October 26, 1984

Docket Nos. 50-361
and 50-362

Mr. Kenneth P. Baskin
Vice President
Southern California Edison Company
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Post Office Box 800
Rosemead, California 91770

Mr. James C. Holcombe
Vice President - Power Supply
San Diego Gas & Electric Company
101 Ash Street
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San Diego, California 92112

Gentlemen:

Subject: Issuance of Amendment No 26 to Facility Operating License NPF-10 and Amendment No. 15 to Facility Operating License NPF-15 San Onofre Nuclear Generating Station, Units 2 and 3

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 26 to Facility Operating License No. NPF-10 and Amendment No. 15 to Facility Operating License No. NPF-15 for the San Onofre Nuclear Generating Station, Units 2 and 3, located in San Diego County, California. The amendments modify the Technical Specifications to (1) allow manual coupling and uncoupling of the control element assemblies (CEAs) to the control element drive motor (CEDM) drive shaft extensions during refueling, (2) allow weighing of CEAs during refueling, (3) allow the water level during refueling to be as low as 23 feet above the fuel assemblies, (4) allow correction of the measured heat dissipation of the fuel handling building post-accident cleanup filter system (FHBPACFS) heaters to the nominal voltage for purposes of determining heater operability, and (5) specify a tube thinning criteria of 44% for all steam generator tubes.

These amendments were requested by your letters of June 27, 1984, June 29, 1984, and July 18, 1984, and are covered by proposed changes numbered 50, 141, 179, and 180.

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

Amendment No. 25 for San Onofre 2, issued September 21, 1984, included a revised Technical Specification page regarding response times for Emergency Feedwater Actuation Signal. Through oversight, one of the response times was not changed in Amendment No. 25. The revised response time was reviewed and discussed in the Safety Evaluation supporting the amendment. Enclosed is a revised page 3/4 3-29 to correct the oversight.

Sincerely,

Original signed by:
George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosures:
1. Amendment No. 26 to NPF-10
2. Amendment No. 15 to NPF-15
3. Safety Evaluation

cc w/enclosures: See next page

MEB
RBosmak
10/26/84

*See previous concurrence.

DL:LB#3	DL:LB#3	OELD*	DL:LB#3	DL:AD/L	SSPB*
*HRood/yt	JLee	LChandler	GWKnighton	TNovak	DBrinkman
10/ /84	10/24/84	10/17/84	10/25/84	10/26/84	10/9/84

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SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. NPF-10

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

   A. The applications for amendment to the license for San Onofre Nuclear Generating Station, Unit 2 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) dated June 27, June 29, and July 18, 1984 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this 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. 26, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 26, 1984

DL:LB#3	DL:LB#3	OELD	DL:LB#3
JLee:yt	HRood	LChandler	GWKnighton
10/2/84	10/2/84	10/1/84	10/6/84

OCT 26 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 26

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

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

Amendment Pages	Overleaf Pages
3/4 4-12	3/4 4-11
3/4 9-6	3/4 9-5
3/4 9-11	3/4 9-12
3/4 9-14	3/4 9-13
B 3/4 4-3	B 3/4 4-4
B 3/4 9-2	B 3/4 9-1

Delete page 3/4 4-15a.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calender months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in both sets of inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 44% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs* or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

APPLICABILITY: During movement of CEAs* and/or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of CEAs* and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of CEAs* or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds.

*Except for movement of four finger CEA's, coupling and uncoupling the CEA extension shafts or verifying the coupling and uncoupling.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet* of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

*Water level may be lowered to a minimum of 23 feet above the top of the fuel for movement of four finger CEA's, coupling and uncoupling of CEA extension shafts or for verifying the coupling and uncoupling.

3/4.9.11 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

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. Restore the inoperable fuel handling building post-accident cleanup filter system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

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.

SURVEILLANCE REQUIREMENTS

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.

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding 44% of the nominal tube wall thickness. This criteria was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 GPM leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

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 1720 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.

3/4.9.6 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.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's 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 CEA's 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.

Coupling and uncoupling of the CEA's and the CEDM drive shaft extensions is accomplished using the gripper operating tool. The coupling and uncoupling is verified by weighing the drive shaft extensions.

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING

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 a single fuel assembly and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

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 capability. 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.

