DEC 2 1985

Docket Nos.: 50-361
and 50-362

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Mr. James C. Holcombe
Vice President - Power Supply
San Diego Gas & Electric Company
101 Ash Street
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Gentlemen:

Subject: Issuance of Amendment No. 39 to Facility Operating License NPF-10 and Amendment No. 28 to Facility Operating License NPF-15 San Onofre Nuclear Generating Station, Units 2 and 3

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 39 to Facility Operating License No. NPF-10 and Amendment No. 28 to Facility Operating License No. NPF-15 for the San Onofre Nuclear Generating Station, Units 2 and 3, located in San Diego County, California. The amendments modify the Technical Specifications concerning maximum enrichment of the fuel assemblies and the criticality requirements for storage of fuel in the fuel storage areas.

These amendments were requested by your letters of August 23, October 10 and 16, 1985, and are covered by Proposed Change Number PCN-199.

A copy of the Safety Evaluation supporting the amendments is also enclosed.

Sincerely,

Original signed by:
George W. Knighton

George W. Knighton, Director
PWR Project Directorate No. 7
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 39 to NPF-10
2. Amendment No. 28 to NPF-15
3. Safety Evaluation

cc: See next page

*Previous concurred on by:

DL:LB#3*	DL:LB#3*	OELD*	DPWRL:B
JLee/yt	HRood	Vogler	GWKnighton
11/14/85	11/14/85	11/20/85	11/29/85

Mr. Kenneth P. Baskin
Southern California Edison Company

San Onofre Nuclear Generating Station
Units 2 and 3

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 39
License No. NPF-10

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 2 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and the City of Anaheim, California (licensees) dated August 23, October 10 and 16, 1985, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the 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 set forth in 10 CPR Chapter I;

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

   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 amendment and Paragraph 2.C(2) of Facility Operating License No. NPF-10 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. 39, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective immediately and is to be fully implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W Knighton

George W. Knighton, Director
PWR Project Directorate No. 7
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: DEC 2 1985

