/o). This results in a soluble boron concentration equal to 806 ppm. Note that this soluble boron concentration includes an allowance for 5 % burnup uncertainty.

2) The design basis fuel assembly for the fresh fuel storage cells in the Region 2 racks was taken to be a conservative representation of the Westinghouse OFA 17 x 17 fuel assembly having a nominal enrichment of 4.9 wt% ^{235}U and no IFBA loadings. The design basis fuel assembly for the burned fuel storage cells in the Region 2 racks was taken to be a conservative approximation to the Westinghouse Standard 17 x 17 fuel assembly. This conservative approximation to the burned fuel assembly envelopes the characteristics of all burned fuel assemblies currently stored in the spent fuel pool. This design basis burned fuel assembly was represented by a four-node axial representation of the assembly burnup and applicable fuel and moderator temperatures.

3) Minimum fuel assembly burnup limits versus fuel assembly initial average enrichment were established for the spent fuel storage cells that have burned fuel assemblies in them. These limits were established on a nominal basis for a repeating array of fuel assemblies arranged in a one-out-of-four storage pattern (one fresh fuel assembly and three burned fuel assemblies in a 2X2 array). The maximum fresh fuel assembly enrichment for storage in these racks is determined to be 4.9 w/o ^{235}U with no credit for any sort of burnable absorbers. Alternatively, fresh fuel assemblies having a minimum of 16 Integral Fuel Burnable Absorbers (IFBAs) may contain a maximum enrichment less than or equal to 5.0 w/o ^{235}U in the above mentioned storage configuration. An IFBA consists of a neutron absorbing material applied as a thin Zirconium Diboride coating on the outside of a UO_2 pellet. (IFBA was modeled in the middle 120 inches of the active fuel with a density of 1.5 grams ^{10}B per inch.) Note that the fuel stack density was modeled as 95 % of theoretical density. This translates into a pellet density approximately equal to 96 % of theoretical density.

4) No credit has been taken for any Boraflex material present in the storage racks. These results are applicable to fuel assemblies with Zirlo cladding also because Zirlo-clad fuel assemblies are slightly less reactive compared to the Zirc-4 clad assemblies (that are analyzed in this report).

The analyses contained herein lead to the conclusion that the total soluble boron concentration required to maintain K_{eff} less than 0.95, after including all biases and uncertainties and assuming the most limiting accident, is less than or equal to 806 ppm.

The most limiting accident condition was determined to be the misloading of a fresh 4.9 wt% ^{235}U fuel assembly just outside of the fuel rack and immediately adjacent to a fresh fuel assembly inside the fuel rack.

Section 4 of this report contains the plots defining the limits on storage of spent fuel assemblies versus assembly burnup and initial enrichment.

Table 1.4-1
Summary of Calculational Results for
Cores X Through XXI of the B&W Close Proximity Experiments

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

(a) - metal separating unit assemblies

(b) - spacing between unit assemblies in units of fuel rod pitch

Table 1.4-2
Summary of Calculational Results for Selected Experimental PNL Lattices, Fuel Shipping and Storage Configurations

Exp't. No.	K_{eff}	Comments
043	0.99787 ± 0.00106	Uniform rectangular array, no poison
044	1.00104 ± 0.00102	"
045	0.99955 ± 0.00101	"
046	0.99960 ± 0.00103	"
061	0.99792 ± 0.00099	2 x 2 array of rod clusters, no poison
062	0.99628 ± 0.00096	"
064	0.99696 ± 0.00103	2 x 2 array of rod clusters, 0.302 cm thick SS-304 cross
071	0.99970 ± 0.00101	2 x 2 array of rod clusters, 0.485 cm thick SS-304 cross
079	0.99463 ± 0.00102	2 x 2 array of rod clusters, cross of 0.3666 g boron/cm^2
087	0.99423 ± 0.00099	2 x 2 array of rod clusters, cross of 0.1639 g boron/cm^2
093	0.99787 ± 0.00098	2 x 2 array of rod clusters, cross of 0.1425 g boron/cm^2

Table 1.4-3
Standard Material Compositions Employed in Criticality Analysis for Diablo Canyon Nuclear Power Plant Spent Fuel Storage Rack

Material	Element	Weight Fraction
Zircaloy-4	Zr	0.9829
Den.= 6.56 g/cc	Sn	0.0140
	Fe	0.0021
	Cr	0.0010
Water	SCALE Standard Composition Library w/ Den. = 0.9982 @ 293 °K	
Concrete	SCALE Standard Composition Library	
Fresh UO_2	Fraction of Theoretical Density = 0.95	
SS304	SCALE Standard Composition Library	
Boron Isotopics	SCALE Standard Composition Library	

2.0 Design Input

As noted in the Introduction Section, the Diablo Canyon Units 1 & 2 spent fuel storage pool configuration and the Region 2 storage racks as analyzed herein are consistent with Reference 1. This section provides a brief description of the spent fuel storage rack with the objective of establishing a basis for the analytical model employed in the criticality analyses described in Section 3.1.

2.1 Spent Fuel Pool Storage Configuration Description

The Spent Fuel Pools in each of the Diablo Canyon Nuclear Power Plants are identical. Therefore, any reference to a spent fuel pool in this analysis is applicable to the spent fuel pool of both units. The spent fuel pool and the fuel storage rack types and orientation are illustrated in Figure 2.1-1. The spent fuel storage area in the pool is presently divided into Regions 1 and 2 – both employing a flux-trap design. Region 1 consists of storage racks nominally employing boraflex panels in each storage cell as parasitic absorbers. Region 2 racks are similar in design to those of Region 1 racks except that they do not contain any Boraflex absorber.

Changes in the spatial distribution of the Boraflex absorber material under irradiation resulted in the decision to take no credit for the presence of the Boraflex in the Region 1 storage modules for purposes of this analysis. Moreover, the calculations are performed for Region 2 racks only and the results are conservatively applied to the less reactive Region 1 racks. Therefore, the results of this analysis will yield burnup versus initial enrichment storage curves for a repeating 2X2 storage configuration (1 fresh, three burned) which are applicable to all cells within both spent fuel pools

2.2 Individual Storage Rack Type Descriptions

Subsequent sections describe the Region 1 and 2 storage racks in greater detail.

2.2.1 Region 1 Racks.

The current Region 1 storage racks are free standing and self-supporting. Each storage cell consists of a Stainless Steel box with an internal dimension of 8.85 inches and a thickness of 0.08 inches. A 0.02-inch stainless steel wrapper holds the 0.047-inch Boraflex poison plate. The nominal cell-to-cell spacing is 10.93 inches. The pertinent

dimensions of the constituent materials for the typical Region 1 rack storage cells are also summarized in Table 2.2-1.

2.2.2 Region 2 Racks

The current Region 2 storage racks are also free standing and self-supporting. Each storage cell consists of a Stainless Steel box with an internal dimension of 8.85 inches and a thickness of 0.09 inches. The nominal cell-to-cell spacing is 10.93 inches. The pertinent dimensions of the constituent materials for the typical Region 2 rack storage cells are also summarized in Table 2.2-2.

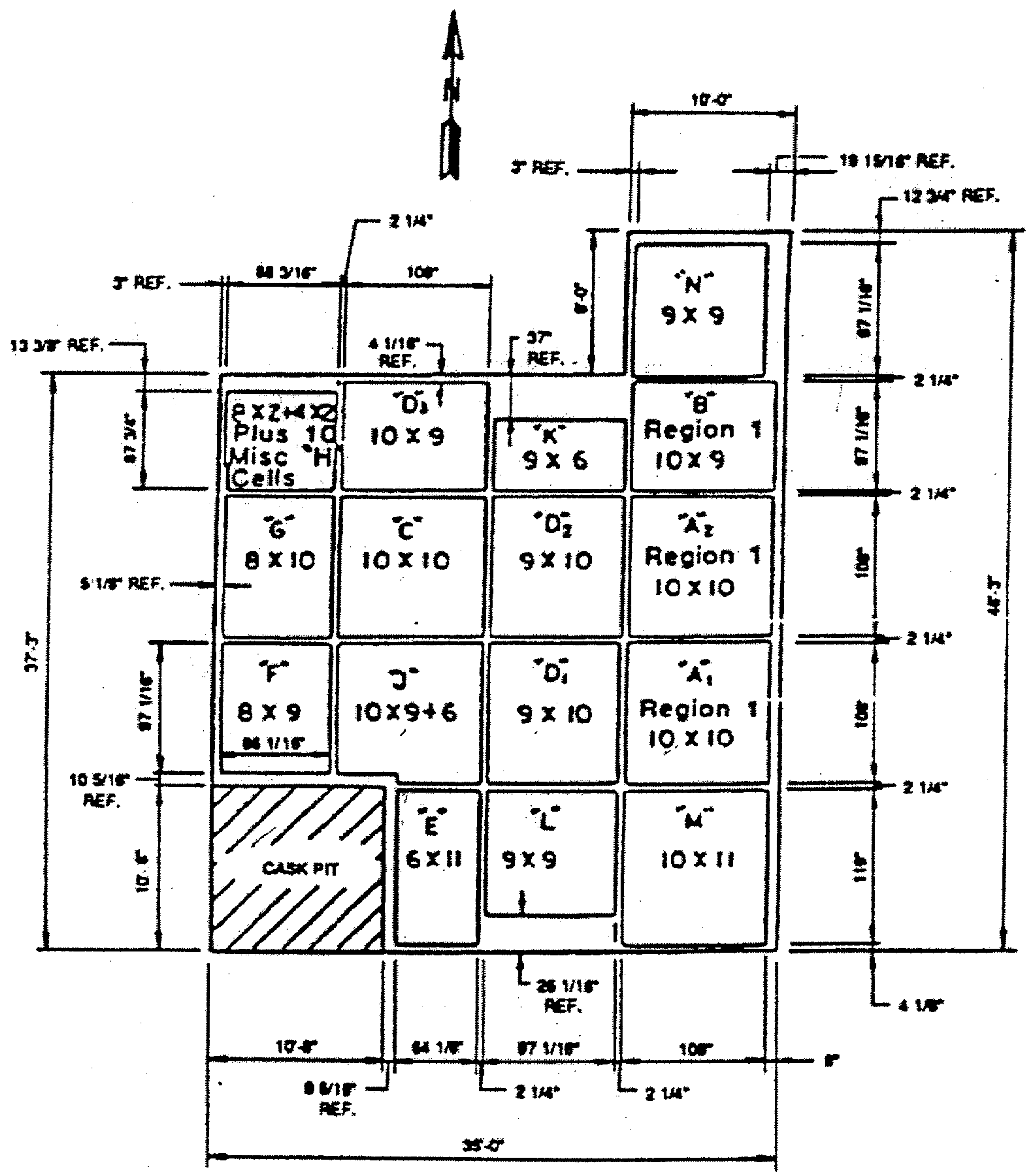
Table 2.2-1
Region 1 Storage Rack Cell Dimensions

Description	Design Dimensions
Cell Pitch, cm.(in.)	27.7622 (10.930)
Cell ID, cm.(in.)	22.479 ± 0.381 (8.850 ± 0.15)
Stainless Steel Wall Thickness, cm.(in.)	0.2032 (0.080)
Boraflex Plate Thickness, cm. (in.)	0.11938 ± 0.01778 (0.047 ± 0.007)
SS cover sheet thickness, cm.(in.)	0.0508 (0.020)
Water gap cm. (in.)	4.53644 ± 0.127 (1.786 ± 0.05)

Table 2.2-2
Region 2 Storage Rack Cell Dimensions

Description	Design Dimensions	Model Dimensions
Cell Pitch, cm.(in.)	27.7622 (10.930)	27.75966 (10.929)
Cell ID, cm.(in.)	22.479 ± 0.0381 (8.850 ± 0.015)	22.479 (8.850)
Stainless Steel Wall Thickness, cm.(in.)	0.2286 ± 0.0127 (0.090 ± 0.005)	0.2286 (0.090)
Water gap cm. (in.)	4.8260 ± 0.127 (1.900 ± 0.05)	4.82346 (1.899)

Figure 2.1-1
Spent Fuel Pool, General Arrangement



POOL LAYOUT UNIT 1

3.0 Analysis

3.1 KENO Models

The purpose of this section is to describe the models employed in KENO to represent either arrays of different types of storage cells or the full spent pool storage configuration.

3.1.1 Region 2

The Region 2 infinite array of storage cells was modeled using the dimensions of Table 2.2-2 except in this case the infinite array is based on a one-out-of-four arrangement of fresh and burned fuel assemblies in a 2X2 array with periodic boundary conditions. As noted previously, the four-zone axial burnup model is employed to simulate burned fuel assemblies. A conservative center-to-center spacing of 10.929 inches was utilized for the model with water at the ambient temperature and soluble boron concentration for the case being analyzed. Figure 3.1-1 shows a sketch of the geometry employed in the KENO representation of the Region 2 storage cell array.

The Region 1 cells are not modeled in this analysis. Based on the geometric and material specifications provided in Table 2.2-1 and Table 2.2-2, it is expected that the results for the Region 2 cells can be conservatively applied to the Region 1 cells. This is due to the reasoning that the Region 1 cells have the same cell-to-cell pitch but were manufactured with slightly larger amounts of structural steel. Therefore, the results based on the KENO model provided in Figure 3.1-1 can be applied to the entire spent fuel pool.

3.1.2 Full Spent Fuel Pool Model

A KENO model of the full spent fuel pool was created to evaluate fuel mishandling accidents and soluble boron requirements. The following are the salient features of the KENO model for the entire Diablo Canyon spent fuel pool.

- A total of 15 different rack modules are included in the spent fuel pool model. They are the "E", "L", "M", "A1", "D1", "J", "F", "A2", "D2", "C", "G", "B", "K", "D3" and "H" modules. Note that module "N" was not included in the full spent fuel pool model because of modeling constraints. Exluding module "N" will not affect the accuracy of the boron and fuel mishandling reactivity results.

- All the rack modules are modeled with the geometrical and material specifications of Region 2 cells.

- The cask pit was modeled as a water cell.

- A nominal inter-module gap of 2.25 inches was used throughout the model

- All the rack types are housed in a pool of water. The pool wall was not modeled in the analysis. A water gap of less than 4.0 inches was utilized between the peripheral modules and the model boundary.

- All the dimensions were based on the as-built dimensions of the Diablo Canyon spent fuel pool racks.

Figure 3.1-2 is a GIF plot from the KENO model of the left half of the spent fuel pool. Figure 3.1-3 is a GIF plot from the KENO model of the right half of the spent fuel pool.

