August 20, 1986

Docket Nos.: 50-361
and 50-362

Mr. Kenneth P. Baskin
Vice President
Southern California Edison Company
2244 Walnut Grove Avenue
Post Office Box 800
Rosemead, California 91770

Mr. James C. Holcombe
Vice President - Power Supply
San Diego Gas & Electric Company
101 Ash Street
Post Office Box 1831
San Diego, California 92112

Gentlemen:

Subject: Issuance of Amendment No. 52 to Facility Operating License NPF-10 and Amendment No. 41 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. 52 to Facility Operating License No. NPF-10 and Amendment No. 41 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 revise Technical Specification 3/4.11.2, "Gaseous Effluents" to allow incineration of radioactively contaminated oil.

These amendments were requested by your letters of April 27 and August 29, 1985 and are covered by Proposed Change Number PCN-111.

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

Page B 3/4 6-4 of the Technical Specifications for NPF 10 and NPF-15 was revised in Amendment Nos. 51 and 40, respectively, issued August 11, 1986. Due to logistical problems, the Unit 3 page contained a paragraph which had been previously revised. Please remove the Page B 3/4 6-4 for Unit 3 and replace it with the enclosed Page B 3/4 6-4 (and its overleaf B 3/4 6-3).

Sincerely,

/s/

Harry Rood, Senior Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 52 to NPF-10
2. Amendment No. 41 to NPF-15
3. Safety Evaluation
4. Page B 3/4 6-4 for Unit 3 T.S.

cc: See next page

*See previous concurrence page

HR for GK

PBD7	PBD7*	OELD*	DIR:PBD7
HRood/es	JLee		GWKnighton
*8/6/86	8/18/86	8/14/86	8/20/86

August 20, 1986

ISSUANCE OF AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NPF-10
AND AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NPF-15
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DISTRIBUTION

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Mr. Kenneth P. Baskin
Southern California Edison Company

San Onofre Nuclear Generating Station
Units 2 and 3

cc:
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Regional Administrator, Region V
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Resident Inspector, San Onofre NPS
c/o U. S. Nuclear Regulatory Commission
Post Office Box 4329
San Clemente, California 92672

cc:
California State Library
Government Publications Section
Library & Courts Building
Sacramento, CA 95841
ATTN: Ms. Mary Schnell

Mayor, City of San Clemente
San Clemente, CA 92672

Chairman, Board Supervisors
San Diego County
1600 Pacific Highway, Room 335
San Diego, CA 92101

California Department of Health
ATTN: Chief, Environmental
Radiation Control Unit
Radiological Health Section
714 P Street, Room 498
Sacramento, CA 95814

Mr. Joseph O. Ward, Chief
Radiological Health Branch
State Department of Health Services
714 P Street, Building #8
Sacramento, California 95814

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52
License No. NPF-10

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

   A. The application 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 April 27 and August 29, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 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 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. 52, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The change in Technical Specifications is to become effective within 30 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensees shall adhere to the Technical Specifications existing at the time. The period of time during change over shall be minimized.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harry Rood

Harry Rood, Senior Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 20, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 52

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 Page	Overleaf Page
3/4 11-9	3/4 11-10
3/4 11-11	3/4 11-12
3/4 11-13	3/4 11-14

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
B. Containment Purge 42 inch	P Each Purge[b,c]	P Each Purge[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
8 inch	M[b] Grab Sample	M[b]	Principal Gamma Emmitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. 1. Condenser Evacuation System	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
2. Plant Vent Stack	W[b,e]	W[b]	H-3	1×10^{-6}
D. All Release Types as listed in B and C above.	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}
E. Incinerated Oil[h]	Each batch[i] Grab Sample	Each batch[i]	Principal Gamma Emitters[g]	5×10^{-7}

TABLE 4.11-2 (Continued)

TABLE NOTATION

The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a particular measurement system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance.

