June 7, 1995

Mr. Gregory M. Rueger
Nuclear Power Generation, B14A
Pacific Gas and Electric Company
77 Beale Street, Room 1451
P. O. Box 770000
San Francisco, California 94106

SUBJECT: ISSUANCE OF AMENDMENTS FOR DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M91508) AND UNIT NO. 2 (TAC NO. M91509)

Dear Mr. Rueger:

The Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. DPR-80 and Amendment No. 103 to Facility Operating License No. DPR-82 for the Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated February 6, 1995, and supplemented by letters dated March 24, 1995, and May 22, 1995.

These amendments would allow the storage of fuel with enrichments up to and including 5.0 weight percent U-235, would clarify that substitution of fuel rods with filler rods is acceptable for fuel designs that have been analyzed with applicable NRC-approved codes and methods, and would allow the use of ZIRLO fuel cladding in the future in addition to Zircaloy-4.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original Signed By

Melanie A. Miller, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-275
and 50-323

Enclosures:
1. Amendment No. 104 to DPR-80
2. Amendment No. 103 to DPR-82
3. Safety Evaluation

cc w/encls: See next page

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CGrimes, 011E22	ACRS (4)
WBateman	GBagchi
LHurley, RIV	CMcCracken
RJones	CMiller

* See previous concurrence

DOCUMENT NAME: DC91508.AMD

OFC	PDIV-2/LA	PDIV-2/PM	OGC	NRR:ECGB*	NRR:SPLB*
NAME	EPeyton	MMiller:ye	AP H	GBagchi	CMcCracken
DATE	5/22/95	5/23/95	5/24/95	5/19/95	5/19/95

OFC	NRR:SRXB	NRR:TERB *
NAME	RJones*	CMiller
DATE	5/22/95	5/23/95

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 7, 1995

Mr. Gregory M. Rueger
Nuclear Power Generation, B14A
Pacific Gas and Electric Company
77 Beale Street, Room 1451
P. O. Box 770000
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Enclosures:
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cc w/encls: See next page

cc w/encls:
NRC Resident Inspector
Diablo Canyon Nuclear Power Plant
c/o U.S. Nuclear Regulatory Commission
P. O. Box 369
Avila Beach, California 93424

Dr. Richard Ferguson, Energy Chair
Sierra Club California
6715 Rocky Canyon
Creston, California 93432

Ms. Nancy Culver
San Luis Obispo
Mothers for Peace
P. O. Box 164
Pismo Beach, California 93448

Ms. Jacquelyn C. Wheeler
P. O. Box 164
Pismo Beach, California 93448

Managing Editor
The County Telegram Tribune
1321 Johnson Avenue
P. O. Box 112
San Luis Obispo, California 93406

Chairman
San Luis Obispo County Board of
Supervisors
Room 370
County Government Center
San Luis Obispo, California 93408

Mr. Truman Burns
Mr. Robert Kinosian
California Public Utilities Commission
505 Van Ness, Room 4102
San Francisco, California 94102

Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Services
Post Office Box 942732
Sacramento, California 94232

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Harris Tower & Pavillion
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

Mr. Peter H. Kaufman
Deputy Attorney General
State of California
110 West A Street, Suite 700
San Diego, California 92101

Christopher J. Warner, Esq.
Pacific Gas & Electric Company
Post Office Box 7442
San Francisco, California 94120

Mr. Warren H. Fujimoto
Vice President and Plant Manager
Diablo Canyon Nuclear Power Plant
P. O. Box 56
Avila Beach, California 93424

Diablo Canyon Independent Safety
Committee
ATTN: Robert R. Wellington, Esq.
Legal Counsel
857 Cass Street, Suite D
Monterey, California 93940

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. DPR-80

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated February 6, 1995, as supplemented by letters dated March 24, and May 22, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 104 , are hereby incorporated in the license. Pacific Gas and Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Melanie A. Miller

Melanie A. Miller, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 7, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 103
License No. DPR-82

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee) dated February 6, 1995, as supplemented by letters dated March 24, and May 22, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 103, are hereby incorporated in the license. Pacific Gas and Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of its date of issuance to be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Melanie A. Miller

Melanie A. Miller, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 7, 1995

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. Overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
xii	xii
xvii	xvii
3/4 9-17	3/4 9-17
3/4 9-18	3/4 9-18
3/4 9-19	3/4 9-19
---	3/4 9-20
---	3/4 9-21
---	3/4 9-22
B 3/4 9-3	B 3/4 9-3
5-5	5-5
5-6	5-6

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

INDEX

BASES

SECTION PAGE

3/4.11 RADIOACTIVE EFFLUENTS

REFUELING OPERATIONS

3/4.9.14 SPENT FUEL ASSEMBLY STORAGE

SPENT FUEL POOL REGION 2

LIMITING CONDITION FOR OPERATION

3.9.14.1 The combination of initial enrichment, pellet diameter, and cumulative burnup for spent fuel assemblies stored in Region 2 shall be within the acceptable area of Figure 3.9-2.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations (with loads in the fuel storage area) except to perform the following: move the non-complying fuel assemblies to Region 1 in accordance with TS 3.9.14.3. Until the requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14.1 The cumulative burnup of each spent fuel assembly stored in Region 2 shall be determined by analysis of its burnup history, prior to storage in Region 2. A complete record of initial enrichment, fuel pellet diameter, and the cumulative burnup analysis shall be maintained for the time period that the spent fuel assembly remains in Region 2 of the spent fuel pool.