Figure 3.1-1
Sketch of KENO Model for Infinite Array of Region 2 Cells

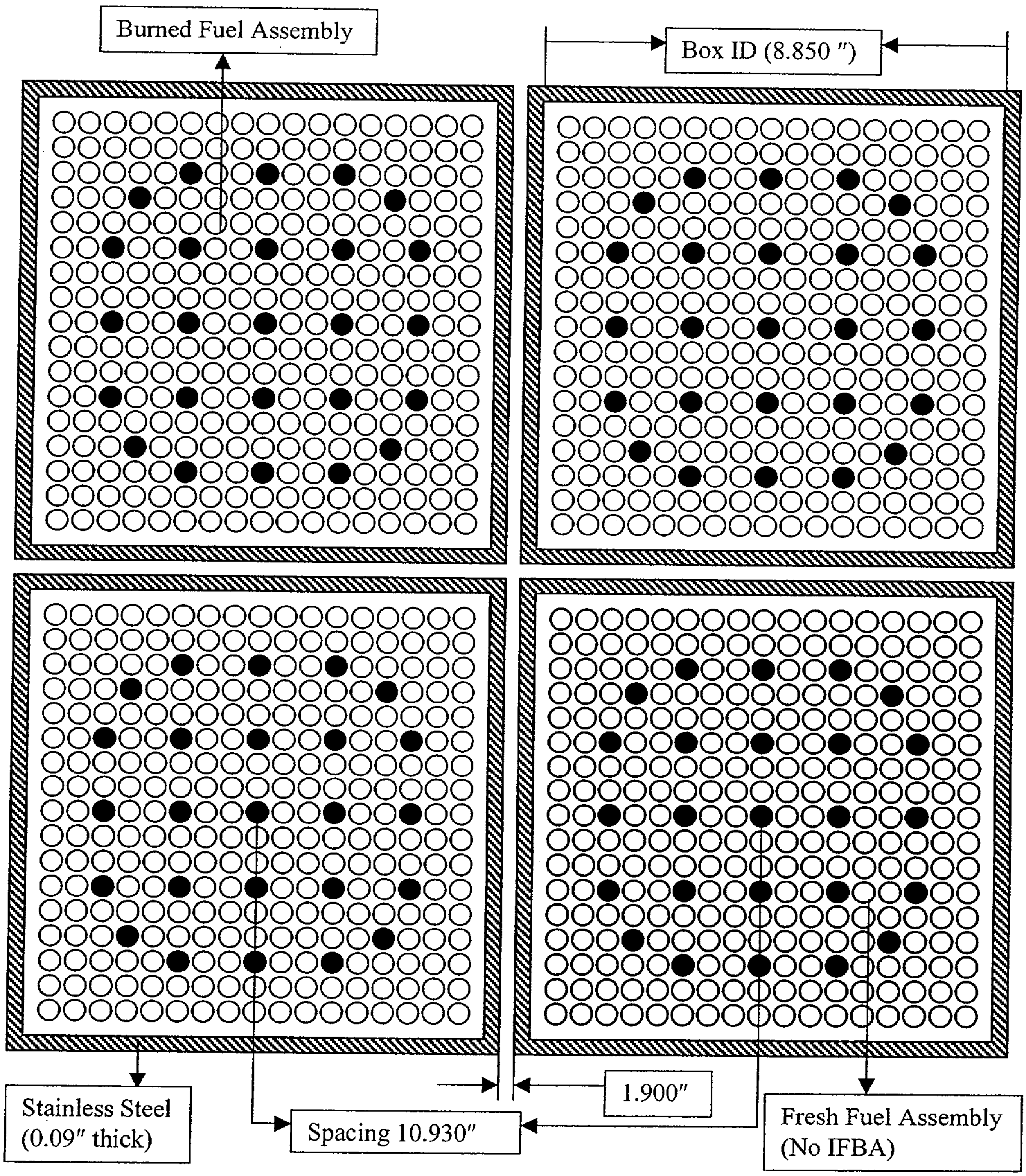


Figure 3.1-2
KENO GIF Plot of the Left Half of the Full Spent Fuel Pool Model

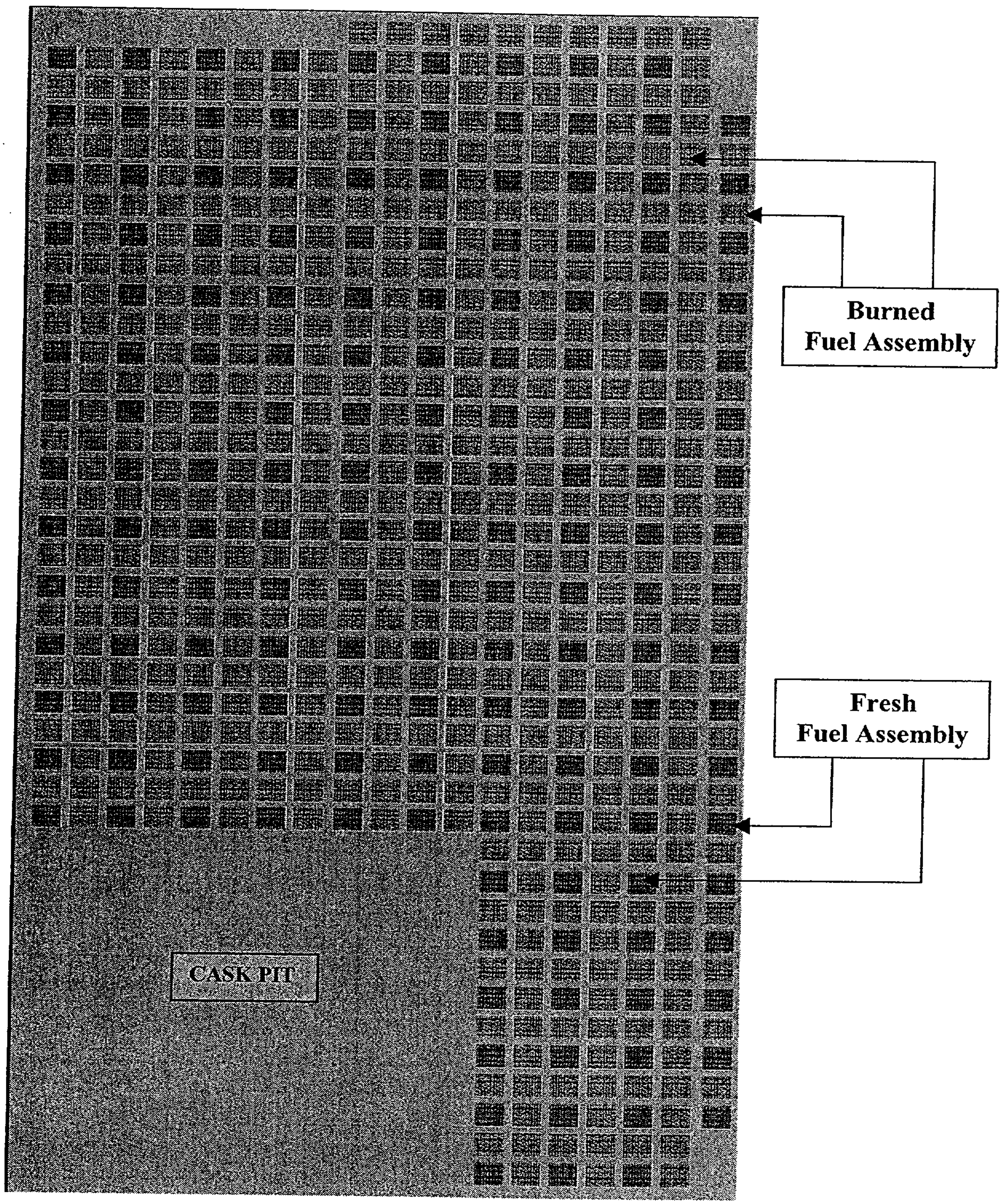
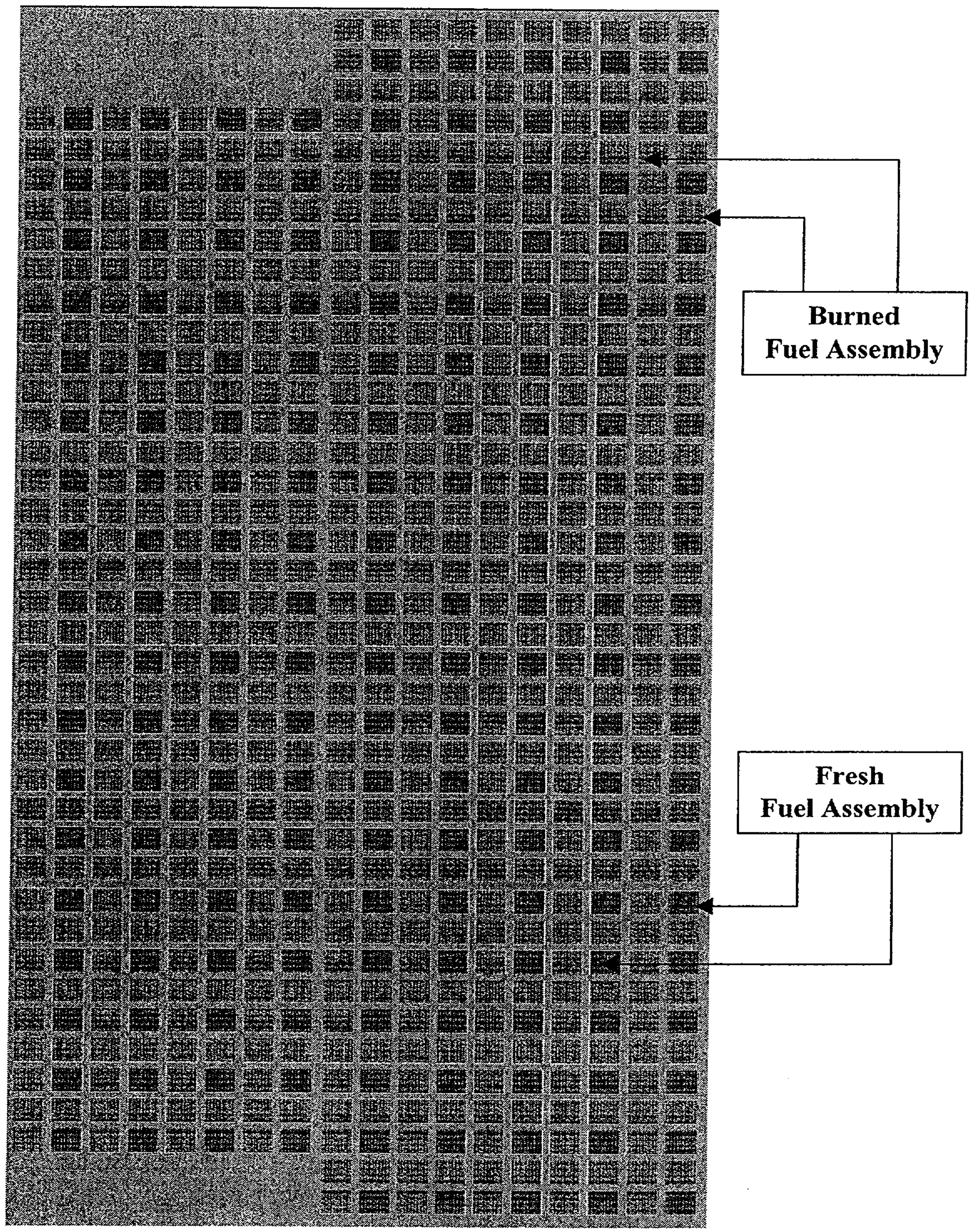


Figure 3.1-3
KENO GIF Plot of the Right Half of the Full Spent Fuel Pool Model



3.2 Design Basis Fuel Assemblies

The Diablo Canyon Nuclear Power Plants (Units 1 & 2) have been in operation for more than 15 years and during that time interval a variety of reload batches containing different fuel assembly designs have been cycled through the reactors. Thus, the criticality safety analysis of their spent fuel pools must take into account possible differences in the reactivity characteristics of the different assembly types. For purposes of this analysis, the different types of fuel assemblies were surveyed so as to define a reference design fuel assembly which would assure conservative results for the analysis.

3.2.1 Design Basis Fuel Assemblies

Table 3.2-1 provides the relevant dimensions required to model the Westinghouse 17X17 design Standard, and Westinghouse OFA fuel assemblies in the spent fuel pool environment. Based on the results of the scoping calculations, the Westinghouse Standard fuel assembly is determined to be the most reactive fuel assembly to represent the burned fuel assemblies in the Diablo Canyon spent fuel pool. The Westinghouse OFA fuel assembly is determined to be the most reactive fuel assembly to represent the fresh fuel assemblies. The results of the IFBA credit calculations, shown in Table 3.5-4 demonstrate that the OFA fresh fuel assemblies are more reactive than the Standard fresh fuel assemblies. Other fuel assembly designs like the LOPAR and Performance Plus are found to be less reactive in the spent fuel pool environment, than these design basis fuel assemblies.

The Standard fuel assembly is shown in Figure 3.2-1 and the OFA fuel assembly is shown in Figure 3.2-2. Note that the OFA fuel assembly is also shown with the 16-rod IFBA arrangement. The presence of IFBA is only credited for fresh fuel assemblies with an enrichment greater than 4.9 w/o ^{235}U. The presence of "Zirlo" cladding in the fuel pins does not lead to any significant change in the assembly reactivity compared to the "Zirc-4" clad fuel assemblies. In fact, the presence of Niobium in the "Zirlo" cladding is only expected to increase the absorption in the clad material. Therefore, the fuel assemblies will be conservatively modeled using Zirc-4 cladding.

Table 3.2-1
Input Parameters for the Westinghouse 17X17 Fuel Assembly Models

Description	OFA	Standard
Rods/Assy.	264	264
Guide Tubes/ Assy.	24	24
Instrument Tubes / Assy.	1	1
Rod Pitch, in.	0.496	0.496
Pellet OD, in.	0.3088	0. 3225
Pellet Density[1], % TD	95 ± 2.5	95 ± 2.5
Max. Enrichment, wt%	5.0 ± 0.05	5.0 ± 0.05
Active Fuel Length, in.	144	144
Clad OD, in.	0.360	0.374
Clad Thickness, in.	0.0225	0.0225
Clad Material [3]	Zirc.-4	Zirc.-4
Guide Tube OD, in.	0.474	0.482
Guide Tube Thickness, in.	0.016	0.016
Guide Tube Mat.	Zirc.-4	Zirc.-4
Inst. Tube OD, in.	0.474	0.482
Inst. Tube Thickness, in.	0.016	0.016
Inst. Tube Mat.	Zirc.-4	Zirc.-4
IFBA Design / Material	16 Rods / ZrB_2	16 Rods / ZrB_2
IFBA loading (1.0X)	1.50 g ^{10}B / Inch	1.57 g ^{10}B / Inch
IFBA Length, in.[2]	120	120

1 Includes no allowance for dishing
2 For the IFBA rods, there is a shim cutback of 12 inches at the fuel top and bottom
3 Conservatively applies for other clad types like Zirc-2 and Zirlo

Figure 3.2-1
Geometrical View of Westinghouse 17x17 Standard Fuel Assembly

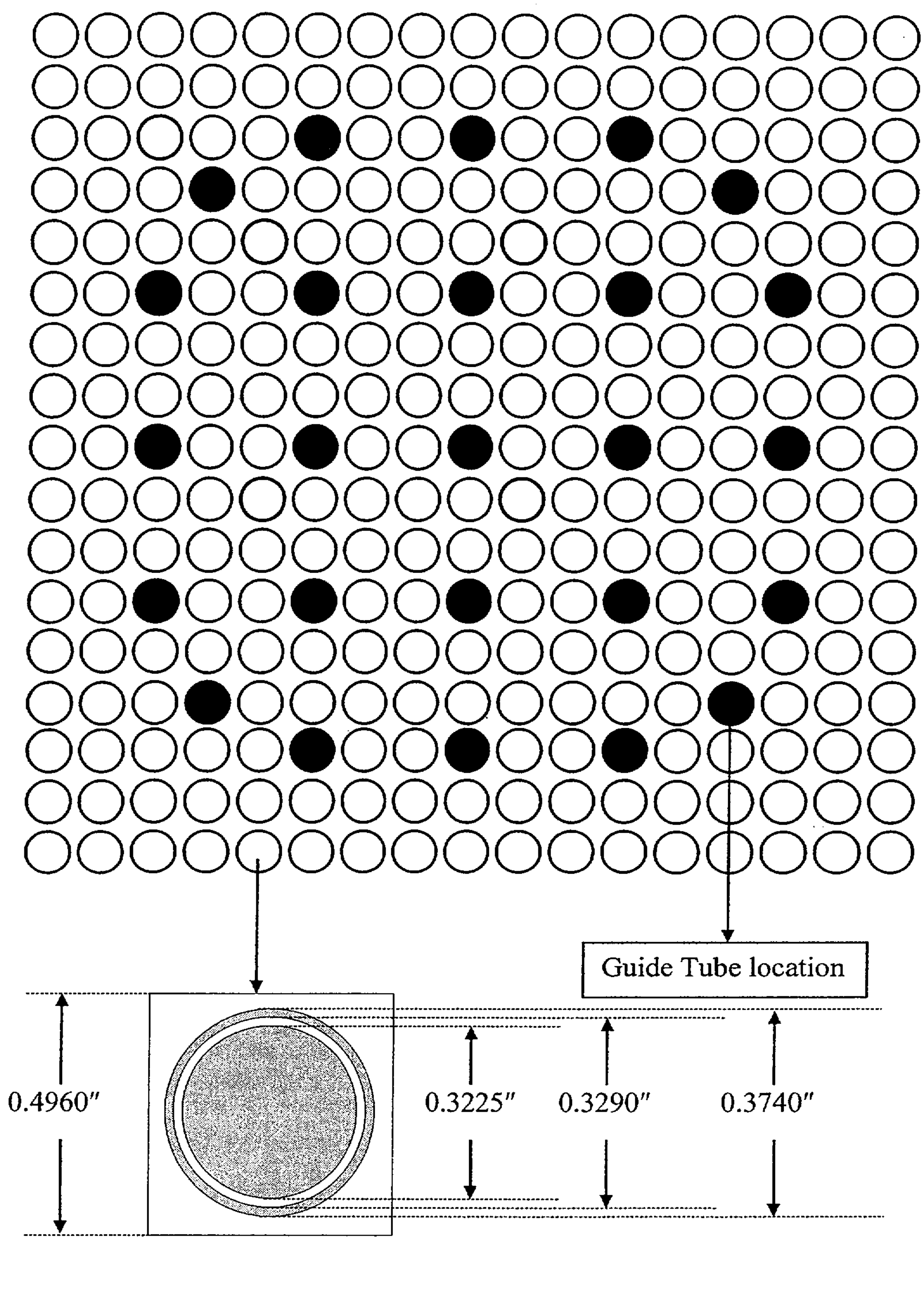
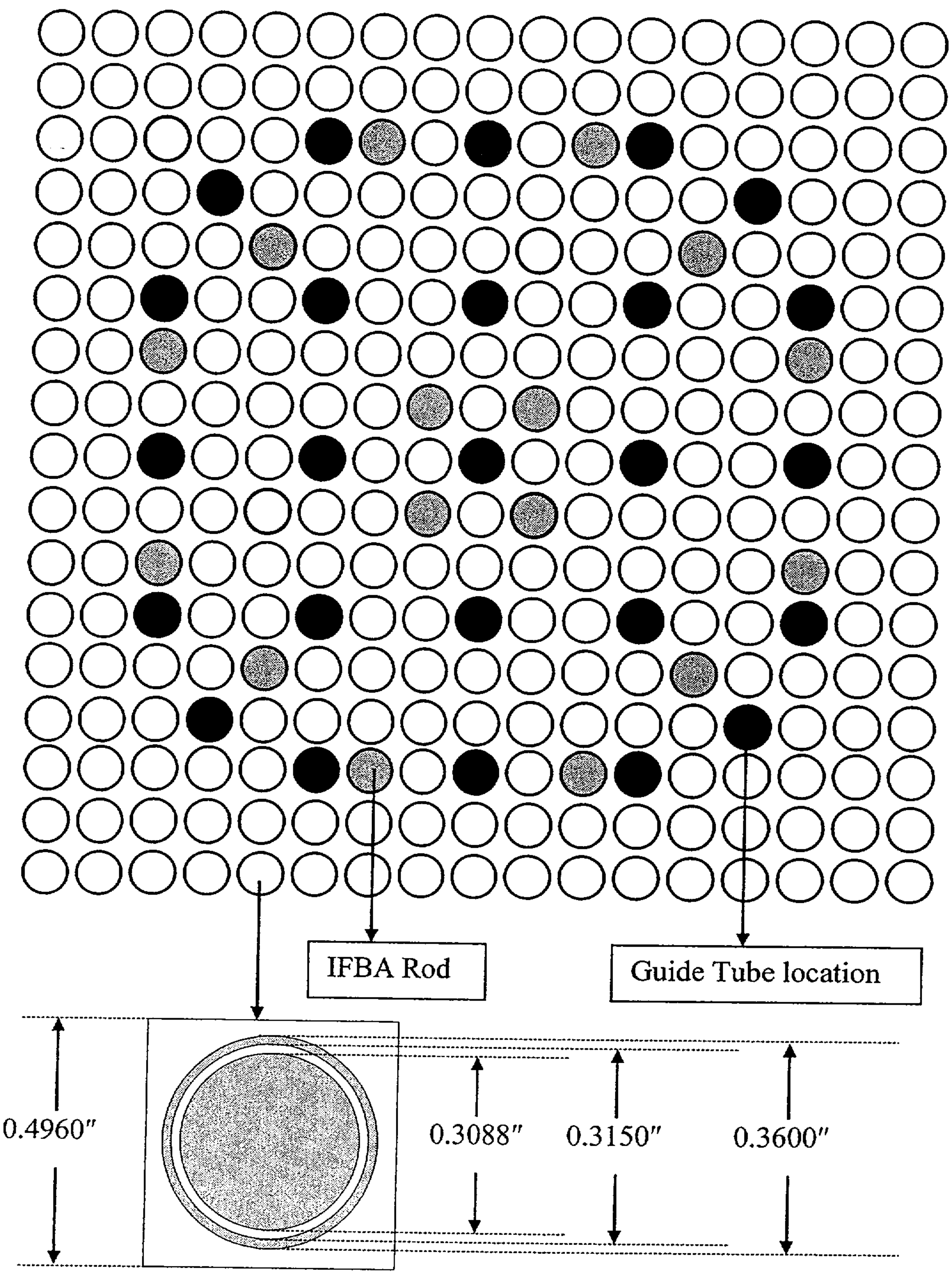


Figure 3.2-2
Geometrical View of Westinghouse 17x17 OFA Fuel Assembly



3.3 Modeling of Axial Burnup Distributions

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

For any given spent fuel assembly, the fuel burnup is a continuous function of axial position. However, from a calculational point of view, this function can be discretized in such a manner that the axial "end-effect" is adequately captured. It is often common practice to divide the fuel assembly into several axial zones with each zone assumed to be uniform in burnup. Moreover, it is required that the size of the top and bottom axial zones be small (typically 6 to 8 inches) so as to capture the steep burnup gradient with axial position while that of the central zone may be larger. In spent fuel pool calculations, a four-zone axial model is found to be adequate (Reference 15, PE&D Report) to represent the spent fuel assembly. Such a four-zone model would have three zones with fine mesh spacing (three at the top of the fuel assembly) and the fourth zone is the remainder of the fuel assembly. Figure 3.3-1 provides a pictorial view of the axial zones employed in the four zone axial model.

