August 14, 1987

Docket Nos.: 50-361
and 50-362

Mr. Kenneth P. Baskin
Vice President
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Mr. James C. Holcombe
Vice President - Power Supply
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Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NPF-10 AND AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NPF-15 SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3 (TAC NOS. 49463, 51663, 54702, 54703, 60720 AND 60721)

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 60 to Facility Operating License No. NPF-10 and Amendment No. 49 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 the operating licenses by deleting License Condition 2.C(5) relating to environmental qualification of electrical equipment. The amendments also revise Technical Specification 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation;" 3/4.3.3.6, "Accident Monitoring Instrumentation;" and 3/4.7.1.5, "Main Steam Isolation Valves."

These amendments were requested by your letters of January 25, 1984; April 19, 1985; July 1, 1985; October 25, 1985; February 7, 1986, and September 6, 1986. The amendments cover Proposed Change Numbers PCN-84, PCN-183 and PCN-207.

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

Sincerely,

Original signed by:
Harry Rood

Harry Rood, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V & Special Projects

Enclosures:
1. Amendment No. 60 to NPF-10
2. Amendment No. 49 to NPF-15
3. Safety Evaluation

cc: See next page
* See Previous Concurrence

*DRSP/PD5	DRSP/PD5	OGC-Bethesda	DRSP/D:PD5
JLee	HRood:cd		GWKnighton
6/10/87	6/3/87	7/23/87	8/12/87

EAK

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ISSUANCE OF AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NPF-10
AND AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NPF-15
SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

DISTRIBUTION
~~Docket File~~ 50-361/362
NRC PDR
Local PDR
PRC System
NSIC
PD5 Plant File
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

DOCKET NO. 50-361

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 60
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 January 25, 1984, April 19, 1985, July 1, 1985, October 25, 1985, and February 7, 1986, as supplemented by letter dated September 6, 1986, 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 amendment and Paragraph 2.C(2) of Facility Operating License No. NPF-10 is hereby amended to read as follows:

    (2) <u>Technical Specifications</u>

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 60 , are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. In addition, the license is amended by changing Paragraph 2.C(5) of Facility Operating License NPF-10, which is hereby amended to read as follows:

    (5) <u>Environmental Qualification</u>

    This paragraph intentionally deleted.

4. The change to Paragraph 2.C(5) of the license is effective as of the date of issuance of this amendment. The changes in Technical Specifications are 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 changeover shall be minimized.

FOR THE NUCLEAR REGULATORY COMMISSION

for EA Licitra
George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III,
VI, V & Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: August 14, 1987

August 14, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 60

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 3-29	3/4 3-30
3/4 3-53	-
3/4 3-53a	-
3/4 3-55	3/4 3-56
3/4 7-9	3/4 7-10
B 3/4 3-3	-
B 3/4 3-4	-
B 3/4 3-5	-

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	8.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown and Sample Isolation (HV8419, HV8421) (HV4053, HV4054, HV4057, HV4058)	20.9
(4) Auxiliary Feedwater Isolation (NOTE 7) (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
(2) ECCS Miniflow Isolation Valves Close	50.7* (Note 8)
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*/52.7**
(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*/52.7**
(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)

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
15. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
16. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to A.C. motor-operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-actuated valves except MSIVs, MFIVs, and CCW Valves 2HV-6211, 2HV-6216, 2HV-6223 and 2HV-6236.

4a. CCW noncritical loop isolation Valves 2HV-6212, 2HV-6213, 2HV-6218, and 2HV-6219 close.

4b. Containment emergency cooler CCW isolation Valves 2HV-6366, 2HV-6367, 2HV-6368, 2HV-6369, 2HV-6370, 2HV-6371, 2HV-6372, and 2HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.
7. Include HV4762 and HV4763 following implementation of DCP 195J.
8. Prior to completion of DCP 6234, valve closure is manually initiated. Following completion of DCP 6234, valves are to close automatically on a RAS coincident with a high-high containment sump signal.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17. Containment Water Level - Wide Range	2	1	20, 21
18. Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19. Cold Leg HPSI Flow	1/cold leg	N.A.	20
20. Hot Leg HPSI Flow	1/hot leg	N.A.	20
21. Heated Junction Thermocouple System-Reactor Vessel Level Monitoring System*	2	1	22, 23

NOTES:

