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PG&E Letter DCL-04-149

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 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 04-07
Revision to Technical Specifications 3.7.17 and 4.3 for Cycles 14-16 for a Cask Pit Spent Fuel Storage Rack

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively. The enclosed license amendment request (LAR) proposes to revise Technical Specifications (TS) 3.7.17 and 4.3 for Cycles 14-16 to allow installation and use of a temporary cask pit spent fuel storage rack (cask pit rack) for Units 1 and 2. The cask pit rack would allow the storage of an additional 154 spent fuel assemblies. The total spent fuel pool (SFP) storage capacity for each unit would be increased to 1478 fuel assemblies for Cycles 14-16.

Based on the current inventory of fuel assemblies stored in the SFP and anticipated discharges of spent fuel, Unit 1 will lose full core offload capability (FCOC) in 2007, and Unit 2 will lose FCOC in 2008. On September 26, 2003, Pacific Gas and Electric Company (PG&E) received Amendment No. 162 to Facility Operating License No. DPR-80, and Amendment No. 163 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant Units 1 and 2. These amendments authorize handling and loading of Holtec International's (Holtec) multi-purpose canisters and transfer cask in the DCPP 10 CFR 50 facilities. On March 19, 2004, PG&E received a 10 CFR Part 72 license for an Independent Spent Fuel Storage Installation (ISFSI) at Diablo Canyon. PG&E is presently in the process of obtaining a Coastal Development Permit (CDP) from the California Coastal Commission, which is required to construct the ISFSI. The delay in the issuance of the CDP has resulted in a delay of the final design, construction, and projected operational date of the ISFSI, such that PG&E finds it necessary to provide additional temporary spent

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fuel storage to ensure that FCOC is retained in accordance with the current operational practice for both Units.

Enclosure 1 contains a description of the proposed change, the supporting technical analyses, and the no significant hazards consideration determination. Enclosures 2 and 3 contain marked-up and revised TS pages, respectively. Enclosure 4 provides the marked-up TS Bases changes for information only. TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Specifications Bases Control Program."

Proprietary and non-proprietary versions of the supporting Holtec Licensing Report HI-2043162, Rev. 1, for the new cask pit rack are provided as Enclosures 5 and 6, respectively. Enclosure 7 contains an affidavit signed by Holtec, the owner of the proprietary information in the report. The affidavit sets forth the basis on which the Holtec information contained in the subject report may be withheld from public disclosure by the Commission, and it addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. PG&E requests that the Holtec proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390.

PG&E has determined that this LAR does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), an environmental assessment does not need to be prepared since the proposed change does not involve a significant change in the types or in the amounts of any effluent that may be released offsite, or a significant increase in the individual or cumulative occupational radiation exposure.

The change in this LAR is not required to address an immediate safety concern. PG&E requests approval of this LAR be assigned a medium priority for review and approval and requests that the amendments be issued no later than October 2005 to allow PG&E to maintain FCOC. PG&E requests the LAR be made effective upon NRC issuance, to be implemented within 90 days of issuance.

If you have any questions or require additional information, please contact Mr. Terence Grebel at (805) 545-4160.

Sincerely,

for

David H. Oatley
Vice President and General Manager

tlg/4160
Enclosures

cc: Edgar Bailey, DHS
Bruce S. Mallett
David L. Proulx
Diablo Distribution

cc/enc: Girija S. Shukla

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT

James R. Becker, of lawful age, first being duly sworn upon oath says that he is Vice President Operations and Station Director – Diablo Canyon of Pacific Gas and Electric Company; that he has executed license amendment request LAR 04-07 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.

James R. Becker
Vice President Operations and Station Director

Subscribed and sworn to before me this 3rd day of November 2004.

Sandra Eatherly
Notary Public
County of San Luis Obispo
State of California

SE 11/3/04

SANDRA EATHERLY
Commission # 1425568
Notary Public - California
San Luis Obispo County
My Comm. Expires Jul 18, 2007

Affidavits

AFFIDAVIT PURSUANT TO 10CFR2.790

I, Charles W. Bullard II, being duly sworn, depose and state as follows:

(1) I am the Project Manager for Holtec International and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the document entitled "Spent Fuel Storage Expansion at Diablo Canyon Power Plant," Holtec Report HI-2043162, Revision 1. The proprietary material in this document is delineated by proprietary designation (i.e., shaded text) on pages 3-1 through 3-5, 3-10, 4-23, 4-25, 5-4, 5-5, 7-3 and 7-4.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.790(a)(4), and 2.790(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b, 4.d, and 4.e, above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures

outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed historical data and analytical results not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed using codes developed by Holtec International. Release of this information would improve a competitor's position without the competitor having to expend similar resources for the development of the database. A substantial effort has been expended by Holtec International to develop this information.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the

information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF NEW JERSEY)

) ss:

COUNTY OF BURLINGTON)

Charles W. Bullard II, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at Marlton, New Jersey, this 28th day of October 2004.

Chul W. Bullard II

Mr. Charles W. Bullard II
Holtec International

Subscribed and sworn before me this 28th day of October 2004.

Marie C. Pepe

NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2005

EVALUATION

1.0 DESCRIPTION

This License Amendment Request (LAR) proposes to amend Operating Licenses DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed changes would revise the Operating Licenses to amend Technical Specifications (TS) 3.7.17 and 4.3 for Cycles 14-16 to allow installation and use of temporary cask pit spent fuel storage racks (cask pit racks) for Units 1 and 2. The cask pit racks would allow the storage of an additional 154 spent fuel assemblies in each unit. The total spent fuel pool (SFP) storage capacity for each unit would be increased to 1478 fuel assemblies for Cycles 14-16.

2.0 PROPOSED CHANGE

This LAR proposes to revise the following two sections of the DCPP Units 1 and 2 Technical Specifications:

- Section 3.7.17, Plant Systems-Spent Fuel Assembly Storage
- Section 4.3, Design Features-Fuel Storage

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

1. TS 3.7.17: LCO 3.7.17 would be revised to add requirements during Cycles 14-16 for a cask pit rack.

2. Figures 3.7.17-1 to 4: Figures 3.7.17-1 to 3 would be changed to note that they are applicable to the permanent storage racks. Figure 3.7.17-4 would be added to provide acceptable/unacceptable enrichment and burnup requirements for the fuel stored in the cask pit rack for Cycles 14-16.

3. TS 4.3.1.1 would be revised to show applicability to the permanent SFP storage racks.

4. TS 4.3.1.3: This new section would add requirements for enrichment, K_{eff} for borated and unborated water, burnup, 10-year decay time, nominal center to center distance between fuel assemblies, and use of Metamic as the neutron absorbing material for the cask pit rack for Cycles 14-16.

5. TS 4.3.3: This section would be revised to clarify the maximum capacity in the permanent SFP storage racks, limit the number of spent fuel

assemblies that may be stored during Cycles 14-16 in the cask pit rack, and limit the total number of fuel assemblies that may be stored in the SFP permanent and cask pit racks.

In summary, the proposed changes would amend TS 3.7.17 and 4.3 for Cycles 14-16 to allow installation of cask pit rack in Units 1 and 2. The cask pit racks would allow the storage of an additional 154 spent fuel assemblies for each unit. The total SFP storage capacity for each unit would be increased to 1478 fuel assemblies for Cycles 14-16.

TS Bases 3.7.16 and 3.7.17 also would be revised to include the cask pit rack and the associated analyses that were performed for use of the cask pit rack during Cycles 14-16. The TS Bases changes are included for information only. TS Bases changes will be implemented pursuant to TS 5.5.14, "Technical Bases Control Program."

The proposed TS changes are noted on the TS marked-up pages provided in Enclosure 2. The revised TS are provided in Enclosure 3. The marked-up TS Bases are contained for information only in Enclosure 4.

3.0 BACKGROUND

3.1 Purpose for Proposed Amendments

Based on the current inventory of spent fuel assemblies stored in the SFP and anticipated discharges of spent fuel, Unit 1 will lose full core offload capability (FCOC) in 2007, and Unit 2 will lose FCOC in 2008.

On September 26, 2003, Pacific Gas and Electric Company (PG&E) received Amendment No. 162 to Facility Operating License No. DPR-80, and Amendment No. 163 to Facility Operating License No. DPR-82, for DCPP Units 1 and 2. These amendments authorize handling and loading of Holtec International's (Holtec) multi-purpose canisters and transfer cask in the DCPP 10 CFR 50 facilities. On March 19, 2004, PG&E received a 10 CFR Part 72 license for an Independent Spent Fuel Storage Installation (ISFSI) at DCPP. PG&E is presently in the process of obtaining a Coastal Development Permit (CDP) from the California Coastal Commission, which is required to construct the ISFSI. Delay in issuance of the CDP has resulted in delay of the final design, construction, and projected operation of the ISFSI, such that PG&E finds it necessary to provide additional temporary spent fuel storage to ensure that FCOC is retained in accordance with the current operational practice for both units until the ISFSI is completed.

3.2 Description of the Spent Fuel Pool and Racks

There are two SFPs located in the fuel handling building at DCPP, one for each unit. They are constructed of reinforced concrete. The overall dimensions of each pool are 48 feet wide, by 58 feet long, and 46 feet deep. The walls of the pool are nominally 5 to 6-feet thick. The foundation slabs have a minimum thickness of 5 feet and are founded on approximately 5 feet of lean concrete that rests on rock strata. The walls and floor of the SFP are lined with a stainless steel liner 1/8-inch and 1/4-inch thick, respectively. This liner serves only as a watertight boundary, not as a structural member.

Each SFP currently contains 16 freestanding spent fuel rack modules, containing a combined total of 1,324 fuel storage locations.

The cask pit racks, will utilize a fuel rack with a 12 by 13 configuration. Two cells of the rack have been eliminated resulting in a capacity of 154 storage cells. One cask pit rack will be installed in the cask pit area of the SFP for each unit, an approximately 10-foot by 10-foot-square area, which is recessed approximately 4.5 feet lower than the SFP floor. The cask pit, located in one corner of each SFP, is enclosed on two sides by extensions of the reinforced concrete SFP walls, and on the other two sides by the edges of the 5-foot-thick reinforced concrete floor of the SFP. The floor of the cask pit is a 4.75-foot-thick reinforced concrete slab, topped by 5 inches of concrete cover. Below the elevation of the SFP floor, the cask pit is lined with 1/4-inch-thick stainless steel plate. In addition, a 3/4-inch-thick carbon steel plate is provided as backing for the stainless steel liner in the floor of the cask pit.

An existing welded stainless steel spent fuel transfer cask restraint frame will prevent overturning of a transfer cask while in the cask pit, protect the adjacent spent fuel racks from interaction with a transfer cask, and prevent sliding of the spent fuel racks into the cask pit during a seismic event. The restraint frame, located approximately 12 feet above the SFP floor, is anchored to two SFP walls.

In order for the top of the cask pit rack to be at a uniform elevation with the existing high density spent fuel racks, a stainless steel support platform (platform) will be installed in the cask pit. The platform is designed with shims on each side to ensure a snug fit with the four cask pit walls in order to anchor the platform and preclude any differential movement between the cask pit walls and the platform. The four corners of the platform are equipped with machined recesses that are coaxial with, and approximately 3/4-inch larger in diameter than, the 5-inch-diameter rack support pedestals. Thus, when the cask pit rack is placed on the platform, sliding of the cask pit rack is limited to 0.375 inches. Two "connector links," made

of precipitation-hardened stainless steel, are installed through storage cells adjacent to each of the four pedestal locations. These "connector links" prevent vertical separation of the cask pit rack and the platform, and excessive horizontal displacement at the top of the cask pit rack during a seismic event. See Enclosure 5 for drawings of the cask pit rack and support platform.

This proposed license amendment would allow an additional use of the cask loading pit as a temporary storage location for relatively low enrichment, high burnup, spent fuel, which has been discharged from the reactor for at least 10 years. Because the cask pit will eventually be needed for loading spent fuel into transfer casks, the cask pit rack will be removed prior to any spent fuel cask loading operations. The platform will remain in the cask pit following removal of the cask pit rack to be used for cask handling.

4.0 TECHNICAL ANALYSIS

4.1 Heavy Loads

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines and recommendations to assure safe handling of heavy loads by prohibiting, to the extent practicable, heavy load travel over stored spent fuel assemblies, fuel in the reactor core, safety-related equipment, and equipment needed for decay heat removal.

The 125-ton-rated Fuel Handling Building (FHB) crane will be used for all heavy loads handling of the cask pit rack and the support platform within the FHB.

The maximum lift weight during installation and removal of the cask pit rack is as follows:

Item	Weight (lbs)
Rack	26,000
Lift rig	1,800
Rigging	500
Total Lift	28,300

The maximum lift weight during installation of the platform is as follows:

Item	Weight (lbs)
Platform	22,825
Lift rig	2,300
Rigging	500
Total Lift	25,625

Therefore, the FHB crane is qualified to handle the weight of the new cask pit rack and platform. As discussed below, prior to installation of the new cask pit rack and associated platform, PG&E intends to upgrade the FHB crane in accordance with the implementation guidelines of NUREG-0612, Appendix C, under the provisions of 10 CFR 50.59, which is one acceptable method of meeting NUREG-0612, Section 5.1.2.

Pursuant to the defense-in-depth approach of NUREG-0612, the following additional safety measures will be undertaken for the cask pit rack installation and removal, and platform installation activities.

The rack designer, Holtec, will develop a set of inspection points which have proven to produce high quality work in numerous prior re-rack projects. Surveys and measurements are performed on the cask pit rack prior to and subsequent to placement into the cask pit to ensure that the as-built dimensions and installed locations will be acceptable.

Measurements of the cask pit and floor elevations were performed to determine the actual pool configuration and to allow height adjustments of the support platform prior to rack installation. These inspections will minimize rack manipulation during placement into the SFP.

4.1.1 NUREG-0612 Section 5.1.1

NUREG-0612 endorses a defense in-depth approach for the handling of heavy loads near spent fuel and safe shutdown systems. General guidelines for overhead handling systems that are used to handle heavy loads in the area of the reactor vessel and SFP are given in Section 5.1.1 of NUREG-0612. They are as follows: (1) definition of safe load paths; (2) development of procedures for load handling operations; (3) training and qualification of crane operators in accordance with Chapter 2-3 of American National Standards Institute (ANSI) B30.2-1976; (4) use of special lifting devices that meet guidelines in ANSI N14.6-1978; (5) installation and use of noncustom lifting devices in accordance with ANSI B30.9-1971; (6) inspection, testing, and maintenance of

cranes in accordance with Chapter 2-2 of ANSI B30.2-1976; and (7) design of crane in accordance with Chapter 2-1 of ANSI B30.2-1976 and Crane Manufacturers Association of America document CMAA-70. Section 5.1.2 of NUREG-0612 provided additional guidelines for control of heavy loads in the SFP area of Pressurized Water Reactors.

DCPP's Control of Heavy Loads Program for meeting NUREG-0612 Section 5.1.1 is discussed below.

