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November 11, 1988

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

Gentlemen:

Subject: Docket No. 50-206
Amendment Application No. 157
San Onofre Nuclear Generating Station
Unit 1

Enclosed is Amendment Application No. 157 to the Provisional Operating License DPR-13 for San Onofre Nuclear Generating Station, Unit 1. Amendment Application No. 157 consists of Proposed Technical Specification Change No. 185.

Proposed Change 185 is a request to revise Appendix A, Technical Specification 3.5.1 and Table 2.1 to incorporate the revised steam/feedwater flow mismatch trip setpoints and revised limiting conditions for operations. The proposed change will provide the high and low steam/fedwater flow mismatch trip setpoints to satisfy the single failure criterion, unconsidered prior to the failure of PT-459, for design basis Loss of Normal Feedwater and Feedwater Line Break events.

This proposed change is required for SONGS 1 Return to Service from Cycle 10 Refueling Outage. Acordingly, SCE requests approval of the proposed change prior to the anticipated need date of February 19, 1989, to prevent delay for Unit 1 startup.

Pursuant to 10CFR 170.22, the required amendment application fee of $150 is enclosed.

If you have any questions regarding this amendment application, please call me.

Very truly yours,

Kenneth P Baskin

Enclosure

cc: J. B. Martin, Regional Administrator, NRC Region V
F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3
J. H. Hickman, California Department of Health Services

A001 w/check $150
1/1 #A-145973

Based on the significant hazards analysis provided in the Description of Proposed Change and Significant Hazards Analysis of Proposed Change No. 185, it is concluded that (1) the proposed change does not involve a significant hazards consideration as defined in 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.

Pursuant to 10 CFR 170.12, the fee of $150 is herewith remitted.

Subscribed on this 11th day of November, 1988.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: Kenneth P. Baskin
Kenneth P. Baskin
Vice President

Subscribed and sworn to before me this 11th day of November, 1988.

OFFICIAL SEAL
C. SALLY SEBO
Notary Public–California
LOS ANGELES COUNTY
My Comm. Exp. Apr. 20, 1990

C. Sally Sebo
Notary Public in and for the County of Los Angeles, State of California

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By: James A. Beoletto
James A. Beoletto

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. 157 was served on the following by deposit in the United States Mail, postage prepaid, on the 11th day of November, 1988.

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DESCRIPTION AND SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS OF
PROPOSED CHANGE NO. 185 TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE NO. DPR-13

This is a request to revise Sections 2.1, "REACTOR CORE-Limiting Combination of Power, Pressure and Temperature" and 3.5.1, "REACTOR TRIP SYSTEM INSTRUMENTATION" of the Appendix A Technical Specifications for San Onofre Nuclear Generating Station, Unit 1 (SONGS 1).

DESCRIPTION OF CHANGE

Technical Specification 2.1 describes the limiting safety settings for systems designed to limit transients in the reactor and reactor coolant system. These settings serve to maintain the integrity of the reactor coolant system and minimize the effects of specified transients on the integrity of the nuclear fuel in the reactor core. Proposed Change No. 185 revises Table 2.1 and the Basis of the technical specification to include the safety setting for the steam/feedwater flow mismatch trip and to delete a provision of the pressurizer level trip when the steam/feedwater flow mismatch trip is credited. The last change is necessary due to design considerations which are described in the significant hazards consideration analysis section of this proposed change.

Technical Specification 3.5.1 describes the Limiting Conditions for Operation (LCOs) for the Reactor Trip System instrumentation, which includes the steam/feedwater flow mismatch trip. Proposed Change No. 185 revises the LCO of the steam/feedwater flow mismatch functional unit to be Mode 1 above 50% nominal power. This operability requirement is consistent with the redesign of this trip functional unit.

EXISTING TECHNICAL SPECIFICATION

See Attachment 1

PROPOSED TECHNICAL SPECIFICATION

See Attachment 2

SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that a proposed license amendment to implement revised provisions for steam/feedwater flow mismatch safety settings and operability for SONGS 1 represents a 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; or 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.

