- Technical Specification 3.9.12 (Bases 3/4.9.12) was revised to allow both trains of the Fuel Handling Building Post-Accident Clean-up Filter System to be out of service during the construction period for reracking the spent fuel pool. This revision to Technical Specification 3.9.12 was required to allow continued operation of the spent fuel handling machine without fuel, temporary gantry crane and cask handling crane with the fuel handling building equipment hatches open. Compliance with this revised Technical Specification 3.9.12 will ensure that with a minimum fuel decay time of 88 days, the radiological consequences of the worst postulated heavy load drop in the pools will not result in releases that exceed 25 percent of the 10 CFR 100 limits at the exclusion area boundary.

- Revision to design features section of 5.6.3 of the Operating License will provide consistency with Technical Specification 3.9.11 value of 23 feet of water to be maintained over the top of irradiated fuel assemblies.

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

Sincerely,

Lawrence E. Kokajko, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

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

cc w/enclosures:
See next page

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4/26/90	4/26/90	4/27/90	4/ /90

No OGC concur

[S02/3 AMD 68308/9]

Messrs Ray and Cotton
Southern California Edison Company

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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 NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
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 Southern California Edison Company (SCE) on behalf of itself and San Diego Gas and Electric Company, the City of Riverside, California and the City of Anaheim, California (licensees) dated March 10, April 19, May 4, May 19, June 1, June 2, September 22, November 2, November 9, 1989, January 18, February 9, February 16, and March 20, 1990, 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 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

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P PDC

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. 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. 87 , are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 1, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

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 area of change. Also enclosed are the following overleaf pages to the amended pages.

AMENDMENT PAGE	OVERLEAF PAGE
3/4 9-7	--
3/4 9-13	--
3/4 9-13a	3/4 9-14
3/4 9-16	3/4 9-15
B 3/4 9-2	B 3/4 9-1
B 3/4 9-2a	--
B 3/4 9-3	--
B 3/4 9-4	--
5-7	--
5-7a	5-8
5-12 (Figure 5.6-1)	5-11 (Table 5.7-1)
5-13 (Figure 5.6-2)	--
5-14 (Figure 5.6-3)	--
5-15 (Figure 5.6-4)	--

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

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

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

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

c. Installation or removal of the cask pool cover over the cask pool with fuel in the cask pool. The cover, fuel, and racks will be removed from the cask pool on completion of the reracking process.

d. The lift of construction loads, including the temporary gantry crane and the old and the new fuel storage racks (including lifting equipment and rigging), above the cask pool with the cask pool cover in place and fuel in the cask pool. This includes temporary storage of these construction loads on the cask pool cover during construction. These lifts are prohibited prior to a minimum fuel decay time of 88 days for all stored spent fuel assemblies.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the fuel handling machine in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.7 Fuel handling machine interlocks and physical stops which prevent fuel handling machine travel with loads in excess of 2000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to fuel handling machine use and at least once per 7 days thereafter during fuel handling machine operation.

REFUELING OPERATIONS

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel handling building post-accident cleanup filter systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building post-accident cleanup filter system inoperable, fuel movement within the storage pool or operation of fuel handling machine over the storage pool may proceed provided the OPERABLE fuel handling building post-accident cleanup filter system is capable for being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no fuel handling building post-accident cleanup filter system OPERABLE, suspend all operations involving movement of fuel within the storage pool or operation of fuel handling machine over the storage pool until at least one fuel handling building post-accident cleanup filter system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

d. Temporary exception to item (a) and (b) above, applicable only during spent fuel pool reracking construction activities:

   With no fuel handling building post-accident cleanup filter system OPERABLE, all spent fuel pool reracking construction activities including continued operation of the fuel handling machine without fuel, cask handling crane or the temporary gantry crane are permitted provided that the irradiated fuel in the storage pools has decayed for a minimum of 88 days and that no more than 480 irradiated fuel assemblies are stored in the pools.

   Fuel assemblies will only be moved with the post accident cleanup filter system OPERABLE per a and b above.

SURVEILLANCE REQUIREMENTS (Continued)

1. Verify that with the system operating at a flow rate of 12925 cfm ± 10% and recirculating through the HEPA filters and charcoal adsorbers, the total bypass flow of the system through the system diverting valves, to the facility vent is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the cleanup filter system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 12925 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. Verifying a system flow rate of 12925 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less that 7.3 inches Water Gauge while operating the system at a flow rate of 12925 cfm ± 10%.

2. Verifying that on a Fuel Handling Isolation (FHIS) test signal, the system automatically isolates normal ventilation and starts recirculation through the HEPA filters and charcoal adsorber banks.

3. Verifying that the heaters dissipate 28.7 ± 1.5 kw for E464, 32.3 ± 1.7 kw for E465, and 3.8 ± 0.2 kw for E652 when tested in accordance with ANSI N510-1975 with the measured heater dissipation corrected to correspond to nominal voltage.

4.9.12 The above required fuel handling building post-accident cleanup filter systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 12925 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 12925 cfm ± 10%.

3/4.9 REFUELING OPERATIONS

3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.13 The boron concentration in the spent fuel pool shall be maintained at a level greater than or equal to 1850 ppm.

Applicability: With fuel assemblies in the spent fuel pool.

Action: With the requirement of the above specification not satisfied:

Immediately suspend all additions or movement of fuel in the spent fuel pool and take action to restore the boron concentration to a value equal to or greater than 1850 ppm.

SURVEILLANCE REQUIREMENTS

4.9.13 A sample of spent fuel pool water shall be collected and analyzed for boron concentration at least:

a. Once per month and

b. Within 72 hours prior to any fuel movement.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

Five finger CEAs are removed from the reactor vessel either along with the associated fuel bundle utilizing the refueling machine or can be removed without the associated fuel bundle utilizing the refueling machine auxiliary hoist. The four finger CEAs are inserted through the upper guide structure with two fingers in each of the two adjacent fuel bundles in the periphery of the core. The four finger CEAs are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane or the refueling machine auxiliary hoist.

Coupling and uncoupling of the CEAs and the CEDM drive shaft extensions is accomplished using one of the gripper operating tools. The coupling and uncoupling is verified by weighing the drive shaft extensions.

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING

A. Refueling Operations

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in six fuel assemblies, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is based on the calculated results which demonstrate that, with credit taken for the fuel handling building filters, the offsite doses would be well within (less than 25%) the 10 CFR 100 limits.

B. Spent Fuel Pool Reracking Construction Activities

The restriction on movement of heavy loads over spent fuel ensures that in the event a heavy load is dropped:

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta K/K conservative allowance for uncertainties. Similarly, the boron concentration value of 2350 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING (Continued)

1. The radiological consequences due to complete rupture of all spent fuel assemblies in the spent fuel pool and the cask pool (480 maximum), will remain below (less than 25% of) the exposure limits of 10 CFR 100 for offsite doses as long as spent fuel has received a minimum of 88 days decay time. This analysis takes no credit for fuel handling building filters (i.e., the fuel handling building hatches are open).

2. Any possible distortion of all fuel assemblies and racks will not result in a critical array and K_{eff} will remain less than 0.95, as long as fuel is stored per Technical Specifications 5.6, "Fuel Storage," and 3.9.13, "Spent Fuel Pool Boron Concentration."

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling train be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capacity. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION

The minimum requirement of 1850 ppm boron ensures that $k_{eff} \leq 0.95$ in the Region II racks in the event of fuel assembly misloading with an enrichment/burnup combination not meeting the criterion for storage in Region II.

Calculations show that with 1800 ppm boron the Region II racks can be completely filled with misloaded fresh unshimmed fuel with an assembly average enrichment of up to 4.1 w/o and maintain $k_{eff} \leq 0.95$, including all uncertainties. Therefore, 1850 ppm of boron is specified to allow for measurement uncertainty.

3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves 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 STORAGE 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 accident analysis.

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

The limitations on the fuel handling building post-accident cleanup filter 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 accident analyses.

An exception to these limitations, the fuel handling building post-accident cleanup filter system can be taken out of service during reracking construction activities based on the new limitation of 88 days minimum decay time and limiting the number of spent fuel assemblies to a maximum of 480. These added limitations ensure that all potential radioactive releases are enveloped by existing accident analysis without regard to the OPERABILITY of this system.

Cumulative operation of the system with the heaters on for at least 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

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 and when fully loaded with fuel which meets the burnup requirement of Technical Specification 5.6.2. This includes a conservative allowance for uncertainties as described in the UFSAR.

b. A nominal 10.40 inch center-to-center distance between fuel assemblies placed in the Region I storage racks and a nominal 8.85 inch center-to-center distance between fuel assemblies placed in the Region II storage racks.

5.6.2 The enrichment-fuel assembly discharge burnup shall be above (greater than) the curves shown on Figure 5.6-1, for Unit 2 fuel assemblies, or Figure 5.6-2 for Unit 1 fuel assemblies for unrestricted storage.

New or burned fuel which does not meet the enrichment versus discharge burnup criteria of Figures 5.6-1/5.6-2 may be stored in Region I. It may also be stored in Region II if all the following conditions are met.

Fuel Type 1 = new or burned fuel which does <u>not</u> meet the criteria of Figures 5.6-1/5.6-2.

Fuel Type 2 = fuel which <u>does</u> meet the criteria of Figures 5.6-1/5.6-2.

a. Fuel Type 1 shall have initial enrichment $\leq$ 4.1 w/o U-235.

5.6.2 (Continued)

b. Fuel Type 1 shall be stored in Region II in either (not both simultaneously) a checkerboard pattern or an alternating row pattern per Figure 5.6-3.

c. Fuel Type 1 shall be separated from Fuel Type 2 by at least one (1) completely empty row of storage cells.

d. Fuel Type 1 shall not be stored in the same row as Fuel Type 2.

e. One (1) completely empty row of Region II storage cells shall separate Fuel Type 1 stored in Region II from fuel storage Region I.

f. Except for the purposes of a fuel reconstitution station described below (g), Fuel Type 1 and Fuel Type 2 storage arrays shall not alternate in Region II.

g. For purposes of fuel reconstitution/inspection work, it is permissible to have the three (3) row (empty - every other Fuel Type 1 - empty) arrangement of Figure 5.6-4 any where in the Fuel Type 2 storage array. Additional empty rows are allowed. This three (3) row pattern may be repeated in the Fuel Type 2 storage array if at least eight (8) rows separate repetitions.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below the Technical Specification 3.9.11 value (23 feet above the top of irradiated fuel assemblies seated in the storage racks).

CAPACITY

5.6.4 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1542 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.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray System		

Calculational Method:

1. The spray cycle is defined as any initiation and termination of main or auxiliary spray flow through the pressurizer spray nozzle.

2. If the maximum temperature difference between the pressurizer and the pressurizer spray during the spray cycle exeeds 200°F, each spray cycle and the corresponding temperature difference is logged.

3. The spray system usage factor is calculated as follows:

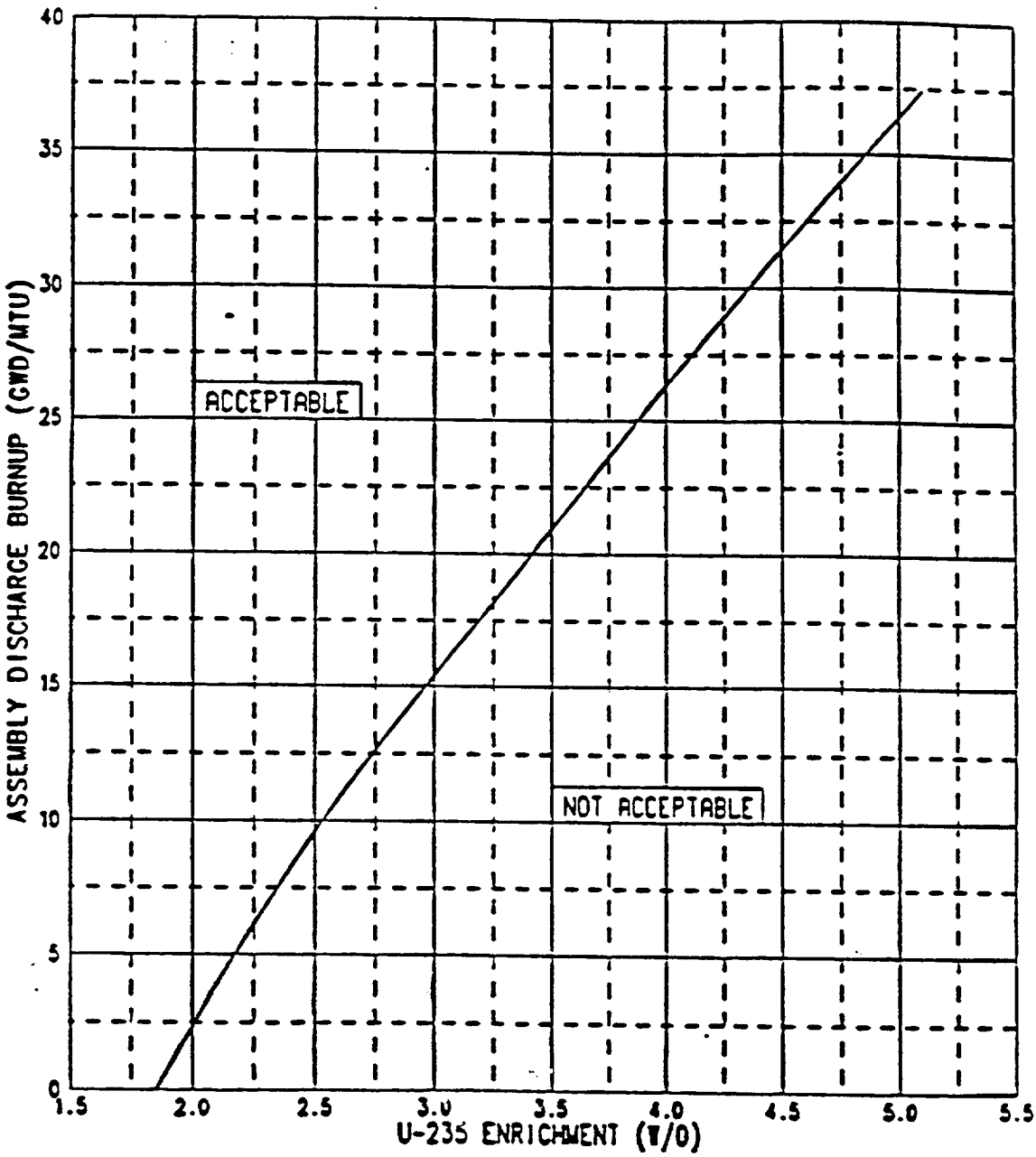
   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N and N_A).

