April 11, 1989

Docket No. 50-361

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

Mr. Gary D. Cotton
Senior Vice President
Engineering and Operations
San Diego Gas & Electric Company
101 Ash Street
P.O. Box 1831
San Diego, California 92112

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. NPF-10 SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2 (TAC NOS. 71174 AND 71175)

The Commission has issued the subject amendment to Facility Operating License NPF-10 for San Onofre Nuclear Generating Station, Unit 2. The amendment consists of changes to the Technical Specifications in response to your application dated November 7, 1988 and your letters dated December 29, 1988 and February 23, 1989. This request was designated by you as PCN-278.

The amendment revises Technical Specifications 3/4.4.8.1, "Pressure/Temperature Limits;" 3.4.1.4.1, "Cold Shutdown-Loops Filled;" 3.4.1.3, "Hot Shutdown;" 3.4.8.3.1, "Overpressure Protection System, RCS Temperature less than or equal to 235°F;" and 3.4.8.3.2, "Overpressure Protection System, RCS Temperature greater than 235°F."

Copies of our related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

original signed by

Donald E. Hickman, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 70 to License No. NPF-10
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	BGrimes
NRC & LPDRs	TMeek (8)
PD5 Plant File	EButcher
MVirgilio	ACRS (10)
JLee	GPA/PA
OGC-WF	ARM/LFMB
DHagan	WJones
EJordan	DHickman

As marked

*SEE PREVIOUS CONCURRENCE.

DRSP/PD5	DRSP/PD5	EMTB	SRXB	OGC	DRSP/D:PD5
JLee	DHickman:rw	CYCheng	MWHodges	*JMoore	GKnighton
4/10/89	4/10/89	4/6/89	3/31/89	4/5/89	4/11/89

8904210330 890411
PDR ADOCK 05000361
P PDC

DF01
1/1

Docket No. 50-361

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

Mr. Gary D. Cotton
Senior Vice President
Engineering and Operations
San Diego Gas & Electric Company
101 Ash Street
P.O. Box 1831
San Diego, California 92112

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. TO FACILITY OPERATING LICENSE NO. NPF-10 SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2 (TAC NOS. 71174 AND 71175)

The Commission has issued the subject amendment to Facility Operating License NPF-10 for San Onofre Nuclear Generating Station, Unit 2. The amendment consists of changes to the Technical Specifications in response to your application dated November 7, 1988 and your letters dated December 29, 1988 and February 23, 1989. This request was designated by you as PCN-278.

The amendment revises Technical Specifications 3/4.4.8.1, "Pressure/Temperature Limits;" 3.4.1.4.1, "Cold Shutdown-Loops Filled;" 3.4.1.3, "Hot Shutdown;" 3.4.8.3.1, "Overpressure Protection System, RCS Temperature less than or equal to 235°F;" and 3.4.8.3.2, "Overpressure Protection System, RCS Temperature greater than 235°F."

Copies of our related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Donald E. Hickman, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. to License No. NPF-10
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	BGrimes
NRC & LPDRs	TMeek (8)
PD5 Plant File	EButcher
MVirgilio	ACRS (10)
JLee	GPA/PA
OGC-WF	ARM/LFMB
DHagan	WJones
EJordan	DHickman

subject to change

DRSP/PD5	DRSP/PD5	EMTB	SRXB	OGC	DRSP/D:PD5
JLee	DHickman:rw	CYCheng	MWHodges		GKnighton
3/31/89	4/10/89	/ /89	/ /89	4/13/89	/ /89

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

April 11, 1989

Docket No. 50-361

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

Mr. Gary D. Cotton
Senior Vice President
Engineering and Operations
San Diego Gas & Electric Company
101 Ash Street
P.O. Box 1831
San Diego, California 92112

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. NPF-10 SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2 (TAC NOS. 71174 AND 71175)

The Commission has issued the subject amendment to Facility Operating License NPF-10 for San Onofre Nuclear Generating Station, Unit 2. The amendment consists of changes to the Technical Specifications in response to your application dated November 7, 1988 and your letters dated December 29, 1988 and February 23, 1989. This request was designated by you as PCN-278.

The amendment revises Technical Specifications 3/4.4.8.1, "Pressure/Temperature Limits;" 3.4.1.4.1, "Cold Shutdown-Loops Filled;" 3.4.1.3, "Hot Shutdown;" 3.4.8.3.1, "Overpressure Protection System, RCS Temperature less than or equal to 235°F;" and 3.4.8.3.2, "Overpressure Protection System, RCS Temperature greater than 235°F."

