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

SOUTHERN CALIFORNIA EDISON COMPANY AND

SAN DIEGO GAS AND ELECTRIC COMPANY

DOCKET NO. 50-206

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 38
License No. DPR-13

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

   A. The application for amendment by Southern California Edison Company and San Diego Gas and Electric Company (the licensees) dated December 30, 1977 (Proposed Change No. 68) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility License No. DPR-13 is hereby amended to read as follows:

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(B) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 38 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas V. Wambach

for Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 17, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 38

PROVISIONAL OPERATING LICENSE NO. DPR-13

DOCKET NO. 50-206

Revise Appendix A Technical Specifications and Bases by removing the following pages and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

	REMOVE	INSERT
	12	12
	13	13
	88	88
	89-95	- -
	8	8*
	22	22*
	25	25*
	54	54*
	56	56*
	58b	58b*
	60L	60L*
	86	86*
(Appendix B)	3-10	3-10*
	5-9	5-9*
	5-11	5-11*

*These pages are included to correct administrative errors which occurred with the issuance of previous license amendments.

3.1.3 COMBINED HEATUP, COOLDOWN, AND PRESSURE LIMITATIONS

Applicability: Applies to heatup and cooldown of the reactor coolant system.

Objective: To maintain the structural integrity of the reactor coolant system throughout the lifetime of the plant.

Specification:

A. Reactor pressure and heatup and cooldown of the reactor coolant system during the first 6 years of equivalent full power operation shall be limited in accordance with Figures 3.1.3a and 3.1.3b. Thereafter, limits shall be based on neutron exposure equivalent to not less than 6 years of full power operation, and Figures 3.1.3a and 3.1.3b shall be updated accordingly.

B. Figures 3.1.3a and 3.1.3b shall be updated in accordance with the following criteria and procedures:

(1) The methods of Appendix G, "Protection Against Nonductile Failure", to Section III of the ASME Boiler and Pressure Vessel Code shall be used to obtain the allowable pressure-temperature relationships for the reactor coolant system.

(2) The curves in Figure 3.1.3c shall be used in predicting the reference nil-ductility temperature increase, ΔRT_{NDT}, unless measurements on the irradiation specimens show ΔRT_{NDT}s greater than those predicted by the curves, in which case a new curve having the same slope as the original shall be constructed.

C. The pressurizer heatup rate of 100°F/hour and cooldown rate of 200°F/hour shall not be exceeded.

D. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line as shown in Figures 3.1.3a and 3.1.3b.

<u>Basis</u>: The initial Reference Nil Ductility Temperature (RT_{NDT}) for all reactor vessel material based on Charpy V-notch data drop weight tests, and conservative estimates* is 160°F or less. The RT_{NDT} at the 1/4 thickness location (location of the Appendix G reference flaw tip) increases as a function of cumulative neutron exposure up to approximately 258°F for the core region of the reactor vessel after 30 years of operation.

A six (6) equivalent full power year service period was chosen for the operational limits given in this specification because at the end of this period the limiting RT_{NDT} of the reactor vessel at the 1/4 thickness location is approximately 207°F in the core region. This provides an approximate 47°F margin over the head region of the vessel which sees negligible radiation exposure throughout the life of the plant.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table 1. The fast neutron (E $\gtrsim$ 1 Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Figure 3.1.3d. Using the applicable fluence at the end of the year period and the copper content of the material in question, the ΔRT_{NDT} is obtained from Figure 3.1.3c.

Values of ΔRT_{NDT} may continue to be determined in this manner unless measurements on the irradiation specimens show ΔRT_{NDTs} greater than those predicted by the curves for the equivalent capsule exposure.

Allowable pressure temperature relationships for various heatup and cooldown rates are calculated using methods derived from Non-Mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and discussed in detail in Reference 1.

The results of these calculations are provided in Reference 2.
The design heatup and cooldown rates for the pressurizer are 100°F/hour and 200°F/hour, respectively.