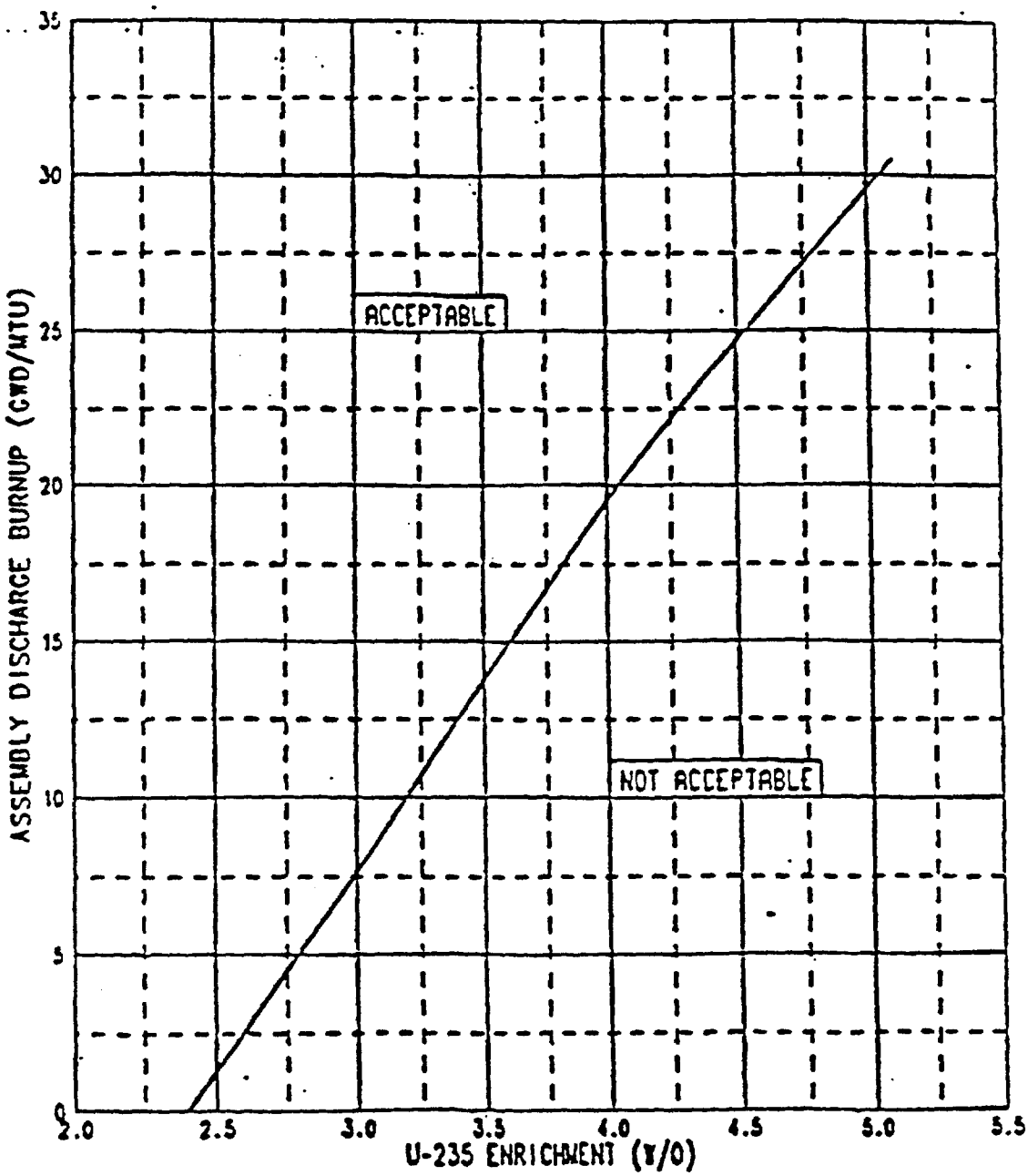
   C. Add Column "N/N_A" to find $\Sigma N/N_A$. This total is the cumulative usage factor.

4. A. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

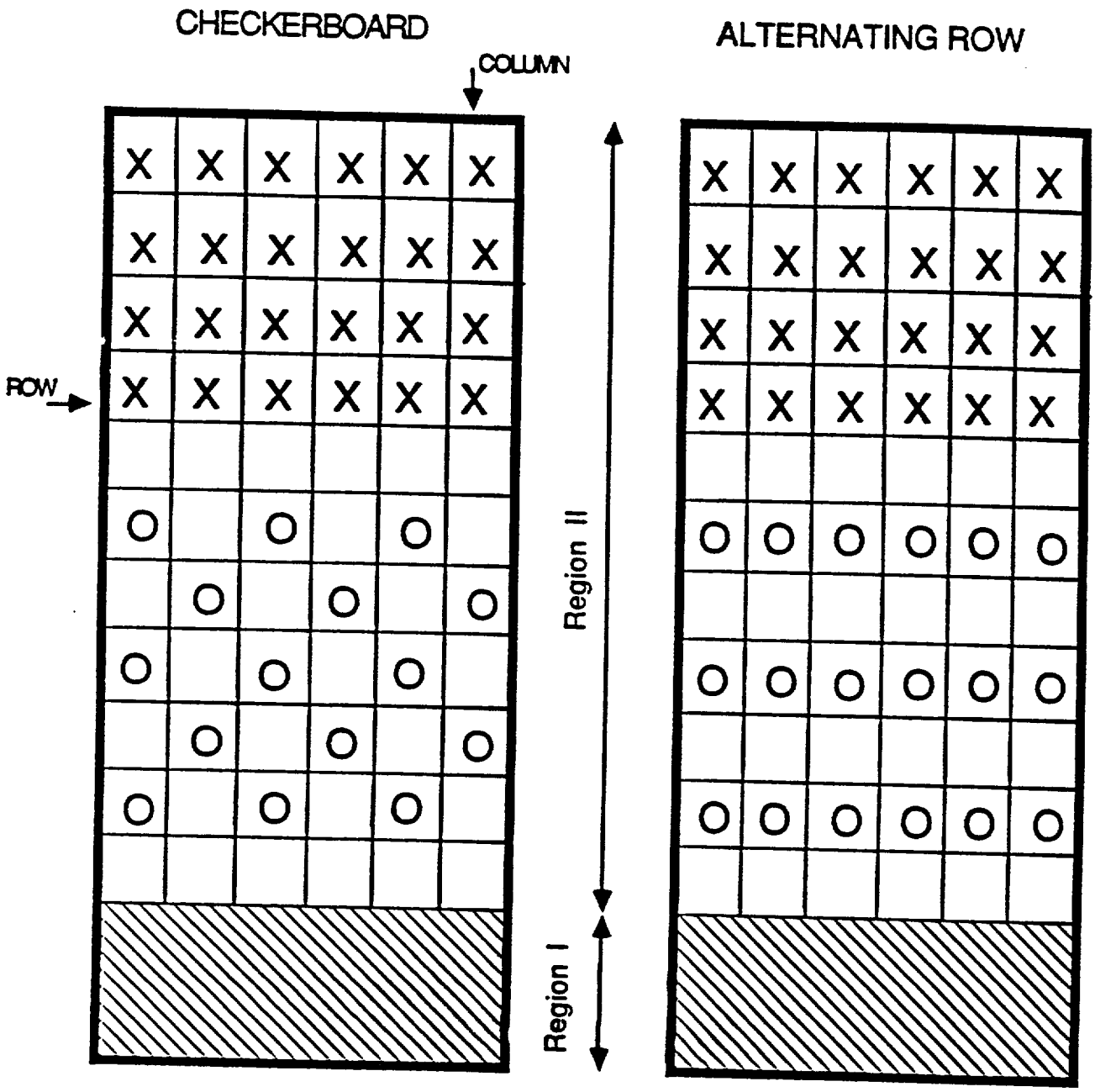
   B. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of spray system fatigue shall be performed within 90 days. The evaluation shall determine that the spray system remains acceptable for additional service beyond the 90 day period or subsequent spray operation shall be restricted so that the maximum temperature difference between pressurizer and pressurizer spray during the spray cycle shall be limited to less than or equal to 200°F.



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
UNITS 2 & 3 FUEL MINIMUM BURNUP VS INITIAL ENRICHMENT FOR REGION II RACKS
FIGURE 5.6-1



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
UNIT 1 FUEL MINIMUM BURNUP VS INITIAL ENRICHMENT FOR REGION II RACKS
FIGURE 5.6-2

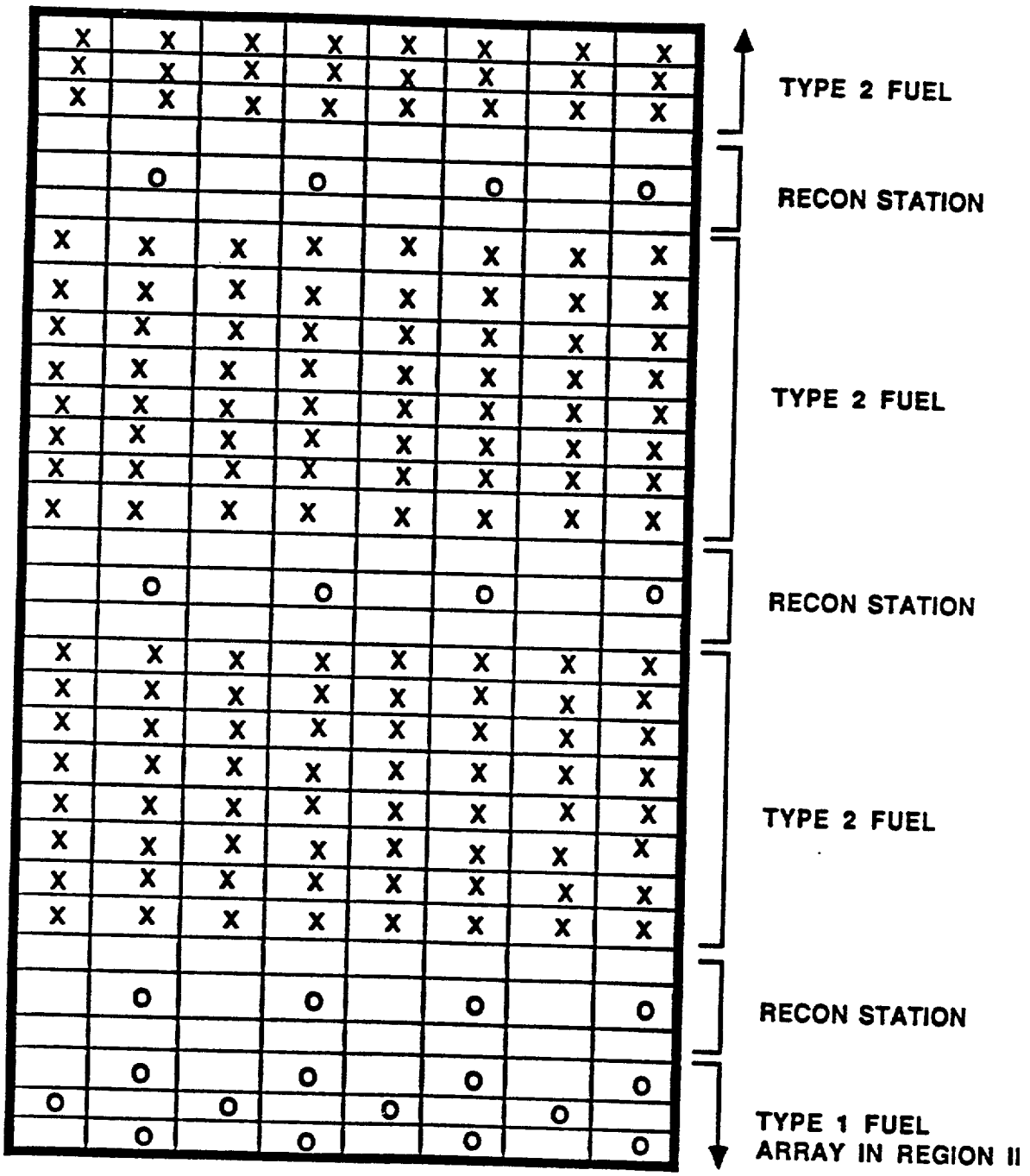


X Type 2 Fuel

O Type 1 Fuel

Empty Storage Location

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
FUEL STORAGE PATTERNS FOR REGION II RACKS
FIGURE 5.6-3



X = TYPE 2 FUEL

O = TYPE 1 FUEL

= EMPTY

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FUEL STORAGE PATTERNS
FOR REGION II RACKS
RECONSTITUTION STATION

FIGURE 5.6-4

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 NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
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 Southern California Edison Company (SCE) on behalf of itself and San Diego Gas and Electric Company, the City of Riverside, California and the City of Anaheim, California (licensees) dated March 10, April 19, May 4, May 19, June 1, June 2, September 22, November 2, November 9, 1989, and January 18, February 9, February 16, and March 20, 1990, 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 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. 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. 77 , are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

John T. Larkins, Acting Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 1, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

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 area of change. Also enclosed are the following overleaf pages to the amended pages.

AMENDMENT PAGE	OVERLEAF PAGE
3/4 9-7	--
3/4 9-13	--
3/4 9-13a	3/4 9-14
3/4 9-16	3/4 9-15
B 3/4 9-2	B 3/4 9-1
B 3/4 9-2a	--
B 3/4 9-3	--
B 3/4 9-4	--
5-7	--
5-7a	5-8
5-12 (Figure 5.6-1)	5-11 (Table 5.7-1)
5-13 (Figure 5.6-2)	--
5-14 (Figure 5.6-3)	--
5-15 (Figure 5.6-4)	--

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

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

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

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

c. Installation or removal of the cask pool cover over the cask pool with fuel in the cask pool. The cover, fuel, and racks will be removed from the cask pool on completion of the reracking process.

d. The lift of construction loads, including the temporary gantry crane and the old and the new fuel storage racks (including lifting equipment and rigging), above the cask pool with the cask pool cover in place and fuel in the cask pool. This includes temporary storage of these construction loads on the cask pool cover during construction. These lifts are prohibited prior to a minimum decay time of 88 days for all stored fuel assemblies.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the fuel handling machine in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.7 Fuel handling machine interlocks and physical stops which prevent fuel handling machine travel with loads in excess of 2000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to fuel handling machine use and at least once per 7 days thereafter during fuel handling machine operation.

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent fuel handling building post-accident cleanup filter systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building post-accident cleanup filter system inoperable, fuel movement within the storage pool or operation of fuel handling machine over the storage pool may proceed provided the OPERABLE fuel handling building post-accident cleanup filter system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filter and charcoal absorbers.

b. With no fuel handling building post-accident cleanup filter system OPERABLE, suspend all operations involving movement of fuel within the storage pool or operation of fuel handling machine over the storage pool until at least one fuel handling building post-accident cleanup filter system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

d. Temporary exception to item (a) and (b) above, applicable only during spent fuel pool reracking construction activities:

   With no fuel handling building post-accident cleanup filter system OPERABLE, all spent fuel pool reracking construction activities including continued operation of the fuel handling machine without fuel, cask handling crane or the temporary gantry crane are permitted provided that the irradiated fuel in the storage pools has decayed for a minimum of 88 days and that no more than 480 irradiated fuel assemblies are stored in the pools.

   Fuel assemblies will only be moved with the post accident cleanup filter system OPERABLE per a and b above.

4.9.12 The above required fuel handling building post-accident cleanup filter systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 12925 cfm ± 10% and recirculating through the HEPA filters and charcoal adsorbers, the total bypass flow of the system through the system diverting valves, to the facility vent is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the cleanup filter system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 12925 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

4. Verifying a system flow rate of 12925 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 7.3 inches Water Gauge while operating the system at a flow rate of 12925 cfm ± 10%.

2. Verifying that on a Fuel Handling Isolation (FHIS) test signal, the system automatically isolates normal ventilation and starts recirculation through the HEPA filters and charcoal adsorber banks.

3. Verifying that the heaters dissipate 28.7 ± 1.5 kw for E464, 32.3 ± 1.7 kw for E465, and 3.8 ± 0.2 kw for E652 when tested in accordance with ANSI N510-1975 with the measured heater dissipation corrected to correspond to nominal voltage.

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 12925 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 12925 cfm ± 10%.

3/4.9 REFUELING OPERATIONS

3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.13 The boron concentration in the spent fuel pool shall be maintained at a level greater than or equal to 1850 ppm.

Applicability: With fuel assemblies in the spent fuel pool.

Action: With the requirement of the above specification not satisfied:

Immediately suspend all additions or movement of fuel in the spent fuel pool and take action to restore the boron concentration to a value equal to or greater than 1850 ppm.

SURVEILLANCE REQUIREMENTS

4.9.13 A sample of spent fuel pool water shall be collected and analyzed for boron concentration at least:

a. Once per month and

b. Within 72 hours prior to any fuel movement.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta K/K conservative allowance for uncertainties. Similarly, the boron concentration value of 2350 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those witha a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

Five finger CEAs are removed from the reactor vessel either along with the associated fuel bundle utilizing the refueling machine or can be removed without the associated fuel bundle utilizing the refueling machine auxiliary hoist. The four finger CEAs are inserted through the upper guide structure with two fingers in each of the two adjacent fuel bundles in the periphery of the core. The four finger CEAs are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane or the refueling machine auxiliary hoist.

Coupling and uncoupling of the CEAs and the CEDM drive shaft extensions is accomplished using one of the gripper operating tools. The coupling and uncoupling is verified by weighing the drive shaft extensions.