Copies of our related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Donald E. Hickman

Donald E. Hickman, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 70 to
   License No. NPF-10
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

cc:

Mr. Gary D. Cotton
Senior Vice President
Engineering and Operations
San Diego Gas & Electric Company
101 Ash Street
Post Office Box 1831
San Diego, California 92112

Charles R. Kocher, Esq.
James A. Beoletto, Esq.
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Orrick, Herrington & Sutcliffe
ATTN: David R. Pigott, Esq.
600 Montgomery Street
San Francisco, California 94111

Alan R. Watts, Esq.
Rourke & Woodruff
701 S. Parker St. No. 7000
Orange, California 92668-4702

Mr. S. McClusky
Bechtel Power Corporation
P. O. Box 60860, Terminal Annex
Los Angeles, California 90060

Mr. Charles B. Brinkman
Combustion Engineering, Inc.
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Mr. Dennis F. Kirsh
U.S. Nuclear Regulatory Commission
Region V
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

Mr. Sherwin Harris
Resource Project Manager
Public Utilities Department
City of Riverside
City Hall
3900 Main Street
Riverside, California 92522

Mr. Hans Kaspar, Executive Director
Marine Review Committee, Inc.
531 Encinitas Boulevard, Suite 105
Encinitas, California 92024

Mr. Mark Medford
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Mr. Robert G. Lacy
Manager, Nuclear Department
San Diego Gas & Electric Company
P. O. Box 1831
San Diego, California 92112

Richard J. Wharton, Esq.
University of San Diego School of
Law
Environmental Law Clinic
San Diego, California 92110

Charles E. McClung, Jr., Esq.
Attorney at Law
24012 Calle de la Plaza/Suite 330
Laguna Hills, California 92653

Regional Administrator, Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane/Suite 210
Walnut Creek, California 94596

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, California 95841
ATTN: Ms. Mary Schnell

Mayor, City of San Clemente
San Clemente, California 92672

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

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

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

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 NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 70
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 Southern California Edison Company (SCE) on behalf of itself and San Diego Gas and Electric Company, the City of Riverside, California and the City of Anaheim, California (licensees) dated November 7, 1988, and supplemental submittals dated December 29, 1988 and February 23, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 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. 70, are hereby incorporated in the license. SCE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and must be fully implemented no later than 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 11, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 70

FACILITY OPERATING LICENSE NO. NPF-10

DOCKET NO. 50-361

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change. Also enclosed are the following overleaf pages to the amended pages.

AMENDMENT PAGE	OVERLEAF PAGE
v	vi
xx	xix
xxii	xxi
3/4 4-3	3/4 4-4
3/4 4-5	3/4 4-5a
3/4 4-27	-
3/4 4-27a	-
3/4 4-29	-
3/4 4-30	-
3/4 4-30a	-
3/4 4-32	3/4 4-31
3/4 4-33	3/4 4-34
B 3/4 4-1	B 3/4 4-2
B 3/4 4-6	B 3/4 4-5
B 3/4 4-7	B 3/4 4-8
B 3/4 4-9	-

Page 3/4 4-28 is reissued without change

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF FIGURES

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant pump,**

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4

ACTION:

a. With less than the above required Reactor Coolant loops and/or shutdown cooling trains OPERABLE, immediately initiate corrective action to return the required loops/trains to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling train, be in COLD SHUTDOWN within 24 hours.

b. With no Reactor Coolant loop or shutdown cooling train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop/train to operation.

---

*All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 when the secondary water temperature of each steam generator is greater than 100°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required Reactor Coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% (wide range) at least once per 12 hours.

4.4.1.3.3 At least one Reactor Coolant loop or shutdown cooling train shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 a. At least one of the following loop(s)/trains listed below shall be OPERABLE and in operation*:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant Pump**

2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant Pump**

3. Shutdown Cooling Train A

4. Shutdown Cooling Train B

b. One additional Reactor Coolant Loop/shutdown cooling train shall be OPERABLE, or

c. The secondary side water level of each steam generator shall be greater than 10% (wide range).

APPLICABILITY: MODE 5, with Reactor Coolant loops filled.

ACTION:

a. With less than the above required shutdown trains/loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required trains/loops to OPERABLE status or restore the required level as soon as possible.

b. With no loop/train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop/train to operation.

*All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 when the secondary water temperature of each steam generator is greater than 100°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The required Reactor Cooling pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.4.1.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$10% (wide range) at least once per 12 hours.

4.4.1.4.1.3 At least one Reactor Coolant loop or shutdown cooling train shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 and Figure 3.4-3 during heatup, cooldown, criticality, boltup, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 10°F in any one hour period with RC cold leg temperature less than 112°F. A maximum heatup of 30°F in any one hour period with RC cold leg temperature less than 163°F. A maximum heat-up of 60°F in any one hour period with RC cold leg temperature greater than 163°F.

b. A maximum cooldown of 10°F in any one hour period with RC cold leg temperatures less than 103°F. A maximum cooldown of 30°F in any one hour period with RC cold leg temperatures less than 145°F. A maximum cooldown of 100°F in any one hour period with RC temperature greater than 145°F.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

d. A minimum temperature of 86°F to tension reactor vessel head bolts.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3. Recalculate the Adjusted Reference Temperature based on the greater of the following:

a. The mean value of shift in reference temperature for plates C-6404-3*, or

b. The predicted shift in reference temperature for weld seams 3-203A or 3-203B as determined by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

*The most limiting material in the reactor vessel in accordance with the new Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988, has changed and are plates C-6404-3. Calculative procedures provided in the new guide should be used to obtain the mean values of shift in RT_{NDT} of C-6404-3 plates. Calculations are based on the actual shift in reference temperature as determined by impact testing on the existing plate C-6404-2 surveillance material.