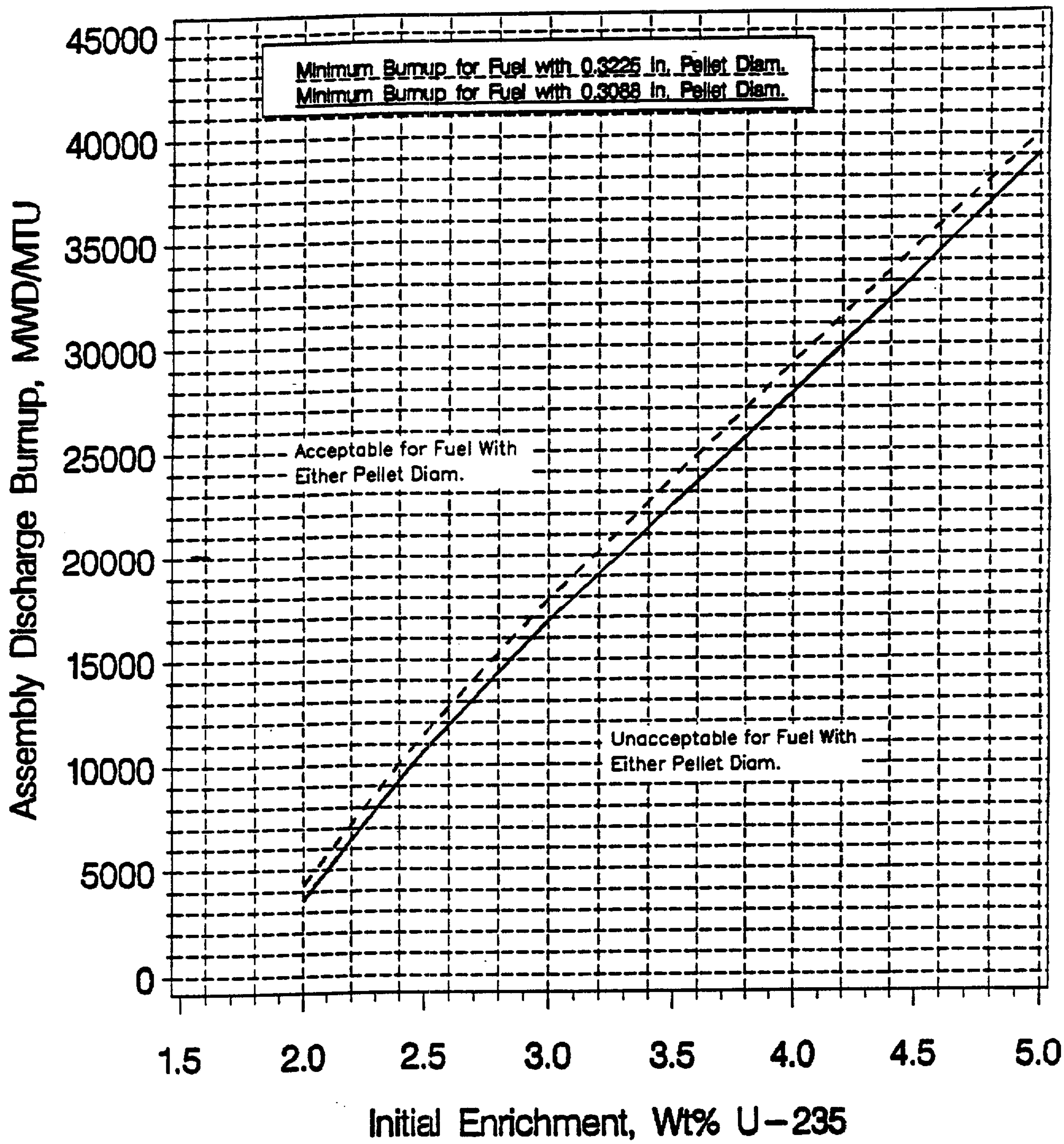

FIGURE 3.9-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCITON OF INITIAL ENRICHMENT AND PELLET DIAMETER TO PERMIT
STORAGE IN REGION 2

REFUELING OPERATIONS

SPENT FUEL ASSEMBLY STORAGE

SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.14.2 The boron concentration of the spent fuel pool shall be greater than or equal to 2000 ppm.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, immediately suspend all movement of fuel assemblies in the spent fuel pool and initiate corrective actions to restore the boron concentration.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14.2 The boron concentration of the spent fuel pool shall be determined by chemical analysis at least once per 31 days.

REFUELING OPERATIONS

SPENT FUEL ASSEMBLY STORAGE

SPENT FUEL POOL REGION 1

LIMITING CONDITION FOR OPERATION

3.9.14.3 The following conditions shall be met for storage of fuel assemblies in Region 1 of the spent fuel pool:

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

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

1) The combination of initial enrichment and cumulative burnup of the assemblies is within the acceptable area of Figure 3.9-3; or

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

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

a) water cells, or

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

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

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

APPLICABILITY: Whenever fuel assemblies are in Region 1 of the spent fuel pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations (with loads in the fuel storage area) except to perform the following: move the non-complying fuel assemblies into a pattern that complies with requirements of the above specification. Until the requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14.3 The cumulative burnup of each fuel assembly stored in Region 1 shall be determined by analysis of its burnup history, prior to storage in Region 1. A complete record of initial enrichment, initial integral boron content, and the cumulative burnup analysis shall be maintained for the time period that the fuel assembly remains in Region 1 of the spent fuel pool.