In calculating the LLD for a radionuclide determine by gamma ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of the measurement system and not as a posteriori (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:
(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a 1-hour period.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

h. Incinerated oil may be discharged at points other than the plant vent stack. Release shall be accounted for based on pre-release grab sample data.

i. Samples for incinerated oil releases shall be collected from representative samples of filtered oil in liquid form.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.2.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND TRITIUM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from tritium, radioiodines and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

c. Less than 0.1% of the limits of 3.11.2.3(a) and (b) as a result of burning contaminated oil.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of tritium, radioiodines, and radioactive materials in particulate form, with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken to reduce releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.3.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the GASEOUS RADWASTE TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site (see Figure 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

    1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.2.4.2 The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 15 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

*These doses are per reactor unit.

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

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-362

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. NPF-15

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

   A. The application 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 April 27 and August 29, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 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 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. 41, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The change in Technical Specifications is to become effective within 30 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensees shall adhere to the Technical Specifications existing at the time. The period of time during change over shall be minimized.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harry Rood

Harry Rood, Senior Project Manager
PWR Project Directorate No. 7
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 20, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 41

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 Page	Overleaf Page
3/4 11-9	3/4 11-10
3/4 11-11	3/4 11-12
3/4 11-13	3/4 11-14

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	$1x10^{-4}$
B. Containment Purge 42 inch	P Each Purge[b,c]	P Each Purge[b]	Principal Gamma Emitters[g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
8 inch	M[b] Grab Sample	M[b]	Principal Gamma Emmitters[g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. 1. Condenser Evacuation System	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	$1x10^{-4}$
2. Plant Vent Stack	W[b,e]	W[b]	H-3	$1x10^{-6}$
D. All Release Types as listed in B and C above.	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	$1x10^{-12}$
			I-133	$1x10^{-10}$
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	$1x10^{-11}$
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$
E. Incinerated Oil[h]	Each batch[i] Grab Sample	Each batch[i]	Principal Gamma Emitters[g]	$5x10^{-7}$

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a particular measurement system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance.

In calculating the LLD for a radionuclide determined by gamma ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of the measurement system and not as a posteriori (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:
(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968)
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

h. Incinerated oil may be discharged at points other than the plant vent stack. Release shall be accounted for based on pre-release grab sample data.

i. Samples for incinerated oil releases shall be collected from representative samples of filtered oil in liquid form.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.2.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND TRITIUM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from tritium, radioiodines and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

c. Less than 0.1% of the limits of 3.11.2.3(a) and (b) as a result of burning contaminated oil.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of tritium, radioiodines, and radioactive materials in particulate form, with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken to reduce releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.3.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the GASEOUS RADWASTE TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site (see Figure 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

    1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.2.4.2 The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 15 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

*These doses are per reactor unit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 52 TO NPF-10 AND AMENDMENT NO. 41 TO NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DOCKET NOS. 50-361 AND 50-362

1.0 INTRODUCTION

Southern California Edison Company (SCE), on behalf of itself and the other licenses, San Diego Gas and Electric Company, The City of Riverside California, and The City of Anaheim, California, has submitted several applications for license amendments for San Onofre Nuclear Generating Station, Units 2 and 3. One such request, Proposed Change PCN-111, is evaluated herein. This change would revise Technical Specification (TS) 3/4.11.2, "Gaseous Effluents," to allow incineration of radioactively contaminated oil.

2.0 DESCRIPTION OF CHANGE

The amendments revise Table 4.11-2, "Radioactive Gaseous Waste Sampling and Analysis Program" of Technical Specification 3/4.11.2, "Gaseous Effluents," and Technical Specification 3.11.2.3, "Dose-Radioiodines, Radioactive Materials in Particulate Form and Tritium," to allow disposal of radioactively contaminated reactor coolant pump (RCP) motor oil, turbine building sump and other waste oil by incineration.

TS 3/4.11.2 provides the maximum dose rates at which radioactive gaseous effluents may be released into the environment. Table 4.11-2 lists the different types of radioactive gaseous releases and specifies sampling and analysis requirements to verify that dose rates are within the limit.

The amendments revise Table 4.11-2 to reflect incineration of oil as a release type and specify sampling and analysis requirements which must be met prior to incineration in order to verify that the dose limit will not be exceeded.

TS 3.11.2.3 specifies limits of dose which an individual may receive due to radioiodines, radioactive materials in particulate form and tritium release from the plant in any calendar quarter and calendar year. The amendments revise TS 3.11.2.3 to limit the dose contribution resulting from the incineration of oil to less than 0.1% of the specified dose limits for radioiodines, particulates, and tritium.

3.0 EVALUATION OF CHANGE

The NRC staff has evaluated the change and has concluded that it is acceptable because the incineration of contaminated oil will be in compliance with 10 CFR 20 Appendix B, Table II, Column I, and will meet the dose objectives of 10 CFR 50 Appendix I. Specifically, concentrations of any radioactivity leaving the Station will be calculated and documented per methods in the Offsite Dose Calculation Manual. The potential dose that could occur as a result of the incineration of contaminated oil has been calculated. The highest radioactivity concentration would probably be less than 4.6 uCi per drum. This value is based on the determination of the highest concentration in reactor coolant pump oil from an 860 MwE net unit owned by another utility (0.2 uCi Co-58, 0.3 uCi Co-60, 1.0 uCi Cs-134, and 3.1 uCi Cs-137 per drum). Assuming this worst case concentration were the average for all 1000 gallons incinerated per year at San Onofre, the dose to any organ of the maximum exposed individual (a child at the nearest residence located 1.3 miles NNW of the plant) was calculated to be 0.001 mrem/yr based on a X/Q of 1.2 E-6 sec/m^2 and a D/Q of 4.5 E-9 m^{-2}. This dose is 0.01% of the limit in TS 3/4.11.2.3 and is considered to be an insignificant contribution to dose via this pathway. Revised Technical Specification 3/4.11.2 will require calculations of dose associated with the incineration of each barrel, and will limit the accumulated dose during a calendar quarter or calendar year to less than 1% of 10 CFR 50, Appendix I limiting dose objectives. This is an appropriate small fraction of such limits for this source and is considered to be As Low As Is Reasonably Achievable. In addition to the above considerations, the equipment used to incinerate waste oil will not be interconnected with or in the immediate vicinity of safety-related systems, and thus will not have an impact on previously evaluated accidents.

In summary, because the results of the change meet all applicable regulatory requirements, and because the revised technical specifications are consistent with NUREG-0472, Revision 2, the Standard Radiological Effluent Technical Specifications for PWRs, the staff finds the amendments to be acceptable.

4.0 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 determination of no significant hazards consideration. No comments were received.

5.0 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 cummulative occupation radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration, and there has been no public comment on such findings. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Sec. 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

The Staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable, and are hereby incorporated into the San Onofre 2 and 3 Technical Specifications.

Dated: August 20, 1986

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 42-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The design of the 8-inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations."

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

The containment spray system and the containment cooling system are redundant to each other in providing post accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 RECIRCULATION FLOW PH CONTROL SYSTEM

The OPERABILITY of the recirculation flow pH control system ensures that there is sufficient trisodium phosphate available in containment to guarantee a sump pH of ≥ 7.0 during the recirculation phase of a postulated LOCA. This pH level is required to minimize the potential for chloride stress corrosion of austentitic stainless steel. The specified amount of TSP will result in a recirculation phase pH of 7.2 assuming complete dissolution and maximum allowed boric acid concentrations from the borated water sources. Similarly, surveillance 4.6.2.2 will produce a pH of 7.2. The specified temperature of 120 ± 10 degrees-F for the surveillance is based is consistent with expected long term recirculation phase sump temperature reported in the FSAR.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The containment cooling system and the containment spray system are redundant to each other in providing post-accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the containment cooling system have been appropriately adjusted. However, the allowable out-of-service time requirements for the containment spray system have been maintained consistent with that assigned other inoperable ESF equipment since the containment spray system also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Containment isolation within the time limits specified for those power operated isolation valves designed to close automatically upon a CIAS signal ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Such valves are listed in Sections A and B of Table 3.6-1 and Surveillance requirements to verify OPERABILITY of these valves are explicitly stated in 4.6.3.1 thru 4.6.3.3. Check valves located inside containment are considered OPERABLE provided their leak rate is within limits when tested pursuant to 10 CFR 50 Appendix J.

Section C of Table 3.6-1 contains a listing of manual valves that are normally closed and assumed to be closed under design basis accident conditions, but which may be opened intermittently for service, maintenance or test during normal operation provided adequate administrative controls are implemented to ensure operator action is taken to close such valves in the event of an accident.