The criticality limit given in Figures 3.1.3a and 3.1.3b are based on RT_{NDT} + 160°F and a 40°F margin over the pressure-temperature limit curves for heatup and cooldown. The criticality limit is designed to provide assurance that the vessel will not exceed the allowable stress intensity factor before yielding under postulated transient conditions.

*"Strong" direction Charpy V-notch data was reduced by 35% to estimate "weak" direction data, and drop weight NDTT of forgings was estimated as 60°F and that of weldments as 0°F or 30 ft-lb temperature, whichever is higher.

f. Records of in-service inspections performed pursuant to these Technical Specifications.

g. Records of Quality Assurance activities as required by the QA Manual.

h. Records of reviews performed for changes made to procedures or equipment or reviews or tests and experiments pursuant to 10CFR50.59.

i. Records of meetings of the OSRC and the NARC.

6.10.3 The following records shall be retained for two years:

a. Records of facility radiation and contamination surveys.

b. Records of training of facility personnel.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

3.1 REACTOR COOLANT SYSTEM

3.1.1 Maximum Reactor Coolant Activity

Applicability: Applies to measured maximum activity in the reactor coolant system at any time.

Objective: To limit the consequences of an accidental release of reactor coolant to the environment.

Specification: A. The specific activity of the reactor coolant shall be limited to:

1. $\leq$ 1.0 μ Ci/gm dose equivalent I-131.

2. $\leq$ 100 /$\bar{E}$ μ Ci/gm, where $\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

B. Action

1. With the specific activity of the coolant determined to be >1.0 μ Ci/gm but < 60 μ Ci/gm dose equivalent I-131, the reactor may be started up or operation may continue for up to 48 hours provided that operation under these circumstances does not exceed 800 hours in any consecutive 12 month period. Should the total operating time at a reactor coolant specific activity > 1.0 μ Ci/gram Dose Equivalent I-131 exceed 500 hours in any consecutive six month period, the licensee shall report the number of hours of operation above this limit to the NRC within 30 days.

2. With the specific activity of the reactor coolant determined to be > 1 μ Ci/gm dose equivalent I-131 for more than 48 hours during one continous time interval or > 60 μ Ci/gm dose equivalent I-131 or > 100/$\bar{E}$ μ Ci/gm, have the reactor subcritical with the average temperature of the reactor coolant (T_{avg}) less than 535°F within 6 hours.

(2) Containment Spray System

a. Two refueling water pumps are operable.

b. Two hydrazine additive pumps are operable.

c. Hydrazine tank level and hydrazine concentration comply with Specification 3.3.4.

(3) Valves and interlocks associated with each of the above systems are operable.

(4) Effective leakage from the recirculation loop outside the containment shall be less than 625 cc/hr. as calculated from the following formula.

Effective Leakage = $a_1 \times L_1 + a_2 \times L_2 + a_3 \times L_3$

where. L_1 = pump and valve leakage which drains to auxiliary building sump

L_2 = valve leakage in auxiliary building or doghouse

L_3 = valve leakage outside

a_1 = iodine release factor for leakage in auxiliary building sump

a_2 = iodine release factor for leakage in auxiliary building or doghouse

a_3 = iodine release factor for leakage outside the auxiliary building or doghouse

If effective leakage from the recirculation loop outside the containment exceeds 625 cc/hr, make necessary repairs to limit leakage to 625 cc/hr within 72 hours or be in cold shutdown within the next 36 hours.

B. During critical operation or when the reactor coolant system temperature is above 200° F, as appropriate per Item A above, maintenance shall be allowed on any one of the following items at any one time:

(1) One motor-operated valve at a time (MOV 1100B or 1100D) in the recirculation loop upstream of the charging pump suction header, for a period of time not longer than 72 consecutive hours.

(2) One refueling water pump and/or its associated discharge valve at a time, for a period not longer than 72 consecutive hours.

(3) One hydrazine pump and/or its associated discharge valve (SV600 or 601) at a time, for a period of time not longer than 72 consecutive hours.

