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 NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 55
License No. DPR-13

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

   A. The applications* for amendment by Southern California Edison Company and San Diego Gas and Electric Company (the licensees) dated December 10, 1980, as supplemented April 10 and 15, 1981, and May 5, 1981, comply 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 applications, 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.

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*Also, see supporting documents submitted by letters identified in the transmittal letter for this amendment

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 Provisional Operating License No. DPR-13 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 55, are hereby incorporated in the license. Southern California Edison Company shall operate the facility in accordance with the Technical Specifications.

3. Provisional Operating License No. DPR-13 is further amended by the reinstatement of Paragraph 3.E (previously deleted by Amendment No. 37 dated October 31, 1978) to read as follows:

   E. Steam Generator Inspections

   Southern California Edison Company shall bring the reactor to a cold shutdown condition to perform an inspection of the steam generators within six equivalent months of operation from the start of Cycle 8 operation. The inspection program shall be submitted to the Commission at least 45 days prior to the scheduled shutdown. Commission approval shall be obtained before resuming power operation following this inspection.

4. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 8, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 55

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 contain the captioned amendment number and vertical lines indicating the areas of change.

Pages

2

6

17

18*

32

39c

60-R

60-U

*Page 18 is included for pagination purposes only; there are no changes to the provisions contained thereon.

2.1 REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature

Applicability: Applies to reactor power, system pressure, coolant temperature, and flow during operation of the plant.

Objective: To maintain the integrity of the reactor coolant system and to prevent the release of excessive amounts of fission product activity to the coolant.

Specification: Safety Limits

(1) The reactor coolant system pressure shall not exceed 2735 psig with fuel assemblies in the reactor.

(2) The combination of reactor power and coolant temperature shall not exceed the locus of points established for the RCS pressure in Figure 2.1.1. If the actual power and temperature is above the locus of points for the appropriate RCS pressure, the safety limit is exceeded.

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 1.