* A channel is eight sensors in a probe. A channel is OPERABLE if four or more sensors, one sensor in the upper head and 3 sensors in the lower head, are OPERABLE.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 22 - With the number of OPERABLE Channels one less than the Required Number of Channels, either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 23 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore both channels of the system to OPERABLE status at the next scheduled refueling.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS (CONTINUED)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
19. Cold Leg HPSI Flow	M	R
20. Hot Leg HPSI Flow	M	R
21. Heated Junction Thermocouple System-Reactor Vessel Level Monitoring System	M	R

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated ~~OPERABLE~~ at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

4.3.3.7.4 Following a seismic event (basemat acceleration greater than or equal to 0.05 g):

a. Within 2 hours each zone shown in Table 3.3-11 shall be inspected for fires, and

b. Within 72 hours an engineering evaluation shall be performed to verify the OPERABILITY of the fire detection system in each zone shown in Table 3.3-11.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 8.0 seconds when tested pursuant to Specification 4.0.5.

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the primary and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generators shall be determined to be less than 200 psig at least once per hour when the temperature of either the primary or secondary coolant is less than 70°F.

room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-6. The high range effluent monitors and samplers (RU-142, RU-144 and RU-146) are in Table 3.3-13. The containment hydrogen monitors are in Specification 3/4.6.5.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.

The Subcooled Margin Monitor (SMM), the Heated Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existence of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If only one channel is inoperable, it should be restored to OPERABLE status in a refueling outage as soon as reasonably possible. If both channels are inoperable, both channels shall be restored to OPERABLE status in the nearest refueling outage. In the event that both HJTC channels are inoperable, existing plant instruments and operator training will be used as an alternate method of monitoring the reactor vessel inventory.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within two hours following an earthquake ($\geq$ 0.02g). Since safe shutdown systems are protected by seismic Category I barriers rated at two and three hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

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. 49
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 January 25, 1984, April 19, 1985, July 1, 1985, October 25, 1985, and February 7, 1986, as supplemented by letter dated September 6, 1986, 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 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. 49, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. In addition, the license is amended by changing Paragraph 2.C(5) of Facility Operating License NPF-15, which is hereby amended to read as follows:

   (5) Environmental Qualification

   This paragraph intentionally deleted.

4. The change to Paragraph 2.C(5) of the license is effective as of the date of issuance of this amendment. The changes in Technical Specifications are 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 changeover shall be minimized.

FOR THE NUCLEAR REGULATORY COMMISSION

for Ed Licitra

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III,
VI, V & Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 14, 1987

August 14, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 49

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 3-29	3/4 3-30
3/4 3-53	-
3/4 3-54	-
3/4 3-55	3/4 3-56
3/4 7-10	3/4 7-9
B 3/4 3-3	-
B 3/4 3-4	-
B 3/4 3-5	-

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
a. MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	8.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown and Sample Isolation (HV8419, HV8421) (HV4053, HV4054, HV4057, HV4058)	20.9
(4) Auxiliary Feedwater Isolation (NOTE 7) (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
a. RAS	
(1) Containment Sump Valves Open	50.7*
(2) ECCS Miniflow Isolation Valves Close	50.7* (Note 8)
7. 4.16 kV Emergency Bus Undervoltage	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/52.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
9. Steam Generator Level - Low (and P - High)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/52.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
10. Control Room Ventilation Airborne Radiation	
a. CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
15. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
16. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-Actuated valves except MSIVs and MFIVs and CCW valves 3HV-6211, 3HV-6216, 3HV-6223 and 3HV-6236.

4a. CCW non-critical loop isolation valves 3HV-6212, 3HV-6213, 3HV-6218 and 3HV-6219.

4b. Containment emergency cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372 and 3HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.
7. Include HV4762 and HV4763 following implementation of DCP 195J.
8. Prior to completion of DCP 6234, valve closure is manually initiated. Following completion of DCP 6234, valves are to close automatically on a RAS coincident with a high-high containment sump signal.