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING

A. Refueling Operation

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in six fuel assemblies, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is based on the calculated results which demonstrate that, with credit taken for the fuel handling building filters, the offsite doses would be well within (less than 25%) the 10 CFR 100 limits.

B. Spent Fuel Pool Reracking Construction Activities

The restriction on movement of heavy loads over spent fuel ensures that in the event a heavy load is dropped:

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING (Continued)

1. The radiological consequences due to complete rupture of all spent fuel assemblies in the spent fuel pool and the cask pool (480 maximum), will remain below (less than 25% of) the exposure limits of 10 CFR 100 for offsite doses. This analysis takes no credit for fuel handling building filters (i.e., the fuel handling building hatches are open).

2. Any possible distortion of all fuel assemblies and racks will not result in a critical array and K_{eff} will remain less than 0.95, as long as fuel is stored per Technical Specification 5.6, "Fuel Storage," and 3.9.13, "Spent Fuel Pool Boron Concentration."

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling train be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling trains OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capacity. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling train, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves 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 STORAGE 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 accident analysis.

3/4.9.12 FUEL HANDLING BUILDING POST-ACCIDENT CLEANUP FILTER SYSTEM

The limitations on the fuel handling building post-accident cleanup filter 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 accident analyses.

An exception to these limitations, the fuel handling building post-accident cleanup filter system can be taken out of service during reracking construction activities based on the new limitation of 88 days minimum decay time and limiting the number of spent fuel assemblies to a maximum of 480. These added limitations ensure that all potential radioactive releases are enveloped by existing accident analysis without regard to the OPERABILITY of this system.

Cumulative operation of the system with the heaters on for at least 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.9.13 SPENT FUEL POOL BORON CONCENTRATION

The minimum requirement of 1850 ppm boron ensure that $K_{eff} \leq 0.95$ in the Region II racks in the event of fuel assembly misloading with an enrichment/burnup combination not meeting the criterion for storage in Region II.

Calculations show that with 1800 ppm boron the Region II racks can be completely filled with misloaded fresh unshimmed fuel with an assembly average enrichment of up to 4.1 w/o and maintain $K_{eff} \leq 0.95$, including all uncertainties. Therefore, 1850 ppm of boron is specified to allow for measurement uncertainty.

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, and when fully loaded with fuel which meets the burnup requirement of Technical Specification 5.6.2. This includes a conservative allowance for uncertainties as described in the UFSAR.

b. A nominal 10.40 inch center-to-center distance between fuel assemblies placed in the Region I storage racks and a nominal 8.85 inch center-to-center distance between fuel assemblies placed in the Region II storage racks.

5.6.2 The enrichment-fuel assembly discharge burnup shall be above (greater than) the curves shown on Figure 5.6-1, for Unit 3 fuel assemblies, or Figure 5.6-2 for Unit 1 fuel assemblies for unrestricted storage.

New or burned fuel which does not meet the enrichment versus discharge burnup criteria of Figures 5.6-1/5.6-2 may be stored in Region I. It may also be stored in Region II if all the following conditions are met.

Fuel Type 1 = new or burned fuel which does <u>not</u> meet the criteria of Figures 5.6-1/5.6-2.

Fuel Type 2 = fuel which <u>does</u> meet the criteria of Figures 5.6-1/5.6-2.

a. Fuel Type 1 shall have initial enrichment $\leq$ 4.1 w/o U-235.

5.6.2 (Continued)

b. Fuel Type 1 shall be stored in Region II in either (not both simultaneously a checkerboard pattern or an alternating row pattern per Figure 5.6-3.

c. Fuel Type 1 shall be separated from Fuel Type 2 by at least one (1) completely empty row of storage cells.

d. Fuel Type 1 shall not be stored in the same row as Fuel Type 2.

e. One (1) completely empty row of Region II storage cells shall separate Fuel Type 1 stored in Region II from fuel storage Region I.

f. Except for the purposes of a fuel reconstitution station described below (g), Fuel Type 1 and Fuel Type 2 storage arrays shall not alternate in Region II.

g. For purposes of fuel reconstitution/inspection work, it is permissible to have the three (3) row (empty - every other Fuel Type 1 - empty) arrangement of Figure 5.6-4 any where in the Fuel Type 2 storage array. Additional empty rows are allowed. This three (3) row pattern may be repeated in the Fuel Type 2 storage array if at least eight (8) rows separate repetitions.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below the Technical Specification 3.9.11 value (23 feet above the top of irradiated fuel assemblies seated in the storage racks).

CAPACITY

5.6.4 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1542 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.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Pressurizer Spray System		

Calculational Method:

1. The spray cycle is defined as any initiation and termination of main or auxiliary spray flow through the pressurizer spray nozzle.

2. If the maximum temperature difference between the pressurizer and the pressurizer spray during the spray cycle exeeds 200°F, each spray cycle and the corresponding temperature difference is logged.

3. The spray system usage factor is calculated as follows:

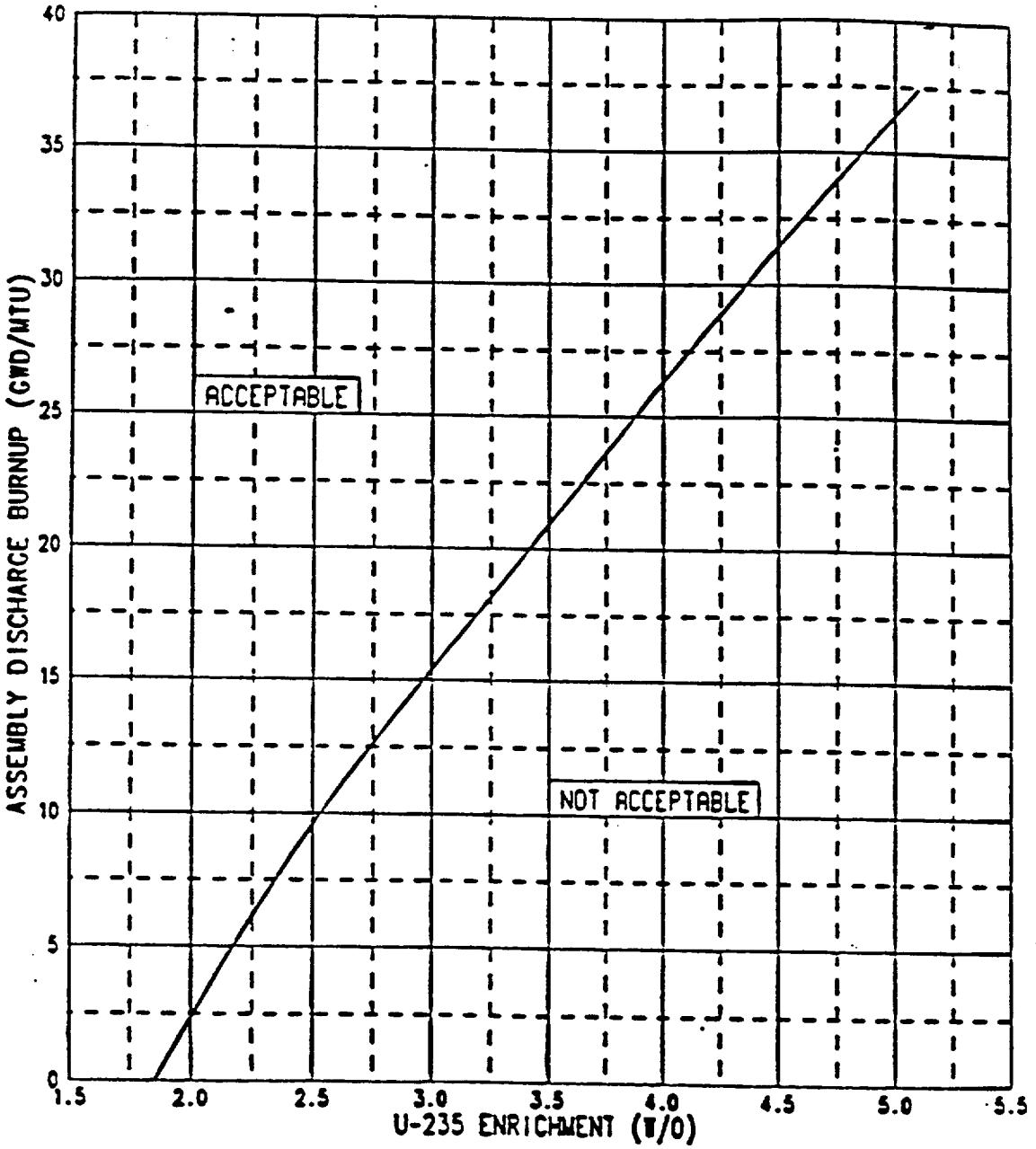
   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N and N_A).

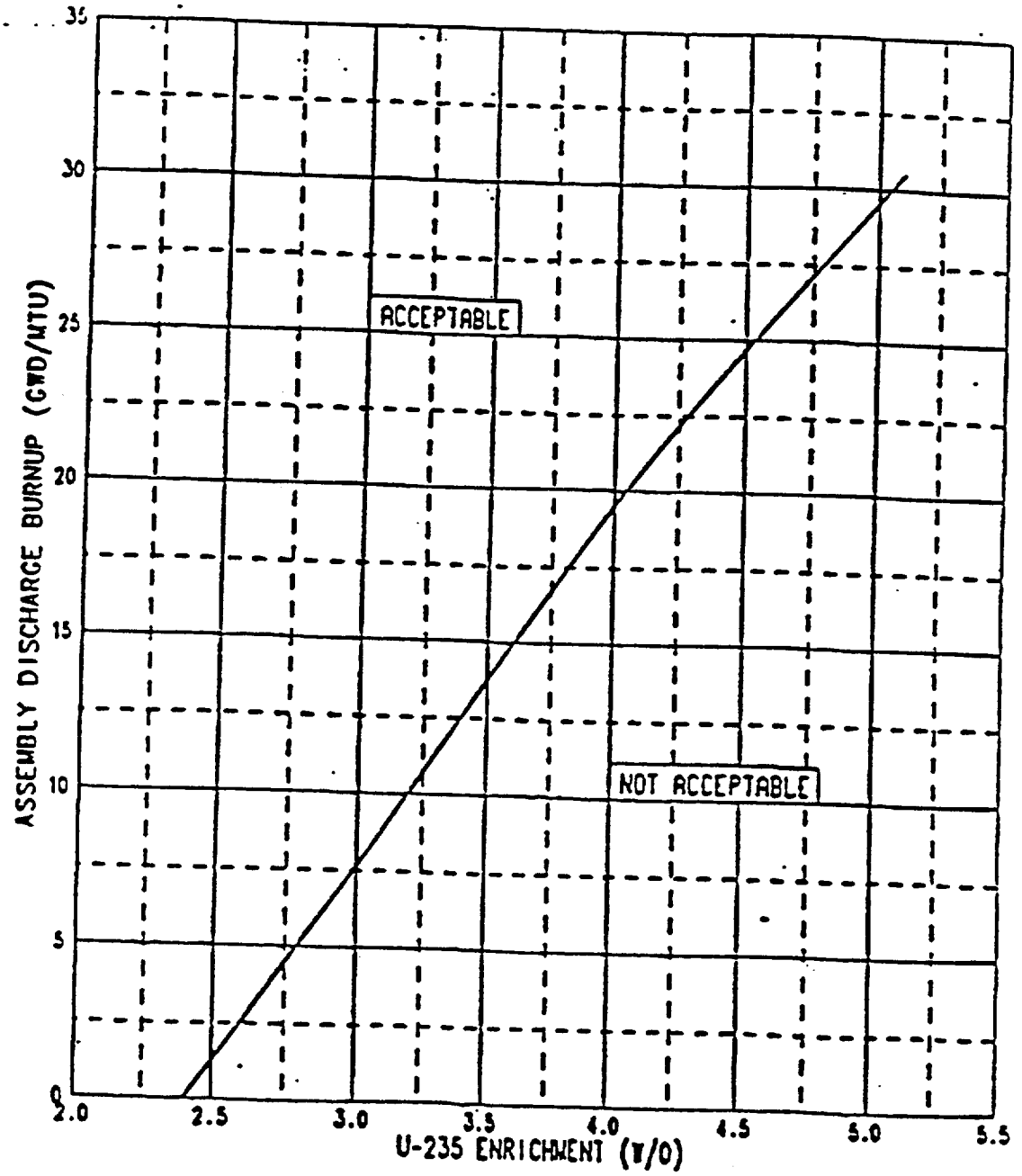
   C. Add Column "N/N_A" to find $\Sigma N/N_A$. This total is the cumulative usage factor.

4. A. If the cumulative usage factor is equal to or less than 0.65 no further action is required.

   B. If the cumulative usage factor exceeds 0.65, subsequent pressurizer spray operation shall continue to be monitored and an engineering evaluation of spray system fatigue shall be performed within 90 days. The evaluation shall determine that the spray system remains acceptable for additional service beyond the 90 day period or subsequent spray operation shall be restricted so that the maximum temperature difference between pressurizer and pressurizer spray during the spray cycle shall be limited to less than or equal to 200°F.



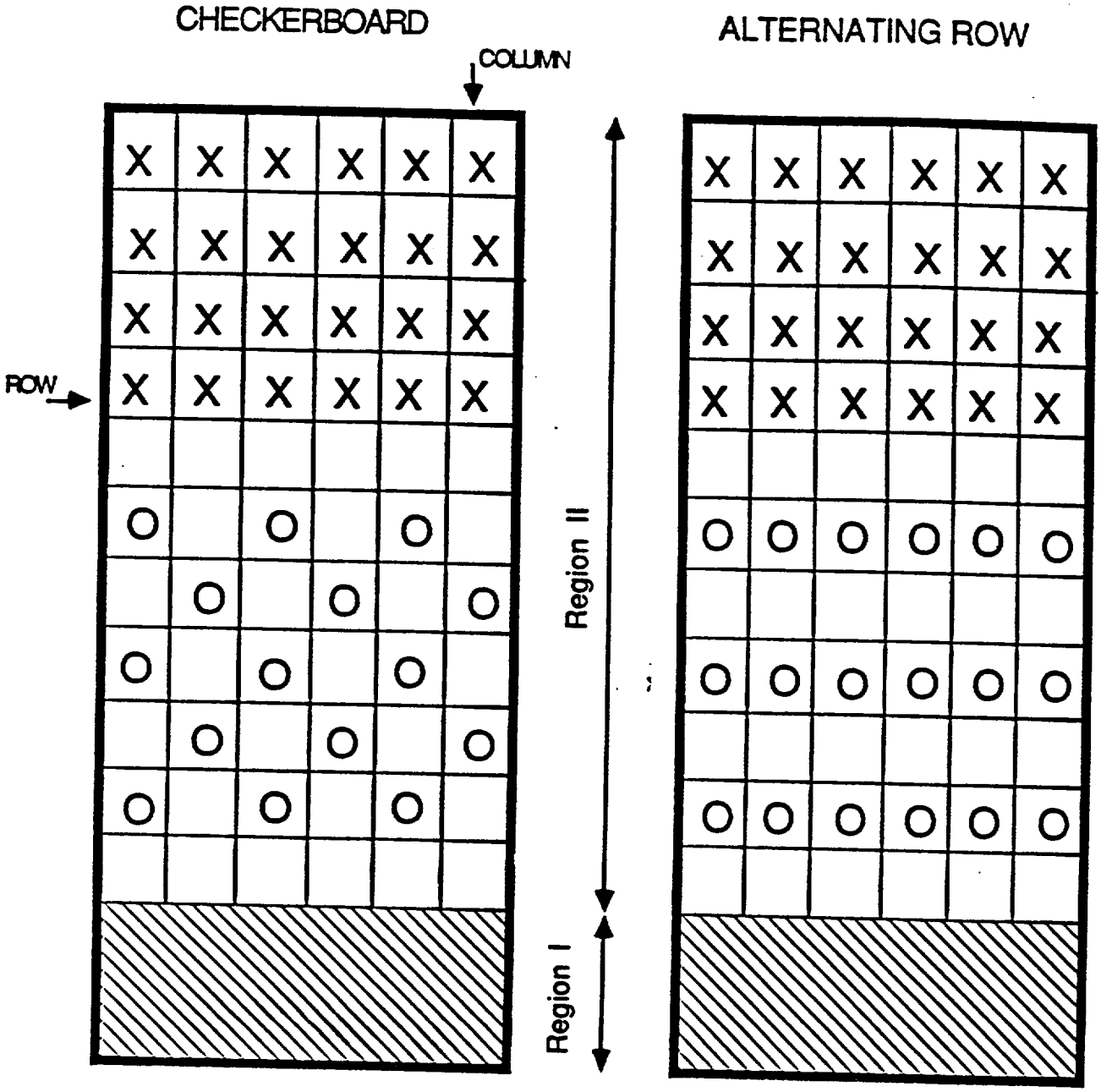
SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
UNITS 2 & 3 FUEL MINIMUM BURNUP VS INITIAL ENRICHMENT FOR REGION II RACKS
FIGURE 5.6-1