DEC 2 1985

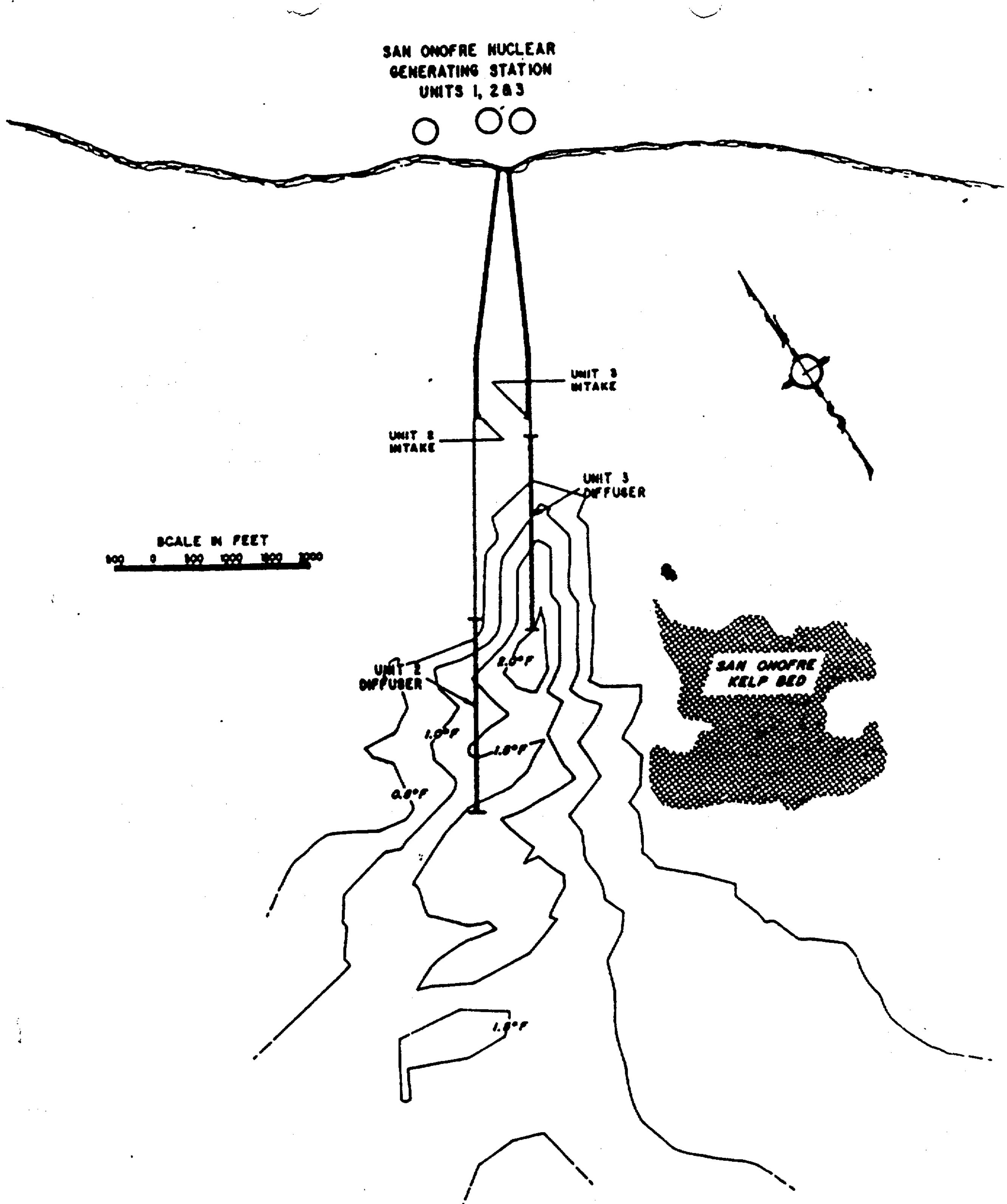
ATTACHMENT TO LICENSE AMENDMENT NO. 39

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
5-6	5-5
5-7	5-8



SITE BOUNDARY FOR LIQUID EFFLUENTS

FIGURE 5.1-4

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1900 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.1 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full length and 8 part length control element assemblies.

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 of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,800 + 600/-0 cubic feet at a nominal T_{avg} of 582.1°F.

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 the FSAR.

b. A nominal 12.75 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.2 The k_{eff} for new fuel for the first core loading stored dry in alternate rows and columns in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 60'6".

CAPACITY

5.6.4 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 800 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

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.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup and cooldown cycles at rates $\leq$ 100°F/hr.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 545°F; cooldown cycle - T_{avg} from $\geq$ 545°F to $\leq$ 200°F.
	500 pressurizer heatup and cooldown cycles at rates $\leq$ 200°F/hr.	Heatup cycle - Pressurizer temperature from < 200°F to > 653°F; cooldown $\geq$ 653°F to $\leq$ 200°F
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psia with RCS temperature in accordance with Specification 3.4.8.
	200 leak testing cycles.	RCS pressured to 2250 psia with RCS temperature greater than minimum for hydrostatic testing, but less than minimum RCS temperature for critically.
	200 seismic stress cycles.	Subjection to a seismic event equal to one half the design basis earthquake (DBE).
	480 cycles (in any combination) of reactor trip, turbine trip with delayed reactor trip, or complete loss of forced reactor coolant flow.	Trip from 100% of RATED THERMAL power; turbine trip (total load rejection) from 100% of RATED THERMAL POWER followed by resulting reactor trip; simultaneous loss of all Reactor Coolant Pumps at 100% of RATED THERMAL POWER.

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 28
License No. NPF-15

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 2 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and the City of Anaheim, California (licensees) dated August 23, October 10 and 16, 1985, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the 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 set forth in 10 CPR Chapter I;

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

   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 amendment and Paragraph 2.C(2) of Facility Operating License No. NPF-15 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. 28, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective immediately and is to be fully implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Director
PWR Project Directorate No. 7
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: DEC 2 1985