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
1	83°	1.15	Standby
2	97°	1.15	3.2 EFPY
3	104°	1.15	13.6 EFPY
4	284°	1.15	24 EFPY
5	263°	1.15	Standby
6	277°	1.15	Standby

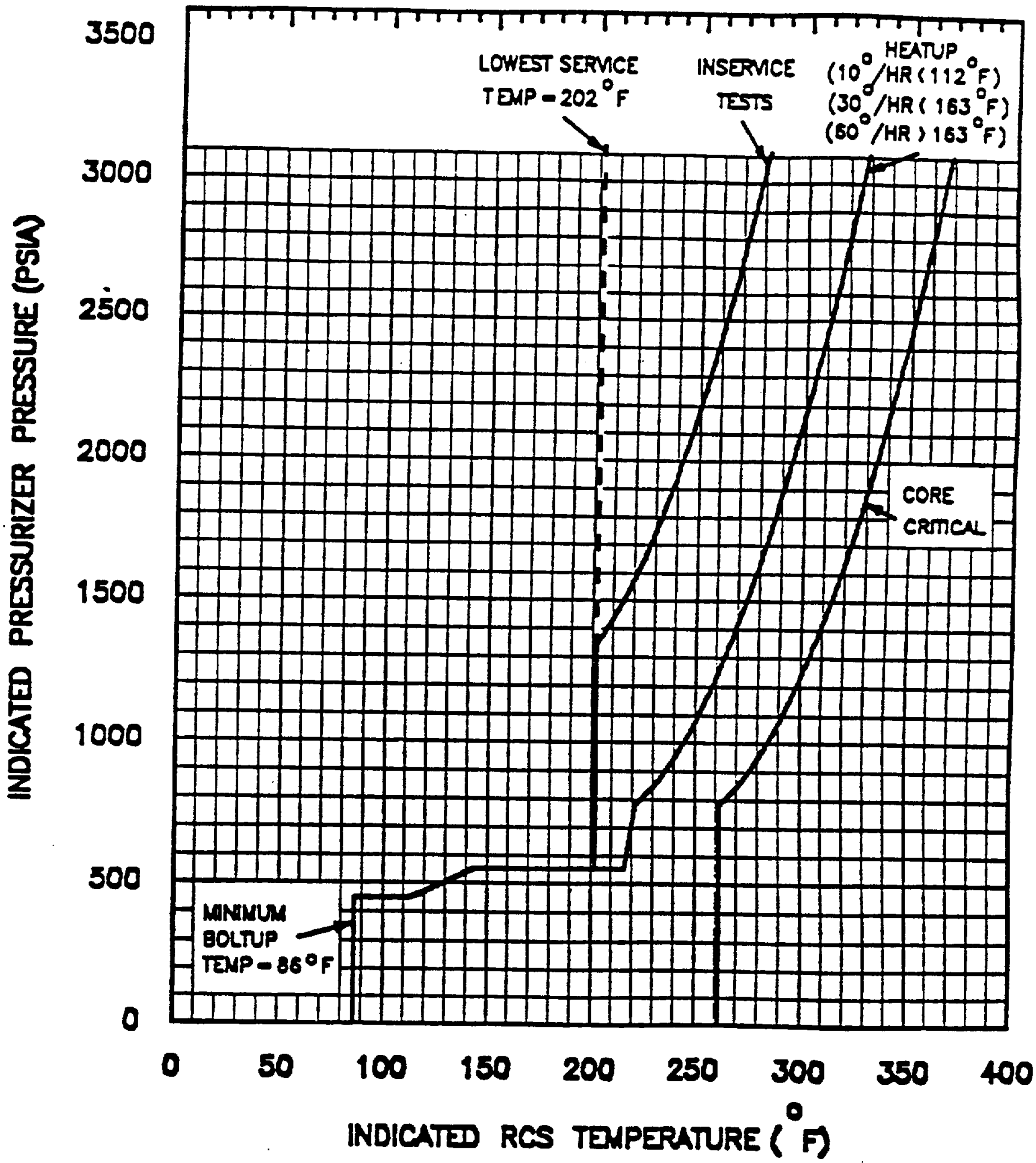

Figure 3.4-2

RCS HEATUP PRESSURE/TEMPERATURE LIMITATIONS FOR 4-8 EFPY

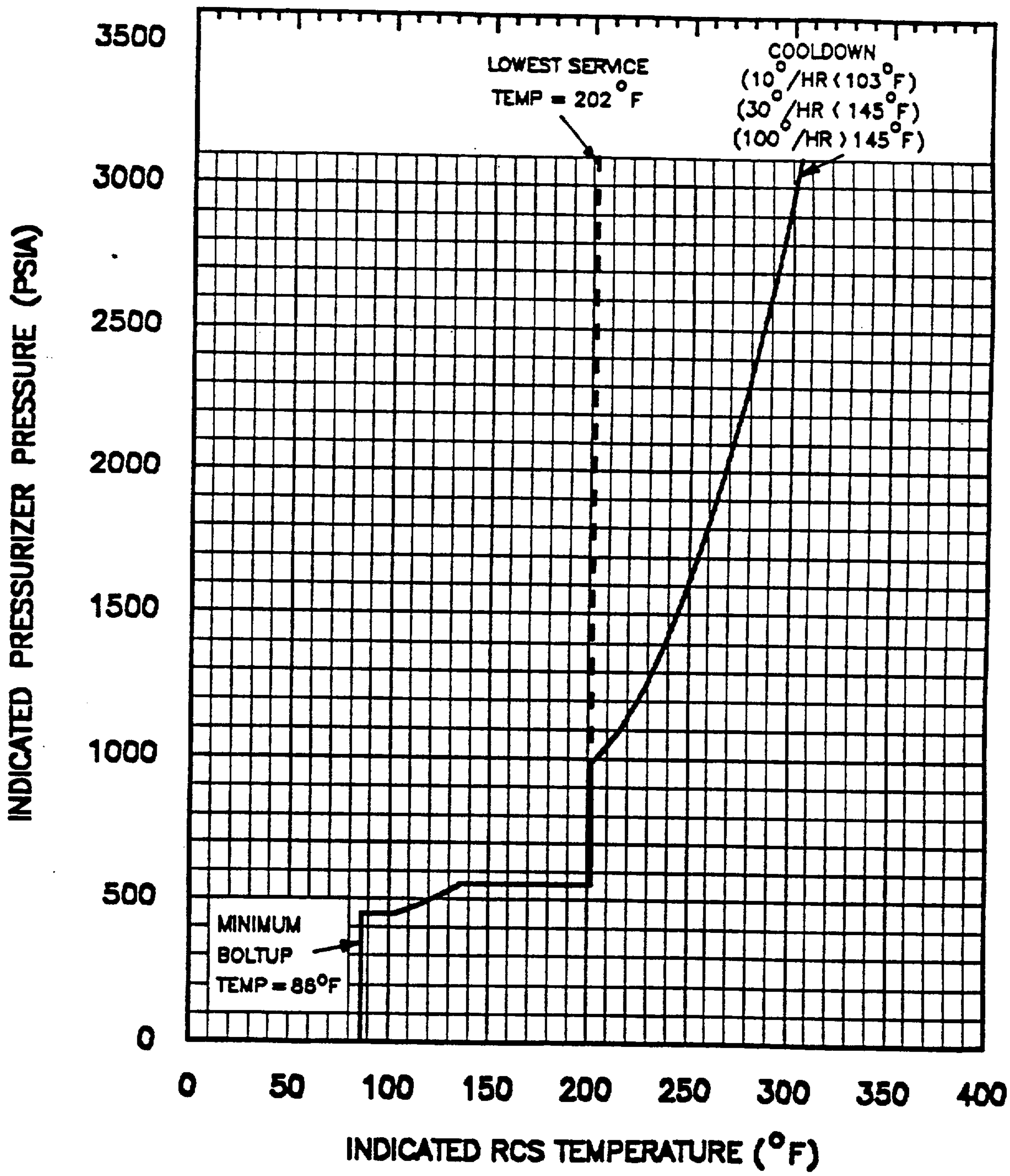

Figure 3.4-3

RCS COOLDOWN PRESSURE/TEMPERATURE LIMITATIONS FOR 4-8 EFPY

Table 3.4-3

Low Temperature RCS Overpressure Protection Range

Operating Period, EFPY	Cold Leg Temperature, °F	
	During Heatup	During Cooldown
4 to 10	$\leq$ 312	$\leq$ 287

PRESSURIZER - HEATUP/COOLDOWN

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer shall be limited to:

a. A maximum heatup of 200°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-1 for each cycle of main spray when less than 4 reactor coolant pumps are operating and for each cycle of auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE ≤ 312°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

   1) A lift setting of 406 ± 10 psig*, and

   2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open, or,

b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

APPLICABILITY: MODE 4 when the temperature of any one RCS cold leg is less than or equal to that specified in Table 3.4-3; MODE 5; MODE 6 with the reactor vessel head on.

ACTION:

a. With the SDCS Relief Valve inoperable, reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.

b. With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) closed, open the closed valve(s) within 7 days or reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.

c. In the event either the SDCS Relief Valve or an RCS vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SDCS Relief Valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours when the SDCS Relief Valve is being used for overpressure protection that SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open.

