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VICE PRESIDENT

March 11, 1989

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Supplement 3 to Amendment Application No. 164
"RCP Bus Undervoltage Trip"
San Onofre Nuclear Generating Station
Unit 1

By letter dated March 4, 1989, SCE submitted Supplement 2 to Amendment Application No. 164 (Revision 2 of Proposed Change 204) to include, among other things, the undercurrent and overcurrent RCP breaker trips in Appendix A, Technical Specifications, Table 2.1, "Maximum Safety System Settings," to prevent core damage from the seized shaft and sheared shaft events. The purpose of this supplement is to provide an additional safety setpoint in Table 2.1 to include the RCP bus undervoltage trip. Technical Specifications 3.5.1, "Reactor Trip System Instrumentation," and 4.1.1, "Reactor Trip System Instrumentation Surveillance Requirements," have also been changed to reflect the addition of this trip.

In loss of flow events caused by a loss of RCP bus, the low flow trip and the undervoltage trip provide diverse and redundant reactor protection. RCP bus undervoltage trip is currently maintained administratively.

If you have any questions regarding this matter, please contact me.

Respectfully submitted,

By: Kenneth P Baskin
Kenneth P. Baskin
Vice President

Subscribed and sworn to before me this 14th day of March, 1989.

Carol A. Gomez
Notary Public in and for the County of Los Angeles, State of California

OFFICIAL SEAL
CAROL A. GOMEZ
NOTARY PUBLIC - CALIFORNIA
LOS ANGELES COUNTY
My Comm. Expires Feb. 26, 1993

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cc: J. B. Martin, Regional Administrator, NRC Region V
F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY (San Onofre Nuclear Generating Station Unit No. 1	)))))	Docket No. 50-206

CERTIFICATE OF SERVICE

I hereby certify that a copy of Amendment Application No. 204, Supplement 3, was served on the following by deposit in the United States Mail, postage prepaid, on the 13th day of March, 1989.

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Southern California Edison Company

DESCRIPTION OF SUPPLEMENTAL CHANGE TO PROPOSED CHANGE
NO. 204 TO THE TECHNICAL SPECIFICATIONS PROVISIONAL
OPERATING LICENSE NO. DPR-13
SUPPLEMENT 1 TO REVISION 2

The following is a supplemental request to revise Section 2.1, "Reactor Core - Limiting Combination of Power, Pressure and Temperature," and Table 3.5.1-1, "Reactor Trip System Instrumentation," and Table 4.1.1, "Reactor Trip System Instrumentation Surveillance Requirements," of the Appendix A, Technical Specifications for San Onofre Nuclear Generating Station, Unit 1 (SONGS 1).

Description of Supplemental Change

The bus undervoltage trip of RCP breakers provides a diverse and redundant reactor protection against potential core damage in addition to the low flow trip. The safety setpoint for the bus undervoltage trip is, however, omitted in Table 2.1, "Maximum Safety Systems Settings." This supplement includes the RCP bus undervoltage trip setpoint in Table 2.1, Table 3.5.1-1, and Table 4.1.1 of the Technical Specifications.

The Attachment 2 changes have single change bars to designate those changes made in previous revisions to this proposed change. This supplement adds changes which are indicated with double change bars.

Existing Technical Specifications

See Attachment 1.

Proposed Technical Specifications

See Attachment 2.

Significant Hazards Consideration Analysis

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that a supplmental license amendment to implement revised provisions for this proposed change and operability for SONGS 1 represents no significant hazards consideration. In accordance with the three factor test of 10 CFR 50.92(c), implementation of the proposed license amendment was analyzed using the following standards and found not to: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The 4 kV Bus-1A supplies power to RCP G-2A and G-2C. The 4 kV Bus-1B is the power source for RCP G-2B. The RCP bus undervoltage trip circuit consists of two detection channels, one on each RCP bus, which opens the respective RCP breakers on detection of bus undervoltage. The RCP circuit breaker trip actuates RCP breaker auxiliary switch contacts which trip the reactor. The undervoltage trip is currently controlled administratively for normal operations.

In loss of flow events caused by a loss of power to the reactor coolant pumps (RCPs), e.g., a loss of bus or manual actions to open an RCP circuit breaker, the low flow trip and the undervoltage trip provide a diverse and redundant reactor protection. Inclusion of the RCP bus undervoltage trip is consistent with assumptions in UFSAR Section 15.7.1.

Additional changes remove from Table 2.1 and provide to Table 3.5.1-1 clarifications of the function of the Reactor Protection System permissives P-7 and P-8. One of three loops reactor trip logic is enabled by the P-8 permissive above 50% power. The P-7 permissive enables the two of three loops reactor trip logic above 10% power. This provides the necessary protection for a single loop failure for sheared shaft and seized shaft events above 50% power and for two loop loss of flow events below 50% power.

Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No.

   Currently the undervoltage trip is administratively controlled for normal operations. The undervoltage trip is necessary to satisfy single failure criterion for loss of flow events caused by loss of power to the RCP. The addition of the undervoltage trip safety setpoint in the Table 2.1 ensures that the RCP bus undervoltage trip is available and OPERABLE, consistent with assumptions in the accident analysis in UFSAR 15.7.1. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The supplemental change includes the safety system setting of the RCP bus undervoltage trip in the Table 2.1. This trip is credited to satisfy the single failure criterion in the event of loss of forced coolant flow due to loss of RCP bus. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The addition of the safety setting of the undervoltage trip in the Table 2.1 provides consistency with assumptions in the accident analysis in UFSAR 15.7.1 and satisfies the single failure criterion for loss of flow events caused by loss of power to the RCPs. Therefore, the proposed supplemental change will not involve any reduction in a margin of safety.

Safety and Significance Hazards Consideration Determination

Based on the Safety Evaluation provided in Amendment Application No. 164 and the information provided above, it is concluded that: (1) the supplemental changes to Proposed Change No. 204 do not involve a significant hazards considerations defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

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Attachment 1

Existing Technical Specifications

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

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

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

SPECIFICATION: Safety Limits

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

(2) The combination of reactor power and coolant temperature shall not exceed the locus of points established for the RCS pressure in Figure 2.1.1. If the actual power and temperature is above the locus of points for the appropriate RCS pressure, the safety limit is exceeded. 60 6/8/81

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 2.1. 117 12/13/88

BASIS: Safety Limits

1. Reactor Coolant System Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the containment atmosphere. In addition, the failure of components of the Reactor Coolant System could result in damage to the fuel and pressurization of the containment. A safety limit of 2735 psig (110% of design pressure) has been established which represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section VIII.

2. Plant Operating Transients

In order to prevent any significant amount of fission products from being released from the fuel to the reactor coolant, it is necessary to prevent clad overheating both during normal operation and while undergoing system transients. Clad overheating and potential failure could occur if the heat transfer mechanism at the clad surface departs from nucleate boiling. System parameters which affect this departure from nucleate boiling (DNB) have been correlated with experimental data to provide a means of determining the probability of DNB occurrence. The ratio of the heat flux at which DNB is expected to occur for a given set of conditions to the actual heat flux experienced at a point is the DNB ratio and reflects the probability that DNB will actually occur.

It has been determined that under the most unfavorable conditions of power distribution expected during core lifetime and if a DNB ratio of 1.44 should exist, not more than 7 out of the total of 28,260 fuel rods would be expected to experience DNB. These conditions correspond to a reactor power of 125% of rated power. Thus, with the expected power distribution and peaking factors, no significant release of fission products to the reactor coolant system should occur at DNB ratios greater than 1.30.(1) The DNB ratio, although fundamental, is not an observable variable. For this reason, limits have been placed on reactor coolant temperature, flow, pressure, and power level, these being the observable process variables related to determination of the DNB ratio. The curves presented in Figure 2.1.1 represent loci of conditions at which a minimum DNB ratio of 1.30 or greater would occur. (1)(2)(3) 49 7/19/79

Maximum Safety System Settings

1. Pressurizer High Level and High Pressure

In the event of loss of load, the temperature and pressure of the Reactor Coolant System would increase since there would be a large and rapid reduction in the heat extracted from the Reactor Coolant System through the steam generators. The maximum settings of the pressurizer high level trip and the pressurizer high pressure trip are established to maintain the DNB ratio above 1.30 and to prevent the loss of the cushioning effect of the steam volume in the pressurizer (resulting in a solid hydraulic system) during a loss-of-load transient.(3)(4)

In the event that steam/feedflow mismatch trip cannot be credited due to single failure considerations, the pressurizer high level trip is provided. In order to meet acceptance criteria for the Loss of Main Feedwater and Feedline Break transients, the pressurizer high level trip must be set at 20.8 ft. (50%) or less. 97 4/7/86

2. Variable Low Pressure Loss of Flow and Nuclear Overpower Trips

These settings are established to accommodate the most severe transients upon which the design is based, e.g., loss of coolant flow, rod withdrawal at power, control rod ejection, inadvertent boron dilution and large load increase without exceeding the safety limits. The settings have been derived in consideration of instrument 127 12/13/88 errors and response times of all necessary equipment. Thus, these settings should prevent the release of any significant quantities of fission products to the coolant as a result of transients.(3)(4)(5)(7) 117 12/13/88

In order to prevent significant fuel damage in the event of increased peaking factors due to an asymmetric power distribution in the core, the nuclear overpower trip setting on all channels is reduced by one percent for each percent that the asymmetry in power distribution exceeds 5%. This provision should maintain the DNB ratio above a value of 1.30 throughout design transients mentioned above. 117 12/13/88

The response of the plant to a reduction in coolant flow while the reactor is at substantial power is a corresponding increase in reactor coolant temperature. If the increase in temperature is large enough, DNB could occur, following loss of flow.