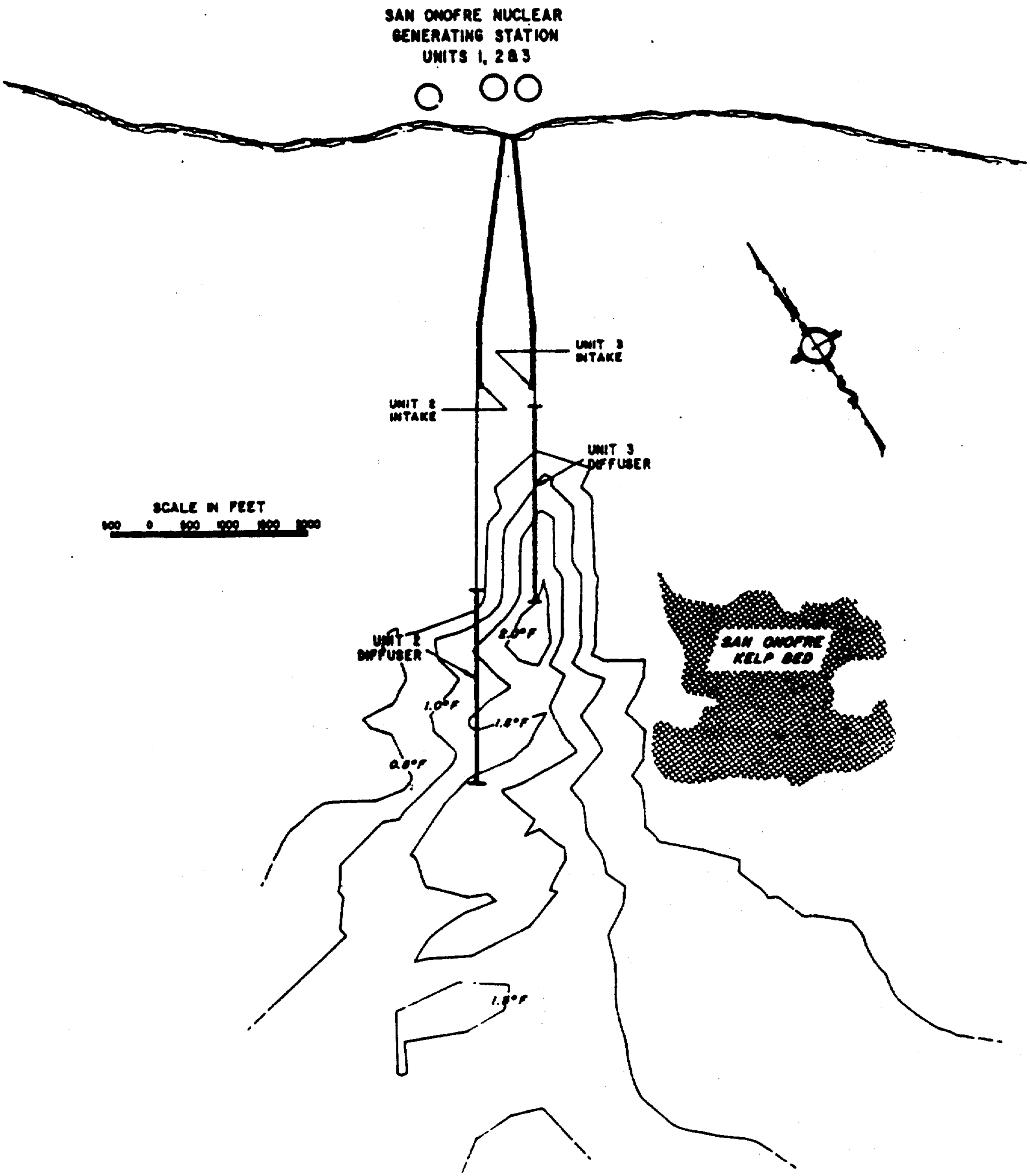
ATTACHMENT TO LICENSE AMENDMENT NO. 28

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
5-6	5-5
5-7	5-8



SITE BOUNDARY FOR LIQUID EFFLUENTS

FIGURE 5.1-4

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1900 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.1 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full length and 8 part length control element assemblies.

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 of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,800 + 600/-0 cubic feet at a nominal T_{avg} of 582.1°F.

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

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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 the FSAR.

b. A nominal 12.75 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.2 The k_{eff} for new fuel for the first core loading stored dry in alternate rows and columns in the spent fuel storage racks shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 60'6".