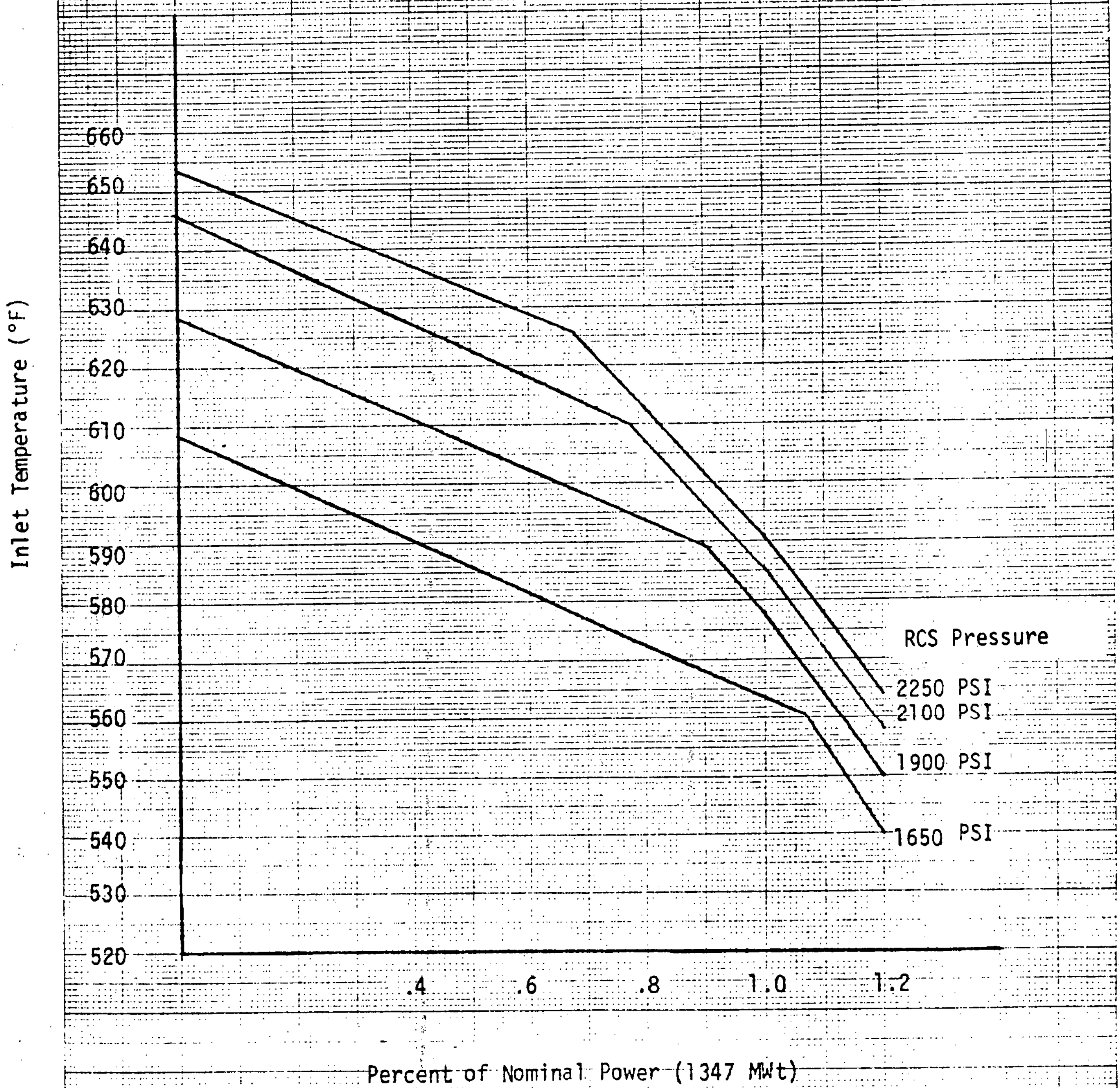
TABLE 1

		TRIP SETTING
1.	Pressurizer High Level	$\leq$ 27.3 ft. above bottom of pressurizer
2.	Pressurizer Pressure: High	$\leq$ 2220 psig
**3.	Nuclear Overpower	$\leq$ 109% of indicated full power
*4.	Variable Low Pressure	$\geq$ $26.15\,(0.894\,\Delta T + T_{avg}) - 14341$
*5.	Coolant Flow	$\geq$ 85% of indicated full loop flow

* May be bypassed at power levels below 10% of full power.

** The nuclear overpower trip is based upon a symmetrical power distribution. If an asymmetric power distribution greater than 10% should occur, the nuclear overpower trip on all channels shall be reduced one percent for each percent above 10%.

Figure 2.1.1
Safety Limits
Temperature, Power, Pressure
RCS Flow ≥ 195,000 GPM



3.1.4 LEAKAGE

Applicability: Applies to reactor coolant system leakage.

Objective: To ensure that leakage from the reactor coolant system does not exceed acceptable limits.

Specification:

A. The reactor coolant system shall be monitored for evidence of leakage.

B. Detectable leakage from the primary coolant system shall be investigated and evaluated. In any event, if the leakage exceeds 1 gpm and the source of leakage is not identified, the reactor shall be shut down. If the sources of leakage have been identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage rate not exceeding 6 gpm shall be permitted.

C. The reactor will be placed in hot standby within six hours and in cold shutdown within the following thirty hours on detection and confirmation of any of the following conditions:

   1. An increase in primary to secondary leakage of 140 gpd (0.1 gpm) over a period of twenty-four hours in any steam generator.

   2. Any primary to secondary leakage in excess of 215 gpd (0.15 gpm) in any steam generator; or

   3. Measured increase in primary to secondary leakage in excess of 15 gpd (0.01 gpm) per day, when measured primary to secondary leakage is above 140 gpd.

   Following reactor shutdown, leaking tubes will be repaired or plugged.

D. In addition, in accordance with the Technical Specifications, the reactor will be placed in hot standby within six hours and in cold shutdown within the following thirty hours on detection and confirmation of primary to secondary leaks in excess of 0.3 gpm in any steam generator. Following reactor shutdown, an eddy current inspection will be performed as required by the Technical Specifications, any leaking steam generator tubes will be repaired or plugged and the NRC will be notified prior to resumption of plant operation.

Basis: Two basic kinds of leakage from the reactor coolant system are possible, namely:

1. To other closed systems.

2. Directly to the containment.

Systems into which leakage from the reactor coolant system could occur are designed to accept such leakage. However, leakage directly into the containment indicates the possibility of a breach in the coolant envelope. For this reason, the acceptable value for a source of leakage not identified was set at one gpm.

Once the source of leakage has been identified, it can be determined if operation can safely continue. Under these conditions, an allowable leakage rate of 6.0 gpm has been established. This is based upon the contingency of sustained loss of all off-site power and failure of the on-site generation. With 6.0 gpm leakage, decay heat removal can safely be accomplished for a period in excess of 12 hours. Within the 12 hour period, the reactor coolant system can be depressurized.

To comply with Paragraph IV.C.1(b)(4) of the "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors" adopted by the AEC on June 19, 1971, the maximum allowable identified leakage rate from the primary coolant system has been established as not exceeding 6.0 gpm. This value is based on operating experience regarding non-safety related equipment limitations which has shown that, under certain circumstances where primary system leakage is directed to the gas handling portion of the radwaste system, the capacity of this system would be exceeded during extended operation with a leakage greater than 6.0 gpm.

Detection of leaks from the reactor coolant system to the containment is accomplished through use of any or all of the following methods:

1. Sump level.

2. Radiation monitoring.

3. Humidity measurements.

With these methods, a leak of one gpm can be detected in a matter of hours. Detection of leaks to other systems is accomplished through the use of radiation monitoring, level indications in the affected system, and water chemistry variations. In both cases, large leaks would be detected by indications from process variables in the reactor coolant and related systems.

The justification for the 0.3 gpm primary to secondary leakage limit is as described in the Basis for Technical Specification 4.16.