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

UNIT 1 FUEL
MINIMUM BURNUP VS INITIAL
ENRICHMENT FOR REGION II RACKS

FIGURE 5.6-2



X Type 2 Fuel

O Type 1 Fuel

Empty Storage Location

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
FUEL STORAGE PATTERNS FOR REGION II RACKS
FIGURE 5.6-3

X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
	O		O		O		O
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
	O		O		O		O
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
X	X	X	X	X	X	X	X
	O		O		O		O
	O		O		O		O
O		O		O		O	
	O		O		O		O

TYPE 2 FUEL

RECON STATION

TYPE 2 FUEL

RECON STATION

TYPE 2 FUEL

RECON STATION

TYPE 1 FUEL ARRAY IN REGION II

X = TYPE 2 FUEL

O = TYPE 1 FUEL

= EMPTY

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

FUEL STORAGE PATTERNS
FOR REGION II RACKS
RECONSTITUTION STATION

FIGURE 5.6-4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.87 TO FACILITY OPERATING LICENSE NO. NPF-10

AND AMENDMENT NO.77 TO FACILITY OPERATING LICENSE NO. NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

By letter dated March 10, 1989 and supplemented by letters dated April 19, May 4, May 19, June 1, June 2, September 22, November 2, November 9, 1989, January 18, February 9, February 16, and March 20, 1990, Southern California Edison Company et al. (SCE or the licensee) requested amendments to the Technical Specifications (TS) of Facility Operating License Nos. NPF-10 and NPF-15 for San Onofre Nuclear Generating Station, Unit Nos. 2 and 3 (SONGS 2/3) in San Diego County, California. The purpose of the requested amendments is to increase the capacity of the fuel storage racks at SONGS 2/3 from 800 to 1542 fuel assembly storage locations and to conduct specific heavy load lifts above the new storage racks that are required for construction and normal fuel pool operations.

The licensee has proposed replacing the existing spent fuel racks, which have 800 storage locations, with new high density spent fuel racks, accommodating 1542 fuel assembly storage locations in both the SONGS 2 spent fuel pool and the SONGS 3 spent fuel pool. The new racks would be free standing and use Boraflex neutron absorbing material for criticality control. A two region design would be used; with 312 storage locations in Region I for storage of all types of SONGS Unit No. 1 and SONGS 2/3 uranium oxide fuel; and 1230 storage locations in Region II for storage of SONGS Unit No. 1 and SONGS 2/3 uranium oxide fuel which either meets specified burnup criteria or is stored in prescribed patterns. Approval of storage of spent fuel produced by operation of SONGS Unit No. 1 in both SONGS Unit Nos. 2 and 3 pools was authorized June 22, 1988 by Amendments 63 and 52, respectively. This change would extend the full core reserve storage capacity for each SONGS 2/3 unit through cycle 11 operation, which is scheduled to begin in 2001 for Unit 2 and 2002 for Unit 3.

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PDR ADOCK 05000361
P PDC

The licensee has proposed the following changes to the Technical Specifications:

- Technical Specification 5.6.1(b) will change the current 12.75 inches center-to-center rack storage location spacing to 10.40 inches center-to-center spacing for Region I, and 8.85 inches center-to-center spacing for Region II.

- Existing Technical Specification 5.6.2 for dry storage of the first core in the fuel pool in alternate rows and columns will be deleted. This Technical Specification was only applicable for the first core and the pool is filled with water which is maintained at a minimum level as prescribed by Technical Specification 3.9.11, "Water Level -- Storage Pool."

- New Technical Specification 5.6.2 and accompanying Figures 5.6-1, 5.6-2, 5.6-3, and 5.6-4 will define the fuel enrichment/burnup limits for storage of Units 1, 2, and 3 fuel in Region II of the high capacity spent fuel storage racks. This new Technical Specification will also define the conditions and storage patterns (checkerboard or alternating row) for which new or burned fuel that does not meet the enrichment vs. burnup criteria for unrestricted storage in Region II may be stored in Region II. Lastly, this new Technical Specification will define the conditions (empty - alternating cells - empty) under which a new/burned fuel reconstitution station may be established in Region II.

- Technical Specification 5.6.4 will be revised to designate that no more than 1542 fuel assemblies may be stored in the spent fuel racks, which is an increase of 742 from the current limit of 800 elements.

- Technical Specification 3.9.7 will be revised to prohibit the lift of construction heavy loads over the spent fuel or cask pools except for the following four cases:

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

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

- Installation or removal of the cask pool cover over the cask pool with fuel in the cask pool.

- The lift of construction loads including the temporary gantry crane and the old and the new fuel storage racks (including lifting equipment and rigging), above the cask pool with the cask pool cover in place and fuel in the cask pool. This includes temporary storage of these construction loads on the cask pool cover during construction. These lifts are prohibited prior to a minimum fuel decay time of 88 days for all stored fuel assemblies.

° The basis for Specification 3.9.7 will be revised to reflect the analysis for the heavy load drops associated with the revised Specification 3.9.7.

° A new Technical Specification 3.9.13 (Bases 3/4.9.13) will be added to specify the boron concentration limit in the pool as 1850 ppm, which includes 50 ppm for measurement uncertainties, prior to any fuel movement.

° A revision to Technical Specification 3.9.12 (Bases 3/4.9.12) is proposed to allow both trains of the Fuel Handling Building Post-Accident Cleanup Filter System to be out of service during the construction period for reracking the spent fuel pool. This revision to Technical Specification 3.9.12 is required to allow continued operation of the spent fuel handling machine without fuel, temporary gantry crane and cask handling crane with the fuel handling building equipment hatches open. Compliance with this revised Technical Specification 3.9.12 will ensure that with a minimum fuel decay time of 88 days, the radiological consequences of the worst postulated heavy load drop in the pools will not result in releases that exceed 25 percent of the 10 CFR 100 limits at the exclusion area boundary.

° A revision to design features section of 5.6.3 of the Operating License will provide consistency with Technical Specification 3.9.11 value of 23 feet of water to be maintained over the top of irradiated fuel assemblies.

2.0 EVALUATION

2.1 Reactor Systems

2.1.1 Allowable Fuel Storage

The spent fuel storage pool will be divided into two regions. Region I will contain 312 storage cells with a nominal center-to-center spacing of 10.40 inches and is designed to accommodate all types of UO_2 fuel

assemblies from Units 1, 2, and 3 with U-235 enrichments up to 4.1 weight percent. Region II will contain 1230 storage cells with a nominal center-to-center spacing of 8.85 inches. The racks will be free standing and use Boraflex neutron absorbing material for criticality control. Placement of fuel in Region II is restricted by burnup and enrichment limits or by prescribed storage patterns. The storage of fuel with initial enrichments up to 4.1 weight percent U-235 in Region 2 requires either that it have a burnup greater than some value that is dependent on initial enrichment, as shown in Figures 5.6-1 and 5.6-2, or that it be stored in a checkerboard or alternating row pattern in the racks, as shown in Figure 5.6-3. For purposes of fuel reconstitution/inspection work, it is also proposed to have the three row (empty - every other location - empty) arrangement shown in Figure 5.6-4.

2.1.2 Calculational Methods

The calculation of the effective multiplication factor, k_{eff}, makes use of the KENO-IV Monte Carlo computer code. This code was benchmarked against a series of critical experiments with characteristics similar to the SONGS spent fuel pool racks. These comparisons resulted in a model bias of + 0.0083 and a 95/95 probability/confidence uncertainty of ± 0.0018. The PHOENIX transport theory code is also used to obtain k_{eff} as a function of burnup for Region II. PHOENIX has been validated by comparisons with experiments where isotopic fuel composition has been examined following discharge from the Yankee Core 5. In addition, an extensive set of benchmark critical experiments has been analyzed with PHOENIX showing good agreement between predictions and measurements.

In order to calculate the criterion for acceptable burnup for storage in Region II, calculations were made for fuel of several different initial enrichments and, at each enrichment, a limiting reactivity value was established. Burnup values that yielded the limiting reactivity values were then determined for each enrichment from which the acceptable burnup domain for storage in Region II was obtained. This is shown in Figure 5.6-1 for CE 16x16 fuel from Units 2 and 3 and in Figure 5.6-2 for Westinghouse 14x14 fuel from Unit 1. The staff has approved this procedure in the past and finds it acceptable for SONGS.

Fuel which does not meet these burnup versus enrichment criteria may, of course, be stored in Region I. However, because of space limitations in Region I, the licensee investigated other means for storage of these fuel assemblies in Region II. Calculations were made for storage of this fuel in a checkerboard pattern or an alternating row pattern as shown in Figure 5.6-3. In these cases, fuel storage surrounding the assembly will be controlled administratively to prevent inadvertent insertion in an unapproved configuration.

Calculations were also performed for the three row arrangement shown in Figure 5.6-4 for Region II. This arrangement would be used for fuel reconstitution or fuel inspection work.

2.1.3. Treatment of Uncertainties

The fuel assemblies are conservatively assumed to be moderated by pure water at the temperature and density within the design limits of the pool which yields the largest reactivity. Although the pool water is normally borated, no dissolved boron is assumed in the calculation.

Additional uncertainties and biases due to manufacturing and mechanical variations in Boraflex thickness, stainless steel thickness, cell inner diameter, center-to-center spacing, asymmetric assembly positioning, Boraflex shrinkage, and Boraflex edge deterioration are treated by using worst case conditions. A bias of + 0.0018 is incorporated to account for poison particle self-shielding in the Boraflex. The uncertainty associated with the reactivity equivalence methodology used in Region II is included in the development of the burnup requirements.

The staff concludes that the appropriate uncertainties have been considered and have been calculated and incorporated in an acceptable manner. In addition, these uncertainties were determined at least at a 95 percent probability 95 percent confidence level, thereby meeting the NRC requirements, and are acceptable.

2.1.4 Results of Analysis

For Region I, the rack multiplication factor is calculated to be 0.9239, including uncertainties at the 95/95 probability/confidence level, when fuel having an enrichment of 4.1 weight percent U-235 is stored therein. Although the pool is normally filled with borated water unborated water was assumed in the analysis.

For Region II, the rack multiplication factor with an equivalent fresh fuel enrichment of 1.85 weight percent U-235 for the CE 16x16 fuel assembly is 0.9468. The value is 0.9315 for an equivalent fresh fuel enrichment of 2.40 weight percent U-235 for the Westinghouse 14x14 fuel assembly. The design will accept CE fuel of 4.1 weight percent U-235 initial enrichment burned to approximately 24 GWD/MTU or Westinghouse 14x14 fuel initially enriched to 4.1 weight percent U-235 burned to approximately 21 GWD/MTU. The analysis for burnup dependent storage in Region II also assumed full flooding by unborated water. Storage in Region II of new or burned fuel assemblies that do not meet the enrichment versus burnup criteria resulted in a multiplication factor of 0.94482 for a checkerboard pattern and 0.94517 for an alternating row pattern. The rack multiplication factor for the fuel reconstitution station in Region II is 0.9339. These calculations were based on the more reactive CE 16x16 fuel assemblies.

The results of the reactivity analyses meet the staff's acceptance criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level.

2.1.5 Accident Analysis

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate accidents, such as an inadvertent misplacement of a fresh fuel assembly into a Region II storage cell, which could lead to an increase in reactivity. For such events, the licensee has applied the double contingency principle, which states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Therefore, the presence of soluble boron in the storage pool water was assumed as a realistic initial condition. The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents. In order to credit the presence of boron, the staff requested the licensee to include a Technical Specification on minimum boron concentration in the spent fuel pool with an associated Surveillance Requirement. This was agreed to by the licensee.

2.1.6 Technical Specification Changes

The following Technical Specification (TS) changes have been made as a result of the proposed spent fuel pool storage modifications. The staff finds these changes acceptable.

1. TS 5.6.1(b) will change the current 12.75 inch center-to-center fuel storage spacing to 10.40 inches for Region I and 8.85 inches for Region II.

2. Existing TS 5.6.2 for dry storage of the first core in the fuel pool is no longer applicable and will be deleted.

3. New TS 5.6.2 and accompanying Figures 5.6-1, 5.6-2, 5.6-3, and 5.6-4 will define the requirements for acceptable storage of fuel assemblies in Region II.

4. TS 5.6.4 will be revised to increase the number of fuel assemblies which may be stored in the spent fuel racks from 800 to 1542.

5. New TS 3/4.9.13 will include a minimum required fuel storage pool boron concentration of 1850 ppm to be verified monthly and within 72 hours prior to any fuel movement.

2.1.7 Conclusion

Based on the review described above, the staff finds that the criticality aspects of the design of the SONGS 2 and 3 spent fuel racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The staff concludes that fuel from Unit 1 and Units 2 and 3 may be safely stored in Region I provided that the enrichment does not exceed 4.1 weight percent U-235. Any of these fuel assemblies may also be stored in Region II

provided it meets the burnup and enrichment limits specified in Figures 5.6-1 or 5.6-2 of the SONGS 2 and 3 Technical Specifications. New or burned fuel which does not meet the burnup and enrichment limits may be stored in Region II in either a checkerboard pattern or an alternating row pattern provided the requirements of Technical Specification 5.6.2 are met.

2.2 Materials and Chemical Engineering

2.2.1 Introduction

Nuclear reactor plants provide storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel storage pool is to maintain the spent fuel assemblies in a sub-critical array during all credible storage conditions. The staff has reviewed the compatibility and chemical stability of the storage rack materials wetted by the pool water in accordance with Section 9.1.2 of the Standard Review Plan (NUREG-0800, July 1981).

The spent fuel storage pool at San Onofre Units 2 and 3 contains air saturated demineralized water, which is borated. The pool is lined with stainless steel. The proposed spent fuel racks are constructed from Type 304 LN stainless steel except for leveling screws, which are SA-564 Type 630 stainless steel. The racks utilize a neutron absorbing material, Boraflex, which is attached to each cell. Boraflex consists of fine boron carbide particles distributed in a polymeric silicone encapsulate.

A Boraflex wrapper (0.020 inch thick) positions the Boraflex on the side of the rack storage cell. The wrapper holds the Boraflex in place on the side of the rack storage cell without pinching, binding, sagging or buckling. This design allows shrinkage during in-service irradiation without developing tears or cracks.

The licensee proposed a long-term surveillance program to monitor the performance of the Boraflex in the spent fuel pool environment. Surveillance coupons, representative of material used in the racks, will be located adjacent to selected racks. At least one coupon will be removed every five years for evaluation. The examination will include visual inspection for overall appearance, dimensional and weight measurements, hardness testing, and neutron attenuation measurements. If degraded Boraflex is found, corrective actions that would be considered include: blockage of affected storage locations to prevent fuel assembly loading, administrative controls on enrichment and/or fuel burnup on fuel subassembly placement in storage rack, or addition of a neutron absorbing material to a fuel subassembly to be placed in the storage rack.