CAPACITY

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TABLE 5.7-1

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	500 pressurizer heatup and cooldown cycles at rates $\leq$ 200°F/hr.	Heatup cycle - Pressurizer temperature from < 200°F to > 653°F; cooldown $\geq$ 653°F to $\leq$ 200°F
	10 hydrostatic testing cycles.	RCS pressurized to 3125 psia with RCS temperature in accordance with Specification 3.4.8.
	200 leak testing cycles.	RCS pressured to 2250 psia with RCS temperature greater than minimum for hydrostatic testing, but less than minimum RCS temperature for critically.
	200 seismic stress cycles.	Subjection to a seismic event equal to one half the design basis earthquake (DBE).
	480 cycles (in any combination) of reactor trip, turbine trip with delayed reactor trip, or complete loss of forced reactor coolant flow.	Trip from 100% of RATED THERMAL power; turbine trip (total load rejection) from 100% of RATED THERMAL POWER followed by resulting reactor trip; simultaneous loss of all Reactor Coolant Pumps at 100% of RATED THERMAL POWER.

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

SAFETY EVALUATION

AMENDMENT NO. 39 TO NPF-10

AMENDMENT NO. 28 TO NPF-15

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

DOCKET NOS. 50-361 AND 50-362

INTRODUCTION

Southern California Edison Company (SCE), on behalf of itself and the other licensees, San Diego Gas and Electric Company, The City of Riverside, California, and The City of Anaheim, California, has submitted several applications for license amendments for San Onofre Nuclear Generating Station, Units 2 and 3. One such request, Proposed Change PCN-199, is evaluated herein. This change would revise the technical specifications relating to the maximum enrichment of the fuel assemblies and the criticality requirements for storage of fuel in the fuel storage areas (reference PCN-199). These technical specifications are being changed because the Cycle 3 fuel enrichment is being changed from 3.7% to 4.1% to accommodate an 18-month refueling cycle. Other amendments have been requested to modify the technical specifications associated with reactor operation with the revised enrichment, and are now being evaluated by the NRC staff. The amendments associated with PCN-199 were requested by the licensee's letters of August 23, October 10, and October 16, 1985. The request includes the results of analyses on the effect of the increased U-235 fuel enrichment on the criticality aspects of both the new and spent fuel storage racks at SONGS 2 and 3. The staff evaluation of the proposed change is given below.

ANALYSIS METHODS

The analysis of the criticality aspects of the storage of SONGS 2 and 3 fuel assemblies having a fuel enrichment of 4.1 wt% U-235 was performed by SCE. The previous analysis for the SONGS 2 and 3 new and spent fuel storage racks was performed by Nuclear Energy Services (NES) for the storage of fuel assemblies having a fuel enrichment of 3.7 wt% U-235. The SCE analysis methods consist of the KENO-IV/S and NITAWL-S codes which are part of the SCALE-2 (Ref. 3) code package and of the EPRICELL-2 and DANCOFF codes which are part of the ARMP code package (Ref. 4). The KENO-IV/S code is a multigroup criticality program for the determination of a system's effective neutron multiplication factor (K_{eff}). This code has the capability of modeling complex, three-dimensional systems. The NITAWL-S code is used to perform the resonance self-shielding calculations for those nuclides with resonance parameters that are important to the criticality analysis. The Nordheim integral treatment (Ref. 5) is used by NITAWL-S. Both NITAWL-S and KENO-IV/S codes were used with the 27 group neutron cross section library furnished with the SCALE-2 code package. The EPRICELL-2 code is a fuel pin cell code used to determine fuel material concentrations for the 4.1 wt% fuel and Dancoff factors for input to the NITAWL-S code. The DANCOFF code was used to calculate Dancoff factors with Sauer's method for checking values computed using EPRICELL-2 and to calculate Dancoff factors for water densities for which the use of EPRICELL-2 is inappropriate.