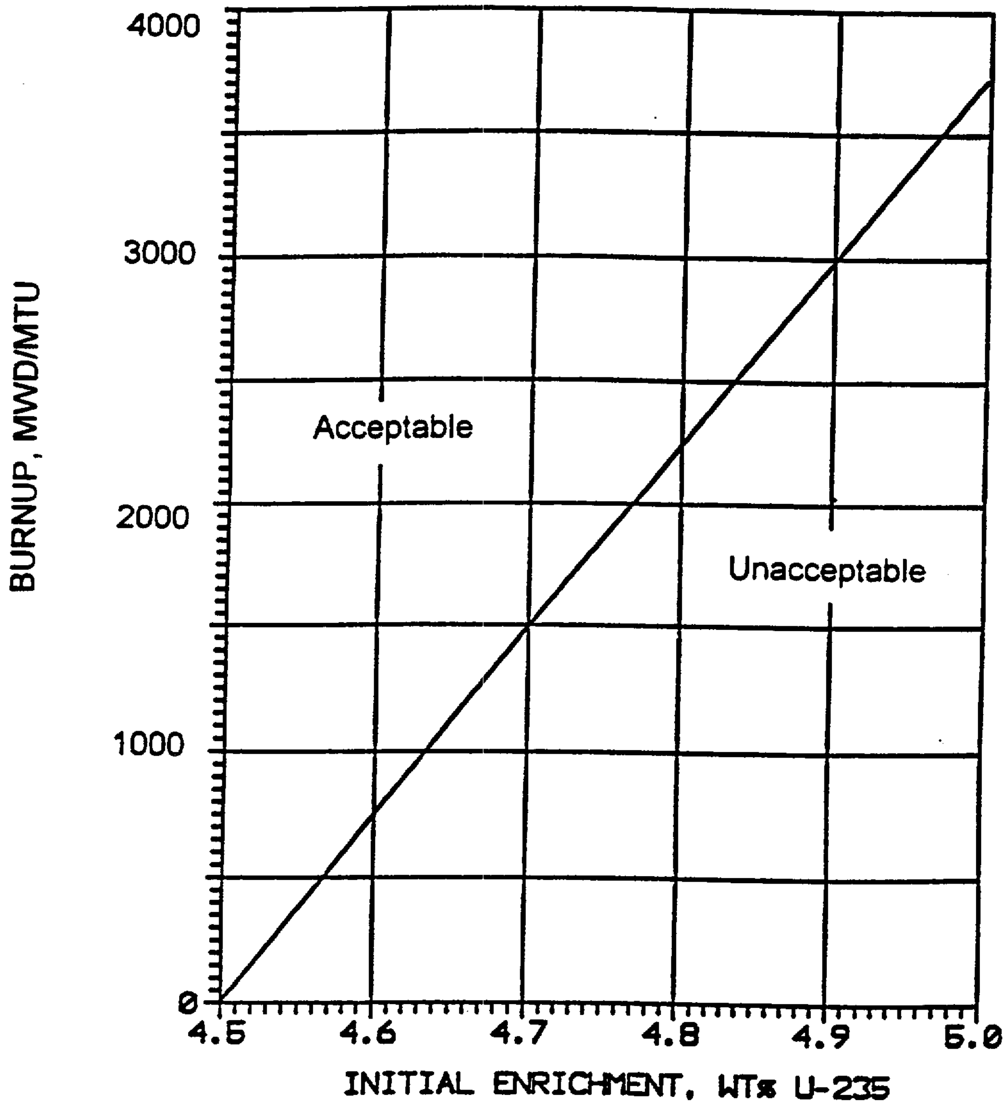

FIGURE 3.9-3
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT (NO IFBA) TO PERMIT
STORAGE IN REGION 1

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment ventilation penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

The minimum water level for movement of fuel assemblies (23 feet above the vessel flange) assures that sufficient water depth is maintained above fuel elements being moved to or from the vessel. With the upper internals in place, fuel assemblies and control rods cannot be removed from the vessel. Operations involving the unlatching of control rods with the vessel upper internals in place may proceed with less than 23 feet of water above the vessel flange provided that 23 feet of water (12 feet above the flange) is maintained above all irradiated fuel assemblies within the reactor vessel.

3/4.9.12 FUEL HANDLING BUILDING VENTILATION SYSTEM

The limitations on the Fuel Handling Building Ventilation System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. Transfer of system operation into the iodine removal mode (exhaust through HEPA filters and charcoal adsorbers) is initiated automatically by either the new fuel storage or spent fuel pool area radiation monitors required by Specification 3.3.3. Following installation of the Fuel Handling Building Ventilation exhaust radiation monitors, the automatic function of the fuel storage area monitors will be removed. Transfer of system operation into the iodine removal mode will be by either of the two Fuel Handling Building Ventilation exhaust radiation monitors required by Specification 3.3.3. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.9.13 SPENT FUEL SHIPPING CASK MOVEMENT

The restriction on spent fuel shipping cask movement ensures that no fuel assemblies will be ruptured in the event of a spent fuel shipping cask accident. The dose consequences of this accident are within the dose guideline values of 10 CFR Part 100.