2.2.2 Evaluation

The stainless steel in the spent fuel storage pool liners and rack assemblies is compatible with the air saturated borated water and radiation

environment of the spent fuel pool. In this environment, corrosion of Type 304L stainless steel is not expected to exceed a rate of 6×10^{-7} inch per year (E.Y. Brush and W.F. Pearl, "Corrosion and Corrosion Product Release in Neutral Feedwater", Corrosion, Volume 28, page 129, April 1972). The corrosion rate is negligible for even the thinnest stainless steel walls in the rack assemblies. Galvanic attack between the stainless steel in the pool liners or rack assemblies and the Inconel/Zircaloy in the fuel assemblies will not be significant since these materials are protected by passivating oxide films. Boraflex is composed of non-metallic materials and, therefore, will not develop a galvanic potential with the metal components.

Space is available to allow escape of any gas that may be generated from the polymer binders in the Boraflex due to heat and irradiation, thus preventing possible bulging or swelling. Boraflex has undergone extensive testing to determine the effects of gamma irradiation in various environments and to verify its structural integrity and creditability as a neutron absorbing material (Bisco Products, Inc., Technical Report No. NS-1-001, "Irradiation Study of Boraflex Neutron Shielding Materials", August 12, 1981). The evaluation tests have shown that Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests were performed at the University of Michigan, exposing Boraflex in 2000 ppm boron solution to 1.03×10^{11} rads of gamma radiation with a concurrent neutron flux of 8.3×10^{13} neutrons/cm^2/sec. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma and neutron irradiation. However, irradiation caused some loss of flexibility and shrinkage of the Boraflex.

Long-term water soak tests at high temperatures were also conducted, "Boraflex Neutron Shielding Material Product Performance Data", August 25, 1981. The tests show that Boraflex will withstand a temperature of 240°F in a solution of 3000 ppm boron for 251 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide. The spent fuel pool water temperature is normally maintained below 140°F, which is well below the 240°F test temperature.

The tests referenced above have shown that neither irradiation, environment, nor Boraflex composition have a discernible effect on the neutron transmission of the Boraflex material. The tests also have shown that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide contained in the Boraflex typically contains 0.1 weight percent of soluble boron. The test results have confirmed the encapsulation capability of the silicone polymer matrix to prevent the leaching of soluble species from the boron carbide.

Recently, anomalies that range from minor physical changes in color, size, hardness, and brittleness to gap formation of up to four inches in width were observed in Boraflex panels that have been used in the spent fuel pools of four nuclear power plants. The exact mechanism that caused the observed physical degradation of Boraflex has not been confirmed. The staff postulates that gamma radiation from the spent fuel may induce crosslinking of the polymer in the Boraflex, producing shrinkage of the Boraflex material. When crosslinking becomes saturated, scissioning (a process in which bonds between atoms are broken) of the polymer predominates as the accumulated radiation dose increases. Scissioning may produce porosity, which allows spent fuel pool water to permeate the Boraflex material, which may cause embrittlement. Gamma radiation from the spent fuel is the most probable cause of the observed physical degradation, such as color change, size, hardness, and brittleness. The staff does not have sufficient information to determine conclusively what caused the gap formation in some Boraflex panels. However, it is conceivable that if two ends of a full-length Boraflex panel are physically restrained, then shrinkage caused by gamma radiation may promote panel tearing and subsequent gap formation.

The staff determined that reasonable assurance exists that the Boraflex panels are not physically restrained in the design of the storage racks at San Onofre Units 2 and 3. The wrapper holds the Boraflex in place on the side of the storage rack cell without pinching, binding, sagging or buckling. Therefore, it is not likely that gaps will form to any significant extent in the Boraflex panels during the design life of spent fuel storage racks. However, minor physical degradation may take place due to irradiation of the Boraflex panels. The Boraflex panels are designed to allow for both shrinkage and edge deterioration and still meet criticality requirements.

The inservice surveillance of the Boraflex panels will monitor the performance of the neutron absorber material in the spent fuel environment. This program will be based on EPRI NP-6159, "An Assessment of Boraflex Performance in Spent Nuclear Fuel Storage Racks", December 1988. In the unlikely event of gap formation in the Boraflex panels that would lead to loss of neutron absorbing capability, the monitoring program will detect such degraded panels, and the licensee would have sufficient time to perform a criticality evaluation.

2.2.3 Conclusions

Based on the above discussion, the staff concludes that corrosion of the proposed fuel storage racks due to the spent fuel pool environment should be of little significance during the life of the facility. The staff finds that implementation of the proposed surveillance program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR 50, Appendix A, General Design Criterion 61, regarding the capability to permit appropriate periodic inspection and testing of fuel storage components, and General Design Criterion 62, regarding prevention of criticality by the structural integrity of components and of the boron absorber material and are, therefore, acceptable.

2.3 Radiation Protection

2.3.1 Occupational Exposure Controls

Both Units 2 and 3 spent fuel pools (SFPs) contain damaged (leaking) fuel. Exposure from discrete radioactive particles (DRP) is a concern when working with damaged fuel. The licensee has a well established DRP control program in place. Some additional control measures proposed for the SFP rerack are: a) minimize the use of divers in the pool, b) minimize the generation of DRPs in the pool by limiting the movement of damaged fuel, c) when used, divers will vacuum their way to and from the job site, and d) periodic surveys of personnel for DRP contamination.

The licensee intends to maximize the use of remote tools where practicable. Divers will be used where the use of remote tools proves impracticable. Some other control features that will be employed to assure that radiation exposures associated with this task are as low as is reasonably achievable (ALARA) include: a) maximize water shielding to reduce dose rates to divers, b) use of high visibility physical barriers to define permissible access areas for divers, c) use of remote dosimetry and radio communication with divers, d) use of the SFP water clean-up system to maintain pool clarity and control radioactive contamination of the pool, and e) under water vacuuming of work areas before work begins. The licensee also has provided a description of contained and airborne radioactivity sources related to the SFP water that may become airborne as a result of failed fuel and evaporation. The staff has reviewed these source terms and finds them acceptable.

The estimated average occupational exposures for the reracking of Unit 2 and 3 fuel pools is 41 person-rem per unit based on a detailed breakdown of occupational dose for each phase of the operation. Although the general dose rates on the refueling floor are low (1 mrem/hr used in estimate) the extensive nature of these modifications requires several thousands of man-hours each to complete. Therefore, the total exposure to complete the reracks will be very sensitive to the source term encountered. The licensee has taken steps to minimize the source terms associated with this evolution. In addition to reducing the concentration of radionuclides in the pool water to as low as reasonably achievable, as discussed above, the licensee will not use above-pool filtration systems and is evaluating the possibility of connecting the pool vacuum systems directly to the plant radioactive waste system, thus minimizing the exposure associated with changing pool clean-up filters.

Based on our review of the San Onofre Report and additional information supplied by the licensee in their letter dated May 4, 1989 and teleconference on March 21, 1990, we conclude that the projected activities and person-rem estimates for this project appear reasonable. The licensee intends to take ALARA considerations into account, and to implement reasonable dose-saving occupational exposures within the applicable limits of 10 CFR Part 20, and maintain doses ALARA, consistent with the guidelines of Regulatory Guide 8.8. Therefore, the proposed radiation protection program for the SFP rerack is acceptable.

2.3.2 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of postulated accidents and included means for their avoidance in the design and operation of the facility, and has provided means for mitigation of their consequences should they occur. The staff independently assessed such so-called design basis accidents (DBAs) and agrees with the licensee that no previously unconsidered DBA would be created by the installation and operation of the reracked spent fuel storage pool.

In its previous Safety Evaluation Report (NUREG-0712, 1981), the staff conservatively estimated offsite doses due to exposures to radionuclides released to the atmosphere from a fuel handling accident. This is the staff's scoping DBA for the spent fuel storage pool. The staff concluded that the plant mitigative features would reduce the DBA doses to well below the doses specified in the applicable regulation at 10 CFR Part 100. Since the applicant intends to utilize higher enrichment fuel, for which higher burnups are intended, the staff reanalyzed the fuel handling DBA for this case. Increased burnup could increase offsite doses from the fuel handling DBA by a factor of 1.2 (NUREG/CR-5009, February 1988). Burnup to 60,000 MWD/T would require the use of fuel initially enriched to about 5.3 weight percent U-235. Thus, we conservatively increased the previously estimated doses by a factor of 1.2. In Table 1.0, the new and old DBA doses are presented and compared to the guideline doses in 10 CFR Part 100. As shown in this table, the DBA doses are still well within the regulatory guideline values and are, therefore, acceptable.

2.3.3 Radioactive Wastes

The plant contains radioactive waste treatment systems designed to collect and process the gaseous, liquid, and solid waste that might contain radioactive material. The radioactive waste treatments systems are evaluated in the Final Environmental Statement (FES). There will be no change in the waste treatment systems described in the FES because of the proposed SFP rerack.

2.3.3.1 Radioactive Material Released to the Atmosphere

The station Technical Specifications, the Offsite Dose Calculation Manual and the Process Control Program limit the total releases of gaseous activity, and require that releases are continuously monitored to assure that releases are within the regulatory limits of 10 CFR Part 20.

With respect to releases of gaseous materials to the atmosphere, the only radioactive gas of significance that could be attributable to storing additional spent fuel assemblies for a longer period of time would be the noble gas radionuclide Krypton-85 (Kr-85). Experience has demonstrated that after spent fuel has decayed four to six months, there is no longer a significant release of fission products, including Kr-85, from stored spent fuel containing cladding defects. To determine the average annual

release of Kr-85, we assume that all of the Kr-85 released from any defective fuel discharged to the SFP will be released prior to the next refueling. Thus, enlarging the storage capacity of the SFP has no effect on the calculated average annual quantities of Kr-85 released to the atmosphere. There may be some small change in the calculated quantities due to a change in the fuel burnup; this is expected to be a small fraction of the calculated annual quantities.

Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

Most of the tritium in the SFP water results from activation of boron and lithium in the primary coolant and this will not be affected by the proposed changes. A relatively small amount of tritium is contributed during reactor operation by fissioning of reactor fuel and subsequent diffusion of tritium thorough the fuel and fuel cladding. Tritium release from the fuel essentially occurs while the fuel is hot, that is, during operations and, to a limited extent, shortly after shutdown. Thus, expanding the SFP capacity will not significantly increase the tritium activity in the SFP.

2.3.3.2 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the SFP cleanup system and by decay of short-lived isotopes. The activity is highest during refueling operations when reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the SFP cleanup system. The increase of radioactivity, if any, due to the proposed modification should be minor because of the capability of the cleanup system to continuously reduce radioactivity in the SFP water to acceptable levels.

We do not expect any significant increase in the amount of solid waste generated from the SFP cleanup systems due to the proposed modification. The expected increase in total waste volume shipped from San Onofre is less than one percent and would not have any significant additional environmental impact.

If the present spent fuel racks to be removed from the SFPs of San Onofre are contaminated, they may be disposed of as low level solid waste. Averaged over the lifetime of the station, this would increase the total waste volume shipped from the station by less than one percent. This will not have any significant additional environmental impact.

2.3.3.3 Radioactive Material Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modifications.

Since the SFP cooling and cleanup systems operate as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity. It is expected that neither the flow rate nor the radionuclide concentration of the floor cleanup water will change as a result of these modifications.

The SFP demineralizer resin removes soluble radioactive materials from the SFP water. These resins are periodically sluiced with water to the spent resin storage tank. The amount of radioactivity on the SFP demineralizer resin may increase slightly due to the additional spent fuel in the pool, but the soluble radioactive material should be retained on the resins. Radioactive material that might be transferred from the spent resin to the sluice water will be effectively removed by the liquid radwaste system.

After processing in the liquid radwaste system, the amount of radioactivity released to the environment as a result of the proposed modification would be negligible.

In summary, the estimated increase in doses due to exposure of individuals and the population to radioactive materials associated with the spent fuel pool modification are not significant.

2.3.4 Radiological Impact Assessment/Occupational Exposure

The occupational exposure for the proposed modification of the SFPs is estimated by the licensee to be less than 41 person-rems per unit based on the detailed breakdown of occupational dose for each phase of the operation. This dose is approximately 12 percent of the average annual occupational dose person-rem experienced at PWRs in the United States, which is currently about 340 person-rem per unit. The total dose incurred during the reracking of the SFPs is expected to be a small fraction of the total occupational radiation dose incurred from operating San Onofre Units 2 and 3.

Additionally, we have evaluated the increase in onsite occupational dose during normal operations, after pool modifications, resulting from the proposed increase in number of fuel assemblies stored in the pool. Based on the present and projected operations in the SFP areas, we estimate that the proposed modifications should add less than one percent to the total annual occupational exposure at both units.

Thus, we conclude that the proposed storage of spent fuel in the modified SFP will not result in any significant increase in doses received by workers.

2.3.5 Radiological Consequence of Potential Accidental Releases

No onsite fuel handling accidents having significant offsite radiological consequence have ever occurred. Such accidents and their potential environmental consequences must be postulated. Potential environmental

consequences of postulated accidents may be bounded realistically by extrapolation of results from conservative estimates. Offsite doses are estimated conservatively in NRC staff safety reviews for plant siting, design and operations evaluations. The combination of assumptions used for the conservative dose estimates assures that doses for such design basis accidents (DBAs) are unrealistically high. This helps to assure safe plant siting, design and operations because the doses so calculated would exceed regulatory limits without the adoption of plant safety features and/or operational controls. The principal regulatory dose limits for safety reviews are embodied in the NRC Regulations as 10 CFR Part 100. For safety reviews, the limiting dose is 300 rem to the thyroid, principally due to inhalation of I-131 postulated to be accidentally released to the atmosphere.

Several bounding accident analyses for this current assessment have been reported previously (NUREG-0712), and the potential consequences have been found acceptable by the NRC staff. The only pertinent credible accident that has not been analyzed for this assessment is the postulated damage of fuel being handled during the reracking period, with a concomitant release of radioactivity to the atmosphere. A postulated design basis fuel handling accident has been analyzed previously in this safety review, and a thyroid dose of 49 rem for a person at the site boundary was estimated conservatively.

For purposes here, it is significant that this very conservative estimate was based on postulated damage to fuel that had decayed for only three days. In the original submittal, however, irradiated fuel will have decayed a minimum of 60 days. I-131 has a half-life of about eight days. During the additional 60 days, I-131 will decay by an additional factor of about 175. The postulated dose will decrease proportionately. Moreover, in a more recent submittal, the licensee committed to an 88 day decay time which will further decrease the dose.

Thus, regardless of the accident probability, which experience says is very low, the offsite thyroid dose due to this bounding postulated accident can be conservatively estimated as 49/175 = 0.3 rem. This dose would be well below the U.S. Environmental Protection Agency Protective Action Guide of five rem (thyroid) for which offsite protective action would be warranted. Thus, based on this bounding analysis, the potential environmental consequences of possible accidents are acceptably low, as are the risks.

2.3.6 Conclusions

Based on its review of the proposed expansion of the SFP at San Onofre Units 2 and 3 the staff concludes that:

1. The estimated additional radiation doses to the general public are:

    a. Much less than those incurred during normal operation of the San Onofre Nuclear Generating Station.

b. Very small in comparison to the dose members of the public receive each year from exposure to natural background radiation.

2. The licensee has taken appropriate steps to ensure that occupational dose will be maintained as low as is reasonably achievable (ALARA) and within the limits of 10 CFR Part 20. The total occupational dose estimated to be associated with the proposed modification of the expanded fuel pool is a small fraction of the total occupational dose expected to operate San Onofre Units 2 and 3 during the life of the plant.

3. The risks of accidents are very low.

On the basis of the foregoing evaluation, it is concluded that there would be no significant additional environmental radiological impact attributable to the proposed reracking and modification to increase the spent fuel storage capacity at Units 2 and 3 of the San Onofre Nuclear Generating Station.

We have concluded, based on the considerations discussed above, that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, with regard to radiation doses to the public and plant workers.

2.4 Structural Engineering

2.4.1. Introduction

This evaluation addresses the adequacy of the structural aspects of the proposed application. The Brookhaven National Laboratory (BNL) assisted the staff in reviewing various structural and seismic analyses, and in auditing the methods and sample calculations. Attached Appendix A is the technical evaluation report (TER) developed by the BNL. The staff accepts the findings of the TER by incorporating the TER as a part of this evaluation.