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 15
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 3 (the facility) filed by the Southern California Edison Company on behalf of itself and San Diego Gas and Electric Company, The City of Riverside and The City of Anaheim, California (licensees) dated June 27, June 29, and July 18, 1984 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, as amended, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this 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. 15, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 26, 1984

DL:LB#3	DL:LB#3	OELD	DL:LB#3
JLee/yt	HRood	LChandler	GWKnighton
10/2/84	10/2/84	10/17/84	10/25/84

OCT 26 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. NPF-15

DOCKET NO. 50-362

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

Amendment Pages	Overleaf Pages
3/4 4-12	3/4 4-11
3/4 9-6	3/4 9-5
3/4 9-11	3/4 9-12
3/4 9-14	3/4 9-13
B 3/4 4-3	B 3/4 4-4
B 3/4 9-2	B 3/4 9-1

Delete page 3/4 4-16.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calender months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in both sets of inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 44% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs* or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

APPLICABILITY: During movement of CEAs* and/or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of CEAs* and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of CEAs* or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds.

*Except for movement of four finger CEA's, coupling and uncoupling the CEA extension shafts or verifying the coupling and uncoupling.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet* of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

*Water level may be lowered to a minimum of 23 feet above the top of the fuel for movement of four finger CEA's, coupling and uncoupling of CEA extension shafts or for verifying the coupling or uncoupling.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

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. Restore the inoperable fuel handling building post-accident cleanup filter system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

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.

SURVEILLANCE REQUIREMENTS

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.

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding 44% of the nominal tube wall thickness. This criteria was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.5.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 GPM leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

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 1720 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 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.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure with two fingers in each of two adjacent fuel bundles in the periphery of the core. The four finger CEA's 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.

Coupling and uncoupling of the CEA's and the CEDM drive shaft extensions is accomplished using the gripper operating tool. The coupling and uncoupling is verified by weighing the drive shaft extensions.

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING

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 a single fuel assembly and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

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 capability. 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.

SAFETY EVALUATION

AMENDMENT NO. 26 TO NPF-10

AMENDMENT NO.15 TO NPF-15

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

DOCKET NOS. 50-361 AND 50-362

Introduction

Southern California Edison Company (SCE), on behalf of itself and the other licensees, San Diego Gas and Electric Company, the City of Riverside, California, and The City of Anaheim, California has submitted several applications for license amendments for San Onofre Nuclear Generating Station, Units 2 and 3. The evaluations of four such requests are presented below.

I. By letter dated June 27, 1984, SCE requested that the NRC revise Technical Specification 3/4.9.6, Refueling Machine (reference PCN-50). Technical Specification 3/4.9.6 defines the operability and surveillance requirements for the refueling machine to ensure that (a) the refueling machine will be used for all movements of fuel assemblies and Control Element Assemblies (CEAs), (b) the refueling machine has sufficient load capacity to lift a fuel assembly, and (c) the core internals and pressure vessel are protected from excessive lifting forces in the event of inadvertent mechanical interference during fuel handling operations. At present, Technical Specification 3/4.9.6 requires that the refueling machine be used for all movement of the CEAs. However, during refueling, the Control Element Drive Motor (CEDM) drive shaft extensions must be manually uncoupled and recoupled. Coupling and uncoupling is verified by weighing of the CEAs. These operations involve small movements of the CEAs. At present, these small movements of CEAs are prohibited by Technical Specification 3/4.9.6 because they are not done by the refueling machine. However, the refueling machine cannot be used for either coupling, uncoupling or weighing of CEAs. Note that the existing technical specifications allow the four-fingered CEAs to be removed without using the refueling machine. The proposed change would add a note which allows coupling, uncoupling and weighing of CEAs during refueling.

II. By letters dated June 27, 1984 and July 18, 1984, SCE requested that the NRC revise Technical Specification 3/4.9.10 (Refueling) Water Level - Reactor Vessel (reference PCN-179). Specification 3/4.9.10 requires that a minimum water level of 23 feet be maintained above the reactor vessel flange during movements of CEAs or fuel assemblies in the reactor vessel. During refueling, the CEDM drive shaft extensions are uncoupled and recoupled to the CEAs, and the four-finger CEAs are removed using the upper guide structure. Coupling or uncoupling of the CEDM drive shaft extensions involves small movements of the CEAs as does the verification of coupling/uncoupling, and the removal of the four-finger CEAs. Therefore, under the current Technical Specifications, a water level of 23 feet must be maintained above the vessel flange during these operations. However, the design of the tools used to couple and uncouple the CEAs from the CEDM drive shaft extensions requires that the work platform be positioned less than 23 feet above the reactor vessel flange. Verification of CEA coupling/uncoupling is most efficiently accomplished at the time the CEAs are coupled/uncoupled. Thus, the existing specification prohibits coupling, uncoupling, and verification of the CEAs using the tools required for these activities. Also, the design of the upper guide structure lift rig and work platform requires the water level to be less than 23 feet above the reactor vessel flange when latching the four-finger CEAs to the work platform and lifting them into the upper guide structure.