DCPP defines a heavy load as a load whose weight is greater than the combined weight of a single fuel assembly and its handling tool. Since the dry weight of the cask pit rack is 26,000 lbs. and the dry weight of the platform is 22,825 lbs., installation and removal of the cask pit rack and installation of its associated platform will involve handling of heavy loads in the vicinity of the SFP. This process will be performed consistent with PG&E's Heavy Loads Program commitments. In Supplemental Safety Evaluation Report (SSER) 27 and SSER-31, the NRC staff concluded that the Diablo Canyon program for control of heavy loads is in compliance with the guidelines of NUREG-0612.

Safe Load Paths

Safe load paths for the limiting load in the FHB are shown in FSAR Figures 9.1-3 and 7. Movement of the cask pit rack and platform will follow these general safe load paths. The cask pit rack will enter the FHB through the roll-up door into the receiving area of the cask wash down area (CWA). The rack module will be removed from the shipping trailer in the horizontal position and then uprighted into a vertical position using lifting devices meeting NUREG-0612 requirements. The cask pit rack and platform will not be suspended over any portions of the SFP containing spent fuel assemblies.

PG&E intends to vacate a minimum of one row of cells in the adjacent permanent racks. Vacating one row surrounding the cask area, combined with the separation provided by the cask seismic restraint, will create a horizontal separation distance between fuel assemblies stored in the pool and the projected vertical lift envelope of the cask pit rack or platform. This distance will provide a margin to ensure that a postulated drop will not impact stored fuel.

Procedures

PG&E procedures covering the handling of heavy loads will be revised as necessary and new procedures will be developed, specifically for the cask pit rack, platform and related heavy load lifts and handling in accordance with PG&E's program requirements.

Procedures for installation and removal of the cask pit rack and installation of the platform will be prepared incorporating experience gained by Holtec previous rack installation and removal projects. Procedures will be developed for the cask pit rack installation and removal for the following activities: 1) mobilization; 2) rack handling; 3) upending; 4) lifting; 5) installation; 6) alignment; 7) dummy gage testing; and 8) ALARA.

These procedures will be comprehensive with respect to load handling, exclusion areas, equipment required, inspection and acceptance criteria before load movement, and steps/sequence to be followed during load movement, as well as safe load paths and special precautions.

Crane Operators

PG&E personnel are required to be trained and qualified for the tasks they perform, including crane operators. This training will be supplemented with training prepared by Holtec specific to the rack installation and removal based on experience in previous rack installation/removal projects. The augmented training will also include instruction on the special lifting devices, heavy load exclusion areas, safe load paths and equipment testing requirements. The training will be completed prior to any heavy load movements of the cask pit rack and its associated platform. This training and qualification meets the requirements of ANSI B30.2-1976.

Special Lifting Devices

The rack and associated platform will be suspended from the FHB crane main hook by Holtec-designed lifting rigs.

These remotely engageable lift rigs, meeting NUREG-0612 stress criteria, will be used to lift the cask pit rack and support platform. The rigs consist of four independently loaded tension rods attached to a frame assembly. The tension rods are designed to prevent loss of engagement with the rack or platform in the locked position.

Moreover, the locked configuration can be directly verified from above the pool water without the aid of a camera.

The stress analyses were performed to demonstrate that the primary stress limits provided in ANSI 14.6 (1978) are met. The individual tension rods have a safety factor of greater than 10. If one of the rods break, the load will continue to be supported by at least two rods, which have a safety factor of greater than 5 against ultimate strength. This ensures that failure of one tension rod will not result in uncontrolled lowering of the load being carried by the rig (which complies with the duality feature in Section 5.1.6(3a) of NUREG-0612).

The lifting rig is load-tested with 300 percent of the maximum weight to be lifted. The test weight is maintained in air for 10 minutes. All critical weld joints are liquid penetrant examined to establish the soundness of all critical joints.

The DCPP cask pit rack-lifting rig is similar to the rigs used in the initial SFP rack installation or re-racking of numerous other plants, such as Hope Creek, Millstone Unit 1, Indian Point Unit 2, Fitzpatrick, and Three Mile Island Unit 1.

General Lifting Devices

The installation and removal of the cask pit rack and platform will not require the use of general lifting devices.

Crane Inspection Testing and Maintenance

DCPP's crane maintenance program meets the requirements of Chapter 2-2 of ANSI B30.2-1976 and NUREG-0612, Section 5.1.16.

Fuel Handling Building Crane Design

PG&E previously described the FHB crane design and qualification in its December 5, 1984, NUREG-0612 submittal. The crane was procured before NUREG-0612 was issued, but it is consistent with the intent of ANSI/CMAA specifications as described and accepted in the previously referenced submittal. As discussed above, in SSER-27 and SSER-31, the NRC staff concluded that the Diablo Canyon program for control of heavy loads is in compliance with the guidelines of NUREG-0612.

4.1.2 NUREG-0612, Section 5.1.2

NUREG-0612, Section 5.1.2 recommends that in addition to satisfying the general guidelines of Section 5.1.1, one of four criteria be met. Prior to installation of the new cask pit rack and associated platform, PG&E intends to upgrade the FHB crane in accordance with the implementation guidelines of NUREG-0612, Appendix C, under the provisions of 10 CFR 50.59, which is one acceptable method of meeting NUREG-0612, Section 5.1.2. In the event that the FHB cranes are not fully compliant with NUREG-0612, Appendix C, at the time of rack/platform installation, PG&E has also performed heavy load drop analyses to demonstrate conformance with NUREG-0612, Section 5.1.2.4. These analyses have been performed in accordance with NUREG-0612, Appendix A, and are also an acceptable method of meeting NUREG-0612, Section 5.1.2. In summary, PG&E will meet the guidance of NUREG-0612, Section 5.1.2 by either installation of a single failure-proof crane or by demonstration that heavy loads have been analyzed and satisfy the criteria of NUREG-0612, Section 5.1.

4.1.3 Heavy Loads Conclusions

Based on the PG&E heavy loads program, there is adequate assurance that the planned actions for the installation of the cask pit racks and platforms are consistent with the "defense-in-depth" approach to safety in the handling of heavy loads described in NUREG-0612, Section 5.1.1. PG&E will meet the guidance of NUREG-0612, Section 5.1.2, by either installation of a single failure-proof crane or by demonstrating that heavy loads have been analyzed and satisfy the criteria of NUREG-0612, Section 5.1.

4.2 Seismic and Structural Design

This section summarizes the structural analyses performed for the new cask pit rack and platform, which will be located in the cask pit area of the DCPP SFP. Evaluations of the auxiliary building, SFP, and liner are also included. More detailed information is contained in Enclosure 5.

The structural evaluations of the cask pit rack, platform, SFP walls, SFP floor, and SFP liner considered loads due to dead weight, live loads, hydrostatic and hydrodynamic forces, seismic inertia, thermal expansion, and mechanical accidents. The loads, load combinations and acceptance criteria for the cask pit rack, platform, and liner were based on the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NF, and on NUREG-0800, Standard Review Plan (SRP), Sections 3.7.1, 3.7.2,

3.7.3, and 3.8.4, Appendix D. The loads, load combinations, and acceptance criteria for the SFP walls and floor were in accordance with the DCPP FSAR Update and the American Concrete Institute building codes, ACI 318-63 and ACI 349-80.

4.2.1 Cask Pit Rack and Platform Structural Evaluation for Seismic Events

The analyses of the cask pit rack and platform use the DCPP licensing basis load combinations, acceptance criteria and methodology summarized in the DCPP FSAR Update. There are three design ground motions for the DCPP:

- Design Earthquake (DE),
- Double-Design Earthquake (DDE), and
- Hosgri Earthquake (HE).

As discussed in the DCPP FSAR Update, the seismic qualification basis for the plant is the two original design basis earthquakes (DE and DDE), plus the HE evaluation, along with their respective analytical methods, acceptance criteria, and initial conditions. In addition, the cask pit rack and support platform were evaluated for the Long Term Seismic Program (LTSP) earthquake ground motions.

Dynamic simulations of the seismic response of the cask pit rack and support platform were performed to demonstrate that the seismically induced stress levels meet the acceptance criteria, and to develop the interface loads on the support platform, SFP floor slab, walls, and liner.

Seismic input was based on synthetic acceleration time histories for the three orthogonal directions (N-S, E-W, and vertical), developed in accordance with the provisions of NUREG-0800, SRP, Section 3.7.1, for each of the three design basis earthquakes (DE, DDE, and HE). The synthetic accelerations time histories cover the full period ranges defined for the DE, DDE, and HE, up to 1.0 sec., 1.0 sec., and 0.8 sec., respectively. A synthetic acceleration time history covering the full period range defined for the LTSP up to 2.0 sec. was also developed. The maximum natural period of the cask pit rack and platform assembly, 0.67 sec., falls within these ranges.

The structural damping values used in the generation of the synthetic time histories are based on the values prescribed in the FSAR. No credit for material or fluid damping is incorporated in the time history generation for conservatism.

The DYNARACK computer program was used to simulate the dynamic behavior of the complex cask pit rack and platform. A three-dimensional dynamic model of the rack and platform assembly was developed to be suitable for time history analysis, and included fluid coupling and mechanical coupling effects. Stress and fatigue analyses of the highly stressed areas were performed, using loads from the limiting rack dynamic analysis cases, in order to demonstrate compliance with the ASME B&PV Code, Section III, Subsection NF, and Subsection NB limits, respectively. Additional information on the modeling methods and assumptions are provided in Enclosure 5.

The results of the dynamic simulations demonstrate that the cask pit rack does not impact the SFP walls or the cask seismic restraint framework. This is due to the high bending rigidity of the honeycomb cellular design of the cask pit rack that prevents any significant flexural bending of the rack, positive mechanical connections between the rack and platform, and the anchorage provided by the shimming of the platform within the cask pit recess. The physical separation provided by the cask seismic restraint framework prevents interaction of the cask pit rack with the adjacent fuel storage rack modules.

The stress levels predicted for the cask pit rack and platform satisfy the applicable criteria of the ASME B&PV Code, Section III, Subsection NF. The calculated cumulative usage factors (CUF) associated with the fatigue analysis of the cask pit rack and platform assembly, considering the combination of twenty DE level events and one DDE/HE/LTSP level event satisfy the applicable criteria of the ASME B&PV Code, Section III, Subsection NB.

4.2.2 Auxiliary Building and Spent Fuel Pool Structural Evaluation for Seismic Events

A structural evaluation was performed to determine the effect that the fully loaded cask pit rack would have on the auxiliary building structure, including the SFP concrete and SFP liner.

Loads applied to the structural analysis and structural capacity assessments were done in accordance with the following:

- The loading in the cask pit area included the dead weight of the platform and cask pit rack fully loaded with fuel assemblies.
- ACI 318-63 is used for the allowable concrete bearing pressure under rack dead loads.
- ACI 349-80 is used for the allowable concrete bearing pressure due to rack impact.
- The allowable foundation pressure is in accordance with FSAR, Section 3.8.4.1.4.
- The allowable liner strain for normal operating and accident thermal conditions are in accordance with ASME Section III, Division II, 1983.

Auxiliary Building Evaluation

The auxiliary building, which includes the SFPs, has been seismically qualified using the criteria outlined in Chapter 3 of the DCPP FSAR update. As described in FSAR, Section 3.7.2.1.7.1, the seismic inertia loads were obtained using time-history analyses of spring and lumped mass models (two horizontal and one vertical) of the auxiliary building. A detailed analytical static model of the auxiliary building was then used to distribute seismic inertia forces and moments to various walls, diaphragms, and columns, as described in FSAR, Section 3.8.2.4. The effect of the change in weight on the seismic models due to the cask pit rack is considered to be insignificant, since the increase in global mass is determined to be less than 1.5 percent. Therefore, there is no change in the seismic responses and forces reported in the FSAR.

Spent Fuel Pool Evaluation

The following load conditions were assessed to evaluate any potential impact from the addition of the cask pit rack and associated support platform:

- Dead load: the cask pit floor is subjected to the dead load of the loaded cask pit rack and platform. This load is applied as a bearing pressure on the platform/floor contact surface.

- Hydrostatic load: this load exerted by the column of water (43 ft nominal height) acting on the cask pit floor, liner, and walls. This is considered to act as a dead load and is not affected by the installation of the cask pit rack and associated platform.

- Thermal load: thermal expansion of the platform will produce horizontal reactions, normal to the cask pit walls, acting at the locations of the platforms.

- Seismic loads: seismic inertial effects on the mass of the cask pit rack and platform will produce lateral and vertical reaction on the cask pit walls at the locations of the shims and on the cask pit floor at the base of the platform. These reactions will be transmitted into the SFP walls and basemat. Note that since it has been demonstrated in the seismic analysis of the cask pit rack (see Section 4.2.1) that there is no impact between the rack and SFP walls or cask restraint framework, seismic loads are applied to portions of the SFP within the cask pit recess.

The combination of the dead weight, thermal expansion, and seismic inertial loads from the cask pit rack and platform will produce additional loads on the stainless steel liner and reinforced concrete SFP structure. These loads are in the form of local bearing and shear, which are combined with the global loading already considered in the design of these elements.

The SFP floor slab, walls and liner plate have been evaluated and found to be adequate to support and transfer the reaction loads from the rack and platform. Stresses in the wall and floor slab are within bearing and shear allowables. Strains in the liner plate are within normal and shear strain allowables. In addition, stresses in the bedrock are within allowable bearing pressure.

4.3 Thermal-Hydraulic Considerations

4.3.1 Spent Fuel Pool Cooling System

The SFP pool cooling system configuration and design basis are described in the DCPP FSAR Section 9.1.3. Each unit has a completely independent SFP cooling and cleanup system. The design of the cooling loop conforms to Design Class I piping criteria. The SFP pool cooling systems consist of two parallel full-capacity pumps discharging to a single shell and tube heat exchanger. Only one pump is operated at a time, providing for a single active failure without any reduction on system capacity.

Although there are no Class 1E electrical loads in the SFP system, the SFP cooling pumps are powered from a Class 1E source.

The SFP cooling system heat exchanger is cooled by the component cooling water system, which in turn is cooled by the auxiliary saltwater system that rejects waste heat to the Pacific Ocean. The SFP water is pumped from the pool through the tube side of the SFP heat exchanger and returned to the pool. The pump suction line is protected by a strainer and is located four feet below the normal SFP water level. The connections to the SFP are provided with anti-siphon devices to preclude possible draining of the pool water. The piping of the SFP cooling system is arranged so that failure of any pipe will not drain the SFP below the level required for acceptable radiation shielding.

4.3.2 Current SFP Cooling System Licensing Basis

The current licensing basis is summarized in DCPP FSAR Update Section 9.1.3.1.1.

The existing analyses supporting the SFP cooling system licensing basis include evaluations of a partial core offload case of 76 assemblies, a partial core offload case of 96 assemblies, and a full core offload case of 193 assemblies. All scenarios evaluated assumed that the core offload starts at 100 hours after reactor shutdown and offload at a rate of four fuel assemblies per hour. The results of the analyses are summarized below.