The individual power fractions of each zone are so modeled that they give the same volume averaged burnup when compared to a uniform burnup model. This model is validated due to the fact that the relative contribution of the bottom zones of the fuel assembly to the K_{eff} is negligible. A benchmarking comparison of the assembly K_{eff}, in the spent fuel pool environment, of the four-zone model and a multi-zone (seven-zone) model, performed in Reference 15 provides adequate validation for the use of such a simplified model for 3D representation of burned fuel assemblies.

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

DIT was used to generate the isotopic concentrations for each segment of the axial profile. Table 3.3-1 lists the fuel and moderator temperatures employed in the spectral calculations for each node in each of the four-zone axial burnup model. These values are based on mid-cycle temperature profiles for a typical 1000 MWe PWR. The fuel temperatures for each axial zone are calculated based on a representative fuel temperature correlation while the moderator temperatures are based on a linear relationship with axial position. These node dependent moderator and fuel temperature data and power profile data were employed in DIT to deplete the fuel to the desired burnup for each initial enrichment and each axial zone. The values of assembly average burnups versus feed enrichment for which burned fuel assemblies were simulated are tabulated in Table 3.3-2.

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

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

Appendix B contains a listing of the isotopic number densities employed in the KENO calculations. The format of the listing is compatible with the KENO input description and can directly be used as part of KENO input for material specification. The isotopic number densities are listed for the combination of initial enrichment and burnup listed in Table 3.3-2. The listing is for the Westinghouse 17X17 Standard fuel assembly design.

Appendix C contains a listing of the ^{10}B number densities determined by matching the DIT K_{eff} and KENO calculated K_{eff} values. There are a total of 16 tables provided in this Appendix. The ^{10}B number density, the DIT calculated K_{eff} and the KENO calculated K_{eff}, for the four zone axial model (and the average fuel assembly model) are listed in each table. The first five tables contain these values for 3.0 w/o, the next five tables contain the data for 4.0 w/o and the next six tables contain data for 5.0 w/o.

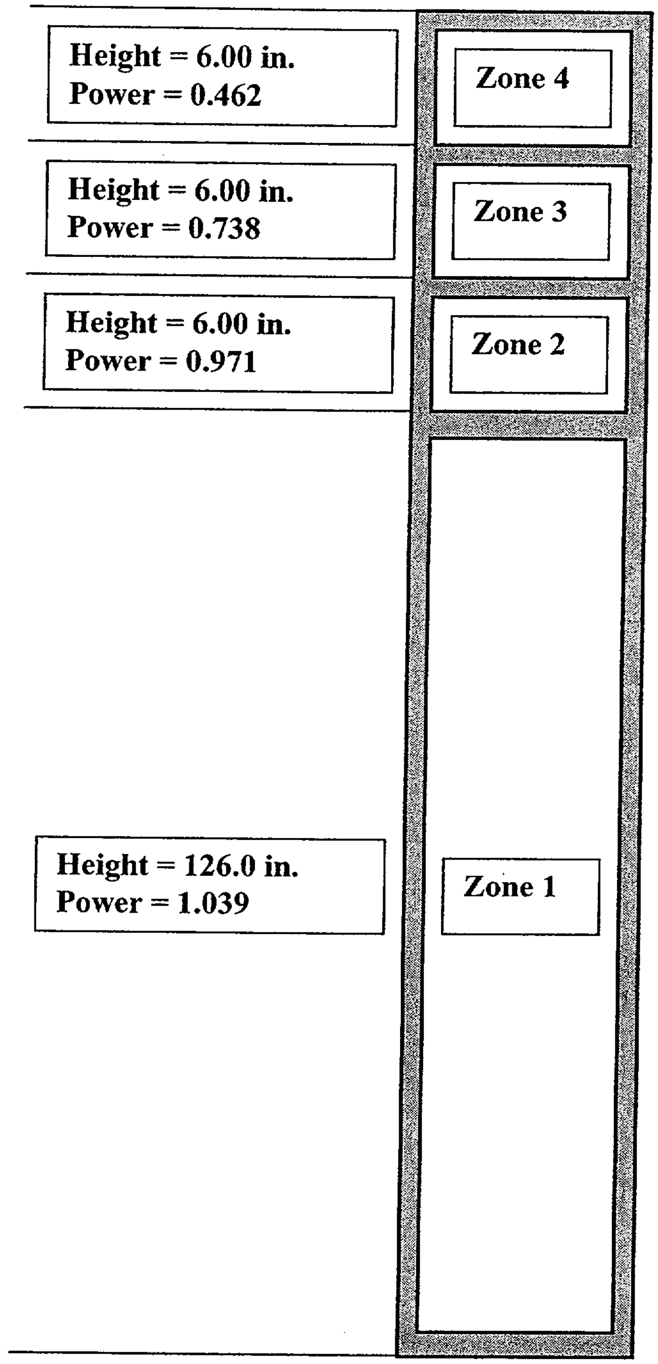
Table 3.3-1
Relative Power, and Fuel and Moderator Temperatures for the Four Zone Model

Zone No.	Height (in.)	Relative Power	Fuel Temperature (°F)	Moderator Temperature (°F)
Average	144.0	1.000	944.12	579.95
1	126.0	1.039	860.55	575.59
2	6.00	0.971	1036.77	605.12
3	6.00	0.738	1226.51	608.99
4	6.00	0.462	1151.87	612.86

Table 3.3-2
Burnup and Initial Enrichments combinations used to determine the Isotopic Number Densities

3 wt%	4 wt%	5 wt%
MWD/MTU	MWD/MTU	MWD/MTU
0	0	0
5,000	15,000	15,000
15,000	25,000	25,000
25,000	35,000	35,000
35,000	45,000	45,000
	55,000	55,000
		65,000

Figure 3.3-1
Sketch of Axial Zones Employed in Fuel Assembly



3.4 Tolerance/Uncertainty Evaluation for Region 2 Cells

Previous sections described the region 2 storage racks within the spent fuel storage pool and the KENO model employed to represent infinite arrays of individual cell types and arrays in these storage racks. In addition, the method of modeling the axial profiles of fuel assembly burnup, moderator temperature, and fuel temperature were discussed in so far as their use in reactivity equivalencing fuel assemblies of different burnup histories are concerned.

Using the above input, analytic models were developed to perform the quantitative evaluations necessary to demonstrate that the effective multiplication factor for the spent fuel pool is less than unity with zero boron present in the pool. Applicable biases to be factored into this evaluation are: (1) the methodology bias deduced from the validation analyses of pertinent critical experiments, and (2) any reactivity bias, relative to the reference analysis conditions, associated with operation of the spent fuel pool over a temperature range of 50 to 212 °F.

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

a) the 95/95 confidence level methods variance,

b) the 95/95 confidence level calculational uncertainty,

c) fuel rod manufacturing tolerance,

d) storage rack fabrication tolerances,

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

Items a) and b) are based on the calculational methods validation analyses. For Item c), the fuel rod manufacturing tolerance for the reference design fuel assembly is assumed to consist of four components: an increase in fuel enrichment from 4.9 to 4.95 wt% ^{235}U and an increase in pellet density from 95 to 97.5 %TD; the individual contributions of each change are combined by taking the square root of the sum of the squares of each component. The pellet density is based on a Uranium loading of 0.134725 g ^{235}U / cm at an enrichment of 4.9 w/o ^{235}U. There is no allowance for dishing and chamfer and therefore the pellet density conservatively represents the stack density of the UO_2 pellets in the fuel rod.

For Item d), the following uncertainty components are evaluated. For the Region 2 racks, the stainless steel box ID is decreased from 8.850 to 8.835 inches, the Stainless Steel box thickness is decreased from 0.09 inches to 0.08 inches and the flux trap water gap is decreased from 1.900 to 1.850 inches.

In the case of the tolerance due to positioning of the fuel assembly in the storage cells, all nominal calculations are carried out with fuel assemblies centered in the storage cells. Table 3.4-1 provides a summary of the KENO cases used in the calculation of biases and uncertainties for the zero soluble boron condition in the infinite array models for each rack type. Table 3.4-2 provides a summary of the biases and uncertainties calculated for the zero soluble boron condition in the infinite array models for the region 2 cells.

Table 3.4-1
Keno Calculated K_{eff} values for the Various Physical Tolerance Cases

Case	Description	K_{eff} from KENO
Region 2 Cells		
1	Nominal Case [1]	0.97606 ± 0.00061
2	Increase in U-235 Enrichment	0.97553 ± 0.00061
3	Increase in Stack Density	0.97957 ± 0.00059
Statistical Sum of F/A Manufacturing Tolerances		0.00476
4	Decrease in SS Box ID	0.97613 ± 0.00060
5	Decrease in SS Box thickness	0.97837 ± 0.00061
6	Decrease in Water Gap	0.97995 ± 0.00060
Statistical Sum of Rack Fabrication Tolerances		0.00633

1 The nominal case is based on a one-out-of four arrangement (one fresh checkerboarded by three burned) of fuel assemblies in a repeating 2X2 array. The fresh fuel assembly is based on the Westinghouse 17X17 OFA design with an enrichment of 4.9 w/o ^{235}U and the burned fuel assembly is based on Westinghouse 17X17 Standard design with an initial enrichment of 5.0 w/o ^{235}U with a burnup of 55,000 MWD/MTU

Table 3.4-2
Summary of Biases and Uncertainties for the Zero Soluble Boron Condition

Description	Region 2
Methodology Bias and Calculational Penalties	
KENO Bias (ΔK_{bias})	0.00259
Penalties	
Pool Temperature Penalty (50 °F – 212 °F)	0.00500
Assembly Off-Center Placement Penalty	0.00000
Sum of Penalties (ΔK_{pen})	0.00500
Total = $\Delta K_{bias} + \Delta K_{pen}$	**0.00759**
Tolerance and Statistical Uncertainties	
Fuel Assembly Manufacturing Tolerance	0.00476
Rack Fabrication Tolerance	0.00633
KENO Methodology Uncertainty	0.00684
Total (Statistically Combined)	**0.01046**
Total Adjustment to K_{eff}	**0.01805**

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

3.5.1 Region 2 Cells

As described in Section 3.3, the spent fuel storage rack analysis model employs a four axial zone representation of spent/burned fuel assemblies in the storage racks in an evaluation of burnup credit for a given spent fuel assembly. For the region 2 storage cells, K_{eff} was evaluated for an infinite array of storage cells over a range of feed enrichment values up to 5 wt% ^{235}U and fuel assembly average burnups up to 55 GWD/T. The storage arrangement that was analyzed, based on Figure 3.1-1, consisted of a one-out-of-four fresh fuel arrangement (one fresh fuel assembly with three burned fuel assemblies in a 2X2 array). The fresh fuel assembly utilized for this purpose has an enrichment of 4.90 w/o ^{235}U with no credit for any type of burnable absorber present in the fuel. These data are then employed to determine the burnup limits versus initial feed enrichment (for the burned fuel assemblies) for a given target K_{eff} value at zero soluble boron. The target value of K_{eff} is selected to be less than unity by an amount sufficient to cover the expected magnitude of analytical biases and uncertainties in these analyses, i.e., approximately 0.02 ΔK_{eff}.

Table 3.5-1 lists the KENO K_{eff} values computed with the four axial zone model for the Region 2 storage cells versus feed enrichment, and fuel assembly average burnup. The burnup limits determined based on a linear interpolation of these results are provided in Table 3.5-2. The first entry in Table 3.5-2 lists the initial enrichment required for fuel assemblies at zero burnup. The required assembly burnups as a function of initial enrichment is fitted to a third order polynomial. This polynomial shown in Table 3.5-3 will be used to determine the burnup for all burned fuel assemblies as a function of initial enrichment.

3.5.2 Credit for IFBA in Fresh Fuel Assemblies

The burnup versus enrichment curve determined in Section 3.5.1 are valid for the one-out-of-four storage of fresh fuel assemblies in Region 2 cells for a fresh fuel enrichment of 4.9 w/o ^{235}U. Above this enrichment limit, the presence of IFBA in the fresh fuel assemblies is credited. The Westinghouse 16-rod IFBA design containing Zirconium Diboride as the burnable absorber is utilized for this purpose. (Note that the 16 IFBA rod configuration contains the minimum number of IFBA rods offered by Westinghouse.) The objective of this analysis is to demonstrate that a fresh fuel assembly with an initial enrichment of 5.0 w/o ^{235}U containing 16 IFBA rods in a one-out-of-four checkerboard configuration in region 2 cells is less reactive than a fresh fuel assembly with 4.9 w/o ^{235}U

with no IFBAs, in the same configuration. The 16 IFBA rod configuration is shown in Figure 3.2-2. The burned fuel assemblies have an initial enrichment of 5.0 w/o ^{235}U and a burnup of 55,000 MWD/MTU. Five cases are considered in this analysis – two cases involving OFA design fresh fuel assemblies with 1.0X and 1.5X IFBA loading and two cases involving the Standard design fresh fuel assemblies with 1.0X and 1.5X IFBA loading. The fifth case represents the "IFBA uncertainty" case in which the ^{10}B loading in the IFBA is reduced by 25% for the most reactive of the first four cases (the OFA 1.0X case). The results of the IFBA credit analyses are shown in Table 3.5-4.

3.5.3 Entire Spent Fuel Pool

A KENO model for the entire spent fuel pool at the zero boron condition was constructed for this analysis. Only the Region 2 cells were used in the entire model. The storage arrangement consisted of a repeating 2X2 array of cells with one fresh fuel assembly at 4.9 w/o ^{235}U and three burned fuel assemblies at 5.0 w/o ^{235}U with a burnup of 55,000 MWD/MTU. The calculated KENO multiplication factor for the spent fuel pool using this model is 0.97361 ± 0.00060, without biases and uncertainties, with no soluble boron. This result along with the result of the infinite array calculation is shown in Table 3.5-5. This is very interesting because it provides an idea of the amount of conservatism that is built into the infinite array results.

Table 3.5-1
KENO K_{eff} Values versus Feed Enrichment, and Assembly Average Burnup for the Region 2 Storage Cells with No Soluble Boron

No.	Burned Fuel Assembly Description	K_{eff} value from KENO
1	Enrichment = 1.28 w/o ^{235}U Burnup = 0 MWD/MTU	0.97888 ± 0.00057
2	Enrichment = 3.00 w/o ^{235}U Burnup = 25,000 MWD/MTU	0.98773 ± 0.00062
3	Enrichment = 3.00 w/o ^{235}U Burnup = 35,000 MWD/MTU	0.97107 ± 0.00064
4	Enrichment = 4.00 w/o ^{235}U Burnup = 35,000 MWD/MTU	0.99001 ± 0.00061
5	Enrichment = 4.00 w/o ^{235}U Burnup = 45,000 MWD/MTU	0.97220 ± 0.00062
6	Enrichment = 5.00 w/o ^{235}U Burnup = 45,000 MWD/MTU	0.99205 ± 0.00059
7	Enrichment = 5.00 w/o ^{235}U Burnup = 55,000 MWD/MTU	0.97606 ± 0.00061

Table 3.5-2
Burnup (MWD/MTU) versus Initial Enrichment for Infinite Array of Region 2 Cells

Enrichment	Burnup (MWD/MTU)
1.28	0
3.000	29,650
4.000	40,650
5.000	52,550

Table 3.5-3
Required Fuel Assembly Burnup (MWD/MTU) versus Initial Enrichment (w/o) for Region 2 Curve expressed as a third-order polynomial

No.	Polynomial Describing the Curve
1	BU = [737.51 $(w/o)^3$ – 8400.1 $(w/o)^2$ + 42513 (w/o) - 42200]

Table 3.5-4
KENO K_{eff} Values for Region 2 Storage Cells Credit for IFBA in the Fresh Fuel Assemblies

No.	Fresh Fuel Assembly Description	K_{eff} value from KENO
1	Enrichment = 4.90 w/o ^{235}U, OFA No IFBAs	0.97606 ± 0.00061
2	Enrichment = 5.00 w/o ^{235}U, OFA 16 IFBAs, 1.0X loading	0.96674 ± 0.00060
3	Enrichment = 5.00 w/o ^{235}U, Standard 16 IFBAs, 1.0X loading	0.96101 ± 0.00060
4	Enrichment = 5.00 w/o ^{235}U, OFA 16 IFBAs, 1.5X loading	0.96133 ± 0.00062
5	Enrichment = 5.00 w/o ^{235}U, Standard 16 IFBAs, 1.5X loading	0.95655 ± 0.00058
6	Same as Case 2 except that the IFBA loading is reduced by 25%	0.96981 ± 0.00060

Table 3.5-5
KENO Results for the Entire Spent Fuel Pool (No Soluble Boron)

Case	Description	K_{eff} from KENO
1	One-out-of-four arrangement in the KENO infinite cell model	0.97606 ± 0.00061
2	One-out-of-four arrangement in the KENO full pool model	0.97361 ± 0.00060

3.6 Soluble Boron K_{eff} Calculational Results

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

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

where:

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

$SBC_{95/95}$ = soluble boron requirement for 95/95 $K_{eff} \leq 0.95$ (ppm),

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

SBC_{PA} = soluble boron required for $K_{eff} \leq 0.95$ under accident conditions (ppm).