3.3.3 MINIMUM WATER VOLUME AND BORON CONCENTRATION IN THE REFUELING WATER STORAGE TANK

Applicability: Applies to the inventory of borated refueling water.

Objective: To insure immediate availability of safety injection and containment spray water of required quality.

Specification: When the Safety Injection System or the Containment Spray System is required to be operable, the refueling water tank shall be filled to at least elevation 50 feet with water having a boron concentration of not less than 3750 ppm and not greater than 4300 ppm.

Basis: The refueling water storage tank serves two purposes; namely:

(1) As a reservoir of borated water for accident mitigation purposes,

(2) As a reservoir of borated water for flooding the refueling cavity during refueling.

Approximately 220,000 gallons of borated water is required to provide adequate post-accident core cooling and containment spray to maintain calculated post-accident doses below the limits of 10 CFR 100[1]. The refueling water storage tank filled to elevation 50 feet represents in excess of 240,000 gallons.

A boron concentration of 3750 ppm is required to meet the requirements of postulated steam line break.[2] A maximum boron concentration of 4300 ppm ensures that the post-accident containment sump water is maintained at a pH between 7.0 and 7.5[3].

The refueling tank capacity of 240,000 gallons is based on refueling volume requirements.

Sustained temperatures below 32°F do not occur at San Onofre. At 32°F, boric acid is soluble up to approximately 4650 ppm boron. Therefore, no special provisions for temperature control to avoid either freezing or boron precipitation are necessary.

Reference:

(1) Enclosure 1 "Post-Accident Pressure Reanalysis, San Onofre Unit 1" to letter dated January 19, 1977 in Docket No. 50-206.

(2) "Steam Line Break Accident Reanalysis, San Onofre Nuclear Generating Station, Unit 1, October 1976" submitted by letter dated December 30, 1976 in Docket No. 50-206

(3) "Addition information, San Onofre, Unit 1" submitted by letter dated March 24, 1977 in Docket No. 50-206.

4.5 RADIOACTIVE LIQUID WASTE RELEASE

Applicability: Applies to release of radioactive liquid waste to the Circulating Water System.

Objective: To verify that discharge of radioactive waste to the Circulating Water System is maintained below the limits set forth in 10 CFR 20.

Specification:

A. Averaged over a year, the release rates of liquid wastes shall not result in concentrations in the circulating water discharge in excess of Part 20 limits for unrestricted areas, except that the maximum release rate over the period of one hour shall not exceed 10 times the yearly averaged limit.

B. At least one circulating water pump shall be in operation whenever radioactive liquid wastes are released.

C. Prior to release of waste, gross activity shall be sampled and determination made of the maximum permissible release rate.

D. All radioactive wastes entering the Circulating Water System shall be monitored for isotopic or gross activity during discharge. Such monitoring may be accomplished by either of the following methods:

1. Continuous monitoring with the in-stream liquid waste monitor channel; or, if the liquid waste monitor channel is inoperable,

2. Analyses of a minimum of three samples of effluent stream taken approximately towards the beginning, midpoint, and end of each release period."

E. The liquid waste monitor and the flow rate meter shall be calibrated at a minimum frequency of once every six months, and normal response of the monitor shall be tested weekly.

F. A record of all liquid waste releases shall be kept in accordance with Specification 6.10.

4.6 RADIOACTIVE GASEOUS WASTE RELEASE

Applicability: Applies to the release of radioactive gaseous waste from the plant stack.

Objective: To verify discharge of radioactive gaseous waste to the atmosphere will not result in ground level radioactivity concentrations outside the plant boundaries in excess of limits established in 10 CFR 20.

Specification: A. Averaged over a year, release rates of gaseous wastes in curies/sec shall not result in a value exceeding that calculated from the following formula:

$$1.8 \times 10^5 \left(\frac{m^3}{sec}\right) \times \sum C_x \left(\frac{uc}{cc}\right)$$

Where C_x is the concentration of any radioisotope X, the values of the concentrations of all isotopes discharged shall be such that $\sum \frac{C_x}{(MPC)_x}$ is less than 1.0. $(MPC)_x$ as defined above shall be that stated in Column 1, Table II of 10 CFR 20. The maximum release rate over any one hour shall not exceed 10 times the yearly averaged limit as stated above.

B. At least one stack fan shall be in operation delivering normal flow whenever radioactive gaseous wastes are released to the vent stack.

C. All radioactive wastes discharged through the stack shall be monitored continuously for gross activity.

D. A record of the above releases shall be kept in accordance with Specification 6.10.

E. The stack gas and particulate monitors shall be calibrated at a minimum frequency of once every six months, and normal response of each monitor shall be tested weekly.

Basis: Prior to release to the atmosphere, gaseous wastes from the ratioactive waste disposal system are mixed in the stack flow of two 20,000 cfm fans.[1] Dilution then occurs in the atmosphere.

The formula prescribed in specification A takes atmospheric dilution into account and ensures that at the point of maximum ground concentration the requirements of 10 CFR 20 will not be

F. Within 130 days from the date of issuance of addenda to Section XI of the ASME Boiler and Pressure Vessel Code, a review for applicability of such addenda shall be made and a proposed change to the specifications as determined by this review shall be submitted to the AEC pursuant to Section 50.59 in 10 CFR Part 50.

Basis: Periodic visual inspection of the rod cluster control assemblies will verify that the structural integrity is maintained.

The inservice inspection program specified conforms (as closely as the as-built condition of the plant permits) to the 1971 edition (including the Summer and Winter 1971 Addenda) of Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems". To the extend applicable, based upon the existing design and construction of the plant, Table 4.7.1 duplicates Table IS-261 of Section XI of the Code for ease of comparison. Table 4.7.2 ensures that the inspections will be made according to the intervals specified in IS-242 of Section XI of the Code. Specification B includes a provision for future updating as required by outages, changes in refueling dates, etc.

Significant exceptions taken to Table IS-261 of Section XI are as follows:

Item 1.5 - This item has been deleted since the plant design and currently available techniques do not permit a volumetric examination of this type of penetration.

Item 2.3 - This item has been modified to delete the surface examination since the pressurizer heater penetrations are not accessible to a surface examination.

Item 2.5 - This item has been deleted since no bolting 2-inch and over is used.

Item 3.4 - This item has been deleted since no bolting 2-inch and over is used.

Item 4.2 - This item has been modified to exclude examination of longitudinal pipe welds since none exist in present plant design.

Item 4.3 - This item has been deleted since no bolting 2-inch and over is used.

Item 4.5 - This item has been deleted since no integrally-welded supports are used.

Item 5.1 - This item has been modified to delete volumetric examinations since currently available techniques do not exist.

4.15 Fire Protection Systems Surveillance

Applicability: Applies to the surveillance of fire detection and extinguishing systems and equipment.

Objective: To ensure the operability of fire detection and extinguishing systems and equipment.

Specifications: A. The Fire Suppression Water System[1] shall be demonstrated to be operable.

(1) With the San Onofre Unit 1 fire water pumps satisfying the pump requirements of Technical Specification 3.[illegible], at least once per seven days by verifying the water supply volume in the San Onofre Unit 1 Service Water Reservoir. With the San Onofre Units 2 and 3 fire water pumps satisfying the pump requirements of Technical Sepcification 3.14.A(1), by initially verifying the water supply volume in the San Onofre Units 2 and 3 service and firewater storage tanks and at least once per seven days thereafter.

(2) At least once per 31 days on a staggered test basis by starting each pump satisying the pump requirements of Technical Specification 3.14.A(1) and operating it for at least fifteen minutes.

(3) At least once per thirty one days by verifying that each valve (manual, power operated or automatic is in its correct position. For valves located inside the containment sphere, verification shall be made consistent with the 31-day requirement when possible during available plant outages or during containment entrances for other reasons.

(4) At least once per 12 months by cycling each testable valve through one complete cycle of full travel. The isolation valve between the screen wash pumps and the Fire Suppression Water System shall be tested only whenever the screen wash pumps are used to satisfy the pump requirements of Technical Specification 3.14.A(1).

(5) At least once per 18 months by performing a system functional test which includes simulated actuation of the system, and:

a. Verifying that each valve in the flow path is in its correct position,

b. Verifying that each pump develops at least 90% of the flow and head at some point on the manufacturer's pump performance curves.

6.9.3 Unique Reporting Requirements

The following special reports shall be submitted as required:

a. Inservice Inspection (Technical Specification 4.7)

b. Reactor Vessel Surveillance Program (Technical Specification 4.9)

c. Fire Protection Systems (Technical Specification 3.14).

The results of required leak tests performed on sealed sources (Technical Specification 4.12) shall be reported annually if the tests reveal the presence of 0.005µCi or more of removable contamination.

Reporting Requirement

A detailed interpretation and analysis of the results of the General Ecological Survey will be presented in the Annual Operating Report, including the California Department of Fish and Game annual fish catch statistics. Alternatively, a summary analysis and preliminary interpretation of the General Ecological Survey data may be presented semiannually, in which case a detailed interpretation and analysis of the complete year of data will be provided annually. Special reports will be prepared in accordance with ETS 5.6.3.d whenever significant or unusual changes are observed.

Bases

The basis of the general ecological survey will be to assure that every reasonable procedure is taken to monitor any significant effects on the marine environment which might be the result of the operation of the Station; and to identify any significant changes to the marine ecology.

The ecological survey will be coordinated with the chemical and oceanographic programs. The locations of the plankton, fish, benthic, intertidal and all physical monitoring stations are based on 9 years of data from the ongoing monitoring program. Divers will occupy all stations prior to their establishment.

c. Radiological Environmental Monitoring

(1) For each medium sampled during the reporting period, e.g., air, baybottom, surface water, soil, fish, include:

(a) Number of sampling locations.

(b) Total number of samples.

(c) Number of locations at which levels are found to be above local backgrounds, and

(d) Highest, lowest, and the mean concentrations or levels of radiation for the sampling point with the highest mean and description of the location of that point with respect to the site.

(2) If levels of radioactive materials in environmental media as determined by an environmental monitoring program indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, 10 CFR Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided.

(3) If statistically significant variations of offsite environmental concentrations with time are observed, correlation of these results with effluent release shall be provided.

(4) Individual samples which show higher than normal levels (25% above background for external dose, or twice background for radionuclide content) shall be noted in the reports.

(5) Results of all radiological samples taken shall be summarized on a quarterly basis following the format of Table 5.6-1 for inclusion in the Annual Operating Report. In the event that some results are not available by March 31 of the following year, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report, but not later than July 1 of the year in which the report was due.

d. Oceanographic and Biological Environmental Monitoring

A detailed annual analysis of oceanographic and biological monitoring will be submitted with each Annual Operating Report. The analysis report will be a summary of the monitoring program results and an assessment of the observed impacts, if any, of the Station operation on the marine environment. In the event that some detailed analyses are not available by March 31 of the following year, the report shall be submitted noting and explaining the reasons for the missing results, but including at a minimum preliminary analyses and conclusions for the reporting period. The missing data shall be submitted in a supplemental report before July 1 of the year that the report was due.

The first Semiannual Operating Report following implementation of this environmental technical specification will identify the procedures to be used in the monitoring program.

5.6.2 Routine Reports - Semiannual

Within 60 days after January 1 and July 1 of each year a report shall be submitted covering the radioactive content of effluents released to unrestricted areas and shipments of solid waste during the previous six months of operation. The data shall be summarized on a monthly basis and shall include as a minimum the following:

a. Radioactive Effluent Releases

A statement of the quantities of radioactive effluents released from the station with data summarized on a monthly basis following the format of USAEC Regulatory Guide 1.21.

(1) Gaseous Effluents

(a) Gross Radioactivity Releases

i. Total gross radioactivity (in curies), primarily noble and activation gases.

ii. Maximum gross radioactivity release rate during any one-hour period.