---

*For valve temperatures less than or equal to 130°F.

RCS TEMPERATURE > 312°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.2 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System Relief Valve (PSV9349) with:

   1) A lift setting of 406 ± 10 psig*, and

   2) Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 open, or,

b. A minimum of one pressurizer code safety valve with a lift setting of 2500 psia ± 1%**.

APPLICABILITY: MODE 4 with RCS temperature above that specified in Table 3.4-3.

ACTION:

a. With no safety or relief valve OPERABLE, be in COLD SHUTDOWN and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.

b. In the event the SDCS Relief Valve or an RCS vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the SDCS Relief Valve code safety valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

SURVEILLANCE REQUIREMENTS

4.4.8.3.2.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours that the SDCS Relief Valve isolation valves 2HV9337, 2HV9339, 2HV9377 and 2HV9378 are open when the SDCS Relief Valve is being used for overpressure protection.

b. Verifying relief valve setpoint at least once per 30 months when tested pursuant to Specification 4.0.5.

4.4.8.3.2.2 The pressurizer code safety valve has no additional surveillance requirements other than those required by Specification 4.0.5.

4.4.8.3.2.3 The RCS vent shall be verified to be open at least once per 12 hours when the vent is being used for overpressure protection, except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

*For valve temperatures less than or equal to 130°F.

**The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR greater than 1.31 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than 4 reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump in Modes 4 and 5 with one or more RCS cold legs less than or equal to that specified in Table 3.4-3 are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than that specified in Table 3.4-3. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to that specified in Table 3.4-3, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

REACTOR COOLANT SYSTEM

BASES

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the San Onofre site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron irradiation damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 60°F/hr or cooldown rate of up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period, and they include adjustments for possible errors in the pressure and temperature sensing instruments.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than I Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper and nickel content of the material in question, can be predicted using FSAR Table 5.2-5 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3, include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel taking into account the location of the sample closer to the core than the vessel wall by means of the Lead Factor. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	@ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb
215-01	C-6403-1	A533GRBCL-1	Upper Shell Plate	40	15	35	130
215-01	C-6403-2	"	"	0	20	25	133
215-01	C-6403-3	"	"	-10	20	45	131
215-03	C-6404-1	"	Intermediate Shell Plate	-30	10	50	145
215-03	C-6404-2	"	"	-20	20	50	155
215-03	C-6404-3	"	"	-20	10	50	131
215-02	C-6404-4	"	Lower Shell Plate	-10	-5	25	124
215-02	C-6404-5	"	"	-20	10	25	134
215-02	C-6404-6	"	"	-10	-20	0	151
238-02	C-6401	A508Cl-2	Vessel Flange Forging	-10	-70	-35	148
209-02	C-6402	"	Closure Head Flange Forging	-10	-90	-40	142
205-02	C-6410-1	"	Inlet Nozzle Forging	20	-40	-35	130
205-02	C-6410-2	"	"	0	-20	-5	135
205-02	C-6410-3	"	"	0	-15	-15	140
205-02	C-6410-4	"	"	0	-65	-50	140
205-06	C-6411-1	"	Outlet Nozzle Forging	-100	-30	-10	140
205-06	C6411-2	"	"	0	-35	-10	140
232-01	C-6424	A533GRBCL-1	Bottom Head Torus	-50	-20	10	122
232-02	C-6425	"	Bottom Head Dome	-50	-30	-20	136
205-03	C-6428-1	A508CL-1	Inlet Nozzle Forging S/E	-30	-70	-50	174
205-03	C-6428-2	"	"	-30	-70	-50	174
205-03	C-6428-3	"	"	-30	-70	-50	174
205-03	C-6428-4	"	"	-30	-70	-50	174
205-07	C-6429-1	"	Outlet Nozzle Ext. Forging	-30	-40	-25	229
205-07	C-6429-1	"	"	-30	-40	-25	229
231-02	C-6430-1	A533GRBCL-1	Closure Head Peels	+10	20	55	118
231-02	C-6431-1	"	"	-20	10	50	100
231-02	C-6432-1	"	"	-10	-15	45	115
231-02	C-6432	"	Closure Head Dome	-10	-15	45	115

PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or a RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to that specified in Table 3.4-3. The Shutdown Cooling System relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and letdown isolated.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM

Reactor coolant system gas vents are provided to exhaust noncondensible gases from the primary system that could inhibit natural circulation core cooling following a non-design bases accident. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The design redundancy of the Reactor Coolant Gas Vent System serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant Gas Vent System are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 70 TO FACILITY OPERATING LICENSE NO. NPF-10