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

3.6.1 Soluble Boron Determination to Maintain K_{eff} Less Than 0.95

Table 3.6-1 contains KENO calculated multiplication factors for the entire Diablo Canyon spent fuel pool at 0, 100, 200 and 300 ppm of soluble boron. Table 3.6-1 contains this data based on a repeating 2X2 array of one of fresh fuel assembly checkerboarded with three burned fuel assemblies. The description of the fuel assemblies utilized in the KENO calculations are also listed in Table 3.6-1.

The last column in Table 3.6-1 is labeled "Delta K_{eff}." Delta K_{eff} is equal to the K_{eff} (plus the one sigma value) of the case with soluble boron minus K_{eff} (less the one sigma value) of the unborated spent fuel pool. The reference K_{eff} value of the unborated spent fuel pool (and the associated one sigma value) is given in this table at zero ppm.

The soluble boron worth in Table 3.6-1 will be employed to determine the soluble boron concentration necessary to maintain K_{eff} less than 0.95 (including biases and uncertainties) and to compensate for the reactivity equivalencing methodologies which could increase the multiplication factor of the spent fuel pool.

The amount of soluble boron required to maintain K_{eff} less than 0.95 including biases and uncertainties is determined based on the results from Table 3.6-1. The soluble boron concentration (ppm) required to reduce the K_{eff} of the entire spent fuel pool by 0.050 delta K_{eff} units is conservatively determined by linear interpolation of the data in Table 3.6-1 (between 200 and 300 ppm) to be equal to 233.2 ppm.

3.6.2 Soluble Boron Determination for Reactivity Equivalencing Methods

The soluble boron credit (ppm) required for reactivity equivalencing methodologies was determined by converting the uncertainty in fuel assembly reactivity and the uncertainty in absolute fuel assembly burnup values to a soluble boron concentration (ppm) necessary to compensate for these two uncertainties. The first term, uncertainty in fuel assembly reactivity, is calculated by employing a depletion reactivity uncertainty equal to 0.005 delta K_{eff} units per 30,000 MWD/T of assembly burnup (obtained from Reference 3) and multiplying by the maximum amount of assembly burnup credited in a Region analysis. The highest assembly burnup credited is 52,550 MWD/T; this value is employed for Region 2 cells at an initial fuel assembly enrichment equal to 5.0 w/o ^{235}U. Therefore, the uncertainty in fuel assembly reactivity is equal to 0.00876 delta K_{eff} units.

The uncertainty in absolute fuel assembly burnup values is conservatively calculated as 5% of the maximum fuel assembly burnup credited in a Region analysis. The maximum fuel assembly burnup credited in this analysis is 52,550 MWD/T. Such a fuel assembly is used in Region 2 cells at an initial fuel enrichment of 5.0 wt% ^{235}U. The uncertainty in the burnup value is determined to be 2,628 MWD/T. The reactivity associated with a delta-burnup of 2,628 MWD/T at 55,000 MWD/T for a Region 2 cell is calculated to be 0.00389 delta K_{eff} units.

The total of these two reactivity effects is equal to 0.01265 (0.00876+0.00389) delta K_{eff} units. By linear interpolation of the data in Table 3.6-1 the soluble boron concentration (ppm) necessary to compensate for this reactivity is equal to 51.6 ppm.

3.6.3 Soluble Boron Determination to Mitigate Accidents

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

Several fuel mishandling events were simulated with KENO to assess the possible increase in the K_{eff} value of the Diablo Canyon spent fuel pool. The fuel mishandling events all assumed that a fresh 4.9 w/o ^{235}U assembly (*Accident* F/A) with no IFBAs was mislocated into any cell of the spent fuel pool intended for less reactive fuel assemblies.

A survey of the various fuel mishandling events indicated that the most disruptive accident would be the misplacement of a fresh fuel assembly with 4.9 w/o ^{235}U outside the module "K", such that it is adjacent to a fresh fuel assembly with 4.9 w/o ^{235}U within module "K". This would then mean that there would be two fresh fuel assemblies adjacent to each other in the spent fuel pool with just the steel box separating them. All the other mishandling events (misplacement of fresh fuel in place of burned assembly within a rack module, accidental drop of a fresh fuel assembly outside the rack module, T-Bone drop etc.,) were also simulated using the full spent fuel pool model. Table 3.6-2 shows the various mishandling accidents (including the location) simulated and the associated KENO calculated K_{eff} value. The description of the fuel assemblies occupying the cells is the same as that shown in Table 3.6-1. Table 3.6-3 shows the worst fuel mishandling accident credited in this analysis with soluble boron credit. In order to determine the amount of soluble boron required to mitigate this accident, the same scenario is simulated, this time at a soluble boron concentration of 400 and 600 ppm.

The last column in Table 3.6-2 is labeled "Delta K_{eff}". Delta K_{eff} is equal to the K_{eff} (plus the one sigma value) of the fuel mishandling scenario minus K_{eff} (less the one sigma value) of the nominal spent fuel pool. The reference K_{eff} value of the spent fuel pool (and the associated one sigma value) provided in Table 3.6-2 was obtained from Table 3.5-5. The reference K_{eff} value for the spent fuel pool for the scenario described in Table 3.6-3 is based on the KENO simulation provided in the Table 3.5-5.

The data in Table 3.6-2 indicates that the highest worth of a fuel mishandling event is based on a accident just outside module "K" with a worth of 0.09760 delta K_{eff}. Simulation of the same accident at a soluble boron concentration of 600 ppm indicates that it is more than sufficient to mitigate the accident.

The soluble boron concentration (ppm) necessary to compensate for this reactivity insertion is conservatively calculated, based on linear interpolation, from the results given on Table 3.6-3, to be equal to 512.5 ppm.

As noted above, the misplaced fuel assembly was determined as being the most adverse postulated mishandling event. Other dropped assembly events such as, for example, the postulated T-bone fuel assembly accident configuration and the postulated deep drop type accidents, have a lower effect on the local K_{eff} of the spent fuel storage rack as indicated by this analyses. In the case of the postulated deep drop accident (which is not analyzed), the deflection of the base plate is limited by the height of the pedestals supporting the rack above the concrete floor and the structural design of the rack. A significant fraction of the base plate deflection distance would be taken up by the fuel assembly structure below the active fuel columns. Thus, one would again conclude the misplaced fuel assembly accident overshadows the reactivity insertion resulting from the postulated deep drop event.

3.6.4 Summary of Soluble Boron Requirements

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

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

where:

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

$SBC_{95/95}$ = soluble boron requirement for 95/95 $K_{eff} \leq 0.95$ (ppm),

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

SBC_{PA} = soluble boron required for $K_{eff} \leq 0.95$ under accident conditions (ppm).

The magnitude of the above components is:

$SBC_{95/95}$ = 233.2 ppm

SBC_{RE} = 51.6 ppm

SBC_{PA} = 512.5 ppm

SBC_{TOTAL} = 798 ppm

Therefore, a total of 798 ppm of soluble boron is required to maintain K_{eff} less than 0.95 (including all biases and uncertainties) assuming the most limiting single fuel mishandling accident. Note that this soluble boron concentration assumes an atomic fraction for ^{10}B equal to 0.199. For a ^{10}B isotopic fraction equal to 0.197 (as recently measured at PG&E), the total soluble boron concentration, required to maintain the same concentration of ^{10}B atoms, would be equal to 806 ppm.

Table 3.6-1
K_{eff} as a Function of Soluble Boron Level

Cell Type	Description	
Region 2	Fresh Fuel Assembly = 4.90 w/o ^{235}U Burned Fuel Assembly = 5.00 w/o ^{235}U, 55 GWD/MTU	
Soluble Boron Concentration, ppm	**K_{eff}**	**Delta K_{eff}**
0	0.97361 ± 0.00060	0.0
100	0.94785 ± 0.00062	-0.02454
200	0.92898 ± 0.00060	-0.04343
300	0.90915 ± 0.00059	-0.06327

Table 3.6-2
K_{eff} for Various Accident Events

(Reference K_{eff} =0.97361 ± 0.00060)

Description of Accident	K_{eff}	Delta K_{eff}
Accident F/A in a Region 2 Cell across a Fresh fuel assembly such that three fresh fuel assemblies are diagonally lined up	0.99179 ± 0.00060	0.01938
Accident F/A in a Region 2 Cell adjacent to a Fresh fuel assembly such that three fresh fuel assemblies are lined up	1.02409 ± 0.00069	0.05177
Accident F/A in T-Bone Drop event	1.02426 ± 0.00066	0.05191
Accident F/A outside module “K”, at the corner of modules “K” and “D3”	1.06179 ± 0.00076	0.08954
Accident F/A outside module “K”, such that it is adjacent to a fresh assembly in module “K”	1.07015 ± 0.00076	0.09760

Table 3.6-3
K_{eff} and Soluble Boron Credit for the Limiting Fuel Mishandling Event

Cell Type	Description
Region 2	Fresh Fuel Assembly = 5.00 w/o ^{235}U Burned Fuel Assembly = 5.00 w/o ^{235}U, 55 GWD/MTU

Description	K_{eff}
Reference Value of K_{eff}	0.97361 ± 0.00060
Accident F/A outside module "K", such that it is adjacent to a fresh assembly in module "K"	1.07015 ± 0.00076
Simulation of the same accident at a soluble boron concentration of 400 ppm	0.98828 ± 0.00077
Simulation of the same accident at a soluble boron concentration of 600 ppm	0.95701 ± 0.00081

4.0 Summary of Results

The following sections contain the criticality analysis results for the Diablo Canyon Units 1 & 2 spent fuel pools.

4.1 Burnup versus Enrichment Storage Curves

The following is a description of the storage curves:

- Table 4.1-1 contains the burnups required to store burned fuel assemblies for various initial enrichments
- Table 4.1-2 contains a third order polynomial which describes the burnup required to store burned fuel assemblies as a function of initial enrichment
- Figure 4.1-1 shows the burnup versus enrichment storage curves.

4.2 Total Soluble Boron Requirement

The total soluble boron (sum of all the three components) required to make the K_{eff} (including all biases and uncertainties, without the adjustment for ^{10}B) less than or equal to 0.95 is determined to be 798 ppm. The soluble boron concentration with the adjustment for the ^{10}B atomic fraction of 0.197 is determined to be 806 ppm.

4.3 Allowable Storage Configurations

Based on the results of the criticality calculations, two storage configurations are allowed in the spent fuel pool. The first one is based on the two by two arrangement (one fresh fuel assembly surrounded by three burned fuel assemblies) and the second one is based on a checkerboard of fresh and empty storage locations. (An empty location is defined as a location without a fuel assembly. An empty location may contain non-fissile components for storage purposes.) The latter storage configuration is possible due to the assertion that this configuration is less reactive compared to the earlier configuration. The KENO result for the infinite array based on this configuration is 0.93412 ± 0.00070. This result is documented in the file "burcurve.tar" as "ofacheck.out."

The allowable storage configurations are illustrated in Figure 4.3-1.

Table 4.1-1
Required Fuel Assembly Burnup (MWD/MTU) versus Initial Enrichment

Enrichment	Burnup (MWD/MTU)
1.28	0
3.000	29,650
4.000	40,650
5.000	52,550

Table 4.1-2
Required Fuel Assembly Burnup (MWD/MTU) versus Initial Enrichment (w/o) expressed as a third-order polynomial

Polynomial Describing the Curve
BU = [696.28 $(w/o)^3$ - 7905.3 $(w/o)^2$ + 40575 (w/o) - 39727]

Figure 4.1-1 Enrichment versus Burnup Storage Curve

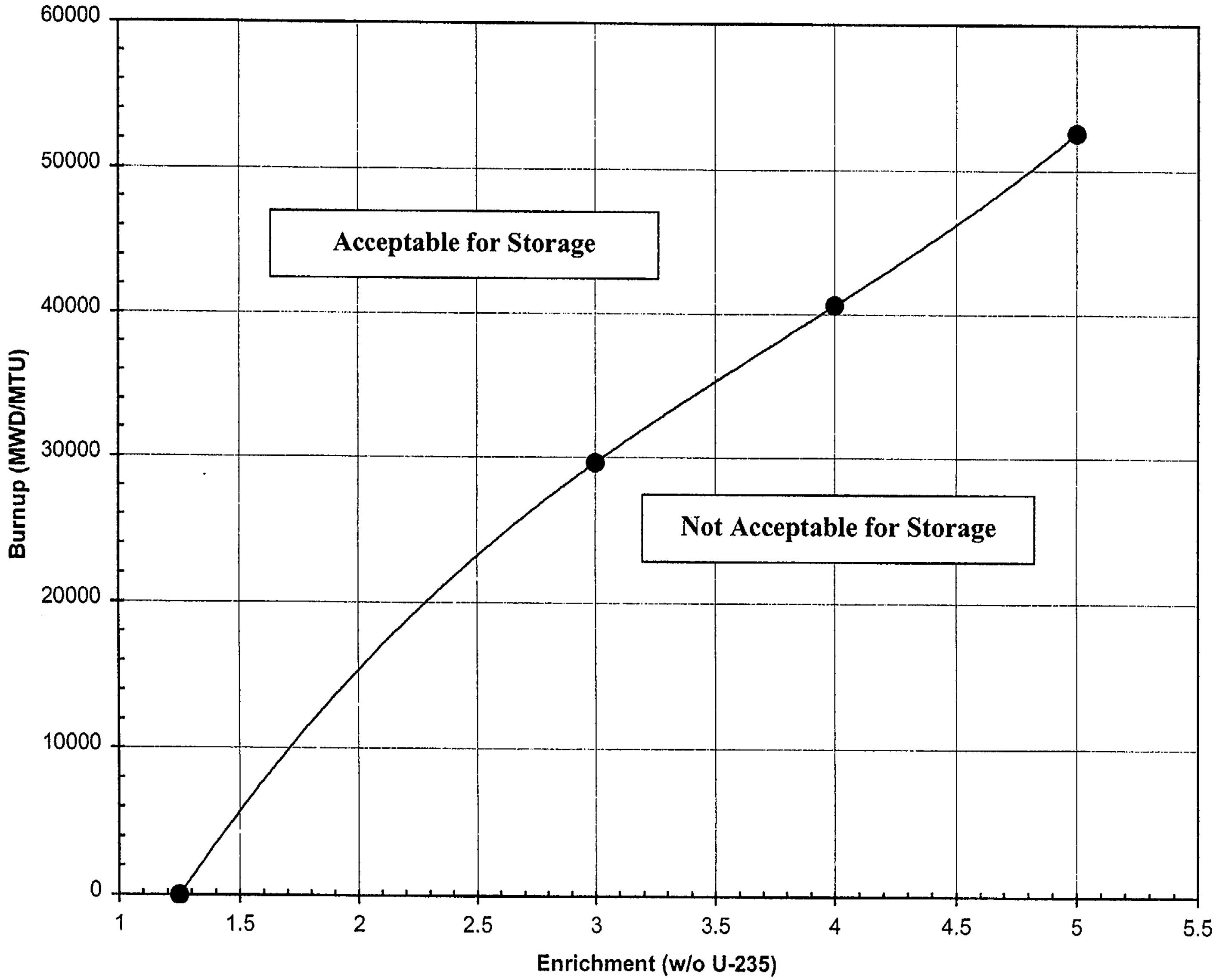
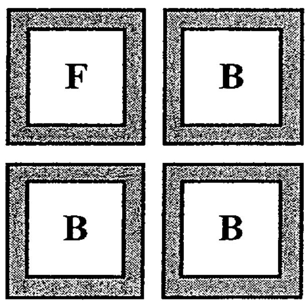


Figure 4.3-1
Allowable Storage Configurations

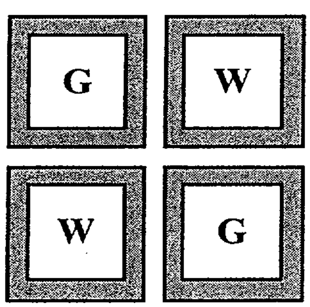


"**F**" represents

- Fresh fuel assembly with an initial enrichment less than or equal to 4.9 w/o ^{235}U with no IFBAs
- Fresh fuel assembly with an initial enrichment less than or equal to 5.0 w/o ^{235}U with 16 IFBAs

"**B**" represents

- Burned fuel assembly that satisfies the burnup-enrichment curve of Figure 4.1-1



"**G**" represents

- Fresh fuel assembly with an initial enrichment less than or equal to 5.0 w/o ^{235}U with no IFBAs

"**W**" represents

- Water cells – locations where fuel assemblies are not present, non-fissile components are permitted

5.0 CDROM and Computer Code Lists

5.1 Computer Codes used in the Design Analyses

Code Name	Computer Type	Revision Identification (i.e. Code Version)
SCALE-PC	PC	Ver4.3/rev00
DIT	HP-UX	Version 1.3/Mod 6