The spent fuel pool (SFP) for each unit is located in the Fuel Handling Building (FHB), between the fuel transfer canal and the cask storage pool. The SFP is 44 ft. 0 in. long and 23 ft. 0 in. wide. The top of the 7 ft. 0 in. thick reinforced concrete basemat constitutes the floor of the SFP. The reinforced concrete walls of the SFP vary in thickness from 4 ft. 0 in. to 5 ft. 6 in. The pool walls and the floor are lined with double stainless steel liner plates (the base liner 3/16 in thick and the reliner 1/8 in thick). A leak chase system is installed in the concrete on the back of the base liner plate.

At present, there are fifteen racks in each SFP. The racks are bolted to the beams, which in turn are anchored to the SFP floor. The proposed high density racks (HDRs) consist of two-Region I racks and six-Region II racks. All the racks are designed to be free standing on the pool

floor. Each rack is provided with multiple leveling pads; 26 pads for Region I racks, and 33 pads for Region II racks. A typical rack cell is 8.64 in by 8.64 in square (inside dimension), fabricated from 0.11 in thick stainless steel plate cold-formed and stitch welded at one edge. Except for the outside faces of Region II racks, all cell faces have boraflex material kept in position by the boraflex wrapper. The wrapper is attached to the outside of the cell by spot welding along the length of the wrapper through its side flanges. Section 3 of Appendix A provides additional structural details of the fuel racks.

This evaluation only pertains to the storage of a single fuel assembly in each storage location of the proposed racks as delineated in the licensee's Spent Fuel Pool Reracking Licensing Report (Revision 6).

2.4.2 Evaluation

The primary areas of the review associated with the proposed application are directed towards assuring the structural integrity of the fuel, fuel cells, rack modules, and the spent fuel pool floor and walls under the postulated load combinations (as delineated in USNRC SRP Section 3.8.4, Appendix D, "Technical Position on Spent Fuel Pool Racks," NUREG-0800, July 1981). The review also included the evaluation of the potential accidents during fuel handling and rerack operations. The major areas of concern and their resolutions are outlined in the following paragraphs.

2.4.2.1 Seismic Input

As the plant is located in a seismically vulnerable area, extensive studies were conducted by the licensee (during the plant licensing) in defining the site specific ground responses in a conservative manner. Though the zero period ground acceleration for Safe Shutdown Earthquake (SSE, termed as Design Basis Earthquake-DBE) was conservatively stipulated at 0.67g, the synthetic free-field horizontal time-history has three low frequency (less than 10Hz) peaks of 0.75g within the 80 sec. duration of the time-history. The smoothed design response spectra (in three directions) are used as input for the reanalyses of the FHB. The parameters of the original lumped mass model of the FHB were adjusted to reflect the increased mass corresponding to the proposed HDRs. The resulting in-structure response spectra at the pool floor level are shown in Figures 22, 23, 24 of Appendix A. These spectra were then used to generate three statistically independent time-histories in the three orthogonal directions. These time-histories (Figures 19, 20, 21 of Appendix A) are used to perform the non-linear dynamic analysis of HDRs. The staff finds the seismic input for the analysis acceptable.

2.4.2.2 Spent Fuel Pool

The finite element model used for analyzing the FHB is refined to include finer elements for the SFP. The remainder of the structure (above the pool deck) is modeled to account for its interaction with the SFP. Hydrodynamic loads created by the oscillating water and the movements of

the HDRs under a DBE are incorporated in the reanalyses of the SFP. The controlling load combinations at the critical SFP sections included the DBE induced loads as the dominant contributors. In stating the adequacy of the concrete sections, the licensee used a concept of utilization factors, defined as the percentage of resistance of a reinforced concrete section that has been utilized under the applied loads relative to the resistance of the section capacity. Tables 10 and 11 of Appendix A provide a summary of the results at critical sections.

The SFP liner plates were evaluated for the vertical pad loads and the horizontal loads that could be induced prior to sliding of the loaded rack under a DBE event. The evaluation indicated that the single support plate loads are acceptable except when applied directly over or adjacent to the leak chase channel weld seams or embedded plates. In such locations, the pads are provided with load spreading floor plates to assure adequate bearing areas.

The effect of the proposed reracking on the soil bearing pressure was also evaluated. The allowable soil bearing pressure is 44 kips/sq. ft. The maximum bearing pressure is computed as 21 kips/sq. ft. There is ample margin to accommodate the increased loads (computed as seven percent of the FHB load) due to the proposed HDRs.

The staff finds the results of the reevaluation of the SFP acceptable.

2.4.2.3 High Density Racks

The racks are analyzed using 3-D finite element models of the rack modules consisting of beam, mass, dynamic gap and friction elements. Section 4.1 of Appendix A provides a detailed description of the dynamic models and the manner in which each of the elements is represented in the models. Time-history analyses of the models were performed using the dynamic analysis capability of the Westinghouse Electric Computer Analysis (WECAN) code. A general review of the code for various applications was performed by the NRC in 1984, and the code had been reviewed for the specific applicability to HDR analysis during the reviews of several rerack applications. However, because of the complex 3-D modeling and high seismic inputs, the licensee had presented the details of code verification during the NRC audit for this specific application. The staff found the application of the code acceptable for the purpose.

The spacings between the individual racks and between the racks and the walls are large enough to assure that under the worst condition of loading the racks would not impact each other or the pool walls. This is demonstrated by the licensee via the results of the rack displacements under DBE loadings with limiting coefficients of friction and various combinations of fuel loadings in the racks. Table 5 of Appendix A shows a summary of the rack displacements versus the spacings provided. Based on the review of the detailed methodology to calculate the displacements, the staff concludes that the rack-to-rack and rack-to-pool impacts should not

occur during the DBE. Also, it is demonstrated that there is an ample margin of safety (greater than 39) against rack overturning. The licensee has committed (Section 4.6.7 of LAR) to perform a walkdown of the pool to check the adequacy of rack location after confirmation of an OBE event.

The stresses in support pads, fuel cells, grid members and cell to cell clips were evaluated under the postulated load combinations and compared against the corresponding ASME allowables (as per Appendix D, SRP 3.8.4). The maximum weld stresses in all weld connections were evaluated. A minimum margin [(allowable/applied)-1] of 0.21 was found to be between the cell and the top grid of Region I rack. A summary of the rack stress margins is provided in Tables 7 and 8 of Appendix A. An additional discussion of the rack stresses is provided in Section 4.2 of Appendix A. The staff finds these computed margins acceptable.

2.4.2.4 Fuel Assembly Drop Accident Analysis

The licensee proposes to store Westinghouse 14x14 (138.5 in. long) fuel assemblies, and Combustion 16x16 (176.8 in. long) fuel assemblies in the HDRs. The licensee performed the drop analysis considering the drop of either one of them from the respective drop heights. Three drop orientations were considered: (1) vertical drop of a fuel assembly on the top of a rack, (2) inclined drop of a fuel assembly on the top of a rack, and (3) vertical drop of a fuel assembly through an empty cell. Each of the three cases was evaluated to determine the velocity of impact with the pool liner. In each case the structure at the lower end of the assembly, i.e., bottom nozzle, guide tubes, etc., had enough strain energy capacity to absorb the kinetic energy associated with the postulated drop. When a fuel assembly was assumed to fall from the baseplate height onto the SFP liner, the stresses imposed on the liner were determined to be 43 percent of the ASME Code allowables for faulted conditions. It was concluded that the SFP liner will not be perforated for any of the fuel drop accidents.

The licensee also performed an analysis to demonstrate that a rack can withstand an uplift load of 6000 lbs. produced by a jammed fuel assembly. The gross stresses produced were found to be within the elastic regime and stresses in localized areas were within the allowable limits.

Based on these analyses, the licensee demonstrated that the structural damage to the rack, the fuel assembly, and the SFP floor due to the postulated fuel drop and for uplift accidents is minimal. The staff finds the licensee demonstration acceptable.

2.4.2.5 Other Drop Accidents

The licensee considered three other conceivable drop accidents that can occur either during reracking operation or during a fuel handling operation: (1) drop of the pool gate, (2) drop of the test equipment, and (3) drop of a HDR. The drop accidents were considered to have occurred on

the top of a HDR loaded with recently discharged fuel assemblies as well as on the pool floor. The drop heights were considered as the maximum height allowed by the administrative controls to be implemented by the licensee.

The results of the various drop analyses indicated that in the worst case scenerio, with conservative assumptions in energy balance calculations, potential fuel damage to six CE (longer) fuel assemblies can occur. The radiological consequences of the recently discharged assemblies were calculated by the licensee. The licensee demonstrated that the doses will be well within the exposure limits of 10 CFR Part 100.

An analysis of a drop of a Region I rack (weight-50,000 pounds) from the administratively controlled height of 19.5 feet above the SFP floor indicated that the stainless steel liners would be perforated, and the concrete basemat could be penetrated about 5 3/4 inches. The licensee determined that the maximum leakage rate from the pool would be limited to 49 gallons/min. The existing SFP makeup water supply is 150 gallons/min. The analysis results of the potential drops of the pool gate and the test equipment were enveloped by the results of the rack drop analysis.

Later on in the review process, the licensee decided to use a single failure proof traveling gantry crane to raise and lower the racks during reracking operation. The procedure will alleviate a potential for a rack drop on the top of a loaded rack.

The staff agrees with the licensee determination that the proposed administrative controls and the use of a single failure proof crane during reracking will limit the structural damage to racks, fuel assemblies and the SFP due to other heavy load accidents to acceptable levels, as stipulated in the licensee's report.

2.4.3 Conclusion

On the basis of the evaluation of the licensee's submittals, information provided by the licensee at meetings, and information audited by the staff and its consultant, the staff concludes that the licensee's structural analyses and design of the proposed spent fuel rack modules and the spent fuel pools are in compliance with the acceptance criteria set forth in the FSAR and are consistent with the current licensing practice. They are, therefore, acceptable.

The conclusion is based on the following because:

(1) The rack modules will be loaded with a single fuel assembly in each storage location of the racks; and

(2) A walkdown will be performed after confirmation of an OBE event to check the adequacy of rack location.

2.5 Plant Systems

Increasing the spent fuel storage capacity of the SFPs for Units 2 and 3 would result in an increased heat load in each of the SFPs. Also, allowing the storage of spent fuel in the cask pool would result in an increased heat load there as well. In its submittals the licensee addressed these and other relevant issues including control of heavy loads, load handling accidents and Technical Specification requirements.

2.5.1 Thermal-Hydraulic Considerations

SFP Bulk Coolant Temperatures

The spent fuel pool cooling system consists of two parallel trains, each containing a pump and heat exchanger (HX). The system contains a cross-over line so as to enable the pump in one train to feed the heat exchanger in the other train if found to be necessary. Each SFP cooling pump is designed to produce a flow rate of 2000 gal/min through the tubes of the HX in that train. Component cooling water (CCW) removes the heat from the shell side of the SFP heat exchanger, with a design flow rate of 3150 gal/min.

The licensee used BTP ASB 9.2 in calculating the decay heat of the spent fuel elements stored in the Units 2 and 3 SFPs, including the use of the long term uncertainty factor. For added conservatism, the licensee used the rated power in lieu of actual power to calculate the decay heat to be expected during normal refueling and also in the event of a full core offload. In addition, heat loads were calculated assuming that the SFP was full of fuel to arrive at postulated heat loads in excess of applicable criteria (24.7 MBTU/HR for normal offload, 51.3 MBTU/HR for a full core offload). Given these considerations and assuming a failure of one SFP cooling pump, the licensee found that the bulk coolant temperature would be less than 140°F for the normal refueling outage; for a full core offload the bulk coolant temperature would be 156°F. These temperatures are below the specified limiting temperatures of less than 140°F (for a normal refueling outage) and less than 212°F (for a full core offload), and are acceptable.

Maximum Fuel Cladding Temperatures

The licensee conducted an analysis of local conditions within spent fuel assemblies to assure against damage to fuel element cladding with possible subsequent failure and discharge of radioactive gases and particulates into the SFP and into the atmosphere surrounding the pool. The licensee calculated the decay heat generated by the spent fuel assemblies (SFA) with the highest heat output to 51.6 BTU/SEC seven days after shutdown. Further, the licensee assumed that the peak fuel rods would have a 60% higher heat generation than the average rods.

In its analysis, the licensee assumed SFP cooling water temperatures (outlet from the heat exchanger) of up to 150°F with up to 80% flow blockage. The licensee also considered a total loss of SFP cooling with a limiting SFP surface temperature of 212°F.

Given these assumptions, the licensee calculated the following fuel cladding temperatures:

(1) 216°F under normal conditions with a 150°F inlet coolant temperature,

(2) 233°F assuming 80% blockage and 150°F inlet coolant temperature, and

(3) 270°F assuming a complete loss of SFP cooling.

The staff finds these cladding temperatures to be well below the maximum allowable and, thus, are acceptable.

Alternate Cooling Sources

In the event of a loss of the SFP cooling system, it would take 7.8 hours for the pool temperature to rise from 140°F to 212°F assuming the maximum normal heat load in the pool and 3.8 hours assuming the maximum abnormal heat load. Without any further cooling, the fuel assemblies would be uncovered in 32 hours with a decay heat load of 51×10^6 BTU/HR and 67 hours with a decay heat load of 25×10^6 BTU/HR. Makeup water could be supplied from the seismic Category I refueling water storage tanks (RWST) in each unit to either the Unit 2 or the Unit 3 SFP via the SFP makeup water pumps. The licensee noted that this makeup path can provide 150 gal/min, which would be enough to replace the expected loss rate of 105 gal/min assuming a pool water temperature of 212°F and a decay heat rate of 51×10^6 BTU/HR. Each RWST contains 245,000 gallons of water. The licensee added that makeup water can also be provided from the nuclear service water tank (25,000 gallon capacity) which is common to Units 2 and 3 and also from the primary plant makeup water tanks in each unit (300,000 gallon capacity) which are also cross tied. Thus, the operators would have sufficient time and resources to correct any problems with the failure of a unit's SFP cooling system.

Aside from the alternate sources of SFP makeup water, during certain conditions the low pressure safety injection (LPSI) pumps may be used to cool the SFP by circulating water through the shutdown heat exchangers. However, the LPSI pumps may not be used for this purpose when there is fuel in the reactor vessel.

The staff finds the availability of methods to maintain the SFP cooling upon loss of the cooling system to be acceptable.

Cask Pool Cooling

In its analysis of cask pool cooling during the reracking process, the licensee determined that existing piping could supply cooling at the rate of 325 gpm to the cask pool. This rate was determined to be capable of removing 6.1 MBTU by discharging the heated water to the SFP. This would result in a cask pool temperature of less than 140°F; the spent fuel pool

temperature would be approximately 107°F. The licensee's heat load analysis assumed that 102 two year old spent fuel assemblies and 108 spent fuel assemblies which had decayed for at least 75 days were being stored. The staff finds this to be acceptable.

Spent Fuel Pool Purification

The SFP purification system is used to maintain the cleanliness of the SFP. Water from the SFP is drawn up into the purification system where it is filtered through an ion exchange bed. The system is not operated if the SFP temperature exceeds 140°F to protect the resins in the ion exchange bed. The licensee plans to remove the purification system distribution piping, located at the bottom of the SFP, during the reracking process. However, the licensee intends to use the purification system as necessary during the reracking evolution to maintain SFP chemistry and clarity. The staff finds this to be acceptable.

Fuel Handling Building HVAC

The licensee evaluated the design of the fuel handling building HVAC system and, based on a maximum SFP temperature of 160°F, found the HVAC system able to meet the original design basis of 104°F in the fuel handling building. The staff finds this to be acceptable.