1. The initial design maximum value of specific power is 15 kW/ft. The values of $F^N_{\Delta H}$ and F_Q total associated with this specific power are 1.75 and 3.23, respectively.

   A more restrictive limit on the design maximum value of specific power, $F^N_{\Delta H}$ and F_Q is applied to operation in accordance with the current safety analysis including fuel densification and ECCS performance. The values of the specific power, $F^N_{\Delta H}$ and F_Q are 13.7 kW/ft., 1.55 and 2.89,, respectively. The control group insertion limits in conjunction with Specification B prevent exceeding these values even assuming the most adverse Xe distribution.

2. The minimum shutdown capability required is 1.25% $\Delta\rho$ at BOL, 1.9% $\Delta\rho$ at BOL and defined linearly between these values for intermediate cycle lifetimes. The rod insertion limits ensure that the available shutdown margin is greater than the above values.

3. The worst case ejected rod accident (8) covering HFP-BOL, HZP-BOL, HFP-EOL and HZP-EOL shall satisfy the following accident safety criteria:

   a) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for nonirradiated fuel and 200 cal/gm for irradiated fuel.

   b) Fuel melting is limited to less than the innermost 10% of the fuel pellet at the hot spot.

Low power physics tests are conducted approximately one to four times during the core cycle at or below 10% power. During such tests, rod configurations different from those specified in Figure 3.5.2.1 may be employed.

It is understood that other rod configurations may be used during physics tests. Such configurations are permissible based on the low probability of occurrence of steam line break or rod ejection during such rod configurations.

Operation of the reactor during cycle stretch out is conservative relative to the safety considerations of the control rod insertion limits, since the positioning of the rods during stretch out results in an increasing net available shutdown.

Compliance with Specification B prevents unfavorable axial power distributions due to operation for long intervals at deep control rod insertions.

3.11 CONTINUOUS POWER DISTRIBUTION MONITORING

Applicability: Applies to axial limit.

Objective: To provide corrective action in the event that the axial offset monitoring system limits are approached.

Specification:

A. The incore axial offset limits shall not exceed the functional relationship defined by:

For positive offsets: $IAO = \dfrac{2.89/P-2.1225}{0.03021} - 3.0$

For negative offsets: $IAO = \dfrac{2.89/P - 2.1181}{-.03068} + 3.0$

where

IAO = incore axial offset

P = fraction of rated thermal power

B. If the incore limit defined by Specification A as measured by the excore axial offset system is exceeded by both axial offset monitoring channels, reactor power shall be reduced until Specification A is satisfied.

C. If it is determined that one of the excore axial offset monitoring channels is inoperable the other axial offset channel shall be used to provide power distribution information. In addition one NIS channel current shall be logged every 2 hours and axial offset information determined from this data until the inoperable channel has been returned to service.

D. If both channels should be declared inoperable, at least three NIS channel currents shall be logged every two hours and offset information determined from three data. If no method for determining axial offset is available, reactor power shall be reduced to 90% of rated thermal power.

2. If two consecutive inspections indicate that less than 10% of the tubes inspected have either (a) new imperfections greater than 20% or (b) previous imperfections that have increased more than 10% since their last inspection, the inspections shall be not less than 10 nor more than 40 calendar months after the previous inspection.

3. Unscheduled inspections shall be conducted in accordance with Specification A in the event of primary-to-secondary leaks exceeding Specification 3.1.4.C, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

E. ACCEPTANCE CRITERIA

1. As used in this specification:

   a. Imperfection means an exception to the dimensions, finish, or contour required by drawing or specification.

   b. Defect means an imperfection of such severity that the tube is unacceptable for continued service.

   c. Plugging limit means the imperfection depth at or beyond which plugging of the tube must be performed. The plugging limit is equal to or greater than 50% of the nominal tube wall thickness, except where sleeves are installed, in which case the plugging limit is equal to or greater than 40% of the nominal sleeve wall thickness.

2. If, in the inspections performed under Specification A,

   a. Less than 10% of the total tubes inspected have imperfections greater than 20% that were not detected during the previous 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 = .3 gallons per minute 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 .3 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require shutdown during which the leaking tubes will be located and plugged and additional inspections performed.

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 the plugging limit of 50% of the tube nominal wall thickness, except where sleeves are installed, in which case the plugging limit is 40% of the nominal sleeve wall thickness. A plugging limit of 50% for tubes and 40% for sleeves ensures that defects will not occur between inspection intervals.

The results of tube ID gauging and dent detection conducted in San Onofre Unit 1 steam generators demonstrate that the denting process has been arrested. Continuing assurance of this condition can be provided by performing a program of limited tube ID gauging and dent detection in either steam generator A or C on a refueling outage frequency. Adequate surveillance of denting related tube restrictions can be maintained at refueling intervals by noting any new restrictions during the conduct of general surveillance and AVB inspections and by gauging tubes which have previously been noted as being restricted. Progression of denting can also be monitored in either steam generator A or C by evaluating third and fourth support plate denting data obtained from the general surveillance and AVB inspections as well as from the ID gauging program and comparing these results with those of previous inspections.