The proposed change adds a note to the applicability for Specification 3/4.9.10 which allows the water level to be lowered to 23 feet above the fuel assemblies rather than the vessel flange (a difference of about 11 feet) during CEA coupling and uncoupling, verification of coupling/uncoupling, and removal of the four-finger CEAs.

III. By letter dated June 27, 1984, SCE requested that the NRC revise Technical Specification 3/4.9.12, Fuel Handling Building Post Accident Cleanup Filter System (reference PCN-180). Specification 3/4.9.12 requires that the Fuel Handling Building Post Accident Cleanup Filter System (FHBPACFS) be operable when irradiated fuel is in the storage pool and defines a number of functional tests which periodically must be conducted to assure such operability. The FHBPACFS includes electrical heaters to maintain the relative humidity at the inlet to the charcoal filters at or below 70% to preserve charcoal adsorber efficiency. Specification 4.9.12.d.3 requires verification that the heater thermal dissipation is within ±5% of the specified rating. The heater ratings contained in the specification are based on the nominal operating voltage. However, when the plant is on line in normal operation, the bus voltages are higher than nominal. Additionally, Specification 4.8.1.1 (Diesel Generator Surveillance Requirements) permits a ±10% bus voltage variation during diesel generator operation. Because the power dissipated by a heater varies with the square of the voltage, small deviations from the nominal voltage (e.g., ±2.5%) will result in heater dissipations outside of the allowable range, thereby rendering the system inoperable.

The proposed change revises Specification 4.9.12.d.3 to allow correction of measured heater dissipation to the nominal voltage for the purpose of determining operability. In addition, the proposed change corrects a typographical error in the specified dissipation for heater E-464. Heater E-464 is actually rated at 28.7 kw versus the 28.4 kw listed currently.

IV. By letter dated June 29, 1984, SCE requested that the NRC revise Technical Specification 3/4.4.4, Steam Generator (reference PCN-141). Technical Specification 3/4.4.4 requires that the steam generator be operable and specifies surveillance requirements to verify steam generator integrity. The current acceptable level of steam generator tube wall thinning shown in Figure 4.4.1 of the Technical Specification is 44% for tube rows 0 through 92 and decreases linearly to 26% in tube row 147. The proposed change will delete Figure 4.4.1 and specify a tube thinning limit of 44% for all steam generator tubes.

Evaluation

I. Revise Technical Specification 3/4.9.6 to allow manual coupling, uncoupling, and weighing of CEAs (PCN-50). The NRC criteria in this area is given in Section 9.1.4 of NUREG-0800 (the Standard Review Plan, or SRP), which discusses acceptance criteria for the fuel handling system. The objectives of the SRP are to preclude criticality accidents and releases of radioactivity. Criticality accidents are, in part, prevented by verification of uncoupling of the CEA extension shafts prior to removal of the upper guide structure, thereby preventing CEA withdrawal when the upper guide structure is removed. The proposed change would permit small movements of the CEAs during refueling due to manual coupling/uncoupling and verification of uncoupling, thereby reducing the probability of accidental criticality.

The proposed specification maintains the requirement to use the refueling machine for significant movements of fuel, requires the refueling machine to have sufficient capacity to lift a fuel assembly and requires an overload cutoff to assure that excessive forces are not applied. The NRC staff has reviewed the proposed change and finds that it meets the SRP acceptance criteria and is therefore acceptable.

II. Revise Technical Specification 3/4.9.10 to allow the water level during refueling to be reduced to 23 feet above the fuel (PCN-179). The NRC criteria for minimum water level in the reactor vessel during refueling is defined in the Bases Section of NUREG-0212, Revision 2, Standard Technical Specifications (STS) for Combustion Engineering Pressurized Water Reactors. Specifically, Bases Section B 3/4.9.10 requires that sufficient water depth (23 feet) be available to remove 99% of the assumed 10% iodine gap activity which would be released by an irradiated fuel assembly striking the reactor vessel flange and rupturing. However, with the fuel assemblies seated in the reactor vessel, as will be the case during CEA coupling, uncoupling, and weighing, and during removal of the four-finger CEAs, no fuel damage could occur above the top of the fuel. The proposed specification requires that 23 feet of water be maintained above the top of the fuel rather than the vessel flange as previously required. This will continue to ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from a fuel assembly damaged by any conceivable accident. Therefore, the proposed change meets the acceptance criteria delineated in the Bases of the STS and is acceptable.