* Emergency diesel generator starting delay (10 seconds) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 seconds) is included.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17. Containment Water Level - Wide Range	2	1	20, 21
18. Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19. Cold Leg HPSI Flow	1/cold leg	N.A.	20
20. Hot Leg HPSI Flow	1/hot leg	N.A.	20
21. Heated Junction Thermocouple System-Reactor Vessel Level Monitoring System*	2	1	22, 23

NOTES:

* A channel is eight sensors in a probe. A channel is OPERABLE if four or more sensors, one sensor in the upper head and 3 sensors in the lower head, are OPERABLE.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 22 - With the number of OPERABLE Channels one less than the Required Number of Channels, either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 23 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;
2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and
3. Restore both channels of the system to OPERABLE status at the next scheduled refueling.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Narrow Range	M	R
2. Containment Pressure - Wide Range	M	R
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
4. Reactor Coolant Inlet Temperature -T_{Cold} (Wide Range)	M	R
5. Pressurizer Pressure (Wide Range)	M	R
6. Pressurizer Water Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Water Level (Wide Range)	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. Safety Valve Position Indicator	M	R
13. Spray System Pressure	M	R
14. LPSI Header Temperature	M	R
15. Containment Temperature	M	R
16. Containment Water Level (Narrow Range)	M	R
17. Containment Water Level (Wide Range)	M	R
18. Core Exit Thermocouples	M	R
19. Cold Leg HPSI Flow	M	R
20. Hot Leg HPSI Flow	M	R
21. Heated Junction Thermocouple System-Reactor Vessel Level Monitoring System	M	R

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TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 8.0 seconds when tested pursuant to Specification 4.0.5.

room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-6. The high range effluent monitors and samplers (RU-142, RU-144 and RU-146) are in Table 3.3-13. The containment hydrogen monitors are in Specification 3/4.6.5.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.)

The Subcooled Margin Monitor (SMM), the Heated Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existence of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If only one channel is inoperable, it should be restored to OPERABLE status in a refueling outage as soon as reasonably possible. If both channels are inoperable, both channels shall be restored to OPERABLE status in the nearest refueling outage. In the event that both HJTC channels are inoperable, existing plant instruments and operator training will be used as an alternate method of monitoring the reactor vessel inventory.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within two hours following an earthquake ($\geq$0.02g). Since safe shutdown systems are protected by seismic Category I barriers rated at two and three hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NPF-10

AND AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NPF-15

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL.

SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 & 3

DOCKET NOS. 50-361 AND 50-362

1.0 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 a number of applications for license amendments for San Onofre Nuclear Generating Station (SONGS), Units 2 and 3. The NRC staff's evaluation of three of these applications is described below.

2.0 DISCUSSION

A. PCN-84

This proposed change deletes License Condition 2.C(5) of the San Onofre Nuclear Generating Station, Units 2 and 3 Operating Licenses, NPF-10 and NPF-15, respectively. These license conditions defined the requirements for the environmental qualification program for safety related electrical equipment. On February 23, 1983, the Commission issued a new regulation, 10 CFR 50.49, which defines the requirements for environmental qualification of safety related electrical equipment and supersedes the previously issued SONGS 2 and 3 license conditions.

Specifically, License Condition 2.C(5)a requires that SCE be in compliance with the provisions of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Equipment," Revision 1 dated July 1981. 10 CFR 50.49 encompasses NUREG-0588 and states that equipment qualified in accordance with NUREG-0588 need not be requalified in accordance with the provisions of 10 CFR 50.49. However, replacement equipment must be qualified under the provisions of 10 CFR 50.49. License Condition 2.C(5)b requires that complete auditable records, which describe the environmental qualification status of all safety related electrical equipment, be available and maintained and that such records be updated and maintained current as equipment is replaced or further tested. 10 CFR 50.49 encompasses the record keeping requirements of this license condition. License Condition 2.C(5)c requires implementation of an environmental qualification maintenance procedures program. Additionally, License Condition 2.C(5)d requires the implementation of an improved environmental qualification surveillance program to detect age-related degradation and any improved maintenance program procedures required by such a surveillance program. 10 CFR 50.49 encompasses these requirements by requiring that qualified equipment will meet its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function, up to the end of its qualified life, and requires that replacement equipment must also be qualified. By letters dated August 23, 1983, and April 10, 1985, SCE affirmed that maintenance program procedures had been implemented and that an improved surveillance program had been implemented.

B. PCN-183

This proposed change revises Technical Specification 3/4.3.3.6 "Accident Monitoring Instrumentation." Technical Specification 3/4.3.3.6 defines types of accident monitoring instrumentation, operability requirements, number of required channels to be operable, actions to be taken in the event that the operability requirements are not met, and periodic surveillance testing to verify operability. The operability of post accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The proposed change adds an additional type of accident monitoring instrumentation subject to these requirements. Specifically, the proposed change adds the reactor vessel level monitoring system (RVLMS) to the technical specifications. The proposed change reflects the addition of the heated junction thermocouple (HJTC) system-reactor vessel level monitoring system.