The SCE analysis methods were benchmarked against critical experiments described in Reference 6. Six experiments were analyzed, that is, experiments number 001, 004, 007, 008, 013, and 014. These experiments used arrays of UO_2 fuel rods with a fuel enrichment of 4.29 wt% U-235. Other physical characteristics of these experiments made them suitable for benchmarking analysis methods for analysing the SONGS 2 and 3 new and spent fuel storage racks. The SCE benchmarking results are presented in a document (Ref. 7) transmitted by Reference 2. These results indicate that the SCE analysis methods underpredict K_{eff} with a bias of -0.01322 at the 95% confidence level. In another document (Ref. 8) transmitted by Reference 2, Torrey Pines Technology (a Division of GA) stated, in a letter attached to Reference 8, that SCE results for the nominal K_{eff} of a reference case with 4.1 wt% fuel were in good agreement with a value obtained by extrapolating the NES results for 3.7 and 3.9 wt% fuel to 4.1 wt% fuel. SCE stated that the Torrey Pines extrapolation was erroneous and that the NES extrapolated value for 4.1 wt% fuel should have been 0.8890 as compared to the SCE value of 0.90111 (Ref. 9). The agreement is still acceptable considering the different analysis methods used by SCE and NES.

Although the SCE benchmarking is not extensive, the results obtained and the documentation provided indicate that the use of these methods by SCE for the analysis of the SONGS 2 and 3 new and spent fuel storage racks is acceptable.

SPENT FUEL STORAGE RACK ANALYSIS

The criticality of fuel assemblies in the SONGS 2 and 3 spent fuel racks is prevented, primarily, by limiting the U-235 enrichment of the uranium in the UO_2 fuel rods. The SCE racks consist of stainless steel cans of square cross section having an outer dimension of 8.81 inches and a nominal wall thickness of 0.125 inches. These storage cans are arranged in a square array with a pitch of 12.75 inches. Songs 2 and 3 are provided with separate spent fuel pools. Each pool has a present capacity to store 800 fuel assemblies. The criticality criterion that the fuel assemblies with a fuel enrichment of 4.1 wt% U-235 must meet is that the effective neutron multiplication factor, K_{eff}, shall be less than or equal to 0.95 for normal and postulated accident conditions. The K_{eff} shall include all biases and uncertainties at least at a 95/95 probability/confidence level. Even though the pools contain borated water, the analysis must assume unborated water when normal conditions are being considered.

SCE performed a number of analyses to determine the sensitivity of a reference K_{eff}, based on nominal rack dimensions and a pool water temperature of 68°F, to variations in pool water temperature, rack pitch, eccentric fuel storage in the racks, and water in the fuel pin gap for 1% failed fuel. The fuel assemblies were assumed to consist of a 16 X 16 array of fuel rods with appropriate spaces for 5 guide tubes, with a fuel enrichment of 4.1 wt% U-235, and with the UO_2 at 94.75% of theoretical density. The effect of a dropped fuel assembly on K_{eff} was also determined. The dropped fuel assembly accident was postulated to occur by dropping a fuel assembly on top of a loaded fuel storage location. All other possible dropped fuel assembly accidents result in greater than 24 inches of water between the dropped fuel assembly and the assemblies in the racks. SCE states that these dropped fuel assembly accidents are not considered further since the dropped fuel assembly is, in effect, isolated from a criticality standpoint from the fuel assemblies stored in the racks. This is an acceptable assumption to make in these analyses.

The SCE analyses determined that the K_{eff} of the reference case is equal to 0.90111 for fresh fuel assemblies containing uranium enriched to 4.1 wt% in U-235. The uncertainties and biases are:

(1)	0.00266	Calculation uncertainty from benchmarking (2s)
(2)	0.01332	Calculation to measured bias from benchmarking
(3)	0.01503	Minimum rack pitch and eccentric fuel load
(4)	0.00192	Most reactive temperature (39°F)
(5)	0.00214	Dropped fuel assembly accident
(6)	0.00008	Waterlogged fuel pins (1% failed)
	0.03515	Total rack biases and uncertainties

The total uncertainties and biases on the K_{eff} associated with SCE's analysis of the Songs 2 and 3 spent fuel racks is 0.03515 and, therefore, the maximum K_{eff} is equal to 0.936 for the 4.1 wt% enrichment fuel assemblies. Since the calculational uncertainty is taken as twice the standard deviation of the data and other uncertainties are for worst case conditions, SCE meets the intent of staff guidance in the determination of the biases and uncertainties at least at a 95/95 probability/confidence level.