Analysis

The scope of steam/feedwater flow mismatch reactor trip upgrade was previously provided to the NRC by letter dated November 20, 1987. The letter described the revised basis for reactor trip on steam/feedwater flow, the single failure design upgrade and the addition of the P-8 permissive feature to preclude operability at nominal power levels below 50%. The definition of the high steam/feedwater flow mismatch setpoint, however, was deferred to some later documentation. It is the purpose of the following information to provide the definition of the high steam/feedwater flow mismatch setpoint, provide justification of the setpoints, and describe functional unit operability requirements. The revised setpoint and operability requirements are proposed in Attachment 2.

Proposed Technical Specification Table 2.1 provides the normalized high and low steam/feedwater flow mismatch setpoints, such that a reactor trip will occur on steam/feedwater flow mismatch if the feedwater flow falls below or exceeds the steam flow by 25%. The existing low steam/feedwater flow mismatch setpoint is included in Table 2.1 and remains unchanged. The high steam/feedwater flow mismatch setpoint is necessary to meet the single failure criterion for the design basis feedwater line break. The high setpoint works in conjunction with the low setpoint such that two of three inputs outside the range specified in Table 2.1 will cause a reactor trip. Hence, two trip inputs from the low setpoints, two from the high setpoints, or one input from each of the high and low setpoints will cause a reactor trip. The trip setpoints take into account the uncertainty in steam and feedwater flow measurements and instrument range limitations.

A Westinghouse analysis conservatively estimated the flow through the faulted line to increase by approximately 50% of the rated feedwater flow, while each of the remaining two intact lines to lose 25% of the rated feedwater flow. Consequently, the high steam/feedwater mismatch setpoint can be any value less than or equal to 1.5; therefore, an added margin is provided in the high setpoint in Table 2.1. The pressurizer high level trip setpoint at 50% narrow range level, equivalent to 20.8 feet, is retained in the proposed Technical Specification, since it is credited in the Westinghouse analysis at less than 50% power with bypassed mismatch trip for Loss of Normal Feed (LONF) and Feedline Break (FLB), and Partial LONF at 100% power. The pressurizer high level trip setpoint at 70% narrow range level, equivalent to 27.3 feet included in the current Technical Specification, is deleted since a pressurizer high level trip setpoint at 70% level does not provide adequate reactor protection for those postulated transients.

Proposed Technical Specification 3.5.1 describes the LCOs for, among other plant features, the steam/feedwater flow mismatch trip instrumentation. The revised operability requirements will allow this functional unit to be inoperable at power levels below 50%. The basis for this revision is provided in the analyses enclosed with the November 20, 1987 letter. The design basis Loss of Normal Feedwater (LONF) and Feedline Break (FLB) cases run at 50% nominal power, assumed that the steam/feedwater flow mismatch trip was unavailable. Accordingly, the steam/feedwater flow mismatch trip instrumentation is not required for LONF and FLB transients at nominal power levels less than 50% and the proposed 3.5.1 Technical Specification is appropriate.

The surveillance of this modification is established in the existing requirements of Table 4.1.1. These requirements remain appropriate. Table 4.1.1 is proposed to be modified by Proposed Change No. 180, currently under NRC review, but the surveillances in the proposed Table 4.1.1 are also appropriate.

Conformance of the proposed changes to the standards for a determination of no significant hazard as defined in 10 CFR 50.92 (three factor test) is shown in the following:

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

   RESPONSE: NO

   Operation of the facility in accordance with this proposed change will not be allowed until completion of the modifications described in the November 20, 1987 letter to the NRC. These modifications will allow for the required credit to be taken for the steam/feedwater flow mismatch trip for certain postulated events. The revised interaction of this instrumentation with design basis events for SONGS 1 was provided with the November 20, 1987 letter. These analyzed transients demonstrate that the combination of the steam/feedwater flow mismatch trip with the high pressurizer level at 50% narrow range trip provide adequate protection for decrease in feedwater flow events, e.g., LONF and FWLB. Therefore, the upgraded design and, consequently, the manner in which the plant is operated are determined to have a net positive effect on probability or consequences of accidents previously evaluated. Therefore, it is concluded that operation of the facility in accordance with this proposed change will not involve any increase in the probability or consequences of an 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