Two channels of the HJTC system are required, one of which must be operable as a minimum. Each channel includes eight sensors in an HJTC probe. A channel is considered to be operable if four or more sensors (one in the reactor vessel upper head region and three sensors in the lower head region) are operable. Should these minimum operability requirements not be met, the proposed change defines actions to be taken. If one channel is inoperable, the proposed change requires that channel to be restored to operable status within seven days if repairs are feasible without shutting down the reactor, or a special report be submitted to the Commission within the following 30 days which outlines the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status. If both channels are inoperable, the proposed change will require that one or both of the channels be restored to operable status within 48 hours if repairs are feasible without shutting down the reactor, or that an alternate means of monitoring reactor vessel inventory be initiated, a special report be submitted outlining actions taken, the cause of the inoperability and plans and schedule for restoring the system to operable status, and that both channels be restored to operable status at the next scheduled refueling outage. In addition, to verify operability of the system, the proposed change requires monthly channel checks and channel calibrations to be performed at refueling outage intervals.

C. PCN-207

This proposed change revises Technical Specification (TS) 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation (ESFAS), and 3/4.7.1.5, "Main Steam Isolation Valves (MSIV's)." TS 3/4.3.2 specifies the number of channels and type of ESFAS instrumentation required to be operable, response times and periodic surveillance tests to verify operability, and actions to be taken when the minimum operability requirements are not met. TS 3/4.7.1.5 defines operability requirements for MSIV's and actions to be taken when one or both MSIV's are inoperable. The operability requirements for the main steam isolation valves ensure that no more than one steam generator will blow down in the event of a main steam line rupture assuming a single failure. Ensuring that only one steam generator blows down prevents the containment design pressure from being exceeded and limits positive reactivity addition due to cool down of the reactor coolant system.

During normal plant operation, the MSIVs are maintained open by hydraulic pressure working against compressed nitrogen gas. The energy stored in the compressed gas provides the motive force for valve closure. Technical Specification 3/4.3.2 currently requires an MSIV closure time of 6.0 seconds*. The pressure required to maintain the valve open and provide 6.0 second response time is high. Dynamic effects on components in the MSIV hydraulic circuits, due in part to the high pressures, have resulted in component failures and spurious MSIV closures during plant operation. A spurious MSIV closure during power operation will result in a reactor trip. Reducing the MSIV operating pressure will result in increased component reliability but will also result in a slower MSIV response time.

The proposed change would increase MSIV closure time from 6.0 to 8.0 seconds. Specifically, the response time listed for the MSIV's in Table 3.3-5, "ESFAS Response Times", under Main Steam Isolation Signal (MSIS) is increased from 6.9 to 8.9 seconds (0.9 seconds is allowed for instrumentation response time, the remainder for the valve). Also, the response time for the MSIV's listed in TS 3/4.7.1.5 is increased from 6.0 to 8.0 seconds.

* At the time PCN-207 was proposed, the required MSIV closure time was 5.0 seconds. Since then, Amendments 46 and 35 to NPF 10 and NPF-15, respectively, dated May 16, 1986, changed the time to 6.0 seconds.

3.0 EVALUATION

A. PCN-84

The NRC staff has evaluated the proposed change and finds it acceptable on the basis that 10 CFR 50.49, which was issued after the SONGS 2 and 3 operating licenses were issued, defines the current NRC requirements for an acceptable environmental qualification program. Specifically, this rule identifies record keeping requirements, implementation schedule, and environmental qualification criteria. The rule supersedes previous staff criteria for environmental qualification such as those contained in NUREG-0588. License Conditions 2.C(5)a and 2.C(5)b identify schedule, environmental qualification criteria and record keeping requirements. Because these requirements have been superseded by 10 CFR 50.49, the proposed deletion of License Condition 2.C(5)a and 2.C(5)b makes the licenses conform to changes in the regulations and is therefore acceptable.

In addition, the Standard Review Plan (SRP) Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," describes the general requirements for the design and environmental qualification of all equipment. The specific acceptance criteria for assessing the acceptability of an environmental qualification program for safety related electrical equipment is provided by reference to NUREG-0588. 10 CFR 50.49 has superseded NUREG-0588 and thus specific SRP acceptance criteria are now found in 10 CFR 50.49. The proposed change deletes License Conditions 2.C(5)c and 2.C(5)d which define the requirements for an environmental qualification maintenance and surveillance program and requires the affirmation of implementation of the improved surveillance and maintenance procedures. 10 CFR 50.49 requires that safety related electrical equipment remain qualified for its qualified life and that replacement equipment also be qualified. These two aspects of 10 CFR 50.49, while not specifically identifying the means of accomplishing these requirements, will achieve the same goal as License Conditions 2.C(5)c and 2.C(5)d; thus, while deletion of these License Conditions removes the specificity of how equipment will be maintained in a qualified condition for its life, the same ultimate requirement is mandated by 10 CFR 50.49. Since the requirements of 10 CFR 50.49 must be met and since they define an acceptable environmental qualification program, the environmental qualification program will continue to meet the acceptance criteria even with the deletion of License Conditions 2.C(5)c and 2.C(5)d. Based on the above, the NRC staff finds the proposed deletion of License Condition 2.C(5) to be acceptable.

B. PCN-183

The NRC staff guidance for this proposed change are given in Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling," of Generic Letter 83-37, "NUREG-0737 Technical Specifications."

The generic letter contains the following statements:

> Subcooling margin monitors, core exit thermocouples, and a reactor coolant inventory tracking system (e.g., the differential pressure measurement system designed by Westinghouse, the Heated Junction Thermocouple system designed by Combustion Engineering, etc.) may be used to provide indication of the approach to, existence of, and recovery from inadequate core cooling (ICC). This instrumentation should be operable during the Power Operation, Startup, and Hot Shutdown modes of operation for each reactor.
>
> Subcooling margin monitors should have already been included in the present Technical Specifications. Technical Specifications for core exit thermocouples and the reactor coolant inventory tracking system should be included with other accident monitoring instrumentation in the present Technical Specifications. Four core-exit thermocouples in each core quadrant and two channels in the reactor coolant tracking system are required to be operable when the reactor is operating in any of the above mentioned modes. A minimum of two core-exit thermocouples in each quadrant and one channel in the reactor coolant tracking system should be operable at all times when the reactor is operating in any of the above mentioned modes. Typical acceptable LCO and surveillance requirements for accident monitoring instrumentation are provided in Enclosure 3.

The licensee responded to the generic letter by proposing Technical Specification 3/4.3.3.6, which added requirements for a HJTC system and a RVLMS to the technical specifications.

The NRC staff has evaluated the proposed technical specification and finds that only the Actions part of the proposed technical specification deviates from the guidance in the generic letter. Specifically the proposed technical specification permits continued operation (after 7 days operation with one less than the required number of channels or after 48 hours operation with one less than the minimum number of channels) provided a special report is submitted to the Commission pursuant to Technical Specification 6.9.2 within 30 days following the event and outlining the cause of the inoperability and plans and schedule for restoring the system to operable status. Generic Letter 83-37 recommends going into the hot shutdown mode if repairs cannot be made after the 7-day or 48-hour period. Although deviating from the generic letter guidance, SCE's Actions for the inoperability of the HJTC and clarification of operability (four or more sensors operable, one sensor in the upper head and three sensor in the lower head) are consistent with the NRC Staff's approval of the Combustion Engineering Owners Group proposed technical specifications for the HJTC system.

For the subcooled margin meter, the proposed technical specifications comply with the guidance in the generic letter in all aspects (Limiting Conditions for Operations, Applicability, Actions and Surveillance Requirements).

The proposed technical specifications for the core-exit thermocouples (TC) exceed the criteria in the guidance for "required number of channels" (seven TCs in the core quadrant versus the guidance of four TCs in the core quadrant), and for "minimum channels operable" (four TCs in the core quadrant versus the guidance of two TCs in the core quadrant). In all other aspects the proposed technical specifications for core-exit thermocouples comply with the generic letter's guidance.

As a result of the review of the cited material, the licensee's response for technical specifications for Item II.F.2, inadequate core cooling instrumentation, is judged to meet the guidance in the generic letter (as approved by the NRC staff for the Combustion Engineering owners group generic technical specification), and is, therefore, acceptable.

C. PCN-207

The NRC staff has evaluated the two principal aspects of the proposed change. These are (1) the effect of the proposed change on the plant safety analyses described in Section 15 of the FSAR, and (2) the effect of the proposed change on peak containment pressure.

With regard to the effect of the proposed change on the plant safety analyses, the worst cases (steam line breaks inside and outside containment) were reanalyzed and found acceptable by the licensee as part of their Cycle 3 reload analysis. This analysis, which conservatively assumed a 10 second MSIV closure time (rather than the proposed 8 seconds) was reviewed and approved by the NRC staff as part of Amendment 47 to NPF-10 and Amendment 36 to NPF-15. These amendments were issued on May 16, 1986, and included a safety evaluation describing the basis for the staff's approval.

With regard to the effect of the proposed change in peak containment pressure, the licensee submitted a reanalysis of containment pressure following a main steam line break (MSLB) based on an 8 second MSIV closure time. The staff compared this analysis with the analysis in the FSAR, which is based on 5 second MSIV closure time. The reanalysis shows that while the peak pressure and temperature within containment are reached earlier in the accident, the peak containment pressure is still bounded by the value of 55.7 psig given in the FSAR analysis.

This is due to the fact that the licensee has made several modifications in the modeling of the MSLB event and valve response. The changes in methodology mainly involved more realistic and detailed (but still conservative) modeling than did the FSAR analysis. The licensee states that "Changes were made to the original FSAR methodology to accommodate longer MSIV response times and achieve acceptable results in the safety Analysis."

In the amendment request and in the licensee's September 6, 1986 response to a staff request for additional information (RAI), the licensee has enumerated the modifications made to the analytical model:

(1) For the revised analysis, the MSIV flow area is assumed not to change during the first second of closure duration. The flow area is then linearly decreased to zero over the remainder of the closure duration.

(2) The rapid depressurization of the ruptured steam generator causes more than 50% of the total feedwater flow to be diverted to this unit. The original analysis for the FSAR assumed that 100% of the total feedwater flow is diverted to this unit. However, subsequent analysis indicated that at full power only 65% of the total flow would be diverted to this unit. The revised analysis assumes that 87.5% of the total flow is diverted to the ruptured unit.

(3) The steam lines were modeled as two separate nodes associated with each steam generator instead of being combined with the steam generator node.

(4) Three separate flow resistances (from each steam generator, to the cross tie, and for the cross tie itself) were used instead of a single, combined flow resistance.

(5) The Darcy equation, used to calculate steam-line flow was modified to account for compressibility.

(6) Choking at various points in the steam lines was considered when appropriate conditions existed.

The staff has reviewed these modifications and found them to be both realistic and conservative. In particular, the changes outlined in (1) and (2) above would have the indicated physical effect of lowering the mass/energy release to the containment. According to the submittal, the peak pressure reported to the FSAR is calculated using the code COPATTA, and is based on mass and energy releases determined by the code SGNIII/CONTRANS. On this basis the NRC staff accepts the calculated results as realistic and conservative.

An additional factor outlined in the responses to the RAI adds to the overall conservatism of the revised analytical model. The need to increase the response time became apparent after licensing San Onofre Units 2 and 3 when there appeared to be little margin between the five second technical specification response times and actual measured response times in the field. Initially, the licensee proposed an increase in MSIV response time from five to six seconds in PCN-96. This response time increase was approved by the NRC staff on May 16, 1986, in Amendments 47 and 36 to the San Onofre 2 and 3 licenses.

This increase in response time was justified using the original FSAR methodology with actual valve flow characteristics instead of those assumed in the original analysis. Because of difficulties with the original valves delineated in the responses to the RAI, valves manufactured by Paul Monroe Hydraulics were used to replace the original Marotta dump valves. It was anticipated that the MSIV response time would increase with these modifications. The NSSS vendor, Combustion Engineering, was requested to reanalyze the main steam line break events to support longer valve closure times. This action was carried out in parallel with the detailed design of the modifications. As it turned out, when modified, the MSIVs close in less than six seconds, the PCN-96 approved response time. This adds to the conservatism of the model assuming the eight second closure time.

Additional conservative assumptions made in both the original analysis and in the revised submittal remain, thus adding further to the overall conservatism of the analytical model.

In summary, the licensee has submitted a proposed change to the technical specifications which increases the required closure time for the MSIVs. A modified form of the model used in the original FSAR analysis has been utilized to demonstrate that no NRC safety requirements are compromised. The NRC staff supports both the realism and the conservatism of the revised model based on the codes used to calculate mass/energy release (SGNIII/CONTRANS) to containment and peak containment pressure (COPPATTA). The staff finds that the information submitted fully supports the amendment request, and the request is, therefore, 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 determinations 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, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumu-occupation radiation exposure. The Commission has previously issued proposed findings that the amendments involve no significant hazards consideration, and there have 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

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 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.

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