SCE considered the dropping of a fuel assembly and included its reactivity effect in its uncertainty allowance. Other postulated accidents that were considered were judged to have an insignificant reactivity effect. In any case, the staff believes that no other postulated accident would cause a criticality accident because credit may be taken for the boron in the pool water by invoking the Double Contingency Principle. This is acceptable.

The previous calculations for SONGS 2 and 3 were performed by NES for 3.7 wt% enrichment fuel assemblies. NES obtained a worst case K_{eff} 0.946. Since the worst case SCE value for 4.1 wt% enrichment fuel assemblies is 0.936, there is an apparent discrepancy between these two results. SCE states (Ref. 9) that this difference can be explained on the basis of its analysis model enhancements. To evaluate this contention, we can extend the SCE and NES results for 4.1% fuel which was discussed previously for nominal K_{eff}s for the reference case. Using NES results provided by SCE (Ref. 9), we construct the following table:

	SCE	NES
Nominal K_{eff} for 4.1 wt% fuel	.9011	.8890*
Calculation to Measurement bias	.0133	.0487
Uncertainties (including pool temperature and various worst case conditions)	.0219	.0265
Total Worst Case K_{eff}	.9363	.9642

The comparison shown in the table demonstrates that the bulk of the reactivity difference is caused by the calculation to measurement bias term. Since the methods used by SCE are clearly an improvement over those used by NES (mainly diffusion theory analysis), the smaller SCE value for the calculation to measurement bias is reasonable as compared to the very conservative NES value. Therefore, we conclude that there is no apparent discrepancy between SCE and NES results based on our review of the available information.

Based on our review, we conclude that the SCE value of K_{eff} equal to 0.936 meets the staff criterion of 0.95, including uncertainties and biases, for the storage of 4.1 wt% enrichment fuel assemblies in the SONGS 2 and 3 spent fuel storage racks.

NEW FUEL STORAGE RACK ANALYSIS

The criticality of fuel assemblies in the SONGS 2 and 3 new fuel racks is prevented, primarily, by limiting the U-235 enrichment of the uranium in the UO_2 fuel rods. The SCE new fuel racks consist of stainless steel structural members that form square cans. The structural material is neglected in the analysis except for the four corner angle pieces. The storage locations are arranged in a square array with two pitches. The first pitch has a nominal value of 29 inches and the second a nominal value of 38 inches. SONGS 2 and 3 are provided with separate new fuel storage racks. Each new fuel storage facility has a present capacity to store 80 fresh (unirradiated) fuel assemblies. The new fuel storage racks are normally in an air (dry) configuration.

*The NES value was extrapolated by SCE from NES results from 3.7 and 3.9 wt% fuel. The NES nominal results included, according to SCE, an arbitrary NES added value of 0.01.

The criticality criteria that the fuel assemblies with a fuel enrichment of 4.1 wt% U-235 must meet are that the effective neutron multiplication factor, K_{eff}, shall be less than or equal to 0.95 for the dry storage racks, that K_{eff} shall be less than or equal to 0.95 when the racks are flooded with pure water, and the K_{eff} shall be less than or equal to 0.98 when the racks are immersed with low-density hydrogenous material due to such causes as, for example, mist, fog, or fire-fighting foam. The K_{eff} shall include all biases and uncertainties at least at a 95/95 probability/confidence level. SCE has performed a comparison of calculational results to criteria in a manner that is different than staff practices. In this evaluation, we will adapt the available SCE results, as required, to compare to the listed criteria.

SCE performed calculations for the dry configuration modeled with a finite lateral geometry including the effect of the concrete walls. The racks are completely enclosed by concrete walls and does not have an open wall in the modeling. SCE calculated K_{eff} for this configuration as 0.384, which includes twice the standard deviation of the KENO-IV/S statistical uncertainty. SCE has not established uncertainties in its benchmarking for this type of configuration. However, its use of the calculational methods in the documentation presented clearly establishes SCE's ability to calculate these configurations. Even assuming large errors and uncertainties, SCE easily meets the criterion on K_{eff} for this dry configuration.