5.2 Computer Jobs on CDROM

KENO Results

Contents of the TAR file : bucurve.tar

File No.	Job Date	File Name	Remarks
1	09/01/2000	ofa493025.in	Input file for 3 w/o, 25 GWD/T, Fresh at 4.9 w/o
2	09/01/2000	ofa493035.in	Input file for 3 w/o, 35 GWD/T, Fresh at 4.9 w/o
3	09/01/2000	ofa494035.in	Input file for 4 w/o, 35 GWD/T, Fresh at 4.9 w/o
4	09/01/2000	ofa494045.in	Input file for 4 w/o, 45 GWD/T, Fresh at 4.9 w/o
5	09/01/2000	ofa495045.in	Input file for 5 w/o, 45 GWD/T, Fresh at 4.9 w/o
6	09/01/2000	ofa495055.in	Input file for 5 w/o, 55 GWD/T, Fresh at 4.9 w/o
7	09/02/2000	ofa49fr.in	Input file for 1.28 w/o, 0 GWD/T, Fresh at 4.9 w/o
8	09/21/2000	ofacheck.in	Input file for checkerboard of fresh and empty cells
9	09/01/2000	ofa493025.out	Output file for 3 w/o, 25 GWD/T, Fresh at 4.9 w/o
10	09/01/2000	ofa493035.out	Output file for 3 w/o, 35 GWD/T, Fresh at 4.9 w/o
11	09/01/2000	ofa494035.out	Output file for 4 w/o, 35 GWD/T, Fresh at 4.9 w/o
12	09/01/2000	ofa494045.out	Output file for 4 w/o, 45 GWD/T, Fresh at 4.9 w/o
13	09/01/2000	ofa495045.out	Output file for 5 w/o, 45 GWD/T, Fresh at 4.9 w/o
14	09/01/2000	ofa495055.out	Output file for 5 w/o, 55 GWD/T, Fresh at 4.9 w/o
15	09/02/2000	ofa49fr.out	Output file for 1.28 w/o, 0 GWD/T, Fresh at 4.9 w/o
16	09/21/2000	ofacheck.out	Output file for checkerboard of fresh and empty cells

Contents of the TAR file : biases.tar

File No.	Job Date	File Name	Remarks
1	09/01/2000	ofaenr.in	Input file for fuel enrichment tolerance
2	09/01/2000	ofarho.in	Input file for pellet density tolerance
3	09/01/2000	biasbox.in	Input file for Box ID tolerance
4	09/01/2000	biasss.in	Input file for Box SS thickness tolerance
5	09/01/2000	biaswgap.in	Input file for Cell Water Gap tolerance
6	09/01/2000	ofaenr.out	Output file for fuel enrichment tolerance
7	09/01/2000	ofarho.out	Output file for pellet density tolerance
8	09/01/2000	biasbox.out	Output file for Box ID tolerance
9	09/01/2000	biasss.out	Output file for Box SS thickness tolerance
10	09/01/2000	biaswgap.out	Output file for Cell Water Gap tolerance

Contents of the TAR file : ifba.tar

File No.	Job Date	File Name	Remarks
1	09/11/2000	ifbaofa10x.in	Input file for OFA fuel with 1.0X IFBA
2	09/11/2000	ifbastd10x.in	Input file for Standard fuel with 1.0X IFBA
3	09/11/2000	ifbaofamin.in	Input file for OFA fuel with reduced IFBA (-25%)
4	09/12/2000	ifbaofa15x.in	Input file for OFA fuel with 1.5X IFBA
5	09/12/2000	ifbastd15x.in	Input file for Standard fuel with 1.5X IFBA
6	09/11/2000	ifbaofa10x.out	Output file for OFA fuel with 1.0X IFBA
7	09/11/2000	ifbastd10x.out	Output file for Standard fuel with 1.0X IFBA
8	09/11/2000	ifbaofamin.out	Output for OFA fuel with reduced IFBA (-25%)
9	09/12/2000	ifbaofa15x.out	Output file for OFA fuel with 1.5X IFBA
10	09/12/2000	ifbastd15x.out	Output file for Standard fuel with 1.5X IFBA

Contents of the TAR file : pooljobs.tar

File No.	Job Date	File Name	Remarks
1	09/01/2000	ofapool.in	Input file for full pool model
2	09/01/2000	ofapoolopp.in	Input file for accident, fresh in diagonal location
3	09/01/2000	ofapooladj.in	Input file for accident, fresh in adjacent location
4	09/01/2000	ofapooltb.in	Input file for T-Bone drop accident
5	09/01/2000	ofapoolmt.in	Input file for accident, outside module corner
6	09/01/2000	ofapoolat.in	Input file for accident, outside module adjacent
7	09/01/2000	ofapool.out	Output file for full pool model
8	09/01/2000	ofapoolopp.out	Output file for accident, fresh in diagonal location
9	09/01/2000	ofapooladj.out	Output file for accident, fresh in adjacent location
10	09/01/2000	ofapooltb.out	Output file for T-Bone drop accident
11	09/01/2000	ofapoolmt.out	Output file for accident, outside module corner
12	09/01/2000	ofapoolat.out	Output file for accident, outside module adjacent

Contents of the TAR file : poolbn.tar

File No.	Job Date	File Name	Remarks
1	09/01/2000	ofapoolp1.in	Input file for full pool model, 100 ppm boron
2	09/01/2000	ofapoolp2.in	Input file for full pool model, 200 ppm boron
3	09/01/2000	ofapoolp3.in	Input file for full pool model, 300 ppm boron
4	09/02/2000	ofapoolat4.in	Input file, worst accident, 400 ppm boron
5	09/02/2000	ofapoolat6.in	Input file, worst accident, 600 ppm boron
6	09/01/2000	ofapoolp1.out	Output file for full pool model, 100 ppm boron
7	09/01/2000	ofapoolp2.out	Output file for full pool model, 200 ppm boron
8	09/01/2000	ofapoolp3.out	Output file for full pool model, 300 ppm boron
9	09/02/2000	ofapoolat4.out	Output file, worst accident, 400 ppm boron
10	09/02/2000	ofapoolat6.out	Output file, worst accident, 600 ppm boron

6.0 References

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2. Newmyer, W.D., "Westinghouse Spent Fuel Rack Criticality Analysis Methodology", WCAP-14416-NP-A, Rev 1 November 1996.

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

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

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

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

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

15. Narayanan P. A., "Isotopic Number Densities for Discharged Westinghouse 17X17 Fuel Assemblies," A-GEN-FE-0118, CE Nuclear Power LLC, Windsor CT, September 26 2000.

APPENDIX B. Isotopic Number Densities employed in KENO calculations

Isotopic Number Densities used in KENO for the Four-Zone Model (3D)

```
'  3.00 W/O,  5 GWD/T, DIT-KINF =  1.31841
U-235     1  0    .57552E-03                  END
U-236     1  0    .23280E-04                  END
U-238     1  0    .22438E-01                  END
PU-239    1  0    .52885E-04                  END
PU-240    1  0    .51616E-05                  END
PU-241    1  0    .13235E-05                  END
SM-149    1  0    .97202E-07                  END
O-16      1  0    .46453E-01                  END
B-10      1  0    .37100E-05                  END
'  3.00 W/O,  5 GWD/T, DIT-KINF =  1.32326
U-235     2  0    .58383E-03                  END
U-236     2  0    .22021E-04                  END
U-238     2  0    .22441E-01                  END
PU-239    2  0    .51927E-04                  END
PU-240    2  0    .48364E-05                  END
PU-241    2  0    .12087E-05                  END
SM-149    2  0    .96544E-07                  END
O-16      2  0    .46453E-01                  END
B-10      2  0    .35000E-05                  END
'  3.00 W/O,  5 GWD/T, DIT-KINF =  1.33901
U-235     3  0    .61091E-03                  END
U-236     3  0    .17176E-04                  END
U-238     3  0    .22459E-01                  END
PU-239    3  0    .42388E-04                  END
PU-240    3  0    .31468E-05                  END
PU-241    3  0    .61895E-06                  END
SM-149    3  0    .87059E-07                  END
O-16      3  0    .46453E-01                  END
B-10      3  0    .28500E-05                  END
'  3.00 W/O,  5 GWD/T, DIT-KINF =  1.35711
U-235     4  0    .64446E-03                  END
U-236     4  0    .11101E-04                  END
U-238     4  0    .22482E-01                  END
PU-239    4  0    .28525E-04                  END
PU-240    4  0    .14117E-05                  END
PU-241    4  0    .17866E-06                  END
SM-149    4  0    .76104E-07                  END
O-16      4  0    .46453E-01                  END
B-10      4  0    .20000E-05                  END
```

```
'  3.00 W/O, 15 GWD/T, DIT-KINF =  1.19975
U-235     1  0    .38057E-03              END
U-236     1  0    .56941E-04              END
U-238     1  0    .22267E-01              END
PU-239    1  0    .10201E-03              END
PU-240    1  0    .23955E-04              END
PU-241    1  0    .12267E-04              END
SM-149    1  0    .11725E-06              END
O-16      1  0    .46453E-01              END
B-10      1  0    .90500E-05              END
'  3.00 W/O, 15 GWD/T, DIT-KINF =  1.21411
U-235     2  0    .40009E-03              END
U-236     2  0    .54248E-04              END
U-238     2  0    .22276E-01              END
PU-239    2  0    .10270E-03              END
PU-240    2  0    .22644E-04              END
PU-241    2  0    .11607E-04              END
SM-149    2  0    .11746E-06              END
O-16      2  0    .46453E-01              END
B-10      2  0    .87750E-05              END
'  3.00 W/O, 15 GWD/T, DIT-KINF =  1.25273
U-235     3  0    .46012E-03              END
U-236     3  0    .44047E-04              END
U-238     3  0    .22335E-01              END
PU-239    3  0    .90682E-04              END
PU-240    3  0    .16212E-04              END
PU-241    3  0    .72798E-05              END
SM-149    3  0    .10327E-06              END
O-16      3  0    .46453E-01              END
B-10      3  0    .68750E-05              END
'  3.00 W/O, 15 GWD/T, DIT-KINF =  1.30042
U-235     4  0    .53994E-03              END
U-236     4  0    .29984E-04              END
U-238     4  0    .22406E-01              END
PU-239    4  0    .67765E-04              END
PU-240    4  0    .84960E-05              END
PU-241    4  0    .27938E-05              END
SM-149    4  0    .86002E-07              END
O-16      4  0    .46453E-01              END
B-10      4  0    .47500E-05              END
```

```
' 3.00 W/O, 25 GWD/T, DIT-KINF =  1.09817
U-235     1  0    .24266E-03              END
U-236     1  0    .78534E-04              END
U-238     1  0    .22080E-01              END
PU-239    1  0    .11913E-03              END
PU-240    1  0    .41508E-04              END
PU-241    1  0    .24354E-04              END
SM-149    1  0    .12443E-06              END
O-16      1  0    .46453E-01              END
B-10      1  0    .14200E-04              END
' 3.00 W/O, 25 GWD/T, DIT-KINF =  1.12239
U-235     2  0    .26786E-03              END
U-236     2  0    .75521E-04              END
U-238     2  0    .22098E-01              END
PU-239    2  0    .12274E-03              END
PU-240    2  0    .39637E-04              END
PU-241    2  0    .23729E-04              END
SM-149    2  0    .12600E-06              END
O-16      2  0    .46453E-01              END
B-10      2  0    .13700E-04              END
' 3.00 W/O, 25 GWD/T, DIT-KINF =  1.17925
U-235     3  0    .34348E-03              END
U-236     3  0    .63775E-04              END
U-238     3  0    .22203E-01              END
PU-239    3  0    .11464E-03              END
PU-240    3  0    .30041E-04              END
PU-241    3  0    .16822E-04              END
SM-149    3  0    .11126E-06              END
O-16      3  0    .46453E-01              END
B-10      3  0    .10800E-04              END
' 3.00 W/O, 25 GWD/T, DIT-KINF =  1.24913
U-235     4  0    .45193E-03              END
U-236     4  0    .45519E-04              END
U-238     4  0    .22327E-01              END
PU-239    4  0    .92517E-04              END
PU-240    4  0    .17190E-04              END
PU-241    4  0    .77730E-05              END
SM-149    4  0    .91718E-07              END
O-16      4  0    .46453E-01              END
B-10      4  0    .72250E-05              END
```

```
'  3.00 W/O, 35 GWD/T, DIT-KINF =  1.00883
U-235     1  0    .14721E-03              END
U-236     1  0    .90987E-04              END
U-238     1  0    .21876E-01              END
PU-239    1  0    .12362E-03              END
PU-240    1  0    .54877E-04              END
PU-241    1  0    .33006E-04              END
SM-149    1  0    .12580E-06              END
O-16      1  0    .46453E-01              END
B-10      1  0    .18700E-04              END
'  3.00 W/O, 35 GWD/T, DIT-KINF =  1.04191
U-235     2  0    .17336E-03              END
U-236     2  0    .88569E-04              END
U-238     2  0    .21906E-01              END
PU-239    2  0    .12986E-03              END
PU-240    2  0    .53136E-04              END
PU-241    2  0    .33073E-04              END
SM-149    2  0    .12856E-06              END
O-16      2  0    .46453E-01              END
B-10      2  0    .18150E-04              END
'  3.00 W/O, 35 GWD/T, DIT-KINF =  1.11345
U-235     3  0    .25244E-03              END
U-236     3  0    .77957E-04              END
U-238     3  0    .22063E-01              END
PU-239    3  0    .12649E-03              END
PU-240    3  0    .42355E-04              END
PU-241    3  0    .25704E-04              END
SM-149    3  0    .11449E-06              END
O-16      3  0    .46453E-01              END
B-10      3  0    .14545E-04              END
'  3.00 W/O, 35 GWD/T, DIT-KINF =  1.20252
U-235     4  0    .37670E-03              END
U-236     4  0    .58353E-04              END
U-238     4  0    .22245E-01              END
PU-239    4  0    .10853E-03              END
PU-240    4  0    .25977E-04              END
PU-241    4  0    .13633E-04              END
SM-149    4  0    .94835E-07              END
O-16      4  0    .46453E-01              END
B-10      4  0    .95000E-05              END
```

```
 '  3.00 W/O, 45 GWD/T, DIT-KINF =   .93488
U-235      1  0     .84688E-04                END
U-236      1  0     .96410E-04                END
U-238      1  0     .21656E-01                END
PU-239     1  0     .12345E-03                END
PU-240     1  0     .63842E-04                END
PU-241     1  0     .38093E-04                END
SM-149     1  0     .12499E-06                END
O-16       1  0     .46453E-01                END
B-10       1  0     .22750E-04                END
 '  3.00 W/O, 45 GWD/T, DIT-KINF =   .97341
U-235      2  0     .10812E-03                END
U-236      2  0     .95234E-04                END
U-238      2  0     .21702E-01                END
PU-239     2  0     .13153E-03                END
PU-240     2  0     .62829E-04                END
PU-241     2  0     .39169E-04                END
SM-149     2  0     .12854E-06                END
O-16       2  0     .46453E-01                END
B-10       2  0     .22350E-04                END
 '  3.00 W/O, 45 GWD/T, DIT-KINF =  1.05390
U-235      3  0     .18204E-03                END
U-236      3  0     .87605E-04                END
U-238      3  0     .21916E-01                END
PU-239     3  0     .13189E-03                END
PU-240     3  0     .52529E-04                END
PU-241     3  0     .32785E-04                END
SM-149     3  0     .11525E-06                END
O-16       3  0     .46453E-01                END
B-10       3  0     .18050E-04                END
 '  3.00 W/O, 45 GWD/T, DIT-KINF =  1.15900
U-235      4  0     .31220E-03                END
U-236      4  0     .68868E-04                END
U-238      4  0     .22160E-01                END
PU-239     4  0     .11888E-03                END
PU-240     4  0     .34300E-04                END
PU-241     4  0     .19454E-04                END
SM-149     4  0     .96250E-07                END
O-16       4  0     .46453E-01                END
B-10       4  0     .12050E-04                END
```

```
'  4.00 W/O, 15 GWD/T, DIT-KINF =  1.28118
U-235     1  0    .58438E-03                    END
U-236     1  0    .64951E-04                    END
U-238     1  0    .22064E-01                    END
PU-239    1  0    .10298E-03                    END
PU-240    1  0    .19451E-04                    END
PU-241    1  0    .98725E-05                    END
SM-149    1  0    .13917E-06                    END
O-16      1  0    .46453E-01                    END
B-10      1  0    .10000E-04                    END
'  4.00 W/O, 15 GWD/T, DIT-KINF =  1.29251
U-235     2  0    .60638E-03                    END
U-236     2  0    .61796E-04                    END
U-238     2  0    .22071E-01                    END
PU-239    2  0    .10330E-03                    END
PU-240    2  0    .18351E-04                    END
PU-241    2  0    .92595E-05                    END
SM-149    2  0    .14025E-06                    END
O-16      2  0    .46453E-01                    END
B-10      2  0    .97500E-05                    END
'  4.00 W/O, 15 GWD/T, DIT-KINF =  1.32749
U-235     3  0    .67581E-03                    END
U-236     3  0    .49403E-04                    END
U-238     3  0    .22123E-01                    END
PU-239    3  0    .89270E-04                    END
PU-240    3  0    .12841E-04                    END
PU-241    3  0    .55355E-05                    END
SM-149    3  0    .12607E-06                    END
O-16      3  0    .46453E-01                    END
B-10      3  0    .78750E-05                    END
'  4.00 W/O, 15 GWD/T, DIT-KINF =  1.37119
U-235     4  0    .76543E-03                    END
U-236     4  0    .32933E-04                    END
U-238     4  0    .22186E-01                    END
PU-239    4  0    .64595E-04                    END
PU-240    4  0    .64870E-05                    END
PU-241    4  0    .19757E-05                    END
SM-149    4  0    .10878E-06                    END
O-16      4  0    .46453E-01                    END
B-10      4  0    .52500E-05                    END
```

```
'  4.00 W/O, 25 GWD/T, DIT-KINF =  1.18742
U-235     1  0    .41245E-03                    END
U-236     1  0    .93813E-04                    END
U-238     1  0    .21899E-01                    END
PU-239    1  0    .12607E-03                    END
PU-240    1  0    .35702E-04                    END
PU-241    1  0    .21726E-04                    END
SM-149    1  0    .14550E-06                    END
O-16      1  0    .46453E-01                    END
B-10      1  0    .15500E-04                    END
'  4.00 W/O, 25 GWD/T, DIT-KINF =  1.20708
U-235     2  0    .44319E-03                    END
U-236     2  0    .89833E-04                    END
U-238     2  0    .21914E-01                    END
PU-239    2  0    .12915E-03                    END
PU-240    2  0    .33937E-04                    END
PU-241    2  0    .20907E-04                    END
SM-149    2  0    .14823E-06                    END
O-16      2  0    .46453E-01                    END
B-10      2  0    .14875E-04                    END
'  4.00 W/O, 25 GWD/T, DIT-KINF =  1.25945
U-235     3  0    .53765E-03                    END
U-236     3  0    .74028E-04                    END
U-238     3  0    .22006E-01                    END
PU-239    3  0    .11776E-03                    END
PU-240    3  0    .24956E-04                    END
PU-241    3  0    .14031E-04                    END
SM-149    3  0    .13370E-06                    END
O-16      3  0    .46453E-01                    END
B-10      3  0    .11795E-04                    END
'  4.00 W/O, 25 GWD/T, DIT-KINF =  1.32419
U-235     4  0    .66620E-03                    END
U-236     4  0    .51196E-04                    END
U-238     4  0    .22116E-01                    END
PU-239    4  0    .91389E-04                    END
PU-240    4  0    .13658E-04                    END
PU-241    4  0    .59529E-05                    END
SM-149    4  0    .11420E-06                    END
O-16      4  0    .46453E-01                    END
B-10      4  0    .80000E-05                    END
```

```
'  4.00 W/O, 35 GWD/T, DIT-KINF =  1.09959
U-235     1  0    .27972E-03                END
U-236     1  0    .11365E-03                END
U-238     1  0    .21721E-01                END
PU-239    1  0    .13418E-03                END
PU-240    1  0    .49636E-04                END
PU-241    1  0    .31676E-04                END
SM-149    1  0    .14511E-06                END
O-16      1  0    .46453E-01                END
B-10      1  0    .20250E-04                END
'  4.00 W/O, 35 GWD/T, DIT-KINF =  1.12837
U-235     2  0    .31498E-03                END
U-236     2  0    .10975E-03                END
U-238     2  0    .21745E-01                END
PU-239    2  0    .14025E-03                END
PU-240    2  0    .47677E-04                END
PU-241    2  0    .31259E-04                END
SM-149    2  0    .14942E-06                END
O-16      2  0    .46453E-01                END
B-10      2  0    .19900E-04                END
'  4.00 W/O, 35 GWD/T, DIT-KINF =  1.19734
U-235     3  0    .42222E-03                END
U-236     3  0    .93432E-04                END
U-238     3  0    .21883E-01                END
PU-239    3  0    .13380E-03                END
PU-240    3  0    .36600E-04                END
PU-241    3  0    .22923E-04                END
SM-149    3  0    .13619E-06                END
O-16      3  0    .46453E-01                END
B-10      3  0    .15750E-04                END
'  4.00 W/O, 35 GWD/T, DIT-KINF =  1.28119
U-235     4  0    .57779E-03                END
U-236     4  0    .67055E-04                END
U-238     4  0    .22043E-01                END
PU-239    4  0    .11023E-03                END
PU-240    4  0    .21281E-04                END
PU-241    4  0    .11082E-04                END
SM-149    4  0    .11694E-06                END
O-16      4  0    .46453E-01                END
B-10      4  0    .10450E-04                END
```

```
 '  4.00 W/O, 45 GWD/T, DIT-KINF =  1.01756
U-235     1  0     .18074E-03                     END
U-236     1  0     .12571E-03                     END
U-238     1  0     .21528E-01                     END
PU-239    1  0     .13492E-03                     END
PU-240    1  0     .60309E-04                     END
PU-241    1  0     .38456E-04                     END
SM-149    1  0     .14170E-06                     END
O-16      1  0     .46453E-01                     END
B-10      1  0     .24690E-04                     END
 '  4.00 W/O, 45 GWD/T, DIT-KINF =  1.05498
U-235     2  0     .21640E-03                     END
U-236     2  0     .12271E-03                     END
U-238     2  0     .21564E-01                     END
PU-239    2  0     .14355E-03                     END
PU-240    2  0     .58710E-04                     END
PU-241    2  0     .38939E-04                     END
SM-149    2  0     .14725E-06                     END
O-16      2  0     .46453E-01                     END
B-10      2  0     .24250E-04                     END
 '  4.00 W/O, 45 GWD/T, DIT-KINF =  1.13873
U-235     3  0     .32631E-03                     END
U-236     3  0     .10827E-03                     END
U-238     3  0     .21753E-01                     END
PU-239    3  0     .14219E-03                     END
PU-240    3  0     .47015E-04                     END
PU-241    3  0     .30800E-04                     END
SM-149    3  0     .13576E-06                     END
O-16      3  0     .46453E-01                     END
B-10      3  0     .19450E-04                     END
 '  4.00 W/O, 45 GWD/T, DIT-KINF =  1.24080
U-235     4  0     .49869E-03                     END
U-236     4  0     .80780E-04                     END
U-238     4  0     .21968E-01                     END
PU-239    4  0     .12344E-03                     END
PU-240    4  0     .28859E-04                     END
PU-241    4  0     .16586E-04                     END
SM-149    4  0     .11781E-06                     END
O-16      4  0     .46453E-01                     END
B-10      4  0     .13050E-04                     END
```

```
 '  4.00 W/O, 55 GWD/T, DIT-KINF =   .94562
U-235      1  0     .11078E-03                     END
U-236      1  0     .13121E-03                     END
U-238      1  0     .21319E-01                     END
PU-239     1  0     .13254E-03                     END
PU-240     1  0     .67638E-04                     END
PU-241     1  0     .42318E-04                     END
SM-149     1  0     .13754E-06                     END
O-16       1  0     .46453E-01                     END
B-10       1  0     .28250E-04                     END
 '  4.00 W/O, 55 GWD/T, DIT-KINF =   .98902
U-235      2  0     .14327E-03                     END
U-236      2  0     .12978E-03                     END
U-238      2  0     .21371E-01                     END
PU-239     2  0     .14300E-03                     END
PU-240     2  0     .66880E-04                     END
PU-241     2  0     .43929E-04                     END
SM-149     2  0     .14385E-06                     END
O-16       2  0     .46453E-01                     END
B-10       2  0     .28000E-04                     END
 '  4.00 W/O, 55 GWD/T, DIT-KINF =  1.08311
U-235      3  0     .24757E-03                     END
U-236      3  0     .11906E-03                     END
U-238      3  0     .21616E-01                     END
PU-239     3  0     .14583E-03                     END
PU-240     3  0     .55881E-04                     END
PU-241     3  0     .37135E-04                     END
SM-149     3  0     .13373E-06                     END
O-16       3  0     .46453E-01                     END
B-10       3  0     .22900E-04                     END
 '  4.00 W/O, 55 GWD/T, DIT-KINF =  1.20206
U-235      4  0     .42799E-03                     END
U-236      4  0     .92562E-04                     END
U-238      4  0     .21890E-01                     END
PU-239     4  0     .13253E-03                     END
PU-240     4  0     .36119E-04                     END
PU-241     4  0     .21977E-04                     END
SM-149     4  0     .11740E-06                     END
O-16       4  0     .46453E-01                     END
B-10       4  0     .15250E-04                     END
```

```
'  5.00 W/O, 15 GWD/T, DIT-KINF =  1.33838
U-235     1  0    .79849E-03                    END
U-236     1  0    .71099E-04                    END
U-238     1  0    .21851E-01                    END
PU-239    1  0    .10289E-03                    END
PU-240    1  0    .16299E-04                    END
PU-241    1  0    .81004E-05                    END
SM-149    1  0    .16358E-06                    END
O-16      1  0    .46453E-01                    END
B-10      1  0    .11100E-04                    END
'  5.00 W/O, 15 GWD/T, DIT-KINF =  1.34773
U-235     2  0    .82204E-03                    END
U-236     2  0    .67614E-04                    END
U-238     2  0    .21857E-01                    END
PU-239    2  0    .10292E-03                    END
PU-240    2  0    .15350E-04                    END
PU-241    2  0    .75403E-05                    END
SM-149    2  0    .16560E-06                    END
O-16      2  0    .46453E-01                    END
B-10      2  0    .10850E-04                    END
'  5.00 W/O, 15 GWD/T, DIT-KINF =  1.37938
U-235     3  0    .89796E-03                    END
U-236     3  0    .53541E-04                    END
U-238     3  0    .21905E-01                    END
PU-239    3  0    .87529E-04                    END
PU-240    3  0    .10569E-04                    END
PU-241    3  0    .43506E-05                    END
SM-149    3  0    .15131E-06                    END
O-16      3  0    .46453E-01                    END
B-10      3  0    .87500E-05                    END
'  5.00 W/O, 15 GWD/T, DIT-KINF =  1.41914
U-235     4  0    .99411E-03                    END
U-236     4  0    .35237E-04                    END
U-238     4  0    .21962E-01                    END
PU-239    4  0    .61907E-04                    END
PU-240    4  0    .52033E-05                    END
PU-241    4  0    .14752E-05                    END
SM-149    4  0    .13389E-06                    END
O-16      4  0    .46453E-01                    END
B-10      4  0    .59250E-05                    END
```

```
'  5.00 W/O, 25 GWD/T, DIT-KINF =  1.25351
U-235     1  0    .60224E-03                END
U-236     1  0    .10572E-03                END
U-238     1  0    .21703E-01                END
PU-239    1  0    .13098E-03                END
PU-240    1  0    .31090E-04                END
PU-241    1  0    .19266E-04                END
SM-149    1  0    .16954E-06                END
O-16      1  0    .46453E-01                END
B-10      1  0    .16825E-04                END
'  5.00 W/O, 25 GWD/T, DIT-KINF =  1.26957
U-235     2  0    .63664E-03                END
U-236     2  0    .10103E-03                END
U-238     2  0    .21715E-01                END
PU-239    2  0    .13349E-03                END
PU-240    2  0    .29468E-04                END
PU-241    2  0    .18356E-04                END
SM-149    2  0    .17335E-06                END
O-16      2  0    .46453E-01                END
B-10      2  0    .16200E-04                END
'  5.00 W/O, 25 GWD/T, DIT-KINF =  1.31702
U-235     3  0    .74472E-03                END
U-236     3  0    .81971E-04                END
U-238     3  0    .21798E-01                END
PU-239    3  0    .11928E-03                END
PU-240    3  0    .21225E-04                END
PU-241    3  0    .11811E-04                END
SM-149    3  0    .15874E-06                END
O-16      3  0    .46453E-01                END
B-10      3  0    .13125E-04                END
'  5.00 W/O, 25 GWD/T, DIT-KINF =  1.37642
U-235     4  0    .88738E-03                END
U-236     4  0    .55572E-04                END
U-238     4  0    .21898E-01                END
PU-239    4  0    .89827E-04                END
PU-240    4  0    .11263E-04                END
PU-241    4  0    .47037E-05                END
SM-149    4  0    .13907E-06                END
O-16      4  0    .46453E-01                END
B-10      4  0    .89500E-04                END
```

```
'  5.00 W/O, 35 GWD/T, DIT-KINF =  1.17304
U-235     1  0    .44078E-03                   END
U-236     1  0    .13186E-03                   END
U-238     1  0    .21543E-01                   END
PU-239    1  0    .14334E-03                   END
PU-240    1  0    .44767E-04                   END
PU-241    1  0    .29813E-04                   END
SM-149    1  0    .16806E-06                   END
O-16      1  0    .46453E-01                   END
B-10      1  0    .22050E-04                   END
'  5.00 W/O, 35 GWD/T, DIT-KINF =  1.19681
U-235     2  0    .48243E-03                   END
U-236     2  0    .12679E-03                   END
U-238     2  0    .21562E-01                   END
PU-239    2  0    .14887E-03                   END
PU-240    2  0    .42775E-04                   END
PU-241    2  0    .29043E-04                   END
SM-149    2  0    .17373E-06                   END
O-16      2  0    .46453E-01                   END
B-10      2  0    .21400E-04                   END
'  5.00 W/O, 35 GWD/T, DIT-KINF =  1.25994
U-235     3  0    .61143E-03                   END
U-236     3  0    .10561E-03                   END
U-238     3  0    .21687E-01                   END
PU-239    3  0    .13899E-03                   END
PU-240    3  0    .31978E-04                   END
PU-241    3  0    .20316E-04                   END
SM-149    3  0    .16078E-06                   END
O-16      3  0    .46453E-01                   END
B-10      3  0    .17000E-04                   END
'  5.00 W/O, 35 GWD/T, DIT-KINF =  1.33711
U-235     4  0    .78979E-03                   END
U-236     4  0    .73774E-04                   END
U-238     4  0    .21832E-01                   END
PU-239    4  0    .11065E-03                   END
PU-240    4  0    .17924E-04                   END
PU-241    4  0    .91522E-05                   END
SM-149    4  0    .14150E-06                   END
O-16      4  0    .46453E-01                   END
B-10      4  0    .11600E-04                   END
```

APPENDIX C. Boron-10 Number Densities Employed in KENO Calculations

Enrichment = 3.000 w/o U-235, Burnup = 5000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	0.3550	1.32115	1.32098 ± 0.00062
1	0.3710	1.31841	1.31847 ± 0.00062
2	0.3500	1.32326	1.32309 ± 0.00062
3	0.2850	1.33901	1.33882 ± 0.00062
4	0.2000	1.35711	1.35742 ± 0.00059

Enrichment = 3.000 w/o U-235, Burnup = 15000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	0.8850	1.20705	1.20740 ± 0.00062
1	0.9050	1.19975	1.19940 ± 0.00062
2	0.8775	1.21411	1.21404 ± 0.00057
3	0.6875	1.25273	1.25294 ± 0.00061
4	0.4750	1.30042	1.29981 ± 0.00059

Enrichment = 3.000 w/o U-235, Burnup = 25000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	1.3800	1.10966	1.10964 ± 0.00056
1	1.4200	1.09817	1. 09813 ± 0.00056
2	1.3700	1.12239	1.12199 ± 0.00057
3	1.0800	1.17925	1.17943 ± 0.00060
4	0.7225	1.24913	1.24928 ± 0.00061

Enrichment = 3.000 w/o U-235, Burnup = 35000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	1.8200	1.02363	1.02353 ± 0.00053
1	1.8700	1.00906	1.00907 ± 0.00053
2	1.8150	1.04191	1.04227 ± 0.00053
3	1.4545	1.11345	1.11341 ± 0.00060
4	0.9500	1.20252	1.20257 ± 0.00061

Enrichment = 3.000 w/o U-235, Burnup = 45000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	2.2350	0.95107	0.95149 ± 0.00051
1	2.2750	0.93488	0.93483 ± 0.00049
2	2.2350	0.97341	0.97335 ± 0.00052
3	1.8050	1.05390	1.05379 ± 0.00056
4	1.2050	1.15900	1.15925 ± 0.00060

Enrichment = 4.000 w/o U-235, Burnup = 15000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	0.9800	1.28737	1.28711 ± 0.00062
1	1.0000	1.28118	1.28110 ± 0.00062
2	0.9750	1.29251	1.29226 ± 0.00064
3	0.7875	1.32709	1.32709 ± 0.00064
4	0.5250	1.37121	1.37121 ± 0.00063

Enrichment = 4.000 w/o U-235, Burnup = 25000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	1.4950	1.19750	1.19796 ± 0.00059
1	1.5500	1.18742	1.18729 ± 0.00059
2	1.4875	1.20708	1.20695 ± 0.00064
3	1.1795	1.25945	1.26003 ± 0.00060
4	0.8000	1.32419	1.32421 ± 0.00059

Enrichment = 4.000 w/o U-235, Burnup = 35000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	1.9750	1.11359	1.11302 ± 0.00061
1	2.0250	1.09959	1. 09918 ± 0.00058
2	1.9900	1.12837	1.12813 ± 0.00054
3	1.5750	1.19734	1.19745 ± 0.00059
4	1.0450	1.28119	1.28162 ± 0.00064

Enrichment = 4.000 w/o U-235, Burnup = 45000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	2.4150	1.03489	1.03466 ± 0.00053
1	2.4690	1.01756	1.01749 ± 0.00054
2	2.4250	1.05437	1.05437 ± 0.00055
3	1.9450	1.13873	1.13837 ± 0.00057
4	1.3050	1.24080	1.24088 ± 0.00062

Enrichment = 4.000 w/o U-235, Burnup = 55000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	2.7750	0.96460	0.96409 ± 0.00054
1	2.8250	0.94562	0.94625 ± 0.00051
2	2.8000	0.98902	0.98866 ± 0.00053
3	2.2900	1.08311	1.08323 ± 0.00056
4	1.5250	1.20206	1.20235 ± 0.00061

Enrichment = 5.000 w/o U-235, Burnup = 15000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	1.0950	1.34380	1.34349 ± 0.00060
1	1.1100	1.33838	1.33777 ± 0.00063
2	1.0850	1.34773	1.34834 ± 0.00067
3	0.8750	1.37897	1.37897 ± 0.00061
4	0.5925	1.41914	1.41871 ± 0.00064

Enrichment = 5.000 w/o U-235, Burnup = 25000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	1.6250	1.26219	1.26192 ± 0.00061
1	1.6825	1.25351	1.25319 ± 0.00055
2	1.6200	1.26957	1.26924 ± 0.00059
3	1.3125	1.31702	1.31664 ± 0.00060
4	0.8950	1.37642	1.37641 ± 0.00064

Enrichment = 5.000 w/o U-235, Burnup = 35000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	2.1500	1.18527	1.18580 ± 0.00056
1	2.2050	1.17304	1.17309 ± 0.00059
2	2.1400	1.19681	1.19620 ± 0.00060
3	1.7000	1.25994	1.26002 ± 0.00061
4	1.1600	1.33711	1.33642 ± 0.00060

Enrichment = 5.000 w/o U-235, Burnup = 45000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	2.6050	1.11032	1.11032 ± 0.00057
1	2.6700	1.09440	1. 09499 ± 0.00058
2	2.6000	1.12673	1.12613 ± 0.00058
3	2.0950	1.20561	1.20596 ± 0.00059
4	1.4150	1.30019	1.30025 ± 0.00065

Enrichment = 5.000 w/o U-235, Burnup = 55000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	3.0000	1.03806	1.03798 ± 0.00053
1	3.0665	1.01899	1.01882 ± 0.00054
2	3.0200	1.05953	1.05931 ± 0.00054
3	2.4650	1.15291	1.15310 ± 0.00057
4	1.6750	1.26469	1.26404 ± 0.00058

Enrichment = 5.000 w/o U-235, Burnup = 65000 MWD/MTU

Zone No.	B-10 Number Density X 10^5 Atoms/(Barn-Cm)	K_{inf} (DIT)	K_{inf} (KENO)
Average	3.3700	0.97141	0.97080 ± 0.00048
1	3.3825	0.95038	0.95052 ± 0.00051
2	3.3800	0.99725	0.99755 ± 0.00052
3	2.8000	1.10166	1.10189 ± 0.00055
4	1.9050	1.23016	1.22992 ± 0.00058

PACIFIC GAS & ELECTRIC COMPANY
DIABLO CANYON UNITS 1 AND 2
SPENT FUEL POOL BORON DILUTION ANALYSIS

JANUARY, 2001

DIABLO CANYON UNITS 1 AND 2

SPENT FUEL POOL BORON DILUTION ANALYSIS

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Rev. 1

January, 2001

Westinghouse Electric Company LLC

Table of Contents

1.0 INTRODUCTION

A boron dilution analysis has been completed for crediting boron in the Diablo Canyon Units 1 and 2 spent fuel rack criticality analysis. The boron dilution analysis includes an evaluation of the following plant specific features:

- Dilution Sources
- Boration Sources
- Instrumentation
- Administrative Procedures
- Piping
- Loss of Offsite Power Impact
- Boron Dilution Initiating Events
- Boron Dilution Times and Volumes

The boron dilution analysis was completed to ensure that sufficient time is available to detect and mitigate the dilution before the spent fuel rack criticality analysis 0.95 k_{eff} design basis is exceeded.