The low flow signal is set high enough to actuate a trip in time to prevent excessively high temperatures and low enough to reflect that a loss of flow conditions exists. Since coolant loop flow is either full on or full off, any loss of flow would mean a reduction of the initial flow (100%) to zero.(3)(6)

__References__:

(1) Amendment No. 10 to the Final Engineering Report and Safety Analysis, Section 4, Question 3

(2) Final Engineering Report and Safety Analysis, Paragraph 3.3

(3) Final Engineering Report and Safety Analysis, Paragraph 6.2

(4) Final Engineering Report and Safety Analysis, Paragraph 10.6

(5) Final Engineering Report and Safety Analysis, Paragraph 9.2

(6) Final Engineering Report and Safety Analysis, Paragraph 10.2

(7) NIS Safety Review Report, April 1988 117 12/13/88

TABLE 2.1

MAXIMUM SAFETY SYSTEM SETTINGS

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	Three Reactor Coolant Pumps Operating
*1. Pressurizer High Level	≤ 20.8 ft. above bottom of pressurizer when steam/feedflow mismatch trip <u>is not</u> credited, or ≤ 27.3 ft. above bottom of pressurizer when steam/feedflow mismatch trip <u>is</u> credited
2. Pressurizer Pressure: High	≤ 2220 psig
3. Nuclear Overpower	
a. High Setting**	≤ 109% of indicated full power
b. Low Setting	≤ 25% of indicated full power
***4. Variable Low Pressure	≥ 26.15 (0.894 $\Delta T + T$ avg.) - 14341
***5. Coolant Flow	≥ 85% of indicated full loop flow

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* Credit can be taken for the steam/feedflow mismatch trip when this system is modified such that a single failure will not prevent the system from performing its safety function.

** The nuclear overpower trip is based upon a symmetrical power distribution. If an asymmetric power distribution greater than 5% should occur, the nuclear overpower trip on all channels shall be reduced one percent for each percent above 5%.

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

TABLE 3.5.1-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	7
2. Power Range, Neutron Flux, Overpower Trip	4	2	3	1, 2	2#
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	4	1**	4	1, 2	28#
4. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
5. Source Range, Neutron Flux					
A. Startup	2	1**	2	2##	4
B. Shutdown	2	1**	2	3*, 4*, 5*	7
C. Shutdown	2	0	1	3, 4, and 5	5
6. NIS Coincidentor Logic	2	1	2	1, 2	29
				3*, 4*, 5*	7
7. Pressurizer Variable Low Pressure	3	2	2	1####	6#
8. Pressurizer Fixed High Pressure	3	2	2	1, 2	6#
9. Pressurizer High Level	3	2	2	1	6#

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TABLE 3.5.1-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Reactor Coolant Flow					
A. Single Loop (Above 50% of Full Power)	1/loop	1/loop in any operating loop	1/loop in each operating loop	1	6#
B. Two Loops (Below 50% of Full Power)	1/loop	1/loop in two operating loops	1/loop in each operating loop	1####	6#
11. Steam/Feedwater Flow Mismatch	3	2	2	1,2	6#
12. Turbine Trip-Low Fluid Oil Pressure	3	2	2	1####	6#

TABLE 4.1.1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.
2. Power Range, Neutron Flux	S	D (2,3) R (3,4)	M	N.A.	N.A.
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	N.A.	N.A.	M	N.A.	N.A.
4. Intermediate Range, Neutron Flux	S	R (3,4)	S/U (1), M	N.A.	N.A.
5. Source Range, Neutron Flux	S	R (3)	S/U (1), M	N.A.	N.A.
6. NIS Coincidentor Logic	N.A.	N.A.	N.A.	N.A.	M (5)
7. Pressurizer Variable Low Pressure	S	R	M	N.A.	N.A.
8. Pressurizer Pressure	S	R	M	N.A.	N.A.
9. Pressurizer Level	S	R	M	N.A.	N.A.
10. Reactor Coolant Flow	S	R	Q	N.A.	N.A.
11. Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.
12. Turbine Trip-Low Fluid Oil Pressure	N.A.	N.A.	N.A.	S/U (1,6)	N.A.