Core Offload Scenario	**Previous Fuel Discharges**	**Peak Bulk SFP Temperature**
76 Assembly – Partial Offload	15 Cycles of 76 Assemblies Discharged at 18 Month Cycles	Less than or equal to 140°F
96 Assembly Partial Offload	12 Cycles of 96 Assemblies Discharged at 24 Month Cycles	Less than 150°F
193 Assembly – Full Core Offload	15 Cycles of 76 Assemblies Discharged at 18 Month Cycles	Less than 175°F

As discussed in FSAR Update, Section 9.1.3.1.1, DCPP's normal refueling is a full core offload. Administrative controls are in place to ensure that the bulk spent fuel pool temperature remains below 140°F for any offload scenario.

Analyses were also performed using conservative assumptions to determine the maximum local water temperatures and maximum fuel cladding temperatures for the above offload scenarios. For the 76-assembly partial offload case, the analysis showed that the maximum local water temperature was 188°F and the maximum cladding temperature was 225°F. For the 96-assembly partial offload case, the analysis showed that the maximum local water temperature was 194°F and the maximum cladding temperature was 231°F. For the full core offload case of 193 assemblies, the analysis showed that the maximum local water temperature was 220°F and the maximum cladding temperature was 254°F. For the limiting full core offload scenario, the maximum cladding temperature would result in localized nucleate boiling but not bulk pool boiling.

A complete loss of the SFP cooling system for an extended period of time was evaluated. For this condition natural surface cooling will maintain the water temperature at or below the boiling point. The analyses demonstrated that boiling would not occur for approximately 9 hours for partial core offload scenarios and would commence after 2.5 hours for a full core offload scenario, allowing sufficient time for corrective operator actions. Determination of maximum boil-off rates were also calculated and shown to be well within the capabilities of the SFP makeup water sources.

4.3.3 Cask Pit Rack Thermal Hydraulic Analyses

This section summarizes the thermal-hydraulic analysis performed by Holtec, the rack vendor, to determine the peak SFP bulk temperatures, maximum local water, and fuel assembly cladding temperatures with a new cask pit rack installed on each unit. Additionally, calculations of minimum time-to-boil and maximum boil-off rates were performed. A more detailed discussion of the thermal-hydraulic analysis methodology, assumptions, and results is included in Section 5.0 of Enclosure 5.

The cask pit rack will be installed during Cycle 14, prior to the 14th refueling outage, and will be removed during Cycle 16, prior to the 16th refueling outage. The cask pit rack thermal-hydraulic analyses are based on the evaluation of three offload scenarios that bound the past and future operating practices at DCPP: 1) a

partial core offload scenario; 2) a full core offload; and 3) an emergency full core offload 36 days after completion of a refueling outage. All scenarios are evaluated to occur during or, in the case of the emergency offload scenario, shortly after the 15th refueling outage. The number of irradiated fuel assemblies assumed to be stored in the SFP in each of the evaluated scenarios conservatively bounds the actual number of irradiated fuel assemblies that will be stored in the SFP when the cask pit rack is installed. Following the removal of the cask pit rack, the number of irradiated fuel assemblies assumed to be in the SFP in all of the above scenarios, exceeds the actual capacity of the SFP.

The partial core offload scenario assumes a discharge of 96 fuel assemblies during the 15th refueling outage. All of the 96 fuel assemblies offloaded are conservatively assumed to have a burnup of 52000 MWD/MTU.

The full core offload scenario assumes a discharge of 193 fuel assemblies during the 15th refueling outage. The 193 offloaded assemblies are separated into two distinct groups; 101 assemblies with 52000 MWD/MTU burnup and 92 assemblies with 25000 MWD/MTU burnup.

The emergency offload scenario assumes that the 15th refueling outage is completed in 30 days, leaving 104 assemblies in the SFP at restart. After 36 days of operation at 100 percent power in Cycle 16, an emergency full core offload is performed, completely filling all available storage locations. The 193 assemblies are separated into two distinct groups: 113 assemblies with 40000 MWD/MTU burnup and 80 assemblies with 3000 MWD/MTU burnup.

All of these scenarios have been evaluated with a base decay heat load contribution from previously discharged fuel assemblies using actual operational data for operating Cycles 1 through 11. The contribution to the base decay heat load from fuel that will be discharged in Cycles 12 through 14 is based on an assumed discharge of 104 assemblies each Cycle using bounding assumptions on fuel assembly burnup and operating power. Cycle lengths assumed for Cycles 12 through 14 are assumed to be 18 months, which conservatively minimizes the decay time and maximizes the base decay heat load.

The transient decay heat contribution for each offload scenario is determined using the Holtec QA-validated computer program BULKTEM, which incorporates the ORIGEN2 code for performing

decay heat calculations. All three of these scenarios assume a core offload rate of four assemblies per hour, starting 100 hours after reactor shutdown, and other appropriately conservative fuel assembly discharge and burnup assumptions.

4.3.3.1 SFP Cooling System Performance Data

The calculated heat transfer rate from the SFPCS to CCW varies with time as a function of several independent variables, including flow rates, temperatures, and heat exchanger fouling and tube plugging. The SFPCS pump and heat exchanger performance data used in the SFP bulk temperature calculations for both units are discussed in Section 5.0 of Enclosure 5.

Conservative values for pump flow and heat exchanger performance were selected to provide bounding calculations for the peak SFP bulk temperature. The thermal performance of the heat exchangers was determined with all heat transfer surfaces assumed to be fouled to their design basis maximum levels and also included an allowance for 5 percent tube plugging. CCW supplied to the heat exchanger was assumed to be 75°F at a flow rate of 3400 gpm.

4.3.3.2 SFP Decay Heat Load

The SFP bulk temperature analysis requires quantifying the total decay heat load as a function of time after reactor shutdown and core offload time. The total decay heat load imposed on the SFP cooling system was evaluated as the sum of two decay heat sources: decay heat from previous offloads already stored in the pool (assumed to be constant), and decay heat from the fuel assemblies recently offloaded from the reactor (variable with time after the reactor shutdown).

The steady-state decay heat load from previously offloaded fuel was calculated using the LONGOR computer program. Inputs to the program were based on known power histories for fuel discharged in Cycles 1 through 11 and a projected fuel offload schedule for Cycles 12 through 14 that conservatively bounds both fuel assembly burnup and the number of fuel assemblies to be offloaded. The decay heat contribution for fuel discharged in Cycles 1 through 11 was based on Unit 2 data since Unit 2 contained more spent fuel

assemblies than Unit 1 following the 11th refueling outages (928 versus 900) and also operated at a higher licensed thermal power than Unit 1 (3411 vice 3338 MWth) during this same time period. Additionally, the assumed reactor thermal power was increased by 5 percent to account for burnup uncertainties. This results in a conservatively high estimate of the base decay heat load, which bounds either Unit.

As discussed earlier, the transient decay heat contribution for each offload scenario is determined using the Holtec QA validated computer program BULKTEM, which incorporates the ORIGEN2 code for performing decay heat calculations.

Each offload is assumed to start at 100 hours after reactor shutdown with an assumed offload rate of 4 assemblies per hour. As described in Section 5.0 of Enclosure 5, conservative assumptions were made with respect to operating power and fuel burnup to determine a bounding decay heat load contribution for the offloaded fuel.

For each scenario, the transient and steady state decay heat loads were then combined to provide a total decay heat load on the SFP cooling system.

4.3.3.3 Maximum SFP Bulk Temperatures

The SFP bulk temperature versus time was calculated for each of the three core offload scenarios using the BULKTEM computer program, based on the time-varying total decay heat load on the SFP cooling system. The calculations also considered passive heat losses to the air above the pool and included several conservative assumptions regarding heat exchanger fouling and tube plugging, SFP thermal capacity, reactor power, and bounding core offload parameters. These assumptions are discussed in Section 5.0 of Enclosure 5.

The partial core offload analysis resulted in a maximum pool bulk temperature of 127°F. The full core offload analysis resulted in a maximum pool bulk temperature of 157°F. Both of the scenarios resulted in maximum pool bulk temperatures less than the analyses supporting the current licensing basis, i.e., 150°F for partial offloads and 174°F for a full core discharge. The emergency core offload analysis resulted in a peak bulk temperature of less than 162°F. Although not part of the current licensing basis, the

emergency offload results were less than the previous analysis results for a full core discharge (174°F).

Assumptions, conservatisms, key input data, and the modeling methodologies are contained in Sections 5.3 and 5.4 of Enclosure 5.

Therefore, with the new cask pit racks installed, the SFP peak bulk temperatures are less than the previous licensing basis analyses for a partial core offload (150°F) and a full core offload (174°F) and no SFP boiling occurs for any offload scenario. As with the current licensing basis, PG&E will continue to maintain administrative controls in place to ensure that peak SFP temperatures remain below 140°F during the normal full core offload scenario.

4.3.3.4 Minimum Time-to-Boil and Maximum Boil-off Rate

The time-to-boil evaluation assumed that forced cooling was lost the moment the peak SFP bulk temperature for each case was reached. The SFP time to boil and corresponding maximum boil-off rates were then determined.

Table 5.8.2 of Enclosure 5, provides the calculated minimum time-to-boil and maximum boil-off rates for each scenario. For the worst-case scenario, the emergency core offload, the calculated time-to-boil was determined to be 3.76 hours after a loss of forced cooling at the peak SFP bulk temperature. The corresponding maximum boil-off rate for this condition was approximately 87 gpm. Both of these values are bounded by the current thermal-hydraulic analysis values of 2.5 hours and 44,905 lbm/hr (~93.6 gpm).

A loss of SFP cooling on each unit will be annunciated in the control room by alarms for high SFP temperature and low SFP water level. The temperature-indicating channel presently has a high alarm setpoint of 125°F. The level-indicating channel has alarms with a high setpoint of 16 inches above normal water level and a low setpoint of 4 inches below normal level. Both of these transmitters are powered from instrument AC sources, which are normally powered from Class 1E vital buses.

The worst-case time-to-boil scenario is 3.76 hrs. Given the conservatisms incorporated into the calculations, actual times-to-boil will be higher than these calculated values and

actual boil-off rates will be lower than calculated. Based on the time-to-boil, plant personnel will have sufficient time to identify elevated SFP temperatures and adequate time to provide makeup to the SFP, if needed.

Makeup water to the pool in the event of a loss of normal cooling is available from various sources as discussed in the DCPP FSAR Update Section 9.1.3.2. Demineralized makeup water can be added directly to the SFP by a Design Class 1 source. Water from the condensate storage tank is pumped to the SFP using the makeup water transfer pumps and the appropriate interconnecting piping and valves. This source has the capability of providing up to 200 gpm of demineralized water, if required. The transfer tank is another Design Class I source of pool makeup, and water can be pumped to the pool by the makeup water transfer pumps. However, the flow path from the transfer tank is completely Design Class I. Additional information of makeup water sources to the SFP is described in DCL-86-020, dated January 28, 1986.

Therefore, makeup to the SFP from a design Class I source is available at flow rates well in excess of that required to make up for boil-off under the worst case offload scenario with a loss of forced cooling. Based on the number and diverse sources of makeup water available, loss of pool makeup capability is not considered credible.

4.3.3.5 <u>Maximum Local Temperatures</u>

The maximum local water and fuel cladding temperatures that may occur in the SFP following the installation of the cask pit rack were determined. The discussion of the maximum local temperature analyses contained in Section 5.6 of Enclosure 5 are summarized below.

The flooded cask pit area for both units is open to the SFP allowing a free exchange of cooling water between the pool and pit areas. The bounding peak local temperature in the fuel rack cell containing the hottest spent fuel assembly must be less than the local saturation temperature of water at the rack depth, and the bounding peak fuel cladding temperature for the hottest fuel assembly should be less than the local saturation temperature of water. If the cladding temperature exceeds the local saturation temperature, then departure from nucleate boiling (DNB) is not permitted to occur.

The close hydraulic coupling between the cask pit and the SFP allows local temperature analysis to model the cask pit rack in a rectangular pool created by combining the SFP and cask pit, using the FLUENT fluid flow and heat transfer modeling program. Quantification of the coupled flow and temperature fields between the cask pit and the SFP was accomplished through use of a computational fluid dynamics (CFD) analysis using FLUENT. The loaded rack internal flow characteristics for the three-dimensional model were chosen based on hydraulic resistance parameters more conservative than the most limiting rack design and fuel assembly type. Although fuel stored in the cask pit rack is limited to fuel assemblies that have a minimum decay time of 10 years, the analysis was performed to bound freshly discharged fuel stored in either the existing racks or the cask pit storage rack. The volumetric decay heat generation rates for the hottest fuel assemblies were extracted from the pool bulk temperature analysis.

To determine the maximum local water temperature in the rack, a single bounding scenario was then evaluated using FLUENT that included the highest bulk SFP temperature and decay heat loads, the highest fuel assembly hydraulic resistance, and the additional resistance of an assumed dropped fuel assembly laying across every cell in the rack. A separate calculation was performed to determine the maximum fuel clad superheat, which was then added to the maximum local water temperature to determine the peak fuel cladding temperature.

The results of the local temperature analysis demonstrate that the calculated worst-case peak local water temperature (188°F) is below the local saturation temperature at the water depth of the cask pit rack (240°F). The results also demonstrate that the peak fuel cladding temperature (213°F) for the hottest fuel assembly is also below the local saturation temperature and that the critical heat flux for DNB is not exceeded. Therefore, no bulk boiling will occur in the cask pit rack and the local water and fuel temperatures are acceptable.

4.3.4 Administrative Controls

Plant procedures are currently in place to limit the peak SFP temperature to within the 140°F limit discussed in DCPP FSAR Update Section 9.1.3.1.1. The procedural controls currently

suspend offload activities at a SFP temperature of 125°F to maintain peak SFP bulk temperatures less than 140°F. Past operating experience at DCPP has shown that peak SFP temperatures are typically less than 115°F during a typical full core offload.

Due to the many variables that can have an impact on peak SFP temperature, DCPP may elect to use a cycle specific offload analysis in lieu of the operating restrictions of the bounding thermal analyses described above. Consideration will be given to the actual core power history, scheduled offload start time, offload rates, actual CCW temperature, and actual CCW and SFP cooling water flow rates to the SFP heat exchanger in the establishment of the specific control values. If DCPP elects to use a cycle specific analysis, plant procedures will require that core offload be suspended at a temperature, which will ensure that the 140°F limit is not exceeded.

4.3.5 Revised DCPP SFP Thermal Licensing Basis

As discussed above, PG&E has updated its SFP thermal-hydraulic analyses as part of the cask pit rack project. These updated analyses were performed using more recent analytical methods that have been previously accepted by the NRC as part of other plant SFP licensing actions. These analytical methods and the associated full core and emergency offload scenarios discussed above will bound both the installation of the cask pit rack and future DCPP SFP fuel storage requirements once the cask pit rack has been removed.

In addition to bounding future offload scenarios following removal of the cask pit rack, the updated analysis includes an evaluation of the emergency core offload scenario, which is not part of the current licensing basis. PG&E requests that the NRC approve the updated thermal-hydraulic analysis as the licensing basis of record for future spent fuel storage requirements, including the temporary supplemental spent fuel storage capacity provided by the cask pit rack.

4.4 Radiological Assessment

4.4.1 Radiation Protection and ALARA Considerations

The existing radiation protection programs at Units 1 and 2 are adequate for the rack-installation operations. All of the operations

involved in the installation and removal of the cask pit rack will be controlled by procedures. These procedures are based on the principle of keeping doses as low as reasonably achievable (ALARA), consistent with the requirements of 10 CFR Part 20.

During the installation and removal of the racks, exposures will be maintained ALARA consistent with the requirements of 10 CFR 20 and the plant's existing ALARA program. Similar operations have been performed in a number of facilities in the past, and experience indicates that the task of installing these cask pit racks and support platforms in locations not previously occupied by other racks can be accomplished with minimum radiation exposure to personnel. Based on a physical survey of the existing spent fuel pool configuration and the design of the cask pit racks and support platforms, which allow remote installation, PG&E has determined that diving operations are not required for the installation and removal of the racks. The radiation protection section will prepare one or more radiation work permits (RWP) for various in-pool and out-of-pool activities. The RWPs and supporting documentation will establish requirements in the following areas:

- Frequency of radiation, contamination, and airborne surveys.
- Individual monitoring devices, typically a TLD and SRD.
- Protective clothing.
- Access and work controls.
- Contamination controls, including controls for radioactive materials that could involve significant shallow dose equivalent and effective dose equivalent exposures.
- Contamination and radiation limits at various steps requiring the evaluation of further controls or activities. (e.g., results of monitoring the rack as it breaks the water may necessitate additional underwater efforts, including pressure washing).

Continuous coverage by a radiation protection technician will be required for all activities involving underwater work, including the removal of items from the SFPs. Radiation protection technicians will be involved in the movement, decontamination, packaging, and storage of the racks. In addition to underwater activities, additional periods of continuous coverage by the radiation protection technician will be specified in the RWP covering the work activities.

During underwater activities and periods of work with high levels of contamination, a portable continuous air monitor with alarming capability will be used to provide warning in the event of increasing airborne radioactivity in the immediate work areas. Respiratory protection may be used if shown to be total effective dose equivalent ALARA consistent with the requirements of 10 CFR 20 and plant procedures.

Radiation protection will perform the following actions prior to and during rack removal:

- Vacuuming of the cask pit area prior to rack/platform installation and removal, if warranted, based in part on pre-job survey data.
- Underwater survey of accessible areas of the rack.
- Underwater pressure wash or rinse, as appropriate.
- Monitoring and rinsing of the rack as it breaks the water surface, as appropriate.
- Survey of the rack prior to wrapping in plastic or bagging.

Once wrapped or bagged the rack will either be placed into a storage container capable of protecting the rack from the elements for on site storage, or into an approved shipping container for shipment to a vendor for volume reduction and/or disposal.

All items or tools used in areas posted as contaminated will either be surveyed and released, if free of contamination, or treated as radioactive material and handled in accordance with DCPP procedures.

Each member of the project team will receive radiation protection training consistent with the requirements of 10 CFR 19. Project specific information, such as the potential for extremity doses when removing and decontaminating items from the SFP, and operating experience with SFP activities will be discussed in pre-job briefings. Radiation protection technicians involved in the project will participate in pre-job briefings involving their activities associated with the work.

4.4.2 Occupational Exposures

The impact on the occupational dose from spent fuel pool operations during the installation and removal of the cask pit rack and installation of the platform is expected to be minimal. Based on previous Holtec experience with rack module removal and decontamination projects and the previous Diablo Canyon re-racking of the SFPs, the removal and storage process will not create significant radiological waste or personnel exposure. The combined occupational exposure for installing a rack and associated platform in each unit is estimated to be a total of approximately 0.3 person-rem. The removal and decontamination process should not result in more than 0.2 person-rem exposure.

Since the airborne radioactivity and water activity will not significantly increase, no design or capacity changes in the SFP ventilation system or SFP water cleanup systems are needed for radiological reasons. Similarly, the radiation monitoring system for the SFPs, as described in the FSAR, is adequate.

An evaluation has been performed to demonstrate that although the water gap between the cask pit racks and the cask pit pool wall is slightly smaller than the gap between the existing racks and the SFP wall, there will be no change in the previously analyzed dose rates at the exterior wall surface. This is attributed to a source term strength reduction due to a minimum cooling time restriction of 10 years for all assemblies stored in the cask pit racks.

4.4.3 Fuel Handling Accidents (FHA)

This section summarizes the impact of installing the cask pit racks on the probability and radiological consequences of a FHA.

The radiological consequences of a FHA occurring in the cask pit area were reviewed to determine if the consequences are bounded by the existing analysis in FSAR Update, Section 15.5.22, as updated by License Amendment 163/165 issued February 27, 2004. The existing analysis determines the consequences of a single fuel assembly drop inside the SFP. The cask pit rack will be installed at the same water depth as the existing racks, providing the same iodine decontamination factors assumed in the FHA analysis. Furthermore, there are no new fuel movement pathways created by the addition of the cask pit racks, since the only fuel movement associated with the cask pit rack is for relocation of fuel from the existing racks to the new rack, and this fuel will be old fuel that has at least 10 years of decay time.

Since the source term for this fuel is much less than the conservative assumption made in the existing FSAR Update analysis, the radioactivity released from a fuel assembly dropped over the new cask pit rack would be less than the DBA. Hence the existing FHA analysis is bounding.

As discussed later in Section 4.6, the criticality analysis concluded that dropping a fuel assembly onto the cask pit rack or into an open cell will not cause an unacceptable increase in reactivity. From a rack structural standpoint, Section 4.7.1.1, shows that a fuel assembly drop will not threaten the structural integrity of the storage rack. Therefore, the reactivity and structural integrity consequences of a FHA occurring in the cask pit area are acceptable.

The probability of a FHA occurring, by the addition of the cask pit racks, is not significantly increased, because the same equipment (e.g., the spent fuel handling crane), procedures and controls will be used to handle fuel assemblies. Furthermore, it is expected that after the cask pit rack is loaded with spent fuel, the fuel will remain in place until dry cask loading operations necessitate their removal. This fuel movement does not significantly increase the normal frequency of fuel movement in the SFP, such as refueling outage and non-outage fuel shuffles.

4.4.4 Radiological Summary

No significant increase in radiation exposure to operating personnel of either unit is expected as a result of adding a cask pit rack; therefore, neither the current health physics programs nor the area monitoring systems need to be modified

4.5 Material Compatibility and Chemical Stability

The proposed cask pit racks and pedestals will be constructed of the following structural materials:

a. ASME SA240 Type 304L for all sheet metal stock and base plate

b. ASME SA240 Type 304L for the internally threaded support pedestals

c. ASME SA564-630, Condition H1100 (precipitation hardened stainless steel) for the externally threaded support spindle

d. Austenitic stainless steel for weld material

The fuel pool liner and rack assembly are stainless steel, which is compatible with the SFP water and radiation environment. In this type of environment of oxygen-saturated borated water, the corrosive deterioration of stainless steel is expected to be negligible. Dissimilar metal contact corrosion (i.e., galvanic attack) between stainless steel of the pool liner or rack/pedestal assembly and the Inconel and Zircaloy in the fuel assemblies stored in the rack, will not be significant because all of these materials are protected by highly passivating oxide films which are at similar galvanic potentials.

Metamic^{TM} panels with a minimum of 25 wt% B_4C are unanodized to minimize risk of surface contamination prior to installation. Reference 9 evaluated the functional performance of the material at elevated temperatures (up to 900°F) and radiation levels (1E 11 rads gamma).

Reference 9 concluded the following:

a. The metal matrix configuration produced by the powder metallurgy process with almost a complete absence of open porosity in Metamic^{TM} ensures that its density is essentially equal to the theoretical density.

b. The physical and neutronic properties of Metamic^{TM} are essentially unaltered under exposure to elevated temperatures (750° to 900°F).

c. No detectable change in the neutron attenuation characteristics under accelerated corrosion test conditions has been observed.

The following acceptance criteria will be used for the manufacturing of the Metamic^{TM}:

1. Spectroscopic examination of the B_4C powder batches to ensure a minimum B-10 content of 25 Wt% and a maximum B_4C content of 31 Wt%.
2. Chemical analysis of the aluminum powder batches to ensure that all trace elements are within prescribed limits.
3. Evaluation of both B_4C and Al powder for compliance with particle size limitations.

4. Determination of the actual B-10 weight density in the as-mixed B_4C-Al powder lots by a determination test (e.g., wet chemistry) to confirm that the minimum B-10 weight density criterion is satisfied.

5. Dimensional verification of every final sized panel produced.

4.6 Criticality

This section summarizes the cask pit rack criticality analyses performed by Holtec, the rack vendor. A more detailed discussion of the analysis methodology, assumptions, and results is located in Enclosure 5.

10 CFR 50 Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," provides a list of the minimum design requirements for nuclear power plants. According to GDC 62, "Prevention of Criticality in Fuel Storage and Handling," the licensee must limit the potential for criticality in the fuel handling and storage system by physical systems or processes."

The NRC regulatory requirements for maintaining subcritical conditions in SFPs are provided in 10 CFR 50.68, "Criticality Accident Requirements." The acceptance criteria for prevention of criticality in the SFP are that if credit is taken for soluble boron, then the effective multiplication factor (keff) shall be less than or equal to 0.95 if fully flooded with borated water, and less than 1.0 if flooded with unborated water.

The NRC defined acceptable methodologies for performing SFP criticality analyses in the following documents:

- NUREG-0800, Standard Review Plan, Section 9.1.2, "Spent Fuel Storage," Revision 3, July 1981

- Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," December 1981

- Memorandum from L. Kopp (NRC) to T. Collins (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998

4.6.1 Computer Codes

The criticality analyses were performed using the Los Alamos National Laboratory three-dimensional Monte Carlo code MCNP4a. This code was selected because it has been used and verified for

criticality analyses extensively and has all the necessary features for this analysis. The critical benchmark experiments considered the effects of varying fuel enrichment, boron-10 loading, lattice spacing, fuel pellet diameter, and soluble boron concentration. The experimental data are sufficiently diverse to establish that the method bias and uncertainty will apply to the DCPP cask pit rack conditions.

In addition to using the MCNP4a code to perform the criticality analyses, the CASMO-4 code was used to perform the fuel depletion analyses. This two-dimensional multigroup transport theory code was used to determine the isotopic composition of the spent fuel and to determine the reactivity effect of the fuel and rack tolerances. From this code, the reactivity effect (delta-k) was determined for each manufacturing tolerance of the fuel assemblies and storage racks.

4.6.2. Methodology

In performing the criticality analysis, K_{eff} was first calculated based on nominal conditions using the limiting (highest reactivity) fuel assemblies allowed to be stored in the cask pit rack. The reactivity analyses were performed for various enrichments, cooling times, burnups, and bounding cladding thickness using appropriately conservative assumptions such as an infinite array, and performing the depletion calculations assuming that all of the fuel assemblies contained burnable poison rodded assemblies (BPRA) which contained the maximum number of poison rods. Additionally, a bounding assembly was used in the depletion analyses with operating parameter assumptions that maximized the residual reactivity in the spent fuel assemblies. These parameters were average fuel pellet temperature, moderator temperature, and average core soluble boron concentration.

The methodology bias as well as a reactivity bias was added to account for the effect of the normal allowable SFP water temperature range. The SFP moderator temperature coefficient of reactivity is negative for the cask pit rack. The reactivity bias was calculated with a decrease in water temperature to 38°F, which is well below actual SFP water temperatures. The heat input to the SFP from spent fuel assemblies keeps the temperature of the pool water elevated well above 38°F. To determine the maximum K_{eff}, a statistical combination of the uncertainties and manufacturing tolerances was performed. The uncertainties included the computer code bias uncertainty and the calculation uncertainty. Both of these uncertainties were determined to a 95/95 threshold,

which is consistent with the requirements of 10 CFR 50.68. For each manufacturing tolerance, a delta-k was calculated between the nominal condition and the most limiting condition, which calculated the highest reactivity effect possible. The manufacturing tolerance reactivity effects were then statistically combined with the 95/95 uncertainties.

The objective of the criticality analysis was to ensure that the effective neutron multiplication factor (k_{eff}) is less than or equal to 0.95 with the cask pit storage racks fully loaded with fuel of the highest permissible reactivity and the pool flooded with borated water at a temperature corresponding to the highest reactivity (38°F which is well below the actual SFP minimum water temperature). Another objective was to demonstrate that k_{eff} is less than 1.0 under the assumed accident of the loss of soluble boron in the pool water (i.e., assuming unborated water in the SFP). A curve for the minimum required fuel burnup as a function of the initial fuel enrichment was developed, which ensures that the above requirements are met.

Additionally, reactivity effects of abnormal and accident conditions were also evaluated to assure that under all credible conditions, the k_{eff} will not exceed the regulatory limit of 0.95, considering the presence of soluble boron.

4.6.3 Criticality Calculations for Normal Conditions

The new cask pit spent fuel racks are designed and analyzed to employ a single-zone storage scheme. The racks will accommodate two types of 17x17 fuel assemblies: Westinghouse Standard Fuel (LOPAR) and Westinghouse OFA Fuel (Vantage 5). Furthermore, all fuel stored in these racks will be fuel assemblies with only relatively low initial enrichments and high burnup (see proposed TS Figure 3.7.17-4). The cask pit racks are designed to maintain the required subcriticality margin when fully loaded and flooded with unborated water at a temperature corresponding to the highest reactivity.

Each cask pit rack (Unit 1& 2) faces the SFP walls on two sides, and face existing permanent racks on the other two sides. The possibility of neutronic coupling between the cask pit racks and the existing permanent racks was considered. Since the distance between the racks is more than 12 inches, and the existing cask restraint structure precludes the racks from being closer than 12 inches, this separation is sufficient to preclude any neutronic interaction between the new cask pit rack and the existing racks.

Therefore, there is no need to consider the existing racks in the criticality calculation.

For reactivity control, Metamic™ absorber panels are utilized having a nominal thickness of 0.077 inches and a nominal B-10 areal density of 0.0185 g/cm^2. The Metamic™ absorber panels are 7-1/2 inches wide and 150 inches in length. These panels are held in place and protected against damage by stainless steel sheathing. The storage cells are assembled into a 12x13 cell array with a nominal lattice center-to-center spacing (pitch) of 8.946 inches, using welded connector bars. The storage cells are designed such that fuel assemblies in each cell will be shadowed (surrounded) by Metamic™ absorber panels on all sides. For neutron leakage, the analysis conservatively assumes an infinite radial array of storage cells, and the 12-inch water reflector is conservatively assumed in the axial direction. The maximum calculated reactivity includes a margin for uncertainty, including manufacturing tolerances (fuel and rack) and fuel depletion uncertainties.

The results of the calculations demonstrate that spent fuel assemblies with less than or equal to 4.1 percent initial enrichment and greater than or equal to 28.53 GWD/MTU may be stored in the cask pit rack. In addition, a burnup vs. initial enrichment curve was developed (refer to proposed TS Figure 3.7.17-4), which ensures that storage of applicable fuel meeting these curve limits will maintain the effective neutron multiplication factor (k_{eff}), less than 1.0 with unborated water and less than or equal to 0.95 with borated water.

4.6.4 <u>Abnormal and Accident Conditions</u>

The effects on reactivity of credible abnormal and accident conditions were examined. None of the abnormal or accident conditions that have been identified as credible would cause the reactivity of the cask pit storage racks to exceed the limiting reactivity value of k_{eff} less than or equal to 0.95, considering the presence of soluble boron. The double contingency principle of ANSI N16.1-1975 (and USNRC letter of April 1978) specifies that it shall require at least two unlikely independent and concurrent events to produce a criticality accident. This principle obviates the necessity of considering the simultaneous occurrence of multiple accident conditions.

Temperature and Void Effects

Temperature effects on reactivity have been calculated with CASMO-4. The results show that the SFP temperature coefficient of reactivity is negative for the cask pit rack, and that introducing voids in the water internal to the storage cell (to simulate boiling) further decreased reactivity. Using the lowest temperature (38°F) in the analysis, therefore, assures that the true reactivity will always be lower than the calculated value regardless of temperature.

Dropped Assembly – Horizontal

For the case in which a fuel assembly is assumed to be dropped on top of a rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the active fuel region of more than 12 inches, which is sufficient to preclude neutron coupling (i.e., an effective infinite separation). Maximum expected deformation under seismic or accident conditions will not reduce the minimum spacing to less than 12 inches. Consequently, the horizontal fuel assembly drop accident will not result in a significant increase in reactivity.

Dropped Assembly – Vertical

It is also possible to vertically drop an assembly into a location occupied by another assembly or that might be empty. Such a vertical impact would, at most, cause a small compression of the stored assembly, if present, or result in a small deformation of the baseplate of an empty cell. These deformations could potentially increase reactivity. However, the reactivity increase would be small compared to the reactivity increase created by the misloading of a fresh assembly discussed in the following section. The vertical drop is therefore bounded by this misloading accident and no separate calculation was performed for the drop accident.

Misloaded Fresh Fuel Assembly

The misplacement of a fresh unburned fuel assembly, in the absence of soluble poison, could result in exceeding the regulatory limit (k_{eff} of 0.95). This could possibly occur if a fresh fuel assembly of the highest permissible enrichment (5.0 wt%) were to be inadvertently misloaded into a storage cell intended for spent fuel. To assure that the regulatory limit is not exceeded under this condition, a soluble boron concentration level of 800 ppm in the SFP is required.

The DCPP TS require that the minimum SFP boron concentration is 2000 ppm. This boron concentration is more than sufficient concentration to maintain the 5 percent subcriticality margin in the SFP during the most limiting SFP accident. Administrative procedures to ensure the presence of soluble boron in the SFP during fuel handling operations preclude the possibly of the simultaneous occurrence of two independent accident conditions such as a fuel assembly misplacement and loss of soluble boron. As described in DCPP FSAR Update, Section 9.1.2.3, a boron dilution analysis was performed to evaluate the time and water volumes required to dilute the SFP from the DCPP TS required minimum boron concentration of 2000 ppm to approximately 800 ppm. The boron dilution analysis demonstrated that adequate time is available to identify and mitigate the dilution event before the spent fuel rack k_{eff} would exceed 0.95. Detection of a SFP boron dilution via pool level alarms, visual inspection during normal operator rounds, significant changes in SFP boron concentration, or significant changes in unborated water source volume, would be expected before a dilution event sufficient to increase k_{ef} above 0.95 could occur.

The reduction in SFP water volume due to installation of the cask pit rack and platform was evaluated and determined to have a negligible effect of the SFP boron dilution analysis.

Mislocated Fresh Fuel Assembly

The mislocation of a fresh unburned fuel assembly, i.e., the accidental placement of an assembly outside of a storage rack adjacent to other fuel assemblies, was also considered. However, there is no area around the rack that provides enough room for an additional assembly to be placed. A mislocated assembly is therefore not a credible condition.

4.6.5 Criticality Assessment Conclusions

The criticality analysis for the storage of spent nuclear fuel in the cask pit storage racks demonstrate that the effective neutron multiplication factor (k_{eff}) is less than 1.0 with unborated water and less than 0.95 with the storage racks fully loaded with fuel of the highest permissible reactivity and the pool flooded with borated water at a temperature corresponding to the highest reactivity at a 95 percent probability with a 95 percent confidence level. Further, the reactivity effects of abnormal and accident conditions were evaluated to assure that under credible abnormal and accident conditions, the reactivity would not exceed 0.95 with soluble boron

credit of 800 ppm at a 95 percent probability with 95 percent confidence level. Calculations performed qualify the cask pit racks for storage of burned fuel assemblies with maximum nominal (initial) enrichments and burnup characteristics as shown in proposed TS Figure 3.7.17-4.

4.7 Accidents and Events Evaluated

4.7.1 Drops

4.7.1.1 Cask Pit Rack and Support Platform Structural Evaluation During Postulated Fuel Assembly Drop Events

The NRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications" specifies that spent fuel rack designs must ensure the functional integrity of racks under all credible fuel assembly drop events. An evaluation of the consequences of fuel assembly drops onto the cask pit rack/platform structure was conducted to demonstrate that the rack could continue to safely store nuclear fuel following the drop. Two categories of accidental drop events were considered: (1) shallow drop, and (2) deep drop. Note that Section 4.6.4 describes the effect on reactivity from a dropped fuel assembly, and Section 4.4.3 summarizes the radiological considerations of a Fuel Handling Accident.

Shallow Drop Scenario

A "shallow drop" of a fuel assembly occurs when the dropped assembly strikes the top of the rack and damages the honeycomb structure, but does not enter an open cell or land directly in a cell already containing a stored fuel assembly. The structural acceptance criterion for this event is that the damage to the rack structure must be limited to the portion of the cell(s) above the top of the active fuel region for stored assemblies, which is approximately 15-15/16 inches below the top surface of the rack. The assumed free-fall height for this event is from 36 inches above the rack and the assumed weight of the dropped fuel assembly, plus its handling tool, is 3000 lbs.

Based on the design of the rack honeycomb structure, the limiting shallow drop scenario that would cause the maximum cell wall deformation occurs at a cell on the rack periphery, rather than at an internal cell. For this limiting

case, the analysis shows that the top of the impacted peripheral cell undergoes plastic deformation to a maximum depth of 15.5 inches, which is less than the 15-15/16-inch distance required to reach the top of active region of a fuel assembly stored in the cask pit rack. It should be noted that the welded connection between the inner surface of the impacted cell wall and the adjacent cell is not considered in the analytical model. The actual depth of plastic crushing would be significantly smaller than the predicted value if this additional welded connection were considered in the rack model. Therefore, the functional integrity of the rack would be maintained.

Deep Drop Scenario

A "deep drop" of a fuel assembly occurs when the dropped assembly enters an empty storage cell and impacts the cask pit rack baseplate. A sufficiently large impact force could adversely affect the structural integrity of the baseplate. Two deep drop locations were selected for evaluation: (1) a drop in a cell located directly above a rack support pedestal (leg), and (2) a drop in an interior cell away from a support pedestal where the baseplate is more flexible. The structural acceptance criteria for a deep drop event are: that the baseplate must remain intact and any deformation of the baseplate from the impact must be acceptable both from a structural and a criticality standpoint. In addition, the impact load from a deep drop onto a rack support pedestal must not penetrate the spent fuel pool liner when the force is transmitted into the liner beneath the cask pit rack and platform.

The analysis shows that the deep drop of a fuel assembly through an interior cell, away from a pedestal, causes a maximum local baseplate deformation of 3.55 inches, which is less than the 4 inches distance from the underside of the baseplate to the top of rack support platform (thickness of the female pedestal). The baseplate, although it deforms plastically, will maintain its structural integrity (i.e., the baseplate is not pierced) since the maximum plastic strain (0.063) is well below the failure limit of 0.38. It should be noted that since the top of rack support platform is approximately 51 inches above the SFP liner surface, the deformed baseplate would not impact the liner.

The scenario of the deep drop of a fuel assembly above a rack support pedestal location was found to be bounded by the rack drop accident, in terms of damage to the rack, SFP liner, and cask pit floor, and therefore, was not explicitly evaluated. Furthermore, the maximum compressive stress applied to the cask pit floor under this drop scenario is less than the concrete compressive strength. Therefore, the SFP liner plate and concrete slab will remain intact without loss of water from the SFP for a deep drop.

4.7.1.2 Spent Fuel Pool Concrete and Liner Evaluation During Postulated Cask Pit Rack and Support Platform Drop

An evaluation was performed of the consequences of an accidental drop of the cask pit rack and platform into the cask pit area of the SFP during installation activities. It was assumed that the rack or platform mass was dropped from the top surface of the SFP water. For this accident, the structural acceptance criterion is that the watertight integrity of the cask pit slab and SFP liner remain intact. Because of the limited gap between the platform and the cask pit, only the normal vertical orientation of the platform was considered. Refer to Section 7 of the Holtec Licensing Report in Enclosure 5.

The results of the rack drop analysis showed the following:

- The cask pit rack baseplate is locally stressed and plastically deformed.
- The SFP liner was shown to maintain structural integrity, since the maximum plastic strain (0.019) is well below the failure limit (0.38).
- There will be some local crack damage to the cask pit floor. These cracks will be limited to a small area on the concrete surface, which imposes no structural threat to the cask pit slab.

The results of the platform drop analysis showed the following:

- The cask pit floor liner plate only experiences minor plastic deformation with a maximum strain of 0.014.

- The drop will cause cracks in the concrete slab, especially on the slab top surface and at the periphery of the slab where the slab is surrounded by the cask pit wall. No through-thickness cracks develop in the slab.

4.7.2 Operational Errors and Mishandling Events

The probability of an operational error occurring during fuel handling activities associated with the cask pit rack is not significantly increased because the same equipment, procedures, and controls that are normally used for handling fuel will be utilized. During the rack installation activity, specific procedures and controls (including an ALARA Plan and Heavy Load Program) will be used to protect personnel, equipment, and the design basis of the SFP.

Section 4.6.4 discusses the criticality impacts resulting from the inadvertent misloading of a fresh fuel assembly into a storage cell. Although administrative controls will be in place to only permit storage of fuel within the limits of this amendment, the analyses demonstrate that with a soluble boron concentration of 800 ppm or greater, reactivity will not exceed 0.95. Therefore, even in the unlikely event that a fresh fuel assembly with the highest permissible enrichment (5.0 percent) was loaded into the rack, the regulatory limit would not be exceeded.

Section 4.4.3 discusses the radiological impacts from a FHA and concludes that the consequences are bounded by the existing analysis in FSAR Section 15.5.22.

4.7.3 Tornados

The impacts of tornados and tornado-borne missiles on the SFP, including the spent fuel racks, are evaluated in the DCPP FSAR, Sections 3.3.2.3.2.3 and 9.1.2.3.2, which concludes that adequate protection against tornado wind forces and tornado-generated missiles has been provided. This evaluation is also applicable to the cask pit rack in the cask-loading pit.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

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 consequences of an accident previously evaluated?

Response: No.

The proposed changes to temporarily increase the spent fuel storage capacity with a cask pit rack were evaluated for impact on the following previously evaluated events:

- A fuel handling accident (FHA)
- A heavy load drop into the cask pit
- A loss of spent fuel pool (SFP) cooling
- A stored fuel criticality event
- A seismic event

The probability of a FHA is not significantly increased by the proposed changes, because the same equipment (e.g., the spent fuel handling crane) and procedures will be used to handle fuel assemblies and the frequency of fuel movement will be essentially the same, with or without a cask pit rack. The FHA radiological consequences are not significantly increased because the source term of a single fuel assembly will remain unchanged, and the cask pit rack will be installed at the same water depth as the existing SFP racks, with the same iodine decontamination factors assumed in the FHA analysis. The structural consequences of dropping a fuel assembly on a cask pit rack were evaluated and found to be acceptable.

In accordance with NUREG-0612, heavy load drops are not required to be postulated if a single failure-proof crane is used for heavy load movements. If drops are postulated, then the consequences must be acceptable. PG&E plans to install a single failure-proof crane in accordance with NUREG-0612, prior to heavy load movements associated with the cask pit rack and platform. In the event that a single failure-proof crane is not available, PG&E has also performed heavy load drop analyses for the cask pit rack and platform, which have shown acceptable results in accordance with NUREG-0612. Therefore, the probability and the consequences of a heavy load drop in the cask pit are not significantly increased.

The probability of a loss of SFP cooling is unaffected and its consequences are not significantly increased with the cask pit rack installed. With the cask pit rack installed, loss of forced cooling results in

a sufficient time-to-boil for the operator to recognize the condition and establish SFP makeup to compensate for water lost due to pool bulk boiling, and thereby maintain a sufficient water blanket over the stored spent fuel.

The probability and consequences of a stored fuel criticality event are not increased by the addition of a cask pit rack. The reactivity analysis for the new cask pit rack demonstrates that reactivity remains subcritical (below 0.95) for the worst-case fuel-mispositioning event with credit for soluble boron.

The probability of a seismic event is unaffected and its consequences are not increased with the cask pit rack installed, because the structural analysis of the cask pit rack demonstrates that the fuel storage function of the rack is maintained during a seismic event.

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

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

Response: No.

The proposed change to add a cask pit rack does not alter the operating requirements of the plant or the equipment credited in the mitigation of design basis accidents, nor do the proposed changes affect any of the important parameters required to ensure the safe storage of spent fuel. A new rack material (Metamic™) is introduced into the pool under these changes; but, based on testing results, there are no mechanisms that create a new or different kind of accident. The NRC has also approved the use of Metamic™ generically for SFPs. The same equipment (e.g., the spent fuel handling crane) and procedures will be used to handle fuel assemblies for the new cask pit rack as are used for existing spent fuel storage. The fuel storage configuration in the cask pit rack will be similar to the configuration in the existing SFP storage racks, and a fuel drop or mispositioning event in the new racks does not represent a new or different kind of accident from fuel handling and mispositioning events previously evaluated.

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

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

Response: No.

The effect of the proposed change on current margins of safety was evaluated for spent fuel storage functionality and criticality, spent fuel and SFP cooling, and SFP/cask pit structural integrity. The design of the new cask pit rack uses proven technology which preserves the proper safety margins for spent fuel storage to provide a coolable and subcritical geometry under both normal and abnormal/accident conditions. The rack design complies with 10 CFR 50 Appendix A General Design Criterion (GDC) 62, the O.T. Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, Regulatory Guide 1.13, and ANSI/American Nuclear Society (ANS) 52.2. Handling of the cask pit rack and its platform in accordance with the defense-in-depth approach of NUREG-0612 with temporary lift devices designed to ANSI N14.6 preserves the proper margin of safety to preclude a heavy load drop in the cask pit.

The proposed SFP cooling system design basis is consistent with the previous licensing basis in FSAR, Section 9.1, for SFP temperature limits during normal and abnormal core offload conditions. The rack and SFP thermal-hydraulic analyses demonstrate that the proposed SFP cooling system design basis is met, and that no bulk boiling will occur in the cask pit rack or SFP with minimum cooling available. In the event of a loss of SFP cooling, there will be sufficient time for operators to identify the condition and initiate makeup flow or restore cooling to preserve fuel-cooling capability.

The criticality analysis demonstrates that the effective neutron multiplication factor (k_{eff}) is less than 1.0 for normal conditions with unborated water and less than 0.95 with 500 ppm of soluble boron, at a 95 percent probability with a 95 percent confidence level. Further, the reactivity effects of abnormal and accident conditions have been evaluated. To assure that under credible abnormal and accident conditions the reactivity will not exceed 0.95 at a 95 percent probability with a 95 percent confidence level, a soluble boron level of 800 ppm will be required to be maintained.

The structural analyses for the cask pit rack and platform and adjacent structures show acceptable results during seismic motion.

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

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

5.2 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. PG&E has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the proposed amendments to the TS, and does not affect conformance with 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants." Applicable regulatory requirements will continue to be met, adequate defense-in-depth will be maintained, and sufficient safety margins will be maintained. The applicable regulatory requirements are addressed in the individual sections of this technical analysis.

Based on the considerations discussed above and within the individual sections of the technical analysis, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security of the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

The following is a summary of the environmental considerations associated with the installation of cask pit rack for each unit during cycles 14 to 16. This temporary modification would increase the licensed storage capacity in each unit from the current 1,324 fuel storage assemblies to 1,478 fuel assemblies. However, maximum storage in the existing SFP racks would not increase and, other than an unexpected event requiring a full core offload, the only times when the actual inventory of spent fuel assemblies in the SFPs would exceed the current licensed limit of 1,324 spent fuel assemblies, would be during the short duration during the 14th and 15th refueling outages on each unit.

6.1 Thermal

Thermal effects on the environment due to adding a cask pit rack to each unit will be negligible. Any additional heat load attributed to the installation of a cask pit rack only occurs during Cycles 14-16 when the racks are installed. The total number of spent fuel assemblies stored in the SFP racks and the cask pit rack will be the capacity of the existing permanent SFP racks alone, at all times except during a reactor offload/refuel

condition. This means that during non-outage periods, the maximum number of spent fuel assemblies stored long-term in the SFP will be the same regardless of whether or not the cask pit racks are installed, resulting in no additional heat load imposed on the environment during non-refueling periods. The most significant contributor to the SFP decay heat load is the number of assemblies discharged for refueling. Since the size of planned refueling discharges are unchanged, there will be no significant increase in SFP heat load. A small additional heat load will be imposed on the SFP cooling system from the oldest spent fuel that is allowed to remain in the SFP longer, due to the additional storage capacity provided by the cask pit rack. However, this additional decay heat load will be insignificant when compared to the total heat rejected to the environment by the plant.

6.2 Radioactive Releases

The storage of additional spent fuel assemblies in the SFP during Cycles 14-16 is not expected to affect the releases of radioactive gases from the SFP. Gaseous fission products such as krypton-85 and iodine-131 are produced by the fuel in the reactor core during reactor operation. Tritium is routinely produced in an operating reactor through neuron capture by deuterium present in the reactor coolant. Small amounts of krypton-85 and iodine-131 fission gases are released to the reactor coolant from the small number of fuel assemblies that develop leaks during reactor operation. During refueling operations, some of these fission products enter the SFP and are subsequently released into the air. There will not be an increase in the amounts of gaseous fission products released to the atmosphere as a result of the temporary increase in SFP storage capacity because the frequency of refueling and the number of freshly off-loaded spent fuel assemblies stored in the SFP, at any one time, will not increase. The dominant source of the tritium inventory in the spent fuel pool at DCPP originates from the Reactor Coolant System (RCS). Normally, DCPP recycles RCS waste and recovers the boron and the high purity water. This water contains moderate concentrations of tritium from the RCS. This water is used to replenish SFP water due to evaporation, and is the major source of tritium in the SFP. The contribution of tritium produced from neutron capture in the boron-10 of the storage racks is insignificant to the total inventory. Therefore, the additional racks will have a negligible effect on the overall tritium level in the SFP.

Normally, the contributions from the fuel storage areas are negligible compared to the other sources of gaseous releases and no significant increases are expected in either unit as a result of the temporary expansion of fuel storage.

The radioactivity and impurities in the SFP water are not expected to increase as a result of the cask pit rack. Removal of radioactivity by filters and demineralizers would offset any anticipated increase of the radioactivity and impurity level of the water. The temporary increase in spent fuel storage capacity during refueling outages is not expected to result in a significant change in long-term generation of solid radioactive waste.

Once removed from the SFPs, the cask pit rack may be shipped to a vendor for volume disposal, disposed of as radioactive waste, stored for temporary use, or stored until decommissioning. The disposal of the used cask pit rack will result in a one-time incremental increase in solid waste. Since the cask pit rack will be packaged, transported, and disposed of in a manner consistent with NRC regulations, there will be no significant radiological impact on the environment. Because ongoing volume reduction efforts have effectively minimized the amount of waste generated at DCPP on an ongoing basis, this incremental increase is bounded by the plant's original licensing basis described in the Final Environmental Statement and, therefore, is acceptable.

The release of radioactive liquids will not be affected directly as a result of the cask pit rack installation. The SFP ion exchanger resins remove soluble radioactive materials from the SFP water. When the resins are changed out, the small amount of resin sluice water is processed by the radioactive waste system before release to the environment. The frequency of resin change-out may increase slightly during the installation and removal of the cask pit rack and installation of the support platform. However, the amount of liquid effluents released to the environment as a result of the cask pit rack and pedestal is expected to be negligible.

The proposed action will not significantly increase the probability or consequences of accidents, there are no significant changes in the types or significant increase in the quantities of effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure.

Alternatives to the proposed action for additional dry cask storage and other storage alternatives were evaluated in the NRC's environmental assessment of PG&E's proposed dry cask storage project (Ref. 10). With the exception of dry cask storage, all the alternatives were evaluated as non-feasible. Dry cask was the preferred option but may not be available on the schedule required.

The "no action" alternative in this case would necessitate operation during cycles 15 and 16 without a FCOC. While this is a feasible alternative, retaining FCOC is desirable from an operational perspective in the event

that unanticipated maintenance activities occur which require the core to be fully offloaded to complete. There would be no safety issue since the fuel could remain in the reactor vessel once the system is cooled down and depressurized; however, the maintenance activities may not be able to be completed until storage is provided. As previously discussed, the installation of the cask pit rack is to provide interim storage to support FCOC while outstanding issues are being resolved to provide dry cask storage for long-term storage of spent fuel.

This LAR has addressed both the safety and environmental aspects of a fuel handling accident. A fuel handling accident may be viewed as a "reasonably foreseeable " design basis event that the pool and its associated structures, systems and components are designed and constructed to prevent and mitigate. The environmental impacts of this accident were found not to be significant.

The NRC staff has previously considered spent fuel storage accidents whose consequences might exceed a fuel handling accident, i.e., "beyond design basis events" (Ref. 11). Such occurrences include a criticality accident and a Zircaloy-clad fire caused by overheating due to a loss of SFP cooling caused by a pool failure. Compliance with General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control," and Part 62, "Prevention of Criticality in Fuel Storage and Handling" of 10 CFR Part 50, Appendix A, and adherence to approved industry codes and standards as described in this LAR, provide assurance that such events are of a very low probability. These conclusions are unaffected by installation of the proposed cask pit rack. The proposed racks would be used only for storage of relatively old spent fuel. The NRC has recently issued guidance on SFPs to mitigate recent security concerns. PG&E will implement this guidance to the extent practicable, as recommended by the NRC.

In summary, PG&E has evaluated the proposed amendment and has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

7.1 References

1. HI-2043162, Rev. 1 (Holtec proprietary), "Spent Fuel Storage Expansion at Diablo Canyon Power Plant for Pacific Gas and Electric Company," dated October 2004.

2. NRC Letter "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

3. NRC Letter "St. Lucie Units 1 and 2 – Issuance of Amendment RE: The Addition of Spent Fuel Cask Pit Storage Racks and the Increase in Spent Fuel Storage Capacity (TAC NOS. MB6627 and MB6628)," dated July 9, 2004.

4. Memorandum from L. Kopp (NRC) to T. Collins (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998.

5. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis (Proposed Revision 2)," dated December 1981.

6. NRC Safety Evaluation for License Amendment No. 8 to Facility Operating License No. DPR-80 and Amendment 6 to Facility Operating License No. DPR-82 on spent fuel pool storage capacity expansion dated May 30, 1986.

7. Supplement to NRC Safety Evaluation on spent fuel pool storage capacity expansion dated October 15, 1987.

8. NRC Safety Evaluation for License Amendment 154 to Facility Operating License No. DPR-80 and Amendment 154 to Facility Operating License No. DPR-82 on crediting soluble boron in the Spent Fuel Pool dated September 25, 2002.

9. Qualification of Metamic for Spent-Fuel Storage Application, EPRI, October 2001.

10. NRC Letter, "Environmental Assessment and Finding of No Significant Impact Related to the Construction and Operation of the Diablo Canyon Independent Spent Fuel Storage Installation (TAC NO. L23399)," dated October 24, 2003.

11. NRC Letter, "Supplement to the Safety Evaluation and Environmental Assessment – Diablo Canyon Re-rack," dated October 15, 1987.

7.2 Precedent

The proposed amendment request for the cask pit rack is similar to that requested by Saint Lucie, and approved in license amendments 192 and 135.

Proposed Technical Specification Changes (mark-up)

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:

a. **In the permanent spent fuel storage racks** any four cells shall be in a configuration as shown in Figure 3.7.17-1**, and**

b. **In the cask pit storage rack, for Cycles 14 – 16, the fuel assemblies shall have:**

1. **An initial enrichment ≤ 4.1 wt% U-235;**
2. **A discharge burnup in the "acceptable" region of Figure 3.7.17-4; and**
3. **A minimum decay time of 10 years since being discharged from the reactor.**

c. **The total combined spent fuel pool capacity in the permanent and cask pit storage racks, for Cycles 14 – 16, is limited to no more than 1433 irradiated fuel assemblies. This limit does not apply for an emergency core offload.**

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 **<u>PERMANENT</u>** SPENT FUEL POOL **<u>STORAGE RACKS</u>**

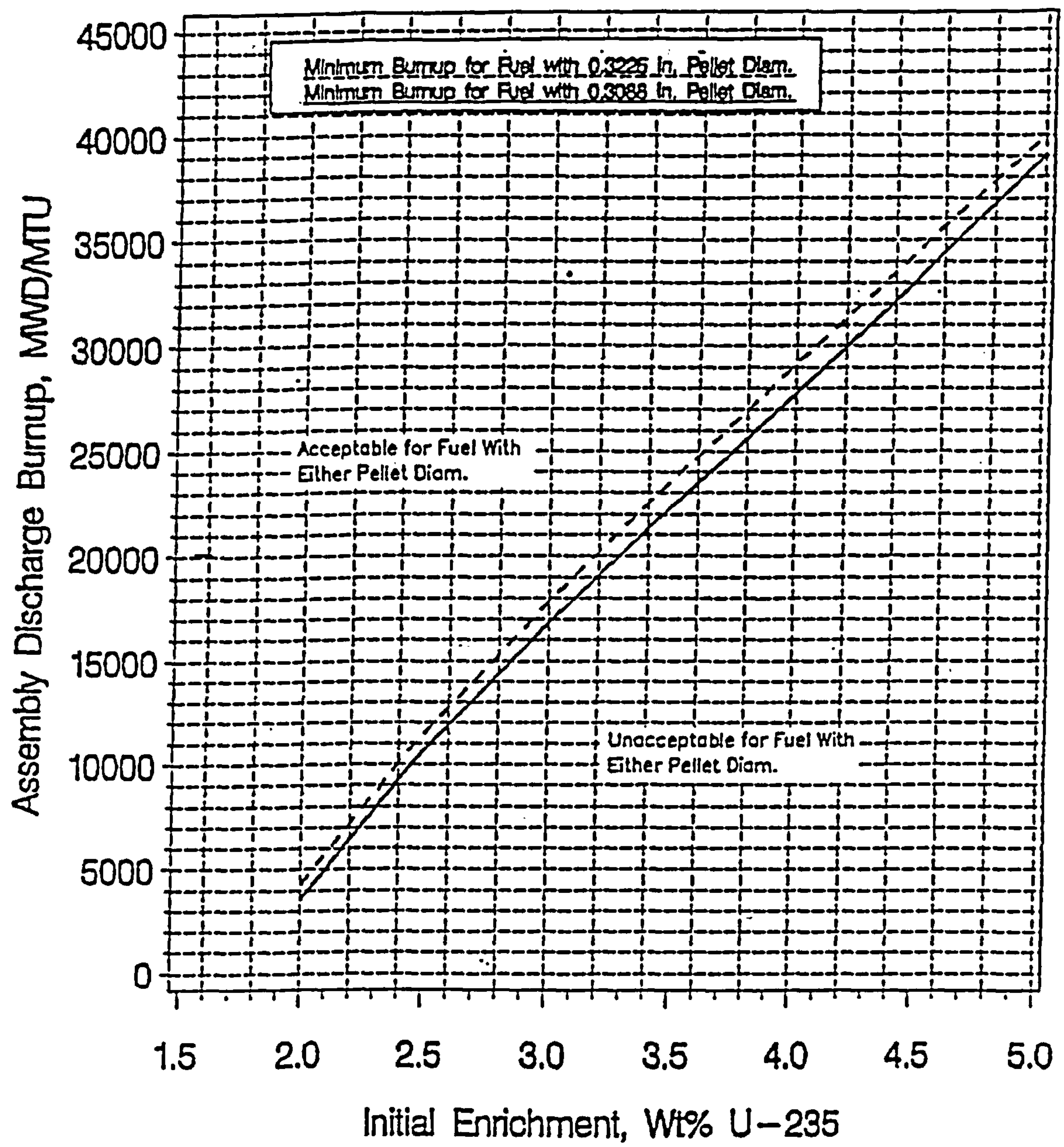

FIGURE 3.7.17-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT AND FUEL PELLET DIAMETER
FOR AN ALL CELL STORAGE CONFIGURATION **FOR THE PERMANENT SPENT FUEL POOL STORAGE RACKS**

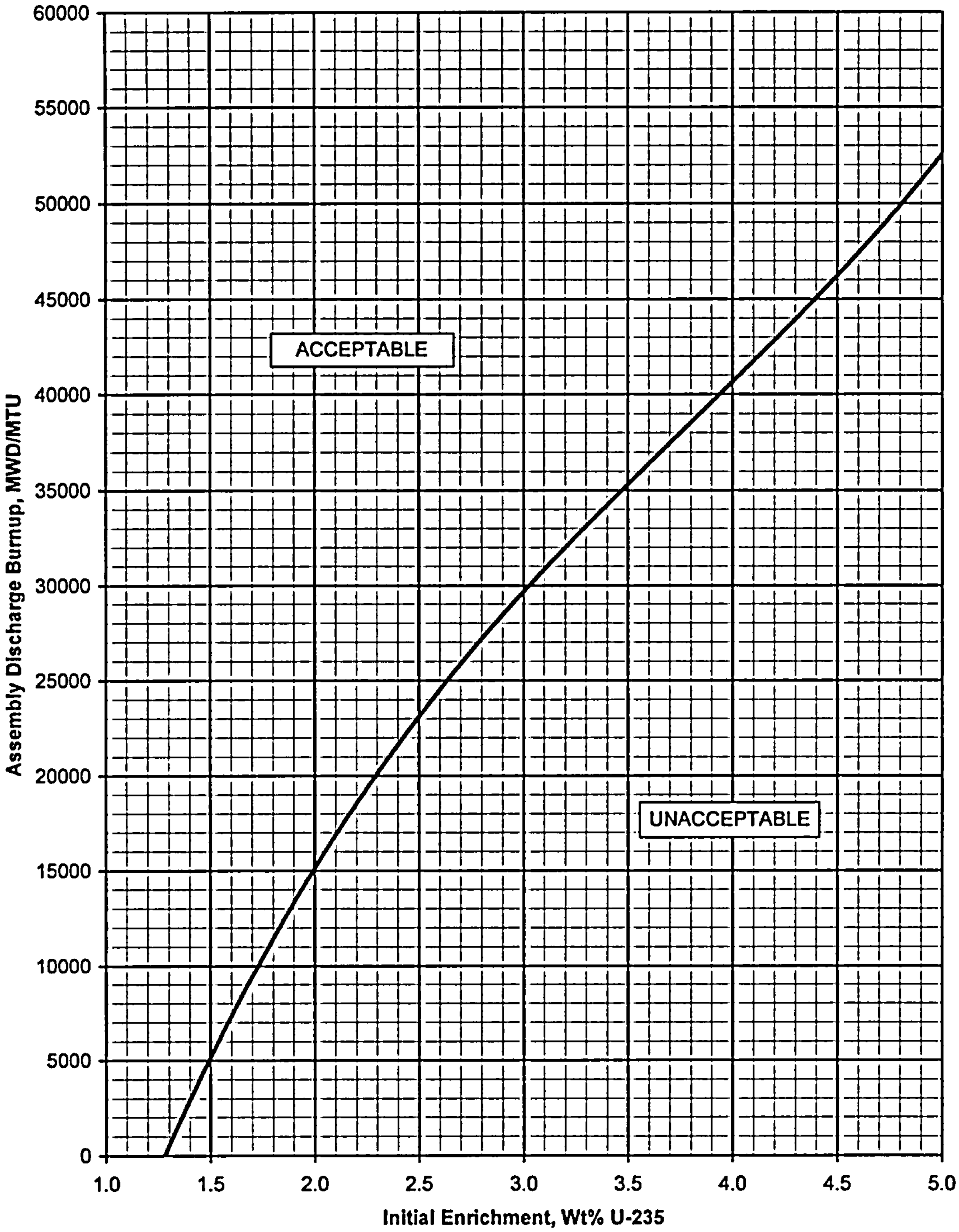

FIGURE 3.7.17-3
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR A 2X2 ARRAY STORAGE CONFIGURATION **FOR THE PERMANENT SPENT FUEL POOL STORAGE RACKS**

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Insert New Figure 3.7.17-4.

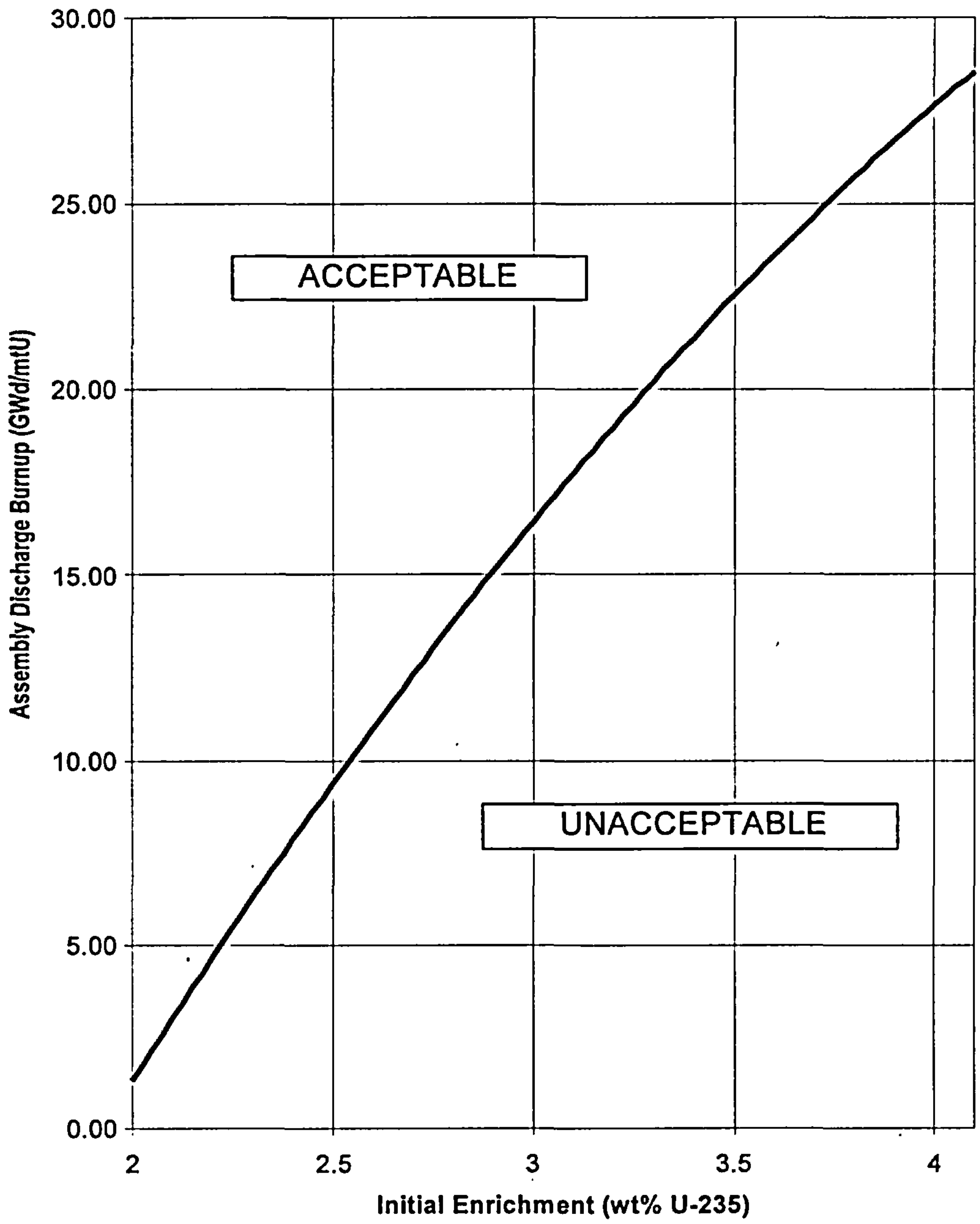

FIGURE 3.7.17-4
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR SPENT FUEL STORAGE IN THE CASK PIT STORAGE RACK

NOTES:

1. INITIAL ENRICHMENT NOT TO EXCEED 4.1 WT %;
2. MINIMUM SPENT FUEL DECAY TIME OF 10 YEARS SINCE BEING DISCHARGED FROM THE REACTOR; AND
3. APPLICABLE DURING CYCLES 14 – 16 WITH CASK PIT RACK INSTALLED

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 **permanent** spent fuel **pool** 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.

Insert 1

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 permanent spent fuel ~~storage~~ pool storage racks are ~~is~~ designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies. **For cycles 14-16, the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 154 fuel assemblies. For cycles 14-16, the total combined spent fuel pool capacity in the permanent and cask pit storage racks is limited to no more than 1478 fuel assemblies.**

Technical Specification Inserts

Insert 1

4.3.1.3 For cycles 14-16, the cask pit storage rack is designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.1 weight percent;

b. k_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 800 ppm, which includes an allowance for uncertainties;

d. A nominal 9 inch center to center distance between fuel assemblies placed in the cask pit fuel storage rack;

e. Fuel assemblies with discharge burnup in the "acceptable" region of Figure 3.7.17-4;

f. Fuel assemblies having a 10 year minimum decay time since being discharged from the reactor; and

g. A neutron absorbing material (Metamic™) between the stored fuel assemblies.

Proposed Technical Specification Changes (retyped)

Remove Page	Insert Page
3.7-28	3.7-28
3.7-29	3.7-29
3.7-30	3.7-30
3.7-31	3.7-31
3.7-32	3.7-32
4.0-1	4.0-1
4.0-2	4.0-2
	4.0-3

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:

a. In the permanent spent fuel storage racks any four cells shall be in a configuration as shown in Figure 3.7.17-1, and

b. In the cask pit storage rack, for Cycles 14 – 16, the fuel assemblies shall have:

1. An initial enrichment ≤ 4.1 wt% U-235;
2. A discharge burnup in the "acceptable" region of Figure 3.7.17-4; and
3. A minimum decay time of 10 years since being discharged from the reactor.

c. The total combined spent fuel pool capacity in the permanent and cask pit storage racks, for Cycles 14 – 16, is limited to no more than 1433 irradiated fuel assemblies. This limit does not apply for an emergency core offload.

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 PERMANENT SPENT FUEL POOL STORAGE RACKS

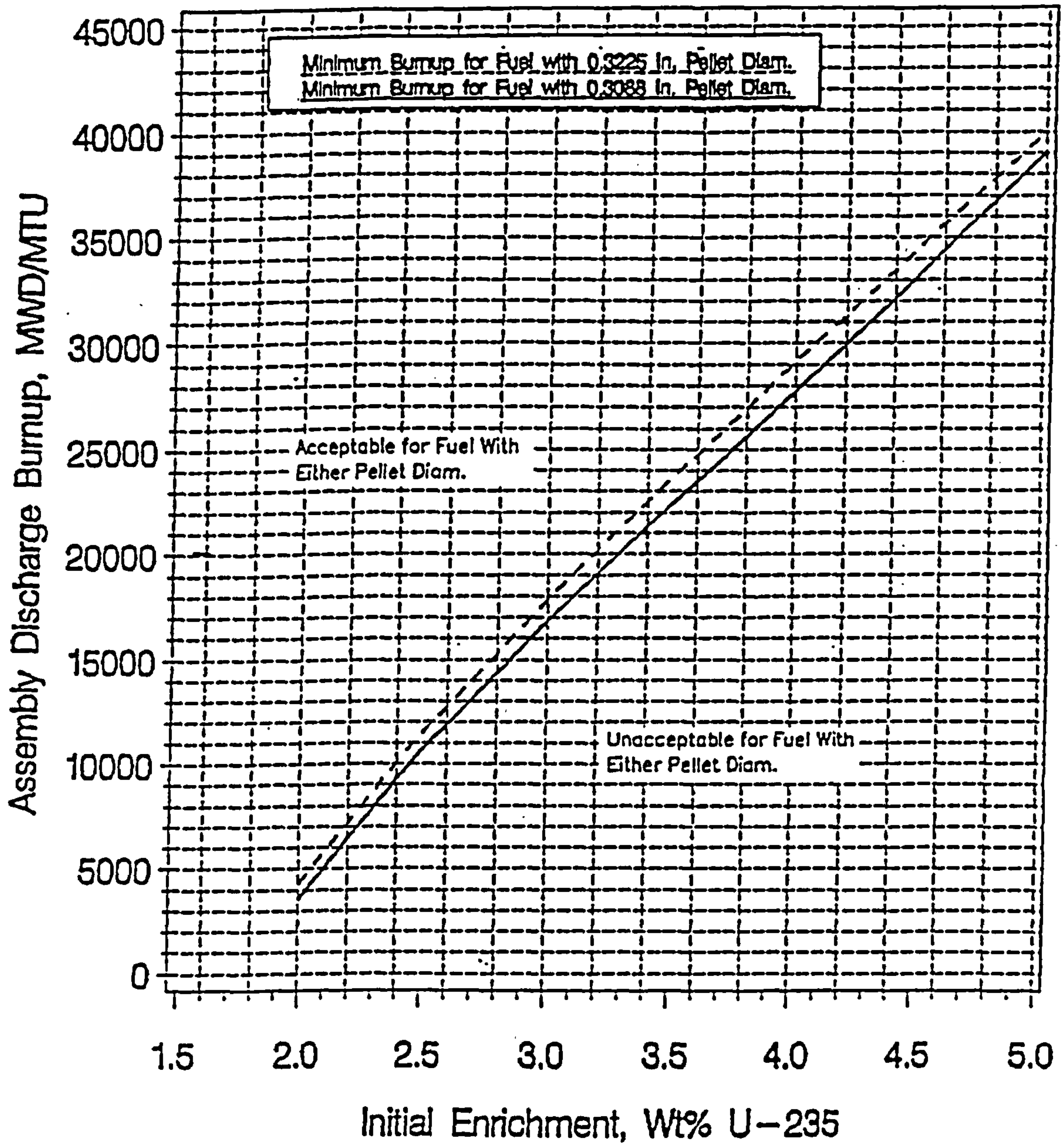

FIGURE 3.7.17-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT AND FUEL PELLET DIAMETER
FOR AN ALL CELL STORAGE CONFIGURATION FOR THE PERMANENT SPENT
FUEL POOL STORAGE RACKS

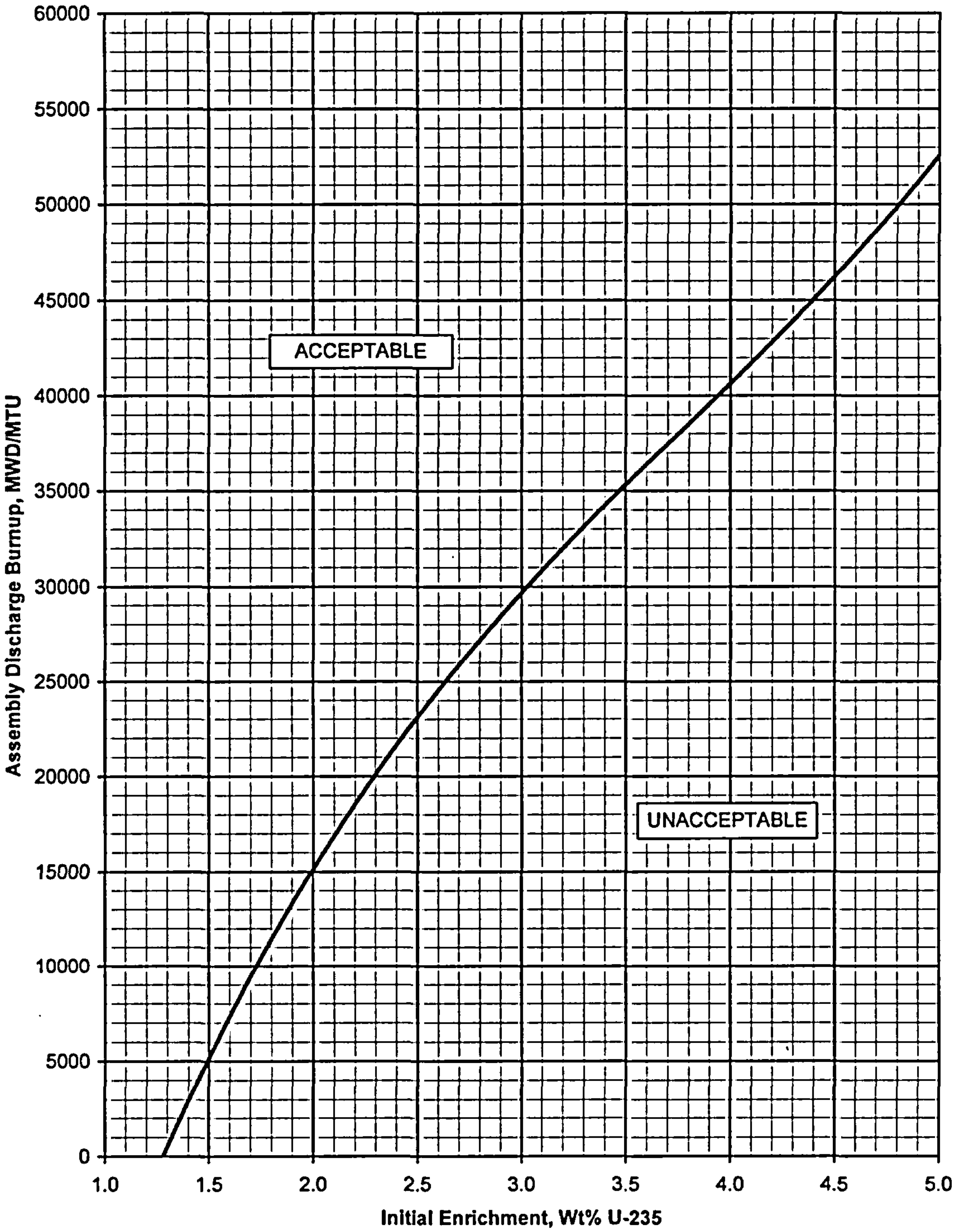

FIGURE 3.7.17-3
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR A 2X2 ARRAY STORAGE CONFIGURATION FOR THE PERMANENT SPENT
FUEL POOL STORAGE RACKS

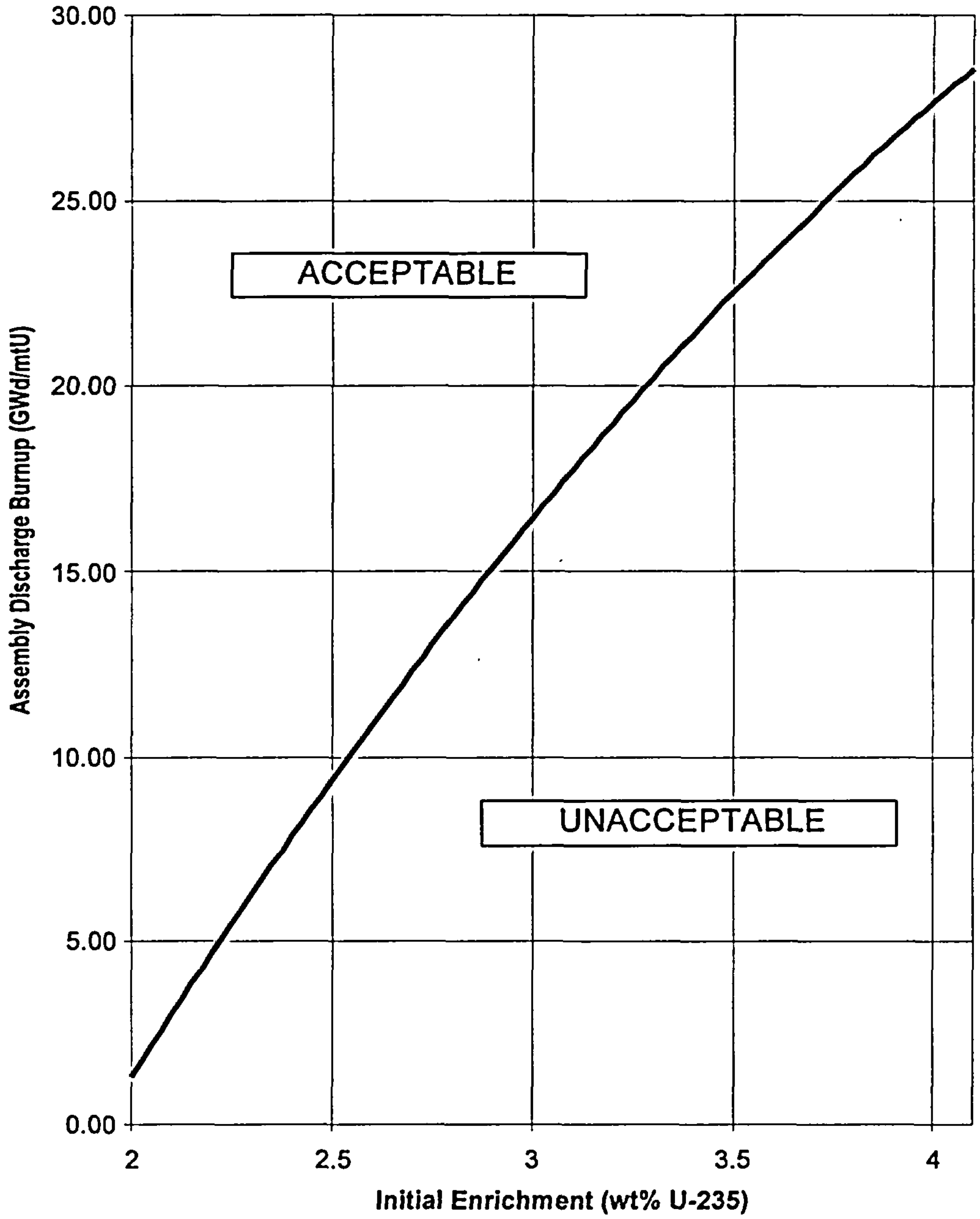

FIGURE 3.7.17-4
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT
FOR SPENT FUEL STORAGE IN THE CASK PIT STORAGE RACK

NOTES:

1. INITIAL ENRICHMENT NOT TO EXCEED 4.1 WT %;
2. MINIMUM SPENT FUEL DECAY TIME OF 10 YEARS SINCE BEING DISCHARGED FROM THE REACTOR; AND
3. APPLICABLE DURING CYCLES 14 – 16 WITH CASK PIT RACK INSTALLED

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 permanent spent fuel pool 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.

4.3.1.3 For cycles 14-16, the cask pit storage rack is designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 4.1 weight percent;

b. $k_{eff} < 1.0$ if fully flooded with unborated water, which includes an allowance for uncertainties;

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 800 ppm, which includes an allowance for uncertainties;

d. A nominal 9 inch center to center distance between fuel assemblies placed in the cask pit fuel storage rack;

e. Fuel assemblies with discharge burnup in the "acceptable" region of Figure 3.7.17-4;

f. Fuel assemblies having a 10 year minimum decay time since being discharged from the reactor; and

g. A neutron absorbing material (Metamic™) between the stored fuel assemblies.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

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 permanent spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies. For cycles 14-16, the cask pit storage rack is designed and shall be maintained with a storage capacity limited to no more than 154 fuel assemblies. For cycles 14-16, the total combined spent fuel pool capacity in the permanent and cask pit storage racks is limited to no more than 1478 fuel assemblies.

Changes to Technical Specification Bases Pages
(For Information Only)

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. 4) each consist of 16 **permanent** stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells. **For cycles 14 – 16, in addition to the 16 racks, a cask pit storage rack with a capacity of 154 cells is installed in each spent fuel pool's cask pit area. This extra rack expands the total storage capacity in each spent fuel pool to 1,478 fuel assembly storage cells.** The spent fuel pool storage racks have been analyzed for the storage of fuel assemblies that meet the requirements of LCO 3.7.17. **The fuel storage capacity during cycles 14 – 16 is 1478 assemblies, of which 1433 assemblies may be irradiated. Unfilled cells in the permanent storage racks may be utilized for the storage of unirradiated fuel assemblies. The limitation of a maximum of 1433 irradiated assemblies is based on the spent fuel pool bulk temperature analysis for cycles 14 – 16, while the cask pit storage rack is installed.**

10 CFR 50.68(b)(4), requires that the ~~high-density~~ spent fuel storage racks are designed to assure that with credit for soluble boron and with fuel of the maximum fuel assembly reactivity, a K_{eff} of less than or equal to 0.95 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with borated water, and a K_{eff} off less than 1.0 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with unborated water.

Criticality analyses have been performed **for the permanent storage racks** (Ref. 3 and 5) **and for the cask pit storage rack (Ref 8),** 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 analyseis.

For the permanent storage racks, Reference 5 provides the analysis for the 2x2 array and checkerboard configurations, and Reference 3 provides the analysis for the all cell configuration. Both criticality analyses **(Ref. 3 and 5)** 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.

Reference 8 provides the analysis for the cask pit storage rack. Storage configurations were defined in the criticality analysies ~~(Ref. 3 and 5)~~ **(Ref. 8)** 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 **minimum** soluble boron concentration of ~~806~~ **500** ppm is required to maintain k_{eff} less than or equal to 0.95 ~~and to~~

~~mitigate the worst accident reactivity insertion for all allowable~~ **under normal** storage conditions **including tolerances and uncertainties**, which is well within the 2000 ppm requirement of LCO 3.7.16.

The criticality analyses considered accident conditions ~~(Ref. 3 and 5)~~ **(Ref. 3, 5, and 8)**. ~~The analyses determined a~~ **Soluble boron credit is then used to maintain k_{eff} less than or equal to 0.95 and to mitigate the worst accident reactivity insertion. For the permanent storage racks,** a soluble boron concentration of 806 ppm is ~~necessary to ensure~~ **required to maintain** k_{eff} ~~will be maintained~~ less than or equal to 0.95 ~~should the most adverse postulated reactivity insertion accident occur~~ **for all allowable storage configurations, which is well within the 2000 ppm requirement of LCO 3.7.16. For the cask pit storage rack, a soluble boron concentration of 800 ppm is required to maintain keff 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.**

For such an occurrence, the double contingency principle of ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 1) 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 reactivity insertion accident condition, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

In addition to consideration of spent fuel pool criticality, a boron dilution analysis (Ref. 6) 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.~~

The results of the boron dilution analysis concluded that an unplanned or inadvertent event that would result in the dilution of the spent fuel pool boron concentration from 2000 ppm to 800 ppm is not a credible event since a dilution event large enough to result in a significant reduction in the spent fuel pool boron concentration would involve the transfer of a large quantity of water from a dilution source and a significant increase in spent fuel pool level, which would ultimately overflow the pool. The overflow of the spent fuel pool would be readily detected and terminated by plant personnel. In addition, because of the large quantities of water required and the low dilution flow rates available, any significant dilution of the spent fuel pool boron concentration would only occur over a long period of time (hours to days). Detection of a spent fuel pool boron dilution via pool level alarms, visual inspection during normal operator rounds, significant changes in spent fuel pool boron concentration, or significant changes in the unborated water source volume, would be expected before a dilution event sufficient to increase K_{eff} above 0.95 could occur.

However, **for the permanent storage racks** 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.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	Most accident conditions result in a negligible reactivity effect in the spent fuel pool ~~(Ref. 3 and 5)~~ **(Ref. 3, 5, and 8)**. However, scenarios can be postulated that could have more than a negligible positive reactivity effect. Examples of such accident scenarios **for the permanent storage racks** 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. 4). 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. **Examples of accident scenarios for the cask pit storage rack are the misplacement of a fuel assembly and a dropped assembly. A soluble boron concentration of 800 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 analysis is provided in the FSAR (Ref. 4). 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 References 3, 4, ~~and~~ 5, **and 8**. The specified boron concentration of 2000 ppm ensures that the spent fuel pool k_{eff} will remain less than or equal to 0.95 at a 95 percent probability, 95 percent confidence level, for a postulated reactivity insertion accident or boron dilution event.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel pool.
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.

BASES (continued)

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. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
2. Not used.
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. FSAR, Section 9.1, 15.4.5, and 15.5.22.
5. "Diablo Canyon Units 1 and 2 Spent Fuel Criticality Analysis," February 14, 2001, Paul F. O'Donnell, Westinghouse Doc. No. A-DP1-FE-0001.
6. "Diablo Canyon Units 1 and 2 Spent Fuel Boron Dilution Analysis," January, 2001, Gary J. Corpora
7. License Amendment 154/154, September 25, 2002.
8. **"Spent Fuel Storage Expansion at Diablo Canyon Units 1 & 2 for Pacific Gas & Electric Co.", October 2004, Holtec Report HI-2043162.**

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. 2) each consist of 16 stainless steel racks of various sizes, with a total of 1,324 fuel assembly storage cells. **For cycles 14 – 16, in addition to the 16 permanent racks, a cask pit storage rack with a capacity of 154 cells is installed in each spent fuel pool's cask pit area. This extra rack expands the total storage capacity in each spent fuel pool to 1,478 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 fuel storage capacity during cycles 14 – 16 is 1478 assemblies, of which 1433 assemblies may be irradiated. Unfilled cells in the permanent storage racks may be utilized for the storage of unirradiated fuel assemblies. The limitation of a maximum of 1433 irradiated assemblies is based on the spent fuel pool bulk temperature analysis for cycles 14 – 16, while the cask pit storage rack is installed.** The **16 permanent** spent fuel storage racks are designed to accommodate three different storage configurations as shown in Figure 3.7.17-1. **The cask pit fuel storage rack is designed to accommodate only fuel with an initial enrichment of ≤ 4.1 weight % U-235, a minimum 10 year decay time and a discharge burnup in the acceptable region of Figure 3.7.17-4.**

10 CFR 50.68(b)(4), requires that the spent fuel storage racks are designed to assure that with credit for soluble boron and with fuel of the maximum fuel assembly reactivity, a K_{eff} of less than or equal to 0.95 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with borated water, and a K_{eff} off less than 1.0 is maintained, at a 95 percent probability, 95 percent confidence level, if the racks are flooded with unborated water.

Criticality analyses have been performed (Ref. 3, 4**, and 6**) 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 3 provides the analysis for the 2x2 array and checkerboard configurations, ~~and~~ Reference 4 provides the analysis for the all cell configuration**, and Reference 6 provides the analysis for the cask pit storage rack.**

For the 16 permanent storage racks, ~~B~~**b**oth criticality analyses **(Ref. 3 and 4)** 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. 3, ~~and~~ 4, and 6) 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 and to mitigate the worst accident reactivity insertion. **For the permanent storage racks,** ~~A~~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. **For the cask pit storage rack, a soluble boron concentration of 800 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.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	**For the permanent storage racks,** ~~T~~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. 2, 3, and 4). **For the cask pit storage rack, accidents that could have significant reactivity effects are misplacement of a fuel assembly and a dropped assembly (Ref. 6).** For these accident occurrences, 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).
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 at a 95 percent probability, 95 percent confidence level, for a postulated reactivity insertion accident or a boron dilution event.
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 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. **In addition, for fuel assemblies stored in the cask pit storage rack, the record will also include fuel assembly decay-time.**

(continued)

BASES (continued)

REFERENCES

1. Not used.
2. FSAR, Section 9.1, 15.4.5, and 15.5.22
3. "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.
4. "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.
5. License Amendment 154/154, September 25, 2002.
6. **"Diablo Canyon Units 1 and 2 Spent Fuel Storage Expansion Licensing Report", October 2004, Holtec Report HI – 2043162.**

Enclosure 5
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