3/4.9.14 SPENT FUEL ASSEMBLY STORAGE

The restrictions placed on spent fuel assemblies stored in the spent fuel pool ensure that keff will not be greater than 0.95 under normal conditions, as discussed in TS 5.6.1.a. The requirement for 2000 ppm boron concentration ensures that k-eff will not be greater than 0.95 under accident conditions. The spent fuel storage has been designed and analyzed for a maximum enrichment of 5.0 weight percent U-235.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 Containment is designed and shall be maintained for a maximum internal pressure of 47 psig and a temperature of 271°F, coincident with a Double Design Earthquake.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4 or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-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 analysis 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 non-limiting core locations.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,811 ± 100 cubic feet at a nominal T_{avg} of 576°F for Unit 1 and 12,903 ± 100 cubic feet at a nominal T_{avg} of 577°F for Unit 2.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal 10.93 inch center-to-center distance between fuel assemblies placed in the storage racks.

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

DRAINAGE

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

CAPACITY

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

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter of February 6, 1995, as supplemented by letters dated March 23, and May 22, 1995, Pacific Gas and Electric Company (PG&E or the licensee) submitted a request for changes to the Technical Specifications (TS). The proposed amendments would allow for the storage of fuel with an enrichment not to exceed a nominal 5.0 weight percent (wt%) U-235 in the new (fresh) and spent fuel storage racks. The proposed changes would also clarify allowed substitution of fuel rods with filler rods and use of ZIRLO fuel cladding. The licensee's supplemental letters provided additional clarifying information and did not change the initial no significant hazards consideration determination that was published in the Federal Register on March 1, 1995 (60 FR 11138).

2.0 EVALUATION

The staff's evaluation of the criticality aspects of the proposed changes follows. The licensee's submittals did not include a request to increase burnup of the fuel.

Fuel Enrichment

The analysis of the reactivity effects of fuel storage in the new and spent fuel storage racks was performed with the three-dimensional multi-group Monte Carlo computer code, KENO-5a, using neutron cross sections generated by the NITAWL code package from the 27 energy group SCALE data library. Since the KENO-5a code package does not have depletion capability, burnup analyses were performed with the two-dimensional transport theory code, CASMO-3. CASMO-3 was also used to determine the reactivity effects of material and manufacturing tolerances. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the Diablo Canyon fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and absorber worth. The intercomparison between two independent methods of analysis (KENO-5a and CASMO-3) also provides an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a reactivity calculations, a minimum of 500,000 neutron histories were typically accumulated in each calculation. Experience has shown that this number of histories is quite sufficient to assure convergence of KENO-5a reactivity calculations. Based on the above, the staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Diablo Canyon new and spent fuel storage racks with a high degree of confidence.

The fresh fuel storage vault contains two 5 x 7 arrays of storage locations with each array providing 35 cells arranged on a 22-inch lattice spacing. The two arrays are separated from each other by about 27.5 inches. The storage vault is intended for the receipt and storage of fresh fuel under dry (air) conditions. However, to assure the criticality safety under normal and accident conditions and to conform to the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling, two separate criteria must be satisfied as defined in NRC Standard Review Plan (SRP), Section 9.1.1. These criteria state that the maximum reactivity of the fully loaded fuel racks shall not exceed a k_{eff} of 0.95 if fully flooded with unborated water or a k_{eff} of 0.98 assuming the optimum hypothetical low density moderation (e.g., fog or foam). The maximum calculated reactivity must include a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true k_{eff} will not exceed the calculated maximum value at a 95 percent probability, 95 percent confidence level (95/95).

Since Diablo Canyon may contain Westinghouse standard or optimized (OFA) fuel designs with a 17 x 17 fuel rod array, calculations were performed to determine the more limiting fuel type from a reactivity standpoint. The Westinghouse OFA fuel is limiting in the fully flooded condition while the standard fuel exhibits the higher reactivity under low-density optimum moderation conditions. The maximum k_{eff} for a fully loaded vault of OFA fuel enriched to 5.0 wt% U-235 was calculated to be 0.945 under fully flooded conditions. For the hypothetical low-density optimum moderation condition, the maximum calculated k_{eff} was 0.900 at a moderator density of approximately 8 percent of full density for a fully loaded vault of standard fuel. The calculations included a calculational bias and uncertainty derived from benchmark calculations, as well as uncertainties due to KENO-5a statistics, lattice spacing, fuel enrichment, and fuel density at the 95/95 probability/confidence level. The results conform to the acceptance criteria of SRP Section 9.1.1 and are, therefore, acceptable.

The storage racks in the spent fuel pool are divided into two regions. Region 1 contains 290 stainless steel storage cells with each cell surrounded on all four sides by Boraflex neutron absorber panels. The cells are spaced 10.93 inches apart with a 1.786 inch water flux-trap between two adjacent Boraflex panels. Region 2 consists of 1034 storage cells and contains no Boraflex. The cells are stainless steel with an inside dimension of 8.85 inches arranged on a 10.929-inch center-to-center spacing, providing a 1.899-inch water gap between the walls of the storage cells. The spent fuel racks are normally fully flooded by water borated to at least 2000 ppm of boron as required by the plant TS. However, to meet the criterion stated in

SRP Section 9.1.2, k_{eff} must not exceed 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity and flooded with unborated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity must include a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true k_{eff} will not exceed 0.95 at a 95/95 probability/confidence level.