SOUTHERN CALIFORNIA EDISON COMPANY

SAN DIEGO GAS AND ELECTRIC COMPANY

THE CITY OF RIVERSIDE, CALIFORNIA

THE CITY OF ANAHEIM, CALIFORNIA

SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 2

DOCKET NO. 50-361

1.0 INTRODUCTION

By letter dated November 7, 1988 Southern California Edison Company (SCE), et al. (the licensees) requested a change to the Technical Specifications for Facility Operating License No. NPF-10 that authorizes operation of San Onofre Nuclear Generating Station, Unit 2 in San Diego County, California. The amendment would revise Technical Specifications 3/4.4.8.1, "Pressure/Temperature Limits;" 3.4.1.4.1, "Cold Shutdown-Loops Filled;" 3.4.1.3, "Hot Shutdown;" 3.4.8.3.1, "Overpressure Protection System, RCS Temperature less than or equal to 235°F;" and 3.4.8.3.2, "Overpressure Protection System, RCS Temperature greater than 235°F."

These changes would revise the pressure/temperature (P/T) and low temperature overpressure protection (LTOP) limits for operation through 10 effective full power years (EFPY). At the time of the submittal, the material analysis report of the first removed surveillance capsule had not been completed. Subsequently, the Battelle Columbus Laboratory, under contract to SCE, completed the report entitled "Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the San Onofre Nuclear Generating Station Unit 2." This report found that the measured neutron fluence was 40% higher than the theoretical neutron fluence estimated in the November 7 submittal. By letter dated December 29, 1988, SCE submitted the report to the NRC and reduced the requested effective period of the P/T and LTOP limits from 10 EFPY to 8 EFPY. By letter dated February 23, 1989 SCE responded to the NRC's November 30, 1988 request for additional information. The staff has reviewed the SCE submittals. Our evaluation and conclusions are described below.

2.0 NEUTRON SURVEILLANCE

Appendix H to 10 CFR Part 50 requires the licensees to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. SCE removed the first surveillance capsule, which was located at 97-degree azimuth, from SONGS 2 on September 20, 1987 after 2.69 EFPY of operation. The 97-degree capsule was analyzed and tested by the Battelle Columbus Laboratory in cooperation with the Ohio State University.

2.1 Evaluation

The objective of the pressure vessel neutron surveillance is to measure the fast neutron flux (fluence) for energies greater than 1.0 MeV at the tensile specimen locations, to extrapolate these measurements to the pressure vessel, to identify the location of the peak fluence and its spectrum, and to verify all of the measurements with appropriate calculations.

Flux measurement is accomplished by the use of activation dosimeters and is covered by several ASTM procedures. The measurements and analyses in this report conform to the requirements of all pertinent ASTM procedures.

The dosimeter monitors covered a wide range of energy and activity half lives. The monitors used included Al-Co, U, Ti, Fe, S, Ni, and Cu. Some were Cd shielded and encapsulated in stainless steel. The dosimeter activity was measured using a 4096-channel Ge(Li) detector with a full width, half-maximum resolution of 1.9 KeV at 1,332.5 KeV. Spectrum data were analyzed using the SPECTRAN-III computer code. For the calculation of the foil activation the daily reactor operating history was taken into account.

The energy distribution and the neutron flux in the reactor, in the surveillance capsule, and in the pressure vessel were calculated using the DOT 4.3 computer program, which is widely accepted for this type of calculation. The principal approximations included third order P3 scattering, S_8 angular quadrature, 48 azimuthal segments and 47 energy groups. The cross sections were based on the BUGLE-80 library which has been derived from ENDF/B-V data. The DOT was used for R- and R-z runs which provided three dimensional neutron flux values in the locations of interest. The power distributions (neutron source) were obtained for each fuel pin from the results of PDQ depletion calculations. In propagating the neutron field from the core to the pressure vessel, both the boron solution in the coolant (a cycle average value) and the coolant temperatures (densities) in the core and the downcomer were taken into account.

The experimental flux determination, however, requires knowledge of the neutron spectra at the location of the measurement. Calculated neutron spectra were used, thus the flux values derived from the measured dosimeter activities are semi-experimental. However, this is the generally accepted method, provided some reasonable checks on the spectra have been performed. Such check calculations have been performed in this report and several adjustments related to the neutron spectra have been made. These adjustments are based on a spectrum which provides the closest possible agreement with the measured values of the flux from each of the dosimeters.

The methodologies, practices and computer programs outlined represent the generally accepted means for such calculations. They also represent what the staff has been requiring from licensees on neutron flux measurements and calculations for licensing actions related to 10 CFR 50.61. They are therefore acceptable.

2.2 Results

Several dosimeters were found to be oxidized, and for their recovery special cutting and cleaning operations were applied. From a total of 27 monitors only three sulfur monitors were discarded. The standard deviation of the unadjusted spectra was about 12%. The calculated flux (with the adjusted spectra) is about 25% less than the measured flux at the inside surface of the pressure vessel. The value accepted as the final value of the flux is the measured value, which is conservative. In general the final values of the flux (and the corresponding fluence) fall in the expected range. The results look reasonably conservative and are acceptable.