   Operation of the facility in accordance with this proposed change will ensure that systems associated with reactor trip are operable and the setpoints assumed in previously performed safety analyses are maintained. The accidents analyzed in the November 20, 1987 letter are variations of previously evaluated accidents and the steam/feedwater flow mismatch trip can now be assumed to provide protective action for particular cases. The steam/feedwater flow mismatch functional unit of the reactor protection system was part of the original plant design and has been previously analyzed for use. Therefore, it is concluded that operation of the facility in accordance with this proposed change does 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

Operation of the facility in accordance with this proposed change assures that new established margins of safety, as described in the November 20, 1987 letter, are maintained. The steam/feedwater flow mismatch trip will now provide a protective action for events previously for which, by design, it could not be postulated to function. As stated in the November 20, 1987 letter the protective action of the steam/feedwater flow mismatch reactor trip is only credited for decrease in feedwater flow events at reactor nominal power levels in excess of 50%. Accordingly, the proposed LCO will now only require operability during Mode 1 above 50% nominal power in lieu of the previous requirements of Modes 1 and 2. However, as this is consistent with the revised analyses, it involves no reduction in a margin of safety. The proposed changes described herein will assure operability and proper setpoint of the instrumentation is maintained. Therefore, it is concluded that operation of the facility in accordance with this proposed change does not involve a significant reduction in a margin of safety.

SAFETY AND SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the preceding analysis, it is concluded that: (1) Proposed Change No. 185 does not involve a significant hazards consideration as defined by 10 CFR 50.92; and (2) the health and safety of the public will not be endangered by the proposed change.

Attachment 1 - Existing Specifications
Attachment 2 - Proposed Specifications

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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 6/8/8

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 2.1

TABLE 2.1

		Three Reactor Coolant Pumps Operating	
*1.	Pressurizer High Level	≤ 20.8 ft. above bottom of pressurizer when steam/feedflow mismatch trip is not credited, or ≤ 27.3 ft. above bottom of pressurizer when steam/feedflow mismatch trip is credited	97 4/7/86
2.	Pressurizer Pressure: High	≤ 2220 psig	
**3.	Nuclear Overpower	≤ 109% of indicated full power	49 7/19/79
***4.	Variable Low Pressure	$\geq 26.15\ (0.894\ \Delta T + T\ \text{avg.}) - 14341$	60 6/8/81
***5.	Coolant Flow	≥ 85% of indicated full loop flow	49 7/19/79

* 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. 97 4/7/86

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

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

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, inadvertent boron dilution and large load increase without exceeding the safety limits. The settings have been derived in consideration of instrument 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)

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 10%. This provision should maintain the DNB ratio above a value of 1.30 throughout design transients mentioned above.

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

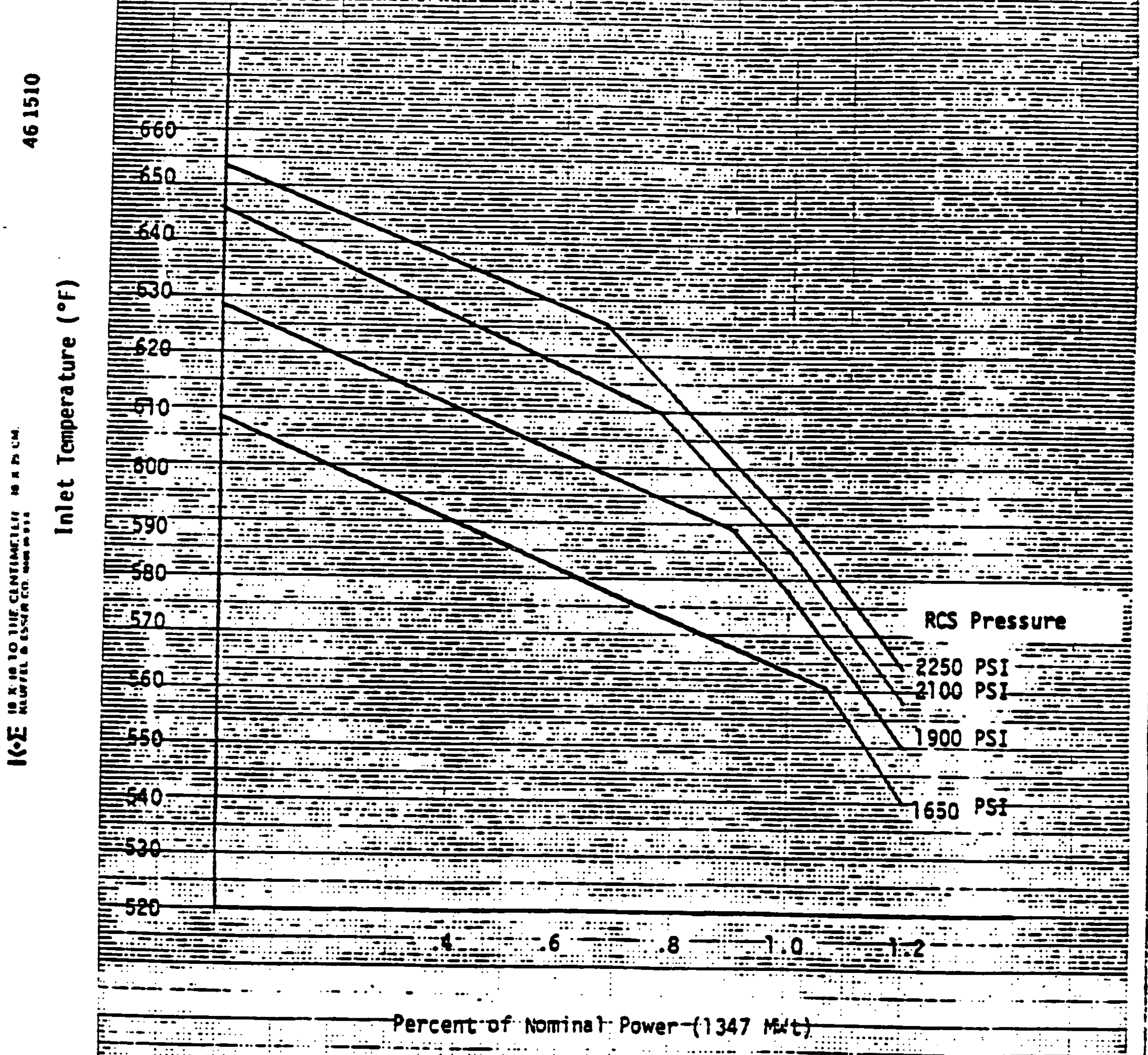
(3) Final Engineering Report and Safety Analysis, Paragraph 6.2

(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

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



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Revised: 6/23/81

TABLE 3.5.1-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL 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	4	2	3	1, 2	2#
3. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
4. 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
5. Pressurizer Variable Low Pressure	3	2	2	1####	6#
6. Pressurizer Fixed High Pressure	3	2	2	1, 2	6#
7. Pressurizer High Level	3	2	2	1	6#
8. 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#
9. Steam/Feedwater Flow Mismatch	3	2	2	1, 2	6#

TABLE 3.5.1-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1####	6#

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TABLE 3.5.1-1 (Continued)

TABLE NOTATION

*	With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal.
**	A "TRIP" will stop all rod withdrawal.
#	The provisions of Specification 3.0.4 are not applicable.
##	Below the Source Range High Voltage Cutoff Setpoint.
###	Below the P-7 (At Power Reactor Trip's Active) Setpoint.
####	Above the P-7 (At Power Reactor Trip's Active) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 8 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.1.

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ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the Source Range High Voltage Cutoff Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the Source Range High Voltage Cutoff Setpoint.

b. Above the Source Range High Voltage Cutoff Setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Mininum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.5.2 as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 8 hours.

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ACTION 7 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.