Initial calculations for Region 1 have shown that OFA fuel gave a slightly higher rack reactivity than the corresponding enrichment for standard Westinghouse fuel. The spent fuel storage racks in Region 1 were reevaluated for 5.0 wt% U-235 enriched fuel moderated by pure water at 20°C with a density of 1.0 gm/cc, which results in the highest reactivity. For the nominal storage cell design in Region 1, uncertainties due to tolerances in fuel enrichment and density, fuel pellet diameter, storage cell inner diameter, stainless steel thickness, water gap thickness, Boraflex width and thickness, and boron-10 (B-10) loading were accounted for as well as eccentric fuel positioning. These uncertainties were appropriately determined at the 95/95 probability/confidence level. In addition, calculational and methodology biases and uncertainties due to benchmarking were included.

The reactivity calculations also considered the effects of Boraflex shrinkage and gap formation. All Boraflex panels were modeled with 4 percent shrinkage. Because of the design of the racks, two different gap assumptions were made, depending on whether the Boraflex panel is located in the rack interior or the rack periphery. The interior panels are held in place by a stainless steel cover plate that is spot welded every 12 inches along each vertical edge through small cutouts in the Boraflex. Because of the localized stresses that would develop by these restraints due to a maximum shrinkage of 4 percent of the Boraflex panel in the 12-inch interval, a gap of 0.48 inches was assumed to occur at the cutout location every 12 inches along the length of the panel. In addition, all perimeter Boraflex panels were assumed to have a 14-inch gap located at the same axial location (top 14 inches). Based on the results of blackness testing performed at other storage facilities, and on upper bound values recommended by Electric Power Research Institute (EPRI), the staff concurs that these assumptions bound the current measured data and future development of additional shrinkage and gaps. The final Region 1 design, when fully loaded with fuel enriched to 4.5 wt% U-235, resulted in a k_{eff} of 0.9421 when combined with all known uncertainties. This meets the staff's criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

To enable the storage of fuel assemblies with nominal enrichments greater than 4.5 wt% U-235, the concept of reactivity equivalencing was used. In this technique, which has been previously approved by the staff, credit is taken for the reactivity decrease due to the integral fuel burnable absorber (IFBA) material coated on the outside of the UO_2 pellet. Based on these calculations, the reactivity of the fuel rack array, when filled with fuel assemblies enriched to 5.0 wt% U-235 with each containing 16 IFBA rods, was found to be 0.9444, thus meeting the acceptance criterion of 0.95. The calculation assumed IFBA rods in the most reactive configuration with 2.25 mg/inch per rod of B-10. Fuel assemblies containing a nominal 36 mg/inch of B-10 are equivalent to assemblies containing 16 IFBA rods at 2.25 mg/inch per rod. Fuel assemblies containing a nominal 72 mg/inch B-10 are equivalent to assemblies containing 32 IFBA rods at 2.25 mg/inch per rod. The calculations included an uncertainty on the B-10 loading in IFBA rods.

As an alternative method for determining the acceptability of fuel storage in Region 1, the concept of burnup credit reactivity equivalencing was used. This is predicated upon the reactivity decrease associated with fuel depletion and has been previously accepted by the staff for spent fuel storage analysis. For burnup credit, a series of reactivity calculations are performed to generate a set of initial enrichment versus fuel assembly discharge burnup ordered pairs which all yield an equivalent k_{eff} less than 0.95 when stored in the spent fuel storage racks. This is shown in Attachment E, Figure 1 of the licensee's submittal dated February 6, 1995, in which a fresh 4.5 wt% enriched fuel assembly yields the same rack reactivity as an initially enriched 5.0 wt% assembly depleted to approximately 3.73 MWD/KgU.

A third alternative for storage of fuel assemblies enriched to 5.0 wt% U-235 and containing no IFBA rods in Region 1 consists of arranging the fuel in an alternating (checkerboard) configuration. Three configurations were analyzed; a checkerboard pattern of fuel assemblies and water-filled cells, a checkerboard pattern of fuel assemblies and assemblies with 32 IFBA rods (72 mg/inch B-10), and a pattern with alternate rows of fuel assemblies and water-filled cells. The licensee has stated that calculations show that the reactivity of an assembly containing a nominal minimum of 72 mg/inch B-10 is equivalent to the reactivity of a fuel assembly with 8,000 MWD/MTU cumulative burnup. The resulting 95/95 k_{eff} values were 0.852, 0.944, and 0.895, respectively, all meeting the NRC acceptance criterion of no greater than 0.95.

The Region 2 spent fuel storage racks were reanalyzed for storage of Westinghouse 17x17 fuel assemblies with nominal enrichments up to 5.0 wt% U-235 using the concept of burnup reactivity equivalencing. For Region 2, the Westinghouse standard fuel assembly design gave a slightly higher reactivity than the OFA. The same initial assumptions, biases and uncertainties as used for the Region 1 analyses were included, except for the design basis temperature and the effects of Boraflex shrinkage and gaps. Since the Region 2 racks contain no Boraflex, the temperature coefficient of reactivity is positive and a temperature of 150°F was assumed. A depletion uncertainty of 0.0005 times the burnup in MWD/KgU was assumed, resulting in an uncertainty of 0.02 Δk for fuel burned to 40 MWD/KgU. This uncertainty is consistent with current practice and is acceptable. The equivalencing showed that fresh standard Westinghouse fuel enriched to 1.74 wt% U-235 yields the same rack reactivity (k_{eff} = 0.9482) as 5.0 wt% fuel irradiated to 40 MWD/KgU. For OFA assemblies, fresh fuel enriched to 1.79 wt% U-235 was equivalent to 5.0 wt% fuel irradiated to 38.75 MWD/KgU, yielding a rack reactivity (k_{eff}) of 0.9462. These values meet the NRC acceptance criterion of 0.95 and are acceptable.

Fuel initially enriched to 5.0 wt% U-235 may also be stored in a checkerboard pattern in Region 2, alternating with cells filled with only water or non-fissile material. For this case, the maximum calculated reactivity, including uncertainties, was 0.9392.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the misloading of an assembly with an enrichment and burnup (or IFBA) combination outside of the acceptable area or pool temperatures exceeding 150°F, which could lead to an increase in reactivity for Region 2. However, for such events credit may be taken for the presence of approximately 2000 ppm of boron in the pool water required by TS 3.9.14.2 since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. In fact, the licensee has determined that only 400 ppm of boron is necessary to mitigate the worst postulated accident in any pool region. Therefore, the staff criterion of k_{eff} no greater than 0.95 for any postulated accident is met.

Use of Filler Rods

In the event that a limited number of fuel rods in an assembly are damaged and cannot be replaced by similar fuel rods, the licensee has proposed using zirconium alloy or stainless steel filler rods with the requirement that the analyses for substituting the filler rods in fuel assemblies must be performed with codes and methods that have been approved by the NRC and must be demonstrated to comply with all fuel safety design bases. This is consistent with NRC Generic Letter 90-02, Supplement 1, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications," and is, therefore, acceptable.

Use of ZIRLO Cladding

Another proposed change would allow the use of ZIRLO, in addition to Zircaloy-4, as an acceptable cladding material. ZIRLO is an improved zirconium-based fuel rod cladding material that has a lower corrosion rate and reduced radiation-induced growth. The staff has previously found ZIRLO to be acceptable and has revised 10 CFR 50.44 and 50.46 to include ZIRLO as an acceptable cladding material. Any use of ZIRLO clad fuel in the core will be evaluated using NRC-approved codes as part of the licensee's cycle-specific core reload safety evaluation. Therefore, this change is acceptable.

Technical Specification Changes

The following Technical Specification changes have been proposed as a result of the requested enrichment increase, as well as the proposed allowance for replacing fuel rods with filler rods, and the addition of ZIRLO as an acceptable fuel cladding. The staff finds these changes and the associated Bases changes acceptable.

(1) TS 3.9.14.1 and Figure 3.9-2 have been revised to allow the storage of spent fuel assemblies with initial enrichments up to 5.0 wt% U-235 in Region 2 of the spent fuel pool. Fuel pellet diameters are considered in combination with initial enrichment and cumulative burnup to encompass both Westinghouse standard and OFA fuel.

(2) TS 3.9.14.3 and Figure 3.9-3 have been added to include the requirements for acceptable fuel storage in Region 1. In addition, an action statement is included requiring suspension of all fuel movement and crane operations except to move the non-complying assemblies into an acceptable pattern.

(3) TS 5.3.1 has been changed to remove reference to the number of fuel rods in each assembly, nominal length of each fuel rod, and maximum fuel enrichment. In addition, the current allowance for fuel rod substitution as justified by analysis is being clarified to specify that the analysis be performed using NRC staff-approved methods, an allowance to use a limited number of lead test assemblies is being added, and ZIRLO fuel cladding is being allowed.

(4) TS 5.6 has been changed to correct the word "borated" with "unborated" and to specify the maximum fuel enrichment allowed to be stored in the racks.

Based on the review described above, the staff finds the criticality aspects of the proposed enrichment increase to the Diablo Canyon new and spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The proposed filler rod substitution and use of ZIRLO fuel rod cladding is also acceptable.

Although the Diablo Canyon TS have been modified to specify the above-mentioned fuel as acceptable for storage in the spent fuel racks, evaluations of reload core designs (using any enrichment) will, of course, be performed on a cycle-by-cycle basis as part of the reload safety evaluation process. Each reload design is evaluated to confirm that the cycle core design adheres to the limits that exist in the accident analyses and TS to ensure that reactor operation is acceptable.

3.0 PUBLIC COMMENTS

Public comments on the staff's proposed no significant hazards consideration determination (60 FR 11138) were provided by Jill ZamEk on behalf of the San Luis Obispo Mothers for Peace (MFP) by letter dated March 30, 1995.

The comments and staff responses follow:

Comment 1

The postulated fuel handling accident offsite thyroid doses could increase by a factor of 1.2. MFP finds these calculations arbitrary, suspect, and not at all reassuring. MFP is looking for an increase in the margin of safety at the plant - not an increase in the risk factor. MFP finds this added risk a significant hazard and unacceptable.

Response

The licensee's submittal only requests an increase in fuel enrichment for new and spent fuel storage. No request has been made at this time to increase fuel burnup and, thus, radioactivity in individual fuel rods and the spent fuel pool will not increase due to this amendment. Therefore, offsite thyroid doses from the postulated fuel handling accident will not change with the increase in fuel enrichment.

The licensee's discussion of the 1.2 factor increase in offsite thyroid doses resulting from a fuel handling accident refers to the bounding analysis contained in NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," for up to 5.0 weight percent U-235 and 60,000 megawatt-days per metric ton Uranium (MWD/MTU) burnup. The licensee has not stated that offsite thyroid doses will increase by a factor of 1.2 but only that the consequences of an enrichment of up to 5.0 weight percent (wt%) U-235 and an unchanged burnup will not approach the 10 CFR Part 100 values and are clearly bounded by the 1.2 factor (which included the higher burnup). The staff agrees with the licensee's conclusion, and further concludes that the offsite thyroid doses will not increase at all based on an increase in fuel enrichment alone.

While the licensee has not requested a higher burnup with this license amendment request, the staff anticipates that the licensee will subsequently submit such a request. In that event, the staff will review the radiological impacts of higher burnup on all design basis accidents, including the fuel handling accident.

Comment 2

The seismic issue is one that continues to jeopardize the safe operation of the plant at DCNPP. The recent earthquake in Japan has undeniably demonstrated that there exists no "earthquake-proof" structure. With the Hosgri Fault within a few miles of the plant, and other nearby faults, DCNPP is clearly vulnerable. MFP argues that increasing the radioactivity in the Spent Fuel Pools at the site unnecessarily increases the risks of a serious accident in the event of a seismic event.

Response

The increase in fuel enrichment (e.g., from 4.5 to 5.0 wt%) alone will not increase fission product inventory in fuel rods. Therefore, increased fuel enrichment will not increase radioactivity in the spent fuel pool. It follows then that the risk of a seismic event is unchanged based on an increase in fuel enrichment alone.

Despite this, the staff wishes to correct unsupported conclusions in the comment regarding the Kobe earthquake and its implications for the Diablo Canyon Nuclear Power Plant (DCPP). DCPP was designed and constructed in the 1970's. Since its original design, DCPP has undergone two extensive and thorough seismic reanalyses, the Hosgri reanalysis of the late 1970's and the Long-Term Seismic Program (LTSP) of the mid-1980's. The LTSP was performed in response to a license condition to conduct a comprehensive geosciences investigation. As part of the LTSP, a major seismic reassessment of the plant was conducted by the licensee and reviewed and approved by the NRC staff.

The scope of the Hosgri and LTSP assessments covered all the safety-related plant structures, systems and components including the spent fuel pool and the building. The spent fuel pool, which is founded on rock and constructed with thick (about 5 feet) reinforced concrete shear walls, is one of the most seismically rugged parts of the plant. Both the Hosgri and LTSP reanalyses assumed the occurrence of a large (magnitude greater than 7) earthquake on the Hosgri fault at a distance of about 4 kilometers from the plant. The seismic demand used in these analyses was based on near-field data recorded from a number of large earthquakes. The reanalyses demonstrated that the seismic capacity of DCPP is greater than the demand of a large nearby earthquake with significant margins.

Most of the loss of life in the recent earthquake in Kobe, Japan was due to the collapse of residential structures that were not seismically designed. Engineered structures that had earthquake damage were generally older and designed to codes that underestimated the size of the earthquake and its proximity to the city. Well engineered structures designed to more recent codes generally performed well with no significant structural damage. For example, the new Kobe City Hall sustained no structural damage. Based on the design and analyses of the DCPP and our review of developments in seismology and earthquake engineering, the NRC continues to have reasonable assurance as to the seismic adequacy of DCPP.

<u>Comment 3</u>

PG&E makes "analyses" to "verify" that an increase in the fuel enrichment would not involve a significant increase in the probability of [sic] consequences of an accident previously evaluated. MFP finds PG&E's assumptions questionable. In its discussion of non-borated water, optimum-density aqueous foam and soluble boron, PG&E's figures of "below 0.88" are dangerously close to criticality - criticality being 1.0. MFP is alarmed by this proposed reduction in the margin of safety.

<u>Response</u>

Normally, fresh fuel is stored temporarily in a dry environment in the new fuel storage vault pending transfer to the reactor core. Under these conditions, the reactivity of the storage racks when filled with fuel of the highest allowed enrichment is extremely subcritical (k-eff is usually less than 0.50 as compared to 1.0 for a critical system). However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation). Therefore, the NRC requires that the criticality safety analyses must address two independent accident conditions in conformance to General Design Criterion 62 of 10 CFR Part 50, Appendix A, which requires the prevention of criticality in fuel storage and handling. These two analysis conditions are:

(a) With the new fuel vault filled with fuel of the maximum permissible reactivity and flooded with pure water, the maximum k-effective shall not exceed 0.95, including mechanical and calculational uncertainties.

(b) With the new fuel vault filled with fuel of the maximum permissible reactivity and containing moderator at the (low) density corresponding to optimum moderation, the maximum k-effective shall be less than 0.98, including mechanical and calculational uncertainties.

The reactivity of the new fuel vault containing 4.5 wt% fuel for the low density accident (analysis condition (b) above) resulted in a k-effective of 0.880. The new analysis with 5.0 wt% fuel resulted in an increase of k-effective to 0.900. Both of these values are well below the NRC requirement of k-effective no greater than 0.98. Therefore, the PG&E analysis, which shows that k-effective remains below 0.98 for this optimum moderation condition, meets the NRC requirement and is not a reduction in the margin of safety, (i.e., defined as the difference between 0.98 and 1.0).

Comment 4

MFP finds that the proposed changes would create new hazards that have not been previously evaluated. MFP asserts that the increased radioactivity of the proposed fuel would impact not only the Spent Fuel Pools at DCNPP, but "low" level radioactive waste and storage, transportation of this waste, and all future handling. Again, MFP finds these increased hazards significant and unacceptable.

Response

Handling, storage, and transportation of low-level radioactive waste are not affected by the increase in fuel enrichment. Based on surveys of operating reactors, the NRC staff has determined that core thermal power is a more accurate indicator of radioactive waste production than fuel enrichment or burnup. Generation of radioactive waste is also dependent on the transport paths from the reactor coolant system to the radioactive waste processing systems. The NRC staff evaluates radioactive waste processing systems using a computer model based on core thermal power and transport paths. Therefore, changes in fuel enrichment or burnup do not alter the basis for staff acceptance of the means of handling, storage, and transportation of radioactive waste.

Comment 5

If PG&E's amendment request were to be granted, the current 18-month cycle for refueling at DCNPP would be extended to up to 24 months. MFP is concerned by this 6 month extension, because it lengthens the period for inspections, surveillances and maintenance for certain safety-related systems and equipment. Because of PG&E's unique rate settlement agreement (1988), PG&E gets paid only when it produces power. This provides PG&E with the incentive to postpone or rush maintenance in order to increase profits. PG&E's most recent outage was completed in an industry record time of 35 days. The Nuclear Regulatory Commission (NRC) sited [sic] 7 violations during this period, and also voiced concern regarding PG&E's rushed work and its severe cuts in staff: "... recent declining trends observed in housekeeping, engineering coordination with the plant, and procedural compliance have raised our concern. Additionally, your efforts to streamline your organization and reduce outage duration may further stress your safety programs." MFP asserts that PG&E's efforts to increase its profits jeopardizes safety. MFP further asserts that the results of the proposed changes in the TS for DCNPP would serve to augment an existing problematic situation and further threaten the safe operation of the plant.

Response

The amendment request in question only requests approval for storage of new and spent fuel with enrichment of up to 5 wt%. There have been no other requests to date from the licensee that support extended cycles. However, the staff does anticipate that the licensee will submit such a request at a future date and for that reason we will address this comment.

The NRC issued Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," on April 2, 1991. This GL provides guidance to licensees for the preparation of amendment requests to support 24-month fuel cycles.

The staff has generically evaluated changes from 18- to 24-month surveillances and has found that the safety impact is small due to redundant components in safety systems and other means to demonstrate during operation that components remain operable. While the staff has found the impact to be small in general, each licensee must perform a technical evaluation which supports this conclusion for the given facility. Also, licensees must demonstrate on a case by case basis that plant component histories based on surveillance and maintenance data support the conclusion that the safety effect is small. Licensees must also show that assumptions in the plant licensing basis remain valid based on an extended surveillance interval. The licensee's evaluation would include an assessment of increased calibration intervals and their effect on instrument errors to ensure that instrument drift will not result in errors that exceed assumptions of the safety analysis. The staff will review the licensee's supporting information to any proposed amendment request to increase the length of fuel cycles to ensure that the proposed changes do not have a significant effect on safety and will only approve amendment requests that are consistent with that conclusion.

In addition, the staff has identified certain benefits associated with extended surveillance intervals. For instance, less frequent testing reduces component wear which, on balance, tends to increase system reliability. On this basis, the staff recommended certain changes to surveillance requirements in GL 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation." While this GL addressed surveillance testing during power operation, similar considerations apply to surveillance requirements conducted during refueling outages.

Likewise, a significant portion of maintenance can be performed while the unit is on-line. The impact on safety of delaying for six months maintenance that can only be done during shutdown is small. In addition, starting July 10, 1996, licensees must meet 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," which requires that important equipment is maintained in accordance with licensee goals such that it can be reasonably assured of performing as required. The maintenance rule is not inconsistent with 24-month cycles.

In our Systematic Assessment of Licensee Performance (SALP) report for DCPP dated September 30, 1994, within the context of our addressing the licensee's overall superior performance the staff mentioned that certain trends were of concern. Our cover letter alerted the licensee to these areas and encouraged them to focus their attention in these areas to "assure continued superior safety performance." These are areas which the NRC will also continue to monitor to ensure that safety performance does not become unsatisfactory.

Conclusion

The NRC has considered MFP's comments and has concluded that there is nothing in them that would cause the staff to change the proposed no significant hazards consideration determination.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on (60 FR 30120). In this finding, the Commission determined that issuance of this amendment would not have a significant effect of the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (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 or to the health and safety of the public.

Principal Contributors: L. Kopp
M. Miller

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