2.3 Conclusion

The measurement and analysis practices, the computer programs and the major approximations in the computations either conform with the relevant ASTM procedures, use the generally accepted methods and practices in nuclear reactor dosimetry, or comply with staff requirements for similar work. In addition the results fall within the expected range. Therefore, we find this report acceptable.

3.0 PRESSURE/TEMPERATURE LIMITS

Appendix G to 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, to test the beltline materials in the surveillance capsules in accordance with Appendix H to 10 CFR Part 50. These tests define the condition of vessel embrittlement at the time of capsule withdrawal in terms of the increase in the reference temperature (RT_{NDT}). Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted RT_{NDT} and upper shelf energy. A method of calculating RT_{NDT} that is acceptable to the NRC staff is described in Regulatory Guide 1.99, Revision 2. An acceptable method for constructing P/T limits is described in Standard Review Plan (SRP) Section 5.3.2.

3.1 Evaluation

The 97-degree capsule contained Charpy impact specimens and tensile specimens that were made from base metal, weld metal, and heat affected zone metal. The specimens were designed and fabricated in accordance with ASTM E23-82 and ASTM E8-81. The Charpy impact, tensile, and hardness tests were conducted in accordance with appropriate sections of ASTM. In particular, the Charpy impact tests satisfied ASTM E23-82 and impact data was prepared according to ASTM E185-82. The Charpy impact tests showed that the longitudinal-orientated base metal (C-6404-3) exhibited the largest increase in transition temperature (RT_{NDT}) of 51.1°F at 30 ft-lb impact energy. This indicates that the base metal is the limiting (controlling) material. To verify this conclusion, the staff calculated the increase in RT_{NDT} of beltline material using methods in Regulatory Guide 1.99, Revision 2. The staff found that the limiting material with the highest adjusted RT_{NDT} is the intermediate shell plate, Code No. C-6404-3 (heat No. C-7595-1). This plate has copper and nickel contents of 0.1% and 0.53%, respectively. The unirradiated RT_{NDT} of the plate is 18°F. The staff's calculation confirmed that the C-6404-3 plate is the limiting material.

3.2 <u>Results</u>

In the November 7, 1988 submittal, SCE used the theoretical neutron fluence in the SONGS 2 FSAR to calculate the adjusted RT_{NDT}. The FSAR neutron fluence was based on the neutron transport analysis and was 40% lower than the measured neutron fluence in the surveillance capsule report. The 40% discrepancy between the measured and theoretical fluence values should not be a safety concern because uncertainties and random error in the transport analysis of this magnitude are expected. In addition, the licensee has implemented a lower neutron leakage fuel management and a longer operating cycle from 18 to 24 months since the beginning of Cycle 4. These measures will eventually reduce the actual neutron fluence to the level of the theoretical neutron fluence predicted by the transport analysis. To be conservative, the staff used the neutron fluence reported in the capsule analysis to calculate a maximum adjusted RT_{NDT} of 112.4°F (Table 1) for the C-6404-3 plate at 8 EFPY and 1/4 T (T = thickness of reactor vessel at beltline). Substituting the RT_{NDT} of 112.4°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits in heatup, cooldown, and inservice tests are within the acceptable values.

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes P/T limits on the reactor vessel closure flanges. Section IV.2 of Appendix G states that when pressure exceeds 20 percent of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the RT_{NDT} of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of -10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the reactor vessel must be thermally annealed if the predicted upper shelf energy (USE) at end of life is below 50 ft-lb. At 2.69 EFPY, the measured limiting USE is 104.7 ft-lb for the transverse-orientated base metal. This is a 16.5% reduction from the unirradiated value of 125.5 ft-lb. Using the method in Regulatory Guide 1.99, Revision 2, the staff predicted that the limiting USE at end of life will still be above 50 ft-lb and thus the USE satisfies the Section IV.B requirement.

3.3 Conclusion

The proposed SONGS 2 P/T limits on the Reactor Coolant System for heatup, cooldown, and inservice tests are valid through 8 EFPY because the limits conform to the requirements of Appendices G and H to 10 CFR Part 50. The SONGS 2 surveillance program also conforms to the requirements of Appendix H to 10 CFR Part 50. The P/T Limits may be incorporated into the SONGS 2 Technical Specifications.

TABLE 1

The staff's prediction of the adjusted RT_{NDT} for the limiting material.

Code No. C-6404-3

Heat No. C-7595-1

Material: Intermediate Shell Plate

Copper	0.1%	
Nickel	0.53%	
Initial RT_{NDT}	18°F	
Fluence at EOL (32 EFPY) and 1/4T	$3.10 \times 10^{19} n/cm^2$	(capsule data)
Fluence at 8 EFPY and 1/4T	$7.75 \times 10^{18} n/cm^2$	(capsule data)
Adjusted RT_{NDT} at 8 EFPY and 1/4T	112.4°F	
Adjusted RT_{NDT} at 32 EFPY and 1/4T	136.3°F	

4.0 LOW TEMPERATURE OVERPRESSURE PROTECTION LIMITS

LTOP is provided by the shutdown cooling system (SDCS) relief valves, which must be aligned to the RCS when the RCS is below the specified temperature to provide assurance that the reactor vessel will be operated in the ductile region in accordance with 10 CFR Part 50, Appendix G, during both normal operation and overpressurization events due to equipment malfunction or operator error. Technical Specifications require alignment of the SDCS relief valves to the RCS whenever RCS temperature is below the temperature corresponding to the P/T curve pressurizer relief valve setpoint of 2500 psia.