2.0 SPENT FUEL POOL AND RELATED SYSTEM FEATURES

This section provides background information on the spent fuel pool and its related systems and features. A one-line diagram of the spent fuel pool related systems is provided as Figure 1. For the purposes of this evaluation, the spent fuel pool and its related systems are sufficiently similar between the two Units that they will be treated as identical. Any significant differences will be identified, so that this report will be bounding for both Units.

2.1 Spent Fuel Pool

The design purpose of the spent fuel pool is to provide for the safe storage of irradiated fuel assemblies. The pool is filled with borated water. The water removes decay heat, provides shielding for personnel handling the fuel, and reduces the amount of radioactive gases released during a fuel handling accident. Pool water evaporation takes place on a continuous basis, requiring periodic makeup. The makeup source can be unborated water, since the evaporation process does not carry off the boron. Evaporation actually increases the boron concentration in the pool.

The spent fuel pool is a reinforced concrete structure with a seam-welded stainless steel liner. The water-tight liner has dedicated drain lines to collect and detect liner leakage. The pool structure is designed to meet seismic requirements. The pool is approximately 39 feet deep. The top of the pit is located on the 140' elevation of the fuel handling building. The bottom of the pit is approximately at the 99' elevation.

In the event of excessive makeup flow into the pool, the pool would first overflow the top of the gate leading to the transfer canal. Once the canal is filled, the water will rise to the top of a 2" curb surrounding the pool, then spill onto the floor and down into several floor drains on the main floor (elevation 140'). This large volume will quickly result in alarms for the liquid waste system to alert the operator of abnormally high inputs.

As shown in Figure 2, the transfer canal lies adjacent to the pool and connects to the reactor cavity during refueling operations. The gate between the pool and the transfer canal is normally closed. The volume of the pool is approximately 54,984 ft^3 to the low alarm level elevation of 137'- 4" less instrument accuracy(1"). The majority of the water volume displaced by objects in the pool is by the spent fuel assemblies. The maximum number of assembly locations is 1324. The water volume

displaced by all 1324 assemblies (3535 ft^3) is subtracted from the total pool volume. The racks themselves occupy a relatively small volume(688 ft^3), but they are subtracted as well. When the above volumes are subtracted from the pool volume, the remaining water volume (50,761 ft^3 = 379,718 gal.) is conservatively rounded down to 379,000 gallons.

2.2 Spent Fuel Storage Racks

The spent fuel racks are designed to support and protect the spent fuel assemblies under normal and credible accident conditions. Their design ensures the ability to withstand combinations of dead loads, live loads (fuel assemblies), and seismic loads.

2.3 Spent Fuel Pool Cooling System

The spent fuel pool cooling system is designed to remove the heat generated by stored spent fuel elements from the spent fuel pool. The system design incorporates redundancy for the only active component, the spent fuel pool cooling pump. System piping is configured so that failure of any pipeline in the cooling system does not drain the spent fuel pool below the top of the stored spent fuel assemblies.

The portion of the spent fuel pool cooling system which, if it failed, could result in a significant release of pool water is seismically designed.

The cooling system train consists of redundant pumps, a heat exchanger, valves, piping and instrumentation. The pump takes suction from the fuel pool at an inlet located below the pool water level, transfers the pool water through a heat exchanger and returns it back into the pool through an outlet located below and on the opposite side of the pool from the cooling system inlet. The return line is designed to prevent siphoning. The heat exchangers are cooled by component cooling water.

2.4 Spent Fuel Pool Cleanup System

The spent fuel pool cleanup system is designed to maintain water clarity and to control borated water chemistry. The cleanup system is connected to the spent fuel pool cooling system. About 100 gpm of the spent fuel pool cooling pump(s) discharge flow can be diverted to the cleanup loop, which includes the refueling water purification filter, spent fuel pool demineralizer, spent fuel pool resin trap filter, and

the spent fuel pool filter. The filters remove particulates from the spent fuel pool water and the spent fuel pool demineralizer removes ionic impurities.

The refueling water purification loop also uses the spent fuel pool demineralizer and filters to clean up the refueling water storage tank after refueling operations. The design flow rate in the loop is limited to 150 gpm to accommodate the design flow of the spent fuel pool demineralizer and filters.

The spent fuel pool has a surface skimmer system designed to provide optical clarity by removing surface debris. The system consists of two surface skimmers, a single strainer, a single pump and three filters. The skimmer pump is a centrifugal pump with a 100 gpm capacity. The pump discharge flow passes through the filter to remove particulates. Skimmer flow returns to the spent fuel pool.

2.5 Dilution Sources

2.5.1 Chemical and Volume Control System (CVCS)

The CVCS connects to the spent fuel pool via a 3" line from the discharge of the outlet of the holdup tanks recirculation pump to the refueling canal. This connection is normally isolated and is used to transfer water from the holdup tanks to the spent fuel pool. The isolation is by two normally closed manual valves.

Since there is also a check valve at the recirculation pump discharge, water will not flow from the spent fuel pool to the holdup tanks. Also, holdup tank water will not gravity-drain to the spent fuel pool because the holdup tank recirculation pump is normally isolated, and the maximum tank water level is below the minimum SFP level.

The recirculation pump can take suction from either of the three holdup tanks. However, by procedure, the pump is aligned to only one holdup tank at a time. Manual valve manipulations are required to switch the pump suction to another tank. Each holdup tank has a nominal volume of 75,000 gallons and can be at a boron concentration from 0 ppm up to 2300 ppm. The flow from this source is estimated to be 375 gpm.

2.5.2 Primary Water Makeup System

The primary water makeup system consists of one sealed primary water storage tank and two primary water pumps per Unit. During normal operation, one primary water pump is running on recirculation to provide primary water on demand to multiple users. Each primary water storage tank contains approximately 200,000 gallons of non-borated, demineralized water.

The primary water makeup system connects to the spent fuel pool indirectly through the spent fuel pool demineralizer outlet and a local tool washing station in the spent fuel pool area. The local station is provided for rinsing items when they are removed from the spent fuel pool. It is isolated by two normally closed valves. Although the flow rate is not measured, the quantity of water is measured and recorded at the station. The flow rate from this 1" line is estimated to be 50 gpm.

When primary water is used to flush spent resin, the spent fuel pool demineralizer is isolated from the cleanup loop by one manual valve. If this valve were left open, primary water could be transferred into the spent fuel pool. The flow from this pathway is estimated to be 170 gpm.

The primary water storage tank can be aligned to directly make up to the spent fuel pool via the primary water transfer pumps. However, this alignment is not addressed in a formal procedure. In any case, the flow through this path would be bounded by the flow through spent resin flushing path described above, which is the pump runout flow (170 gpm).

2.5.3 Makeup Water System

The makeup water system is supplied from a rental demineralizer/deoxygenation unit. Demineralized water can be aligned directly to the spent fuel pool, or stored in the 425,000 gallon condensate storage tank or the fire water transfer tank, which include a 300,000 gallon compartment for fire water and a 150,000 gallon compartment for other uses. Demineralized water is provided directly to the spent fuel pool cooling loop and indirectly via the spent fuel pool cooling return piping.

The direct connection to the spent fuel pool is from the condensate storage tank or fire water transfer tank via the makeup water transfer pumps to a 2" pipe isolated by one normally closed manual valve. The flow from this path is estimated to be 320 gpm.

The indirect connection from the rental processing unit via the cooling return header is a 2" pipe isolated by two normally closed manual valves. The flow from this path is estimated to be 494 gpm.

2.5.4 Liquid Radwaste System

This system contains two 14,735 gallon processed waste receiver tanks and two small transfer pumps which are used as an alternate to the primary water system for flushing spent resin from the spent fuel pool demineralizer. Because the source volume of this flowpath is very limited, it is not considered further in this analysis.

2.5.5 Component Cooling Water System

Component cooling water is the cooling medium for the spent fuel pool cooling system heat exchanger. There is no direct connection between the component cooling system and the spent fuel pool cooling system. If, however, a leak were to develop in a heat exchanger that is in service, the connection would be made. Since the component cooling system normally operates at a slightly higher pressure than the spent fuel pool cooling system, it is expected that a breach in a spent fuel pool cooling system heat exchanger tube would result in non-borated component cooling water entering the spent fuel pool cooling system.

It would be expected that the flow rate of any leakage of component cooling water into the spent fuel pool cooling system would be very low due to the small difference in operating pressures between the two systems. Even if there was significant leakage from the component cooling water system to the spent fuel pool, the impact on the spent fuel pool boron concentration would be minimal because a loss of water from the component cooling water surge tank would initiate an alarm and control room indication to alert the control room operators that the automatic level control makeup valves have opened.

If the alarms which would alert the control room operators of a component cooling water system leak were to fail and leakage from the component cooling water system to the spent fuel pool cooling system were to continue undetected, the component cooling water surge tank would be automatically refilled with makeup water. Until makeup is initiated, the volume added to the spent fuel pool would be limited by the component cooling water surge tank volume of 2000 gallons.

Because of the limited dilution volume from this source relative to the spent fuel pool volume, it is not considered further in this analysis.

2.5.6 Drain Systems

The equipment drain system connects directly to the spent fuel pool cooling system and skimmer system at the drain connections for the spent fuel pool pumps, heat exchangers (tube side), filters, demineralizer, the skimmer pump, and skimmer filter. Each connection has a normally closed valve to isolate it. Backflow through these paths is not considered credible, because if the drain valves were left open, the pressurized spent fuel pool cooling system would flow into the drain system, not vice versa.

2.5.7 Fire Protection System

In an emergency loss of spent fuel pool inventory, a fire hose station is available outside the spent fuel pool area door. This station is capable of providing 125 gpm of non-borated water. Although an available source, the fire hose is specified as a last resort due to the low water quality in a procedure for operator response to a low level alarm.

2.5.8 Spent Fuel Pool Demineralizer

The spent fuel pool demineralizer has a nominal capacity of 39 ft^3 of 1:1 equivalent mixed bed resin. This implies a volume ratio of 60%/40% anion to cation resin. If we assume the bed was loaded with 100% anion, it would bound the capacity to remove boron when it is first aligned to the system. The demineralizer would be operated at a nominal 100 gpm flow rate. Dilution of the spent fuel pool resulting from operation of the demineralizer will only continue until the anion resin is saturated and will not result in an increase in the spent fuel pool level.

2.5.9 Dilution Source and Flow Rate Summary

Based on the evaluation of potential spent fuel pool dilution sources summarized above, the following dilution sources were determined to be capable of providing a significant amount of non-borated water to the spent fuel pool. The potential for these sources to dilute the spent fuel pool boron concentration to the design basis boron concentration (800 ppm) will be evaluated in Section 3.0.

SOURCE	APPROXIMATE FLOW RATE (GPM)
CVCS	
- Holdup Tank to Cleanup Loop	375
Primary Water System	
- To SFP via demineralizer sluice line	170
- 1" PW station near SFP	50
Demineralized Water System	
- To SFP via valve 803	494
- To SFP via valve 8755	320
Fire Protection System	
- Fire hose station in SFP room	125
SFP Demineralizer	100

2.6 Boration Sources

Borated water may be transferred to the spent fuel pool from the refueling water storage tank or the boric acid storage tanks. It is also possible to borate the spent fuel pool by the addition of dry boric acid directly to the spent fuel pool water.

2.6.1 Refueling Water Storage Tank

The refueling water storage tank (RWST) connects to the spent fuel pool through the purification loop via the refueling water purification pump. This connection is normally used to purify the RWST water when the purification loop is isolated from the spent fuel pool cooling system. This connection can supply borated water to the spent fuel pool via the refueling water purification pump to the inlet to the spent fuel pool cooling system purification loop. The refueling water purification pump is powered from a non-vital bus power supply. The RWST is required by Technical Specifications to be kept at a minimum boron concentration of 2300 ppm.

2.6.2 Boric Acid Storage Tanks

Concentrated boric acid may be transferred to the CVCS holdup tanks, then to the spent fuel pool using existing procedures.

2.6.3 Direct Addition of Boric Acid

If necessary, the boron concentration of the spent fuel pool can be increased by emptying bags of dry boric acid directly into the spent fuel pool. However, boric acid dissolves very slowly at room temperature and requires that the spent fuel pool cooling pumps be available for mixing the spent fuel pool water (see section 3.1 for further discussion on spent fuel pool mixing.) An existing procedure specifies that this method be used if the boric acid makeup system is not available or is too slow.

2.7 Spent Fuel Pool Instrumentation

Instrumentation is available to monitor spent fuel pool water level and temperature. Additional instrumentation is provided to monitor the pressure and flow of the spent fuel pool cleanup system, and pressure and temperature of the spent fuel pool cooling system.

The water level instrumentation alarms, high or low level, are annunciated in the control room. Two area radiation monitors annunciate locally and in the main control room.

A change of one foot in spent fuel pool level above the top of the racks and fuel assemblies with the transfer canal isolated requires approximately 10,425 gallons of water. If the pool level was raised from the low level alarm point to the high level alarm (22", including instrument error), a dilution of approximately 19,113 gallons could occur before an alarm would be received in the control room. If the spent fuel pool boron concentration were at 2000 ppm initially, such a dilution would only result in a reduction of the pool boron concentration of approximately 96 ppm.

2.8 Administrative Controls

The following administrative controls are in place to control the spent fuel pool boron concentration and water inventory:

1. The procedures for loss of inventory (other than evaporation) specify that a borated makeup source be used as the makeup source. The procedures specify the RWST, the CVCS holdup tanks, and the boric acid makeup system as the sources in order of preference.

3. In accordance with procedures, plant personnel perform rounds in the spent fuel pool room

once every 12 hours. The personnel making rounds to the spent fuel pool are trained to be aware of the change in the status of the spent fuel pool. They are instructed to check the temperature and level in the pool and conditions around the pool during plant rounds.

4. Procedures require that the chemistry department sample the pool boron concentration prior to and after makeup is added. The makeup procedure requires an administrative minimum boron concentration of 2300 ppm, even though the Technical Specification minimum is 2000 ppm boron.

2.9 Piping

There are no systems (other than those listed in section 2.5.1 to 2.5.8) identified which have piping in the vicinity of the spent fuel pool which could result in a dilution of the spent fuel pool if they were to fail.

2.10 Loss of Offsite Power Impact

Of the dilution sources listed in Section 2.5.9, only the fire protection system is capable of providing non-borated water to the spent fuel pool during a loss of offsite power.

The loss of offsite power would affect the ability to respond to a dilution event. The spent fuel pool level instrumentation is not powered from vital power supplies.

The refueling water purification pump is not powered from a safeguards supply and would not be available to deliver borated water from the RWST. However, at a water level above approximately 50%, the RWST can be gravity-drained to the spent fuel pool through the refueling water purification pumps, if necessary, to provide a borated water source. Finally, manual addition of dry boric acid to the pool could be used if it became necessary to increase the spent fuel pool boron concentration during a loss of offsite power.

3.0 SPENT FUEL POOL DILUTION EVALUATION

3.1 Calculation of Boron Dilution Times and Volumes

For the purposes of evaluating spent fuel pool dilution times and volumes, the total pool volume available for dilution, as described in section 2.1, is conservatively assumed to be 379,000 gallons.

Based on the criticality analyses (Reference 1), the soluble boron concentration required to maintain the spent fuel pool boron concentration at $k_{eff} < 0.95$, including uncertainties and burnup, with a 95% probability at a 95% confidence level (95/95) is 800 ppm.

The spent fuel pool boron concentration is typically in the range of 2000 and 2300 ppm. However, for the purposes of evaluating the dilution times and volumes, the initial spent fuel pool boron concentration is assumed to be at the current Technical Specification minimum limit of 2000 ppm. The evaluations are based on the spent fuel pool boron concentration being diluted from 2000 ppm to 800 ppm. To dilute the combined pool volume of 379,000 gallons from 2000 ppm to 800 ppm would conservatively require 347,000 gallons of non-borated water, based on a feed-and-bleed operation (constant volume).

This analysis assumes thorough mixing of all the non-borated water added to the spent fuel pool with the contents of the spent fuel pool. Refer to Figure 3. Based on the design flow of 2300 gpm per spent fuel pool pump, the 379,000 gallon system volume is turned over approximately every two to three hours with one pump running, which is the normal alignment. It is unlikely, with cooling flow and convection from the spent fuel decay heat, that thorough mixing would not occur. However, if mixing was not adequate, it would be conceivable that a localized pocket of non-borated water could form somewhere in the spent fuel pool. This possibility is addressed by the calculation in Reference 1 which shows that the spent fuel rack K_{eff} will be less than 1.0 on a 95/95 basis with the spent fuel pool filled with non-borated water. Thus, even if a pocket of non-borated water formed in the spent fuel pool, K_{eff} would not exceed 1.0 anywhere in the pool.