2.5.2 Heavy Load Handling Considerations

Temporary Gantry Crane Design and Construction

The licensee elected to provide a single failure proof crane, the temporary gantry crane (TGC), for use in handling all heavy loads over the SFP. The TGC will be designed in accordance with the specifications of Generic Licensing Report EDR-1(P)-A, "Ederer's Nuclear Safety-Related Extra Safety and Monitoring (XSAM) Cranes," Revision 3 dated October 8, 1982, Amendment 3. The staff understands that the licensee will have the TGC designed and constructed in accordance with the design approved by the staff in its SER of August 26, 1983, and that the TGC will be built totally in accordance with the criteria approved by the staff. Accordingly, the staff finds the licensee's plans for construction of the TGC to be in accordance with applicable criteria for a single failure proof crane. It is noted that the TGC has an auxiliary hoist with a 2-ton capacity, which is not single failure proof. However, the licensee notes that the maximum lift with the auxiliary hoist will be limited to 1500 lbs. This load, together with the load block, will not exceed the 2000 lb. limit for loads lifted over racks containing spent fuel as permitted for the handling machine in the Technical Specifications. Therefore, the use of the auxiliary hoist on the TGC for loads no greater than 1500 lbs. over the SFP is also found to be acceptable.

Cask Handling Crane

The cask handling crane (CHC) will be used for heavy load handling outside of the main spent fuel pool (i.e., up to and over the cask pool). The CHC

will be used to install Region II racks in the cask pool portion of the spent fuel pool to be used for storage of spent fuel assemblies during the reracking process. The CHC has been found acceptable in a previous SER, dated August 27, 1984, for use in the fuel handling building (FHB) in that it meets the intent of applicable criteria.

Safe Load Paths

The licensee stated that heavy loads would not be carried over unprotected safe shutdown equipment. The licensee will remove old racks from the FHB and place new racks in the FHB by way of the spent fuel cask access hatch using the TGC and CHC. The CHC will handle racks from the access hatch to the cask pool cover. Racks will be transferred from the SFP by means of the TGC and stored on the cask pool cover, which is designed for this purpose. The CHC will lift the old racks from the cask pool cover for removal. New racks will be transferred to the cask pool cover from the access hatch by means of the CHC and from the cover to the SFP via the TGC. Old racks will also have an intermediate move from the cask pool cover to the washdown pool, where loose material is removed by washing (hydrolazing) before being removed from the FHB.

Racks will be moved in the FHB at a height of 1 foot over the operating deck. A rack drop from this height will not cause spalling of the concrete floors or cause damage to any safety related equipment located under the floor. The only deviation from this 1 foot height limitation will occur when racks are moved in or out of the access hatches. However, no safety related equipment is located in the area below the access hatches. The licensee intends to move the fire water tankers out of the way of heavy load lifts in either FHB. The fire water pumper will also be relocated, similarly, in order to protect it.

The only potential problem area is in the open cask pool when installing a Region II rack for temporary storage of spent fuel during the reracking process. A drop could damage the bottom of the cask pool. However, damage to the cask pool is found to be acceptable. Therefore, the staff considers this issue to be resolved.

Procedures

The licensee intends to revise seven exiting plant procedues and to prepare five new procedures. These, together with four existing plant procedures, will comprise the procedures to be used during the reracking process. The staff finds this to be acceptable.

Training

The licensee stated that crane operators for the TGC and CHC will be trained in accordance with the provisions of station procedure S0123-I-7.15. The licensee stated, further, that this procedure complies with the requirements of Chapter 2.3 (Qualification for Operators) of ANSI B30.2.0-1976 (Overload and Gantry Cranes).

The licensee asked for exemption from provisions of Chapter 2.3 as contained in paragraphs 2-3.1.7(o) and 2-3.2.4(a) which are required operations to be performed at the beginning of each new work shift. These operations are:

(1) 2-3.1.7(o) - All controls shall be tested by the operator before beginning a new shift.

(2) 2.3.2.4(a) - At the beginning of each shift the operator shall try out the upper limit of each hoist.

The licensee cites the unadvisability of conducting these operations when removing existing racks (where radiological conditions do not permit rapid removal), when installing new racks (where the TGC remains attached to the rack while leveling), when installing the TGC (where the CHC remains hooked while seismic restraints are attached), and when cribbing under laydown loads (where the crane remains attached for safety reasons). The licensee committed to develop operating procedures to clearly indicate that the conditions of paragraphs 2-3.1.7(o) and 2-3.2.4(a) will be followed except in certain specific cases, and when not followed these operations will be conducted before the next lift.

The staff finds this to be acceptable.

Special Lifting Devices and Lifting Devices Not Specially Designed

The reracking process will require special lifting devices. These include the following:

(1) New rack lift rig
(2) Old rack lift rig
(3) TGC lifting device
(4) CHC adaptor
(5) Cask pool cover lifting device

The new rack lift rig will be used for moving new racks in the FHB and installing them in the SFP. This rig has four lift rods, which are inserted through oblong holes in the bottoms of the new racks and are rotated to secure the rods to the rack. Two diagonally opposite lift rods are each attached to a lift frame with the remaining two lift rods attached to a separate lift frame. Finally, the two lift frames are secured to a single hanger assembly. Each lift frame will be designed to hold a static load equivalent to three times the load capacity based on yield strength and five times capacity based on ultimate strength. The hanger assembly will be designed to sustain six times the rated load based on yield strength and ten times the rated load based on ultimate strength. Thus, the new rack lifting rig qualifies as a single failure proof design in accordance with the guidance of Section 5.1.6(1)(a) of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants". The old rack lift rig is of a similar, though not identical, design and also qualifies as a single failure proof design. Note that the design load for both of these rigs includes a dynamic load factor in accordance with the guidance in Section 5.1.1(4) of NUREG-0612.

The TGC lifting device is used with the CHC for lifting and installing the TGC on the SFP rails. It is designed to sustain a static load three times the rated load based on yield strength and five times the rated load based on ultimate strength. The CHC adaptor may be used with either the old or new rack lift rig. It is designed in a manner similar to that of the TGC lifting device (three times rated load to yield strength, five times rated load to ultimate strength).

The cask pool cover lifting device is designed as a single load path handling device designed to hold a load equal to six times that rated based on yield strength and ten times the rated load based on ultimate strength. Thus, it qualifies as a single failure proof handling device in accordance with Section 5.1.6(1)(a) of NUREG-0612. The licensee states that the CHC, which will be used to lift the cask, qualifies as single failure proof for this lift (i.e., cask pool cover) because the CHC can sustain a load equivalent to six times the weight of the rig and cask pool cover based on yield strength and ten times the weight of rig and cask pool cover based on ultimate strength.

Each pair of lift rods for the old and new rack lift rigs will be tested at 150% of the rated load. Each single path portion of the old and new rack lift rigs will be tested at 300% of rated load, as will the cask pool cover lifting device. The TGC lifting device and CHC adaptor will be tested at 150% of rated load.

The licensee stated that all of these fixtures meet the provisions of guideline 4 of Section 5.1.1 of NUREG-0612. In addition, the old rack lifting rig, the new rack lifting rig and the cask pool cover lifting rig comply with the provisions of Section 5.1.6 for single failure proof special lifting devices.

Testing, inspection, maintenance and repair of the special lifting devices described above will be performed in accordance with existing station procedures. These procedures will be modified to include these lifting devices. The licensee stated that these provisions will comply with Section 5.3, "Testing to Verify Continuing Compliance," and 5.4, "Maintenance and Repair," of ANSI N14.6-1978, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More for Nuclear Materials."

The licensee stated that all other lifting devices (i.e., not specially designed) used for handling heavy loads in the FHB will comply with the guidelines of Section 5.1.1(5) of NUREG-0612.

The staff finds these lifting devices to comply with the guidance of the applicable sections of NUREG-0612 and, thus, to be acceptable.

Testing, Inspection and Maintenance of Cranes

The CHC will be tested, inspected and maintained in accordance with the existing NUREG-0612 program. TGC in-place inspection and maintenance will be conducted in accordance with the guidelines of Section 5.1.1(6) of NUREG-0612. The TGC will be tested in the following manner:

(1) The TGC will be tested operationally in accordance with the provisions of paragraph 2-2.2.1 of ANSI B30.2-1976 at the factory prior to shipment, at the site prior to installation and over the SFP prior to initial rise. These tests include hoisting and lowering tests, trolley and bridge travel, and testing of limit switches.

(2) A rated load test (1.25x35 tons) will be conducted in accordance with paragraph 2-2.2.2 of ANSI B30.2-1976 at the site prior to final installation.

(3) A full performance test will be performed using the maximum critical load (MCL) of 29.75 tons at the site in accordance with Section 8.2 of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants."

(4) A modified load test (1.25xMCL of 29.75T) will be conducted with the hoist and trolley over the covered cask pool before use.

Note that no tests will be conducted over unprotected spent fuel elements as a matter of safety.

The staff finds the testing, inspection, and maintenance of cranes used in the reracking operation to be acceptable.

2.5.3 Evaluation of Proposed Changes to Technical Specifications (TS)

The proposed Technical Specifications for Units 2 and 3 are identical and the following discussion applies to both units.

TS 3.9.7

The LCO (limiting condition for operation) has been amended to allow carrying loads in excess of 2000 lbs. for 4 excepted cases, as follows:

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

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

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

c. Installation or removal of the cask pool cover over the cask pool with fuel in the cask pool.

d. The lift of construction loads, including the temporary gantry crane and the old and the new fuel storage racks (including lifting equipment and rigging), above the cask pool with the cask pool cover in place and fuel in the cask pool. This includes temporary storage of these construction loads on the cask pool cover during construction. These lifts are prohibited prior to a minimum fuel decay time of 88 days for all stored spent fuel assemblies.

Applicability: With fuel assemblies in the storage pool."

Specifications 3.9.7a and b allow carrying heavy loads over spent fuel assemblies in the fuel pool during normal operation at elevations limited to those analyzed by the licensee and found to meet applicable criteria, as noted above.

Specification 3.9.7c is intended to permit installation and removal of the cask pool cover during the reracking process. In a telephone discussion with the staff on April 18, 1990, the licensee agreed to a modification of TS 3.9.7c by addition of the following: "The cover, fuel and racks will be removed from the cask pool on completion of the reracking process." This addition is required in order to clarify the intent and to ensure continued safety during normal operation of the cask pool. The licensee agreed to permit the staff to make such modifications in the telephone conversation cited above.

Specification 3.9.7d permits the carrying of heavy loads over spent fuel assemblies in the cask pool and over the cask pool (with the cask pool cover in place). Heavy load drops of these loads were analyzed by the licensee and found to comply with applicable criteria, as noted above.

The staff finds these amended specifications, 3.9.7a, 3.9.7b, 3.9.7c, and 3.9.7d to be in compliance with applicable criteria and, therefore, to be acceptable.

TS 3.9.12

The licensee has proposed the following specification (3.9.12d):

"Temporary exception to item (a) and (b) above, applicable only during spent fuel pool reracking construction activities:

With no fuel handling building post-accident cleanup filter system OPERABLE, all spent fuel pool reracking construction activities including continued operation of the fuel handling machine without fuel, cask handling crane or the temporary gantry crane are permitted provided that the irradiated fuel in the storage pool has decayed for a minimum of 88 days and provided that no more than 480 irradiated fuel assemblies are stored in the pools.

Fuel assemblies will only be moved with the post accident cleanup filter system OPERABLE per a and b above."

The first portion of this specification allows reracking operations with both filter trains inoperable and with a maximum of 480 irradiated sub-assemblies decayed for a period of no less than 88 days in the storage pool. This specification corresponds to the heavy loads handling guidance specified in NUREG-0612 and is satisfactory, as noted above.

The requirements for moving fuel assemblies (in specification 3.9.12a and 3.9.12b) comply with those in the Standard Technical Specifications.

In view of the foregoing, the staff finds specification 3.9.12d to be acceptable.

TS 3.9.7 and 3.9.12 Bases

The bases for these proposed technical specifications have been added or corrected, as necessary, to explain the bases for these specification. The staff finds these to be acceptable.

TS 5.6.3

This TS has been changed to specify a level of at least 23 feet above the tops of irradiated spent fuel assemblies (as per TS 3.9.11) in lieu of specifying a specific maintenance level (60'6"). The staff finds this to be acceptable.

TS 5.6.4

TS 5.6.4 states that the spent fuel pool capacity shall be limited to 1542 fuel assemblies, the limit of the proposed design. The staff finds this to be acceptable.

2.5.4 Conclusion

The licensee for the San Onofre Nuclear Generating Station provided plans for removing the old storage racks for spent fuel and replacing them with new racks so as to increase the storage capacity from 800 to 1542 spent fuel assemblies in the Unit 2 and Unit 3 spent fuel pools. The licensee provided new or amended technical specification requirements to permit such replacement or "reracking" in order to allow continued operation with full core reserve capacity until the year 2001 for Unit 2 and 2002 for Unit 3. The staff of the Plant Systems Branch has reviewed the licensee's submittal in those areas for which the Plant Systems Branch has nominal review responsibility, as discussed above. The Plant Systems Branch staff finds the licensee's plans in these areas, as discussed above in Section 2.5 of this SER, to be acceptable.

2.6 Other

In discussions with the licensee, other staff concerns were addressed. These were the construction activities with the hatches open and the monitoring of person-rem exposure data. As a result the following commitments were made by the licensee:

2.6.1 Construction Activities With Hatches Open

During a telephone conversation on March 22, 1990, licensee committed to writing two procedures prior to beginning licensed reracking activities. This commitment included the training of all appropriate staff prior to procedural implementation.

The first procedure is an emergency procedure for responding to a heavy load drop when the fuel handling building hatches are open during rerack construction. Rerack activities with the hatches open render the fuel handling building post accident cleanup filter inoperable. Therefore, this procedure must include provisions for evacuation of the fuel handling building and closing the new fuel and cask hatches to minimize potential radiation exposure or releases. This will enhance the margin of safety that is already established with a minimum decay time of 88 days and the limitation of 480 irradiated fuel assemblies stored in the pools.

Additionally, the second procedure concerns heavy load lifts. A heavy load lift procedure requires that the fuel handling building hatches be closed prior to a lift of the spent fuel pool gates to the cask pool or the transfer pool. This procedure must be in effect if spent fuel is in the vicinity of the gate lift.

2.6.2 Reracking Construction Activities Exposure Report

During a telephone conversation on March 27, 1990, the licensee committed to providing person-rem exposure reports to the staff of the U.S. Nuclear Regulatory Commission. The first report will be an interim report and will involve the San Onofre Unit 2 rerack construction activities only. The second and final report will involve San Onofre Unit 3 rerack construction activities and will provide totals for San Onofre Units 2 and 3. These reports will follow as closely as possible the breakdown of construction activities as defined in Table 5.2-4 in revision 6 of the licensee's Spent Fuel Pool Reracking Licensing Report. Moreover, the data in both the interim and final reports will be compared to the annual occupational exposure data for San Onofre Units 2 and 3.

2.7 Conclusion

Based upon the application, as supplemented, for amendments to provide for spent fuel pool reracking construction and operation activities, and the foregoing analysis including any restrictions and commitments, the staff approves the changes to the San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, Technical Specifications.

3.0 NOTICE OF CONSIDERATION OF ISSUANCE AND CONTACT WITH STATE OFFICIAL

Notice of Consideration of Issuance of Amendments and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on April 24, 1989 (54 FR 16438-B). No request for a hearing or petition for leave to intervene was filed following this notice.