III. Modify Technical Specification 3/4.9.12 to allow correction of the measured heat dissipation by the FHBPACFS heater to the nominal voltage (PCN-180). The NRC criteria for this change is given in SRP Section 9.4.2, Spent Fuel Pool Area Ventilation System (SFPAVS). This section references Regulatory Guide 1.52 which recommends that heaters be installed in the SFPAVS upsteam of the charcoal adsorbers. The heaters must have sufficient capacity to maintain the relative humidity below 70%, thereby preserving the efficiency of the charcoal adsorber. The proposed change effects the manner in which the results of the heater dissipation surveillance tests are evaluated to accommodate the allowed variations for the nominal bus voltage which may exist at the time the surveillance is conducted. The proposed change does not reduce the heater dissipation requirements and maintains the 70% relative humidity acceptance criteria. Therefore, the proposed change satisfies the SRP acceptance criteria and is acceptable.

Additionally, the proposed change increases the required dissipation for heater E-464 from 28.4 kw to 28.7 kw. This corrects a typographical error. Therefore, this part of the proposed change is acceptable.

IV. Revised Technical Specification 3/4.4.4, Steam Generator, to change the tube thinning criteria to 44% for all tubes (PCN-141). The original analysis upon which this Technical Specification is based established the structural adequacy of the San Onofre 2 and 3 steam generator tubes and tube supports, when subjected to various hypothetical accident conditions. It was determined that the limiting event was a combination of a Loss of Coolant Accident (LOCA) and Safe Shutdown Earthquake (SSE). The calculated stresses occurring in the steam generator tube walls as a result of the limiting event were compared to the maximum allowable stresses as defined by the NRC staff's criteria (Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes). The analysis indicated that degradation of up to 64 percent is acceptable for both the straight portion of all tubes and the "U" bend region for the majority of tube rows. The outer tubes rows experienced significant stress in the "U" bent region, due to the combination of hydraulic loads associated with blowdown of the primary system as a result of the LOCA and earthquake-induced accelerations resulting from the SSE. Thus, in the outer tube bundle bend region, the allowable degradation decreased linearly from 64 percent to a minimum of 46 percent at the outermost row. The values in the technical specification vary from 44% to 26%. The 20% differences between the allowable degradation and the Technical Specification limit represents margin for instrument error and degradation between inspections.

However, a revised analysis has shown that tube degradation of 64% is acceptable for all rows of tubes. Thus, the proposed change will remove unnecessary conservatism from the assumptions used in the original analysis and thereby establish a more accurate steam generator tube thinning limit. In addition, the proposed change prevents unnecessary (a) plugging of tubes, (b) associated high personnel radiation exposure and (c) decreases in the steam generator heat transfer surface area.

The revised analysis of the limiting LOCA/SSE scenario includes the frictional or binding restraint on the tubes provided by the vertical tube supports in the horizontal tube run on top of the "U" tube span; this was previously neglected in the original calculations. In addition, the LOCA and SSE peak loads were combined by the square-root-of-the-sum-of-the-squares (SRSS) method; these loads were combined by addition in the original calculations. The use of SRSS, under appropriate circumstances, is an acceptable method under the SRP for combining loads. The combination of a LOCA and SSE is still the limiting event. The revised analysis shows that tube degradation of up to 64 percent is acceptable for all steam generator tubes in meeting the criteria of Regulatory Guide 1.121, which results in the proposed Technical Specification limit of 44% for all tubes, based on the 20% margin previously used.

Section 5.4.2.2 of the SRP reference Regulatory Guide 1.83 which specifies inservice inspection criteria for determining steam generator operability. The proposed change prescribes steam generator tube thinning criteria which was developed in accordance with Regulatory Guide 1.83 and the SRP.

The NRC staff has reviewed the revised analysis of tube stresses and has concluded that the proposed change meets the above-described staff criteria for steam generator tube thinning, and therefore is acceptable.

Contact With State Official

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

Environmental Consideration

These amendments involve changes in the installation or use of facility components located within the restricted area. The staff has determined that the amendments involve no significant increase in the amounts of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on such findings. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec. 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

Conclusion

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

Dated: October 26, 1984

DL:LB#3
HRood/yt
10/2/84

DL:LB#3
GWKnighton
10/25/84

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV4054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kv Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

ISSUANCE OF AMENDMENT NO. 26 TO FACILITY OPERATING LICENSE NPF-10
AND AMENDMENT NO. 15 TO FACILITY OPERATING LICENSE NPF-15
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

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