SCE performed calculations for the fully flooded new fuel storage racks with the same finite lateral geometry as for the dry configuration. For this configuration, the maximum K_{eff} as a function of water density (0 to 1.0 gm/cc water density) occurs at the maximum water density and was calculated to be 0.86294. Since the flooded new fuel storage racks are similar to the spent fuel storage racks, the spent fuel pool uncertainty in K_{eff} of 0.0352 may be used as a good approximation. The worst K_{eff} for the flooded new storage racks is thus equal to 0.898. Therefore, the fully flooded racks meet the criterion on K_{eff} of being equal to or less than 0.95.

For the extreme, low-density hydrogenous moderator conditions, the SCE finite lateral geometry calculations indicate a second peak in the K_{eff} versus water density curve. This K_{eff} is equal to 0.813 at a water density of about 0.05 gm/cc. Even though uncertainties have not been clearly established for this configuration and assuming even large uncertainties, SCE meets the criterion of the worst K_{eff} being less than or equal to 0.98 for this configuration.

A fuel handling accident for the dry configuration would increase the K_{eff} of the dry configuration by a delta-K of 0.08527. Adding this to the K_{eff} of the dry configuration would give a K_{eff} of 0.469. Again this fuel handling accident would remain, including assumed uncertainties and biases, well below the criterion on K_{eff} of 0.95. This fuel handling accident was modeled conservatively in an infinite lateral geometry and by misloading of every storage location with fuel assemblies.

Based on our review and interpretation of SCE calculations, we conclude that SCE meets all the criteria for the storage of 4.1 wt% fuel assemblies in the new fuel storage racks and we also conclude that the uncertainty analysis assumed meets the intent of the 95/95 probability/confidence level requirement.

FUEL TRANSFER CARRIER ANALYSIS

Although the fuel transfer carrier is not normally a part of our review of new and spent fuel pool racks, we have reviewed the SCE results for the 4.1 wt% enriched fuel assemblies. The calculations were performed for two assemblies and for a water density of 1 gm/cc at 68°F. The SCE results indicate that the K_{eff} is equal to 0.908 including a bias term. This result is certainly acceptable since it predicts a margin to criticality.

TECHNICAL SPECIFICATIONS

Technical Specification 5.3 REACTOR CORE

Based on our review, it is acceptable to change this specification from a maximum enrichment of 3.7 to 4.1 wt% U-235.

Technical Specification 5.6.1 CRITICALITY

Our review indicates that the licensee needs to revise this specification to include the new value of the uncertainty of 0.035 delta K/K.

SUMMARY OF EVALUATION

Based on our review, we conclude that SONGS 2 and 3 16X16 fuel assemblies having a maximum enrichment of 4.1 wt% uranium-235 may be stored in the new and spent fuel racks. Our conclusion is based on the following:

1. The criticality calculations have been performed with acceptable methods and have been benchmarked,
2. Uncertainties have been accounted for,
3. Postulated accidents have been considered, and
4. The effective neutron multiplication factor, including a consideration of the various uncertainties and biases, meets our acceptance criteria.

CONTACT WITH STATE OFFICIAL

The NRC staff has advised the Chief of the Radiological Health Branch, State Department of Health Services, State of California, of the proposed determinatins of no significant hazards consideration. No comments were received.

ENVIRONMENTAL CONSIDERATION

These amendments involve changes in the installation or use of facility components located within the restricted area. The licensee has stated that the average level of irradiation of the irradiated fuel discharged from the reactor will not exceed 33,000 megawatt-per metric ton. The staff has determined that the amendments involve no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or commulative occupation radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on such findings. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec. 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of these amendments.

CONCLUSION

Based upon our evaluation of the proposed changes to the San Onofre Units 2 and 3 Technical Specifications, we have concluded that: there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable, and are hereby incorporated into the San Onofre 2 and 3 Technical Specifications.

Dated: DEC 2 1985

REFERENCES

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2. Letter from M. O. Medford (SCE) to Director (NRR), October 10, 1985.

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8. "Criticality Calculations for Fuel Storage Systems," C. W. Gabel, DC-1858, June 22, 1985.

9. Letter from M. O. Medford (SCE) to G. W. Knighton (NRC) October 16, 1985.

ISSUANCE OF AMENDMENT NO. 39 TO FACILITY OPERATING LICENSE NPF-10
AND AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NPF-15
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

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