The staff has advised the State Department of Health Services, State of California, of the proposed amendment. No comments were received.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has been prepared and a notice of issuance

was published in the Federal Register on March 7, 1990 (55 FR 8248). Additionally, a Supplement to the Environmental Assessment and Finding of No Significant Impact has been prepared and published in the Federal Register on April 6, 1990 (55 FR 12971). Accordingly, based upon the Environmental Assessment, as supplemented, the Commission has determined that the issuance of these amendments will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

We have 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: Table 1.0
Figure 5.6.1
Figure 5.6.2
Figure 5.6.3
Figure 5.6.4
Appendix A (Technical Report A-3841-3/90)

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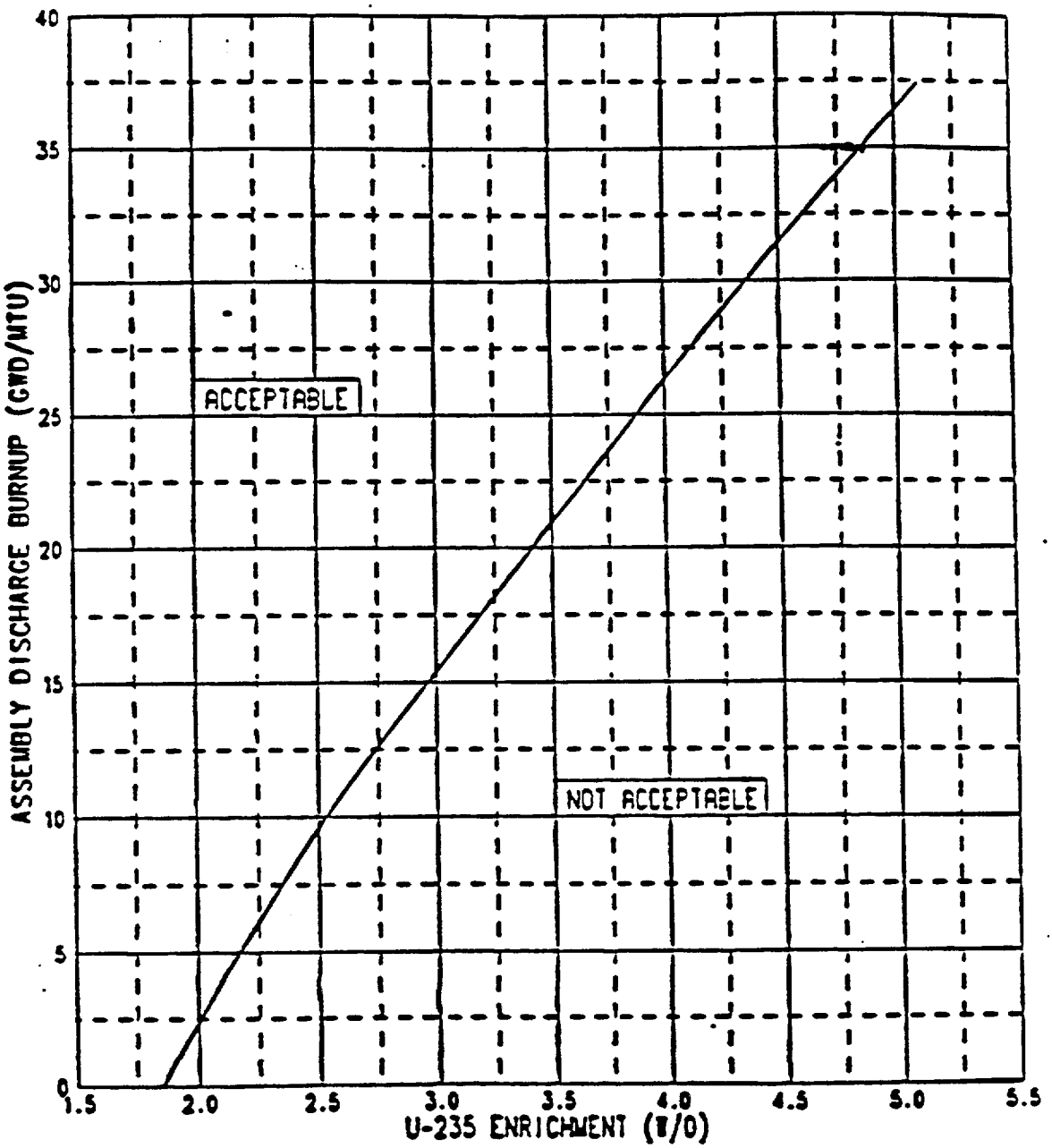
Dated: May 1, 1990

TABLE 1.0

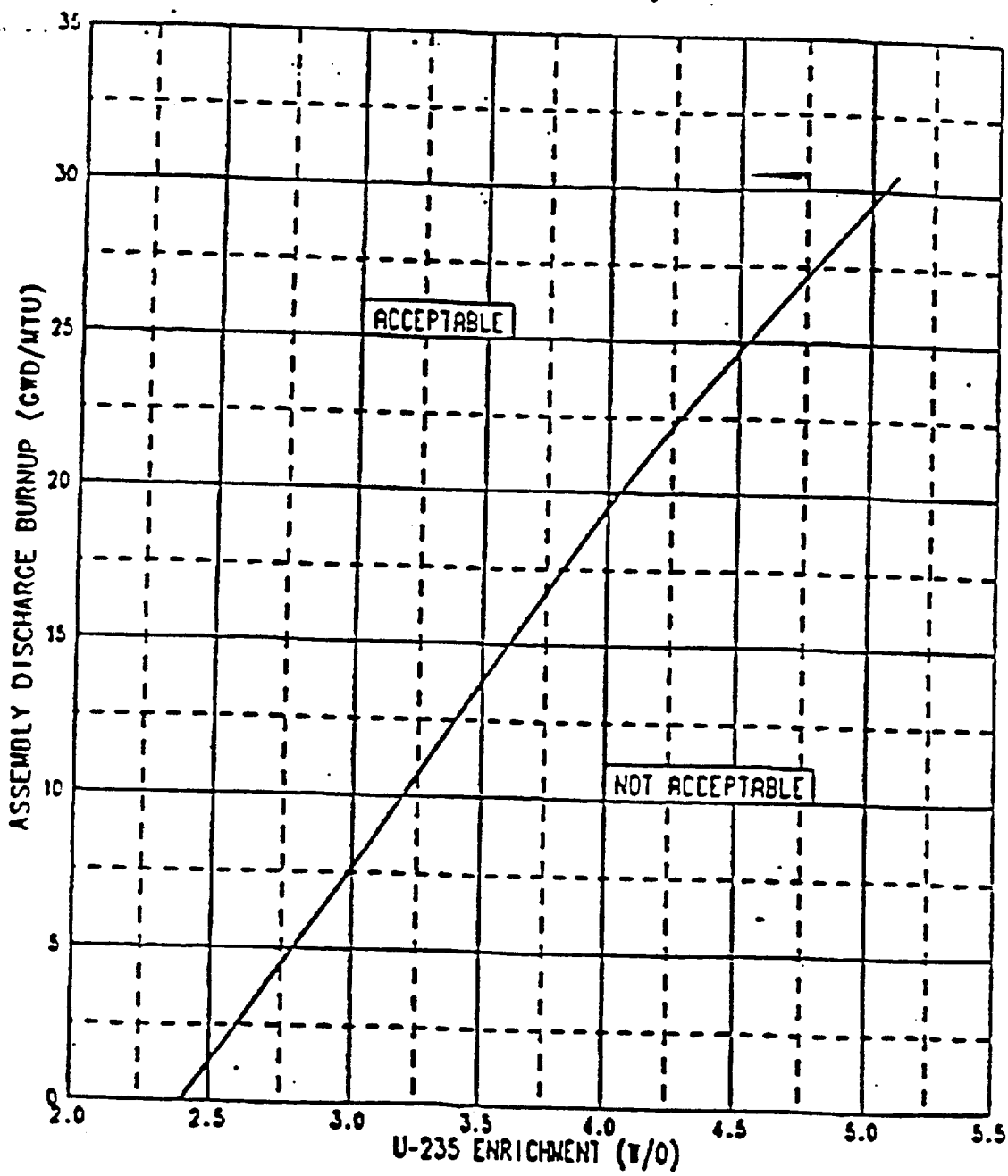
Radiological Consequences of Fuel
Handling Design Basis Accident (REMs)

	Exclusion Area		Low Population Zone	
	Thyroid	Whole Body	Thyroid	Whole Body
Original Estimates (NUREG-0712)	41	7	3	1
Estimates for Higher Fuel Burnup*	49	8.4	3.6	1.2
Regulatory Requirement (10 CFR Part 100)	300	25	300	25

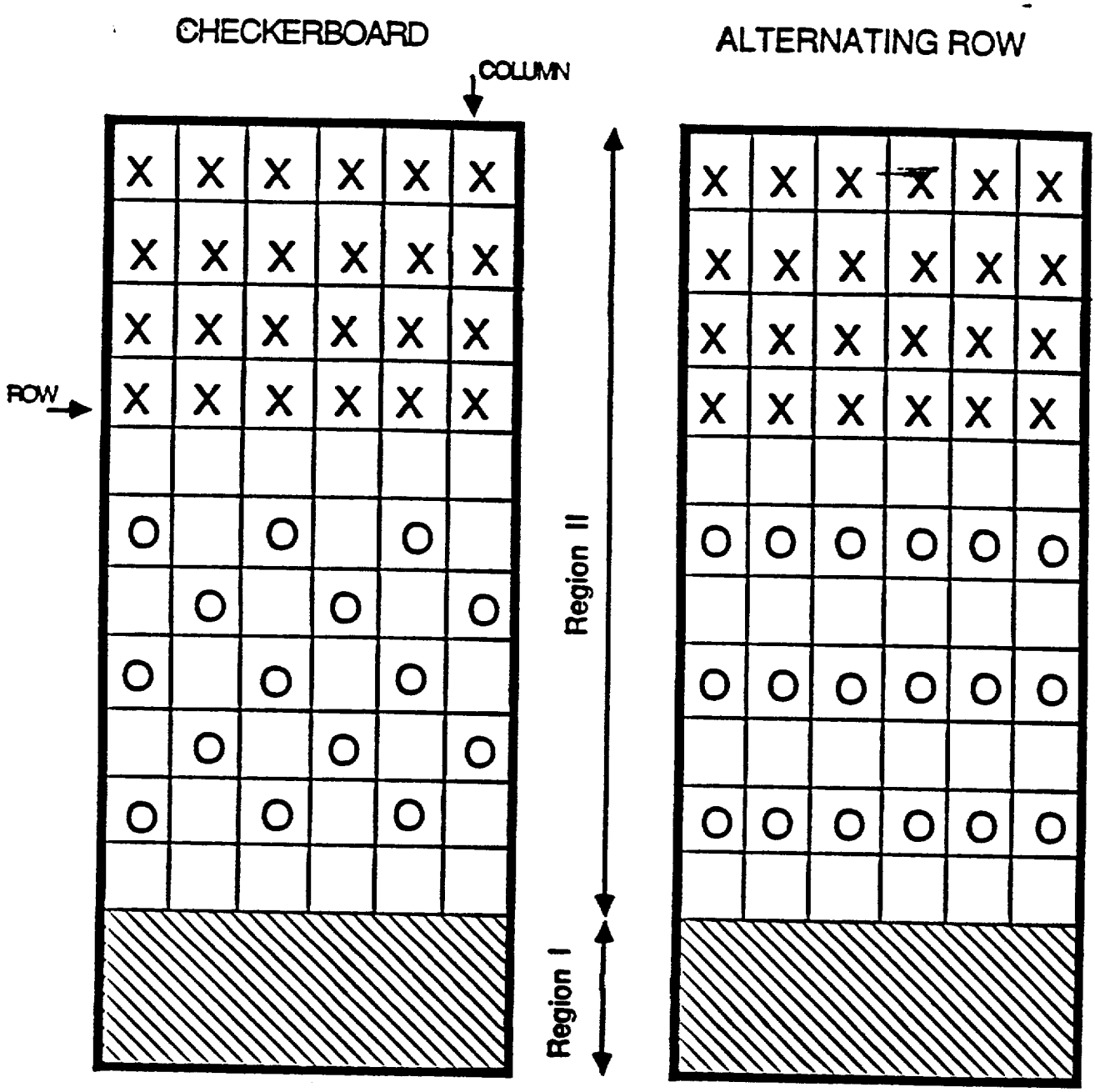
*Factor of 1.2 greater than original estimate



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
UNITS 2 & 3 FUEL MINIMUM BURNUP VS INITIAL ENRICHMENT FOR REGION II RACKS
FIGURE 5.6-1



SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
UNIT 1 FUEL MINIMUM BURNUP VS INITIAL ENRICHMENT FOR REGION II RACKS
FIGURE 5.6-2

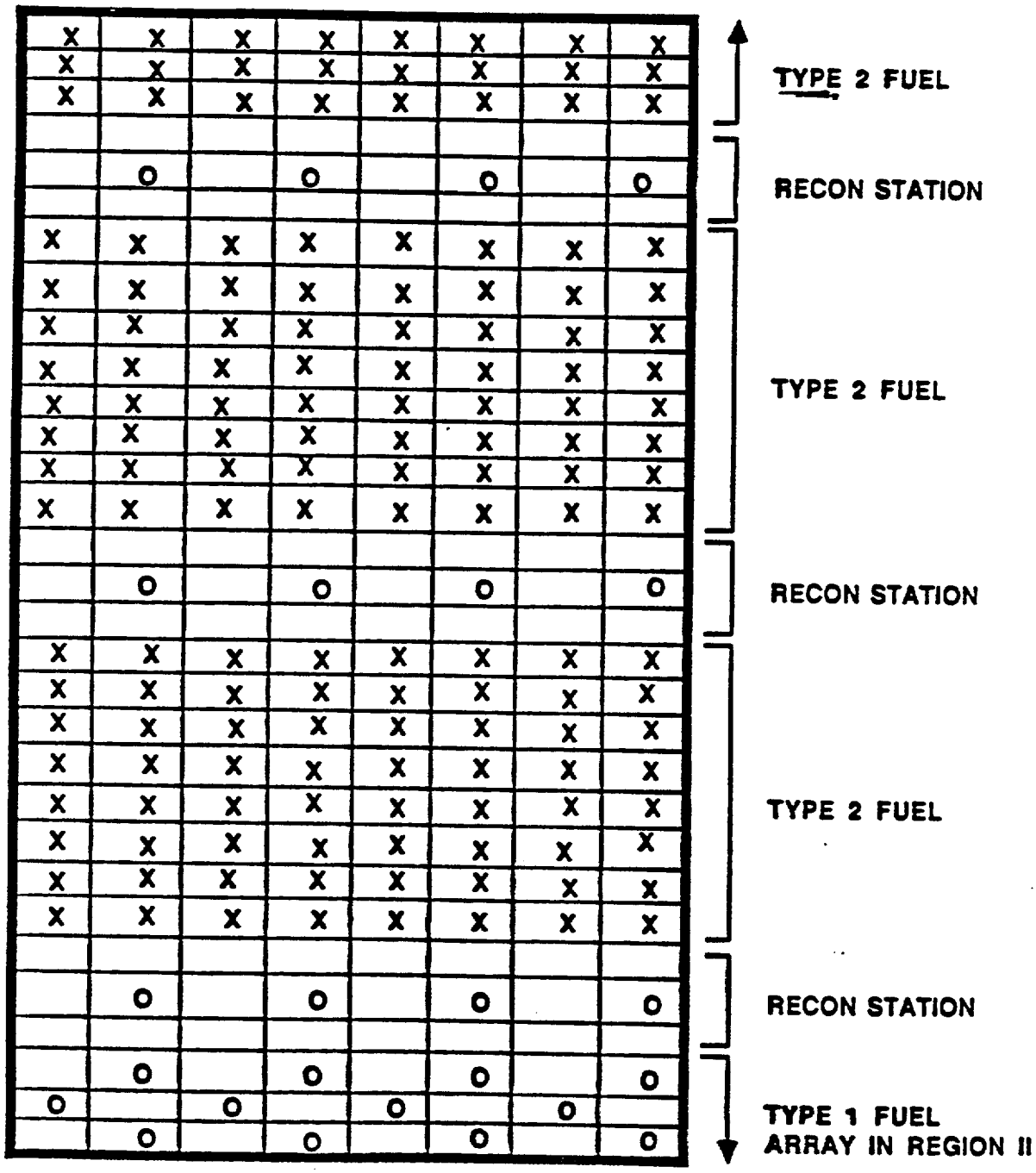


X Type 2 Fuel

O Type 1 Fuel

Empty Storage Location

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
FUEL STORAGE PATTERNS FOR REGION II RACKS
FIGURE 5.6-3



X = TYPE 2 FUEL

O = TYPE 1 FUEL

= EMPTY

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
FUEL STORAGE PATTERNS FOR REGION II RACKS RECONSTITUTION STATION
FIGURE 5.6-4