The time to dilute the spent fuel pool depends on the initial volume of the pool and the postulated rate of dilution. The dilution volumes and times for the dilution scenarios discussed in Sections 3.2 and 3.3 are calculated based on the following equation:

$$t_{end} = \ln (C_o / C_{end}) V/Q \qquad \text{(Equation 1)}$$

Where:

C_o = the boron concentration of the pool volume at the beginning of the event (2000 ppm)

C_{end} = the boron endpoint concentration (800 ppm)

Q = dilution rate (gallons/minute)

V = volume (gallons) of spent fuel pool (379,000)

t_{end} = time to reach C_{end} (minutes)

3.2 Evaluation of Boron Dilution Events

The potential spent fuel pool dilution events that could occur are evaluated below:

3.2.1 Dilution From CVCS Holdup Tanks

The contents of a CVCS holdup tank can be transferred via the recirculation pump to the spent fuel pool via the cleanup loop. The flow path to the transfer canal is through a line that is isolated by one normally closed valve. This connection is a designated source of makeup water in a loss of spent fuel pool inventory event. Each of the three CVCS holdup tanks has a total volume of approximately 75,000 gallons. The water in the tanks can have a boron concentration from 0 ppm to approximately 2000 ppm. Any amount of boron in the CVCS holdup tank water would reduce the dilution of the spent fuel pool resulting from the transfer of CVCS holdup tank water to the spent fuel pool. To dilute the spent fuel pool volume from 2000 ppm to 800 ppm would require 347,000 gallons of unborated water. The combined contents of the three CVCS holdup tanks (approximately 225,000 gallons) is less than the required dilution volume. Normally, these tanks contain letdown flow diverted from the reactor coolant system at the prevailing boron concentration. Although the concentration is normally less than that of the spent fuel pool, it would typically contain several hundred ppm of boron and is therefore less severe than an unborated makeup source. Because of these factors, the CVCS holdup tanks are not considered a credible dilution source for the purposes of this analysis.

3.2.2 Dilution From Primary Water Storage Tanks

The contents of the primary water storage tank can be transferred via the primary water pumps directly or indirectly to the spent fuel pool.

The primary water system consists of a primary water storage tank and two primary water pumps per Unit. Primary water can be supplied to the spent fuel pool cooling system from the tank and pumps associated with either Unit. The two primary water storage tanks each contain approximately 200,000 gallons of non-borated reactor grade water. The tanks are normally not cross-connected. It would require both tanks to be emptied to provide sufficient volume to dilute the spent fuel pool from 2000 to 800 ppm (347,000 gal. required).

The path from the primary water pumps to the local tool washing station near the spent fuel pool is approximately 50 gpm. If the hose were left spilling in the pool unattended, it would take 6.4 hr. to increase the spent fuel pool level from the low to high alarm setpoints, and 5 days to provide the 347,000 gallons required to dilute the pool from 2000 to 800 ppm boron.

The path from the primary water pumps to the spent fuel pool via the spent fuel pool demineralizer resin flushing connection can provide approximately 170 gpm. If the manual isolation valve were left unattended, it would take 2 hr. to increase the spent fuel pool level from the low to high alarm setpoints, and 34 hr. to provide the 347,000 gallons required to dilute the pool from 2000 to 800 ppm boron.

3.2.3 Dilution From Makeup Water System

Flow from the rental water treatment unit is administratively controlled. Although its source of raw water is essentially infinite (raw water), its capacity is limited to a flow of 494 gpm. If flow from this unit were left aligned to the spent fuel pool, it would take 37 minutes to increase the spent fuel pool level from the low to high alarm setpoints, and 12 hours to provide the 347,000 gallons required to dilute the pool from 2000 to 800 ppm boron.

The path from the condensate storage tank or fire water transfer tank via the makeup water transfer pumps to the spent fuel pool via the 2" connection can provide approximately 320 gpm. If the path were left unattended, it would take 1 hr. to increase the spent fuel pool level from the low to high alarm setpoints, and 18 hours to provide the 347,000 gallons required to dilute the pool from 2000 to 800 ppm boron.

3.2.4 Dilution from Fire Protection System

The fire protection system draws from a firewater transfer tank with a total capacity of 450,000 gallons. This capacity is slightly more than the 347,000 gal. required to dilute the spent fuel pool from 2000 ppm to 800 ppm boron. The path from the fire water pump to the spent fuel pool via the fire hose station outside the spent fuel pool area can provide approximately 125 gpm. If the hose were left unattended, it would take 2.5 hr. to increase the spent fuel pool level from the low to high alarm setpoints, and 2 days to provide the 347,000 gallons required to dilute the pool from 2000 to 800 ppm boron.

3.2.5 Dilution Resulting From Seismic Events or Random Pipe Breaks

A seismic event could cause piping ruptures in the vicinity of the spent fuel pool in piping that is not seismically qualified. The only piping within the immediate vicinity of the spent fuel pool that could result in dilution of the spent fuel pool if it ruptures during a seismic event are 1" primary water station for tool washing, and the 2" piping from the makeup water system.

For a seismic event with offsite power available, rupture of the 1" primary water station piping in the spent fuel pool area is bounded by runout flow of the pump (170 gpm), which is already addressed in the analyses in Section 3.2.2. If offsite power is not available, the primary water systems would not operate, and thus, there would be no dilution source.

For a seismic event with offsite power available, rupture of the 2" makeup water piping into the spent fuel pool will not result in flow entering the pool because the path is isolated by a valve (MU-1-803) located outside the pool area. If offsite power is not available, the makeup water system would not operate, and thus, there would be no dilution source.

3.2.6 Dilution From Spent Fuel Pool Demineralizer

When the spent fuel pool demineralizer is first placed in service after being recharged with fresh resin, it can initially remove boron from the water passing through it. In the worst case, assuming 30 ft^3 of anion resin in the demineralizer, up to 12 ppm of boron could be removed from the spent fuel pool water before the resin would become saturated. Since the demineralizer normally utilizes a mixed bed of anion and cation resin, less boron would actually be removed before saturation. Because of the small amount of boron removed by the demineralizers, it is not considered a credible dilution source for the purposes of this evaluation.

3.2.7 Review of Licensee Event Reports(LER)

No LERs related to the spent fuel pool and cooling system were found which could identify any extraordinary mechanisms of dilution or other flow paths not previously addressed in the previous sections.

3.3 Summary of Dilution Events

The four available water sources for spent fuel pool dilution are primary water, makeup water, CVCS holdup tank fluid, and fire protection. Fire protection is the least likely source, since it is specified as the last resort by procedure for use as makeup. The CVCS holdup tank source is the next least likely because it is normally borated to some degree, and because the volume of one tank is less than that required to dilute the spent fuel pool from 2000 to 800 ppm boron. The primary water source is the next least likely source because it is used only for resin sluicing by procedure, and although only one valve misalignment could initiate a dilution accident, spent fuel pool demineralizer resin replacement operations are typically very infrequent, and the volume of one primary water storage tank is less than that required to dilute the spent fuel pool from 2000 to 800 ppm boron.

The makeup water system is the most likely source for dilution. The procedure for makeup for evaporation includes the path from the rental makeup water system and from the makeup water transfer pump. Dilution from either the condensate storage tank (425,000 gal.) or the firewater tank (450,000 gal.) is sufficient to dilute the spent fuel pool from 2000 to 800 ppm boron. In theory, it would be possible for sufficient volume from the rental water treatment unit to eventually provide enough dilution volume to the spent fuel pool. However, at the estimated flow rate of 494 gpm from the unit to the spent fuel pool, it would take approximately 12 hr. to provide 347,000 gal. of dilution volume. Dilution via the makeup water transfer pumps would take even longer. Assuming, of course, that the spent fuel pool level instrumentation high alarm has failed, the operator would easily detect pool water overflowing during rounds (every 12 hours). Such a large overflow volume of water would quickly enter the floor drain system and actuate alarms in the liquid waste system.

4.0 CONCLUSIONS

A boron dilution analysis has been completed for the spent fuel pool. As a result of this spent fuel pool boron dilution analysis, it is concluded that an unplanned or inadvertent event which would result in the dilution of the spent fuel pool boron concentration from 2000 ppm to 800 ppm is not a credible event. This conclusion is based on the following:

The primary water storage tank volume is less than that required for dilution of the spent fuel pool from 2000 to 800 ppm boron. The firewater storage tank volume is only slightly larger than the required dilution volume.

The only makeup source of relatively unlimited volume is via the rental water treatment system. However, even at the high flow rate available from this system to the spent fuel pool, it would take approximately 12 hr. for the dilution event to complete. During this time, at least one operator round (every 12 hr.) and waste system alarms would notify the operator of a dilution event.

If an inadvertent dilution were to be initiated, administrative procedures are in place to address a high level alarm in the spent fuel pool, or to add borated water to the pool to recover from the event. Borated water from the RWST is available via the refueling water purification pump with normal power available, and by gravity feed to the pool should offsite power be lost. As a last resort, boric acid powder can be added directly to the pool.

It should be noted that this boron dilution evaluation was conducted by evaluating the time and water volumes required to dilute the spent fuel pool from 2000 ppm to 800 ppm. The 800 ppm endpoint was utilized to ensure that K_{eff} for the spent fuel racks would remain less than or equal to 0.95. As part of the criticality analysis for the spent fuel racks (Reference 1), a calculation has been performed on a 95/95 basis to show that the spent fuel rack K_{eff} remains less than 1.0 with non-borated water in the pool. Thus, even if the spent fuel pool were diluted to zero ppm, which would take significantly more water than evaluated above, the spent fuel would be expected to remain subcritical and the health and safety of the public would be assured.

5.0 REFERENCES

1. A-DP1-FE-0001,"Diablo Canyon Units 1 & 2 Spent Fuel Pool Criticality Analysis," rev. 00, February, 2001

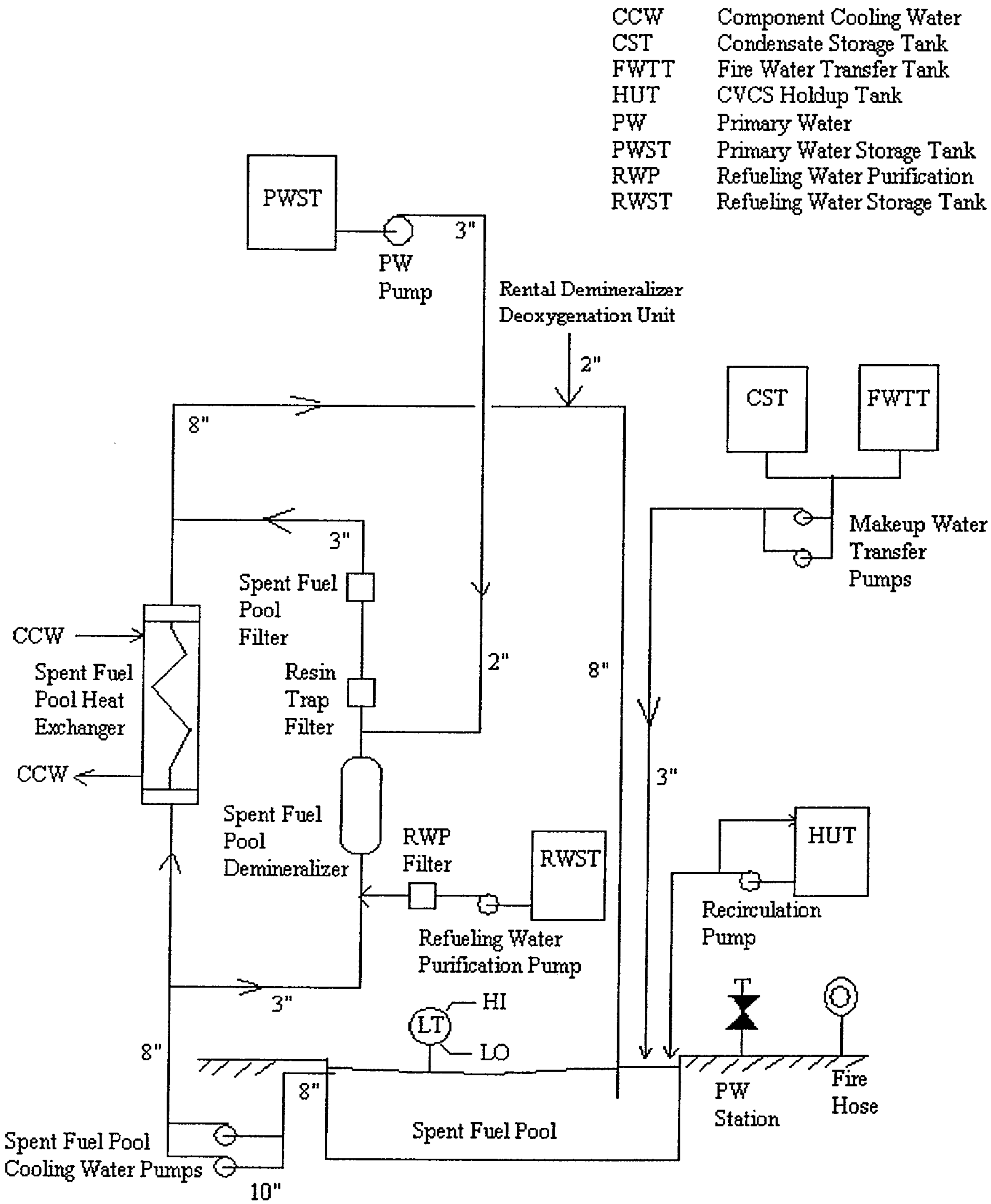

Figure 1 – Spent Fuel Pool and Related Systems

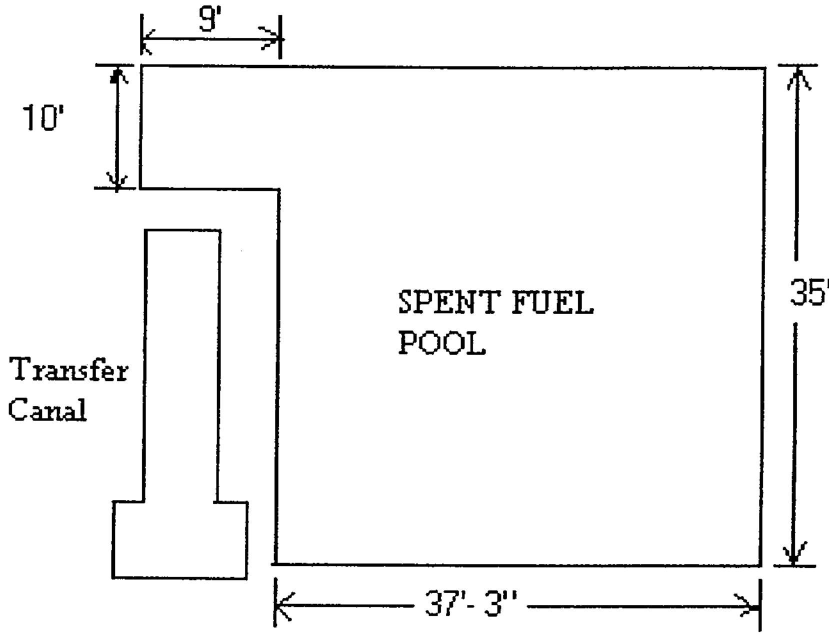

Figure 2 - SFP Plan View

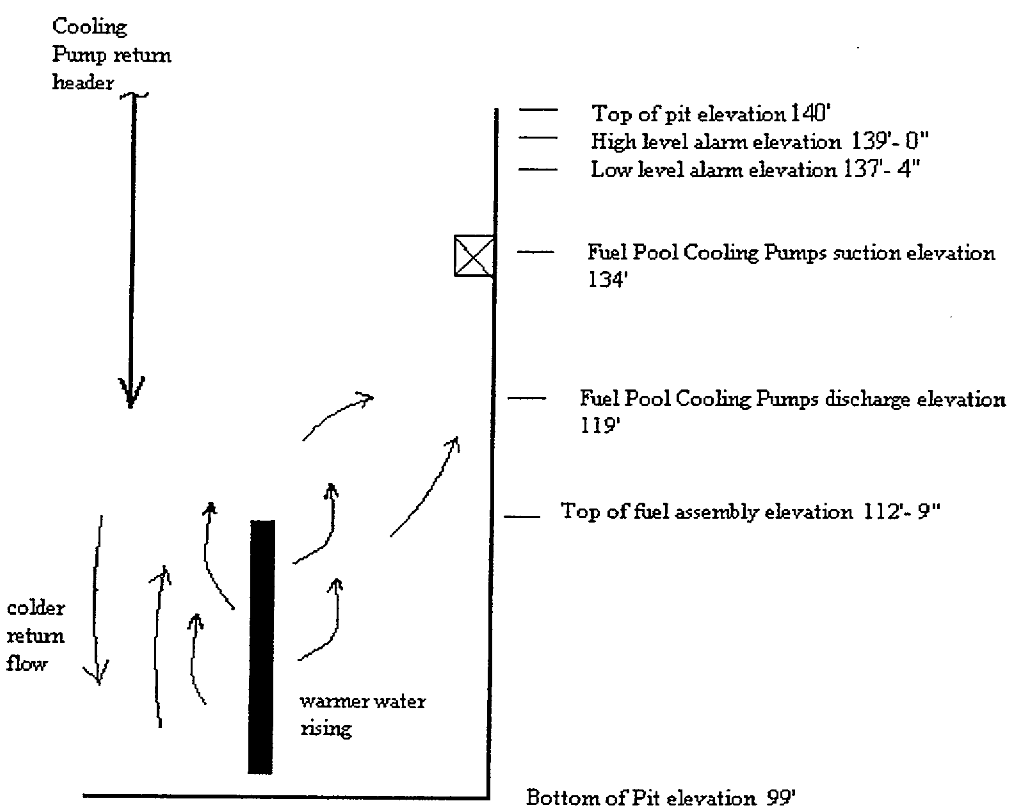

Figure 3 - SFP Mixing