The updated P/T limits require the LTOP system to be aligned whenevr RCS temperature is less than 287°F during RCS cooldown and less than 312°F during RCS heatup. The current Technical Specification P/T limits only require LTOP alignment when RCS temperature is less than 235°F during RCS cooldown and heatup. However, the licensee asserts that the LTOP analyses performed and documented in the FSAR remain applicable to support the proposed Technical Specifications.

4.1 Evaluation

The staff questioned whether or not the original LTOP analysis was still bounding for all transients involving LTOP system design. This concern arose because the initial RCS temperatures in postulated LTOP transients will be higher in accordance with the proposed LTOP alignment temperatures.

The licensee responded to the staff concern in its letter dated February 23, 1989. As documented in the FSAR, the two limiting pressure transients for the LTOP system are (1) mass addition transient which assumes an inadvertent actuation of safety injection (startup of two HPSI pumps and three charging pumps), and (2) energy addition transient which assumes RCP start with a temperature difference of 100°F between the steam generator and the reactor coolant system. For the mass addition transient, the SDCS relief valve was sized to accommodate this LTOP transient for SDCS temperatures from 120°F to 400°F and assuming a 417 psia relief valve setpoint. Therefore, the change in the LTOP alignment temperature from 235°F to 312°F is still bounded by the original relief valve sizing calculation. For the energy addition transient, the original LTOP analysis assumed that one RCP was started with a maximum allowed differential temperature of 100°F between the primary and secondary systems. The most limiting energy addition transient would be with the secondary system at 350°F and the primary system at 250°F. However, this condition is prevented from occurring through administrative controls. Changing the RCS temperature at which LTOP must be aligned from 235°F to 312°F would not change the results of the most limiting energy addition transient. The energy addition transient is driven by the differential temperature between the primary side and secondary side rather than RCS initial energy. For a primary side temperature of 312°F and the maximum allowed secondary temperature of 350°F (SDCS maximum alignment temperature), the differential temperature would be reduced from 100°F to 38°F thus reducing the severity of the pressure transient.

4.2 Results

Based on the above, the staff agrees with the licensee's conclusion that changing the LTOP alignment temperature from 235°F to 312°F would not increase the severity of the most limiting LTOP transient. Therefore, the proposed Technical Specifications regarding LTOP system are bounded by the original FSAR analysis.

4.3 Conclusion

The staff finds that the proposed Technical Specification changes are reasonably conservative and acceptable to support the updated P/T limits applicable for the period of 4-8 EFPY.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published (54 FR 14303) in the Federal Register on April 10, 1989. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

7.0 CONCLUSION

We have 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, (2) such activities will be conducted in compliance with the Commission's regulations and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Lois
J. Tsao
C. Liang

Dated: April 11, 1989

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL.

DOCKET NO. 50-361

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 70 to Facility Operating License No. NPF-10 issued to Southern California Edison Company, San Diego Gas and Electric Company, The City of Riverside, California and The City of Anaheim, California (the licensees), which revised the Technical Specifications for operation of the San Onofre Nuclear Generating Station, Unit 2, located in San Diego County, California.

The amendment was effective as of the date of issuance.

This amendment revised Technical Specifications 3/4.4.8.1, "Pressure/Temperature Limits;" 3.4.1.4.1, "Cold Shutdown-Loops Filled;" 3.4.1.3, "Hot Shutdown;" 3.4.8.3.1, "Overpressure Protection System, RCS Temperature less than or equal to 235°F;" and 3.4.8.3.2, "Overpressure Protection System, RCS Temperature greater than 235°F." These changes revised the pressure/temperature and low temperature overpressure protection limits for operation through 8 effective full power years. The amendment was issued in response to an application for an amendment designated as PCN-278.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendments and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on February 24, 1989 (54 FR 8039). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined that an environmental impact statement will not be prepared and that issuance of the amendment will have no significant adverse effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated November 7, 1988, and subsequent submittals dated December 29, 1988 and February 23, 1989, (2) Amendment No. 70 to License No. NPF-10, (3) the Commission's related Safety Evaluation and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street NW., Washington, DC 20555, and at the General Library, University of California, P.O. Box 19557, Irvine, California 92713. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects III, IV, V and Special Projects.

Dated at Rockville, Maryland this 11th day of April, 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

D E Hickman

D. E. Hickman, Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation