BEFORE THE UNITED STATES NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY for a Class 104(b) License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 1 of the San Onofre Nuclear Generating Station	)	DOCKET NO. 50-206 Amendment No. 161

SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY, pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 161.

This amendment consists of Proposed Change No. 183 to Provisional Operating License No. DPR-13. Proposed Change No. 183 modifies the Technical Specifications incorporated in Provisional Operating License No. DPR-13 as Appendix A.

Proposed Change No. 183 is a request to revise Technical Specifications associated with the Reactor Protection System instrumentation. This proposed change incorporates Limiting Conditions for Operation and Surveillance requirements into the technical specifications that are currently performed by procedure. In addition, surveillance intervals and out of service times have been increased in accordance with Westinghouse recommendations as documented in WCAP-10271.

In the event of conflict, the information in Amendment Application No. 161 supersedes the information previously submitted.

Based on the significant hazards analysis provided in the Description of Proposed Change and Significant Hazards Analysis of Proposed Change No. 183, 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 29th day of December, 1988.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: ______________________
Dr. L. T. Papay
Senior Vice President

Subscribed and sworn to before me this
29th day of December, 1988.

OFFICIAL SEAL
AGNES CRABTREE
Notary Public-California
LOS ANGELES COUNTY
My Comm. Exp. Sep. 14, 1990

Agnes Crabtree
Notary Public in and for the County of
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By: ______________________
James A. Beoletto

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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. 161 was served on the following by deposit in the United States Mail, postage prepaid, on the 29th day of December, 1988.

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DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGE NO. 183
TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE NO. DPR-13

This is a request to revise Section 3.5.1, "Reactor Trip System Instrumentation," and Section 4.1.1, "Operational Safety Items," of Appendix A Technical Specifications for San Onofre Nuclear Generating Station Unit 1.

DESCRIPTION

In response to concerns of the impact of current testing and maintenance requirements on plant operations, particularly as related to instrumentation systems, the Westinghouse Owners Group (WOG) initiated a program to develop a justification to be used to revise generic and plant specific instrumentation technical specifications. Operating plants experienced inadvertent reactor trips during performance of instrumentation surveillance, causing unnecessary transients and challenges to safety systems. Significant time and effort on the part of the operating staff was devoted to performing, reviewing, documenting and tracking the various surveillance activities, which in many instances seemed unwarranted based on the high reliability of the equipment. Significant benefits for operating plants appeared to be achievable through revision of instrumentation test and maintenance requirements.

Background

On February 3, 1983, the Westinghouse Owners Group submitted (letter OG-86) WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" to the NRC as the first step in gaining approval of the instrumentation program. WCAP-10271 documents the justification to be used to justify revisions to technical specifications. The justification consists of the deterministic and numerical evaluation of the effects of particular technical specification changes with consideration given to such things as safety, equipment requirements, human factors and operational impact. The objective is to reach a balance in which safety and operability are ensured. The technical specification revisions evaluated were increased test and maintenance times, less frequent surveillance, and testing in bypass.

In July 1983, the NRC requested additional information from the WOG (letter to J. J. Sheppard from Cecil O. Thomas dated July 28, 1983) required for continued review. The WOG responded in October 1983 (letter OG-106 dated October 4, 1983) with responses to the NRC concerns and Supplement 1 to WCAP-10271 which contains information in addition to that in WCAP-10271. Specifically, Supplement 1 demonstrates the applicability of the justification contained in WCAP-10271 to reactor protection systems for two, three and four loop plants with either relay or solid state logic.

In conjunction with the completion of the WOG efforts in the area of RPS instrumentation, resolution of Systematic Evaluation Program (SEP) Topic VI-10.A, Testing of Reactor Trip System and Engineered Safety Features, has been integrated with this change. The changes resulting from SEP Topic VI-10.A include incorporation of channel testing, checking and calibration requirements currently specified only by procedure.

DESCRIPTION OF CHANGES

Revisions to San Onofre Unit 1 Reactor Trip System Instrumentation Technical Specifications are proposed as follows:

1. Increase the surveillance interval for RPS analog channel operational tests from once per month to once per quarter (WCAP-10271),

2. Increase the time an inoperable RPS analog channel may be bypassed to allow testing of another channel in the same function from two hours to four hours (WCAP-10271),

3. Incorporate channel testing, checking and calibration requirements currently specified only by procedure (SEP Topic VI-10.A), and

4. Incorporate requirements to independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers (Generic Letter 85-09).

Each of the above changes is discussed in detail in the following sections. The proposed technical specifications that incorporate these changes are provided in attachment 2. Change bars are used to illustrate the proposed revisions.

I. The proposed changes to Technical Specification Table 3.5.1-1 "Reactor Trip System Instrumentation" would revise the instrumentation and control requirements as described below.

   1. Manual Reactor Trip

      No change to the existing specification.

   2. Power Range, Neutron Flux, Overpower Trip

      No change to the existing specification.

   3. Power Range, Neutron Flux, Dropped Rod Rod Stop

      No change to the existing specification.

   4. Intermediate Range, Neutron Flux

      No change to the existing specification.

5. Source Range, Neutron Flux

   No change to the existing specification.

6. NIS Coincidentor Logic

   No change ot the existing specification.

7. Pressurizer Variable Low Pressure Calculator

   The title of this reactor trip parameter will be changed to include the word "calculator." This change is proposed to clarify that this trip signal is originated by a calculation that includes RCS delta T and Tave (per Technical Specification 2.1).

   The ACTION statement that corresponds to this instrumentation channel will be changed from ACTION 6 to ACTION 2. This change allows returning an inoperable channel to the untripped condition for surveillance testing of other channels and is consistent with the WOG guidelines for RPS technical specifications.

8. Pressurizer Fixed High Pressure

   The ACTION statement will be changed from ACTION 6 to ACTION 2. This change allows returning an inoperable channel to the untripped condition for surveillance testing of other channels. This is consistent with the bypass provision in the WOG guidelines for RPS technical specifications. A footnote has also been added to require compliance with the provisions of Specification 3.5.5 for any portion of this channel required to be OPERABLE by Specification 3.5.5. This footnote is added to ensure compliance with the more restrictive requirements for the ESFAS function of this channel.

9. Pressurizer High Level

   The ACTION statement will be changed from ACTION 6 to ACTION 2. This change allows returning an inoperable channel to the untripped condition for surveillance testing of other channels and is consistent with the WOG guidelines for RPS technical specifications.

10. Reactor Coolant Flow

    No change to the existing specification.

11. Steam/Feedwater Flow Mismatch

    The ACTION statement will be changed from ACTION 6 to ACTION 2 (see discussion for Item 9).

12. Turbine Trip - Low Fluid Oil Pressure

    The ACTION statement will be changed from ACTION 6 to ACTION 2 (see discussion for Item 9).

13. Reactor Trip Breakers

In conjunction with the resolution to SEP Topic VI-10.A, an LCO for this parameter will be incorporated into the technical specifications. This parameter is currently controlled by plant procedures. The proposed LCO mode applicability requirements are consistent with the Westinghouse STS and the WOG guidelines for RPS technical specifications. The basis for the corresponding ACTION statements is provided below.

ACTION 30 has been added to Table 3.5.1.1 in order to establish appropriate controls for inoperability of the reactor trip breakers (RTBs). This action is similar to ACTION 29 with the exception that no bypass capability is provided for the SONGS 1 RTBs. The configuration of the RTBs is two breakers in series with no bypass breakers. Therefore, should one breaker become inoperable, the plant must be brought to HOT STANDBY within 6 hours. During this time, the remaining OPERABLE breaker will be available to trip the plant if necessary. Once HOT STANDBY has been achieved, ACTION 7 will be imposed. ACTION 7 allows 48 hours to restore the inoperable breaker to OPERABLE status or open the RTBs within the following hour. This requirement is consistent with the WOG guidelines.

14. Sequencer Input to Reactor Trip

In conjunction with the resolution to SEP Topic VI-10.A, an LCO for this parameter will be incorporated into the technical specifications. This parameter is currently controlled by plant procedures. The configuration for this reactor trip signal is as follows. SONGS 1 utilizes two sequencers to automatically start emergency loads in the event of a safety injection actuation signal or automatically sequence emergency loads onto the emergency diesel generators in the event of a coincident safety injection signal and loss of offsite power. Each sequencer has two subchannels. These subchannels each send a signal, initiated by SIS or LOP, to a logic gate. When positive signals are sent from both subchannels, a reactor trip signal is initiated. This process is the same for both sequencers. The proposed channel operability requirements stipulate that both sequencers be OPERABLE with one sequencer required to send a trip signal. The operability of the sequencers includes the individual subchannels.

A footnote has also been added to require compliance with the provisions of Specification 3.5.5 for any portion of this channel required to be OPERABLE by Specification 3.5.5. This footnote is added to ensure compliance with the more restrictive requirements for the ESFAS function of this channel.

II. The proposed changes to Technical Specification Table 4.1.1 "Minimum Frequencies for Testing, Calibrating, and/or Checking of Instrument Channels" would revise the instrumentation and control requirements as described below. Several new requirements have been added to Table 4.1.1 in accordance with resolution to SEP Topic VI-10.A. These changes, in addition to the extended surveillance frequencies for certain RTS channels, are all described below.

1. Manual Reactor Trip

   Footnote 8 has been included to require that the TRIP ACTUATING DEVICE OPERATIONAL TEST independently verify the operability of the undervoltage and shunt trip circuits for the Manual Reactor Trip function. This footnote is incorporated in response to NRC Generic letter 85-09. It is noted that the WOG requirement to verify the operability of bypass breakers has not been included since SONGS 1 does not utilize bypass breakers.

2. Power Range, Neutron Flux

   The test frequency for this channel has been extended from monthly to quarterly and includes reference to testing on a STAGGERED TEST BASIS consistent with the WOG guidelines. Also, consistent with the WOG guidelines a requirement to test this channel prior to each startup, if not performed in the previous 31 days, has been added.

3. Power Range, Neutron Flux, Dropped Rod Rod Stop

   The test frequency for this channnel has been extended from monthly to quarterly and includes reference to testing on a STAGGERED TEST BASIS consistent with the WOG guidelines.

4. Intermediate Range, Neutron Flux

   Testing of this channel will be performed prior to startup if the test has not been performed in the previous 31 days versus the existing requirement of 7 days. In addition, the test frequency has been extended from monthly to quarterly and includes reference to testing on a STAGGERED TEST BASIS. These revisions are consistent with the WOG guidelines. It is noted San Onofre Unit 1 uses a P-7 permissive as opposed to the Westinghouse STS P-10 permissive.

5. Source Range

   Similar to Item 4 above, testing of this channel will be performed prior to startup if the test has not been performed in the previous 31 days versus the existing requirement of 7 days, and quarterly on a STAGGERED TEST BASIS.

6. NIS Coincidentor Logic

   This channel is essentially the same as the Westinghouse STS Automatic Trip Logic channel. Consistent with the requirements for the STS channel, the NIS Coincidentor Logic will be tested monthly on a STAGGERED TEST BASIS.

7. Pressurizer Variable Low Pressure Calculator

   Consistent with the WOG guidelines and the Westinghouse STS, the test frequency for this channel will be extended from monthly to quarterly and performed on a STAGGERED TEST BASIS.

A footnote has also been added to require compliance with the provisions of Specification 4.1.4 for any portion of this channel required to be OPERABLE by Specification 3.5.5. This footnote is added to ensure compliance with the more restrictive requirements for the ESFAS function of this channel.

8. Pressurizer Fixed High Pressure

Consistent with the WOG guidelines, the test frequency for this channel will be extended from monthly to quarterly and performed on a STAGGERED TEST BASIS. A footnote has also been included to clarify that the CHANNEL CHECK surveillance refers to monitoring the parameter not the trip setpoint (i.e., check pressurizer pressure not pressurizer high pressure trip). This footnote also applies to the Pressurizer High Level, Reactor Coolant Flow, and Steam/Feedwater Flow Mismatch channels.

9. Pressurizer High Level

Similar to Item 8 above, the test frequency for this channel will be extended from monthly to quarterly and performed on a STAGGERED TEST BASIS.

10. Reactor Coolant Flow

The current testing interval for this channel is quarterly. PCN 183 will not change this testing interval. It is noted that the WOG guidelines recommend staggered testing of this channel. In order to perform this test at SONGS 1 it is necessary to reduce reactor power and trip a reactor coolant pump. Implementation of staggered testing would require a load reduction to below 50 percent power every 31 days (versus the present requirement of quarterly). SCE considers the potential benefits gained from staggered testing to be outweighed by the increased frequency of load reductions. Therefore, PCN 183 will not implement staggered testing for the Reactor Coolant Low Flow channels.

11. Steam/Feedwater Flow Mismatch

The frequency for testing flow mismatch has been extended from monthly to quarterly and will be performed on a STAGGERED TEST BASIS consistent with the WOG guidelines.

12. Turbine Trip on Low Fluid Oil Pressure

This channel will be tested prior to start-up if not performed in the previous 31 days versus the existing requirement of within the previous 7 days. This requirement is consistent with the WOG guidelines and the Westinghouse STS.

13. Reactor Trip Breakers

As part of the resolution of SEP Topic VI-10.A, this channel, which is currently tested by procedure only, will now be included in the technical specifications. The proposed refueling outage test frequency is based on the SONGS 1 RTB configuration of two breakers in series with no bypass breakers. Without bypass breakers, no capability exists to test the breakers while the unit is operating. Footnote 9 will be included to require that the surveillance tests of the Reactor Trip Breakers independently verify the OPERABILITY of the undervoltage and shunt trip attachments. This footnote is incorporated in response to Generic Letter 85-09.

14. Sequencer Input to Reactor Trip

As part of the resolution of SEP Topic VI-10.A, this channel, which is currently tested by procedure only, will be incorporated into the technical specifications. This channel is essentially the same as the STS Safety Injection Input from ESF channel. The proposed test frequency is consistent with this STS channel.

A footnote has also been added to require compliance with the provisions of Specification 4.1.4 for any portion of this channel required to be OPERABLE by Specification 3.5.5. This footnote is added to ensure compliance with the more restrictive requirements for the ESFAS function of this channel.

EXISTING TECHNICAL SPECIFICATIONS

The existing Technical Specifications are provided in Attachment 1.

PROPOSED TECHNICAL SPECIFICATIONS

The proposed Technical Specifications are provided in Attachment 2. Change bars are used to illustrate the proposed revisions.

SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

As required by 10 CFR 50.91(a)(1), this analysis is provided to demonstrate that the proposed license amendment to implement technical specifications associated with Reactor Protection System instrumentation at SONGS 1 represents a no significant hazards consideration. The analysis provided below is based primarily on the WOG documentation provided in letter OG-158 dated September 3, 1985. In accordance with the three factor test of 10 CFR 50.92(c), implementation of the proposed amendment was analyzed using the following standards and found not to: 1) involve a significant increase in the probability or consequences for 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 proposed change discussed above shall be deemed to constitute 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 an accident previously evaluated?

   Response: No

   Implementation of the proposed changes is expected to result in an acceptable increase in total Reactor Protection System yearly unavailability. This increase, which is primarily due to less frequent surveillance testing, results in an increase of similar magnitude in the probability of an Anticipated Transient Without Scram (ATWS) and in the probability of core melt resulting from an ATWS. Based on the following, these slight increases are judged to be acceptable.

   Implementation of the proposed changes is expected to result in a significant reduction in the probability of core melt from inadvertent reactor trips. This is a result of a reduction in the number of inadvertent reactor trips (0.5 fewer inadvertent reactor trips per unit per year) occurring during testing of RPS instrumentation. This is primarily attributable to testing in bypass and less frequent surveillance.

   The reduction in inadvertent core melt probability is sufficiently large to counter the increase in ATWS core melt probability resulting in an overall reduction in total core melt probability.Incorporation of additional controls not currently in the technical specifications does not impact the probability or consequences of an accident previously evaluated, as these additional surveillances are currently maintained administratively by plant procedures.

   The proposed changes do not result in an increase in the severity or consequences of an accident previously evaluated. Implementation of the proposed changes affects the probability of failure of the RPS but does not alter the manner in which protection is afforded nor the manner in which limiting criteria are established.

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 proposed changes do not result in a change in the manner in which the Reactor Protection System provides plant protection. No change is being made which alters the functioning of the Reactor Protection System (other than in a test mode). Rather, the likelihood or probability of the Reactor Protection System functioning properly is affected as described above. Therefore, the proposed changes do not create the possibility of a new or different kind of accident.

The proposed changes do not involve hardware changes except those necessary to implement testing in bypass. Some existing instrumentation is designed to be tested in bypass and current technical specifications allow testing in bypass. Testing in bypass is also recognized by IEEE Standards. Therefore, testing in bypass has been previously approved and implementation of the proposed changes for testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. Furthermore since the other proposed changes do not alter the functioning of the RPS, the possibility of a new or different kind of accident from any previously evaluated has not been created.

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

Response: No

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. The impact of the reduced testing other than as addressed above is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. Experience at two Westinghouse plants with extended surveillance intervals has shown the initial uncertainty assumptions to be valid for reduced testing.

Implementation of the proposed changes is expected to result in an overall improvement in safety by:

a. 0.5 fewer inadvertent reactor trips per unit. This is due to less frequent testing and testing in bypass which minimizes the time spent in a partial trip condition.

b. Higher quality repairs leading to improved equipment reliability due to longer repair times.

c. Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation. This is due to less frequent distraction of the operator and shift supervisor to attend to instrumentation testing.

Additional Information

The NRC has imposed five conditions on utilities seeking to implement the technical specification changes approved generically as a result of their review of WCAP-10271. These conditions are as follows:

1) Implement a staggered test plan for RPS channels,

2) Develop a common mode failure evaluation program for RPS channels,

3) Install hardware capability for testing in the bypass mode,

4) Cautionary provisions for channels that feed both the RPS and ESFAS, and

5) Confirmation that instrument setpoint methodology accounts for anticipated drift.

SCE's response to each of these conditions is provided in Attachment 3.

SAFETY AND SIGNIFICANT HAZARDS DETERMINATION

Based on the safety analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as 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.

MJT:8337F

ATTACHMENT 1

EXISTING TECHNICAL SPECIFICATIONS

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#

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#

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TABLE 3.5.1-1 (Continued)

TABLE NOTATION

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* With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal. 11/2/84

** A "TRIP" will stop all rod withdrawal.

*** Below the Low Setting setpoint

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 Defeat) Setpoint. 117

Above the P-7 (At Power Reactor Trip Defeat) Setpoint. 12/13/88

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. 83 11/2/84

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 met:

a. The inoperable channel is placed in the tripped condition within 1 hour. 117 12/13/88

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be returned to the untripped condition for up to 2 hours for surveillance testing of other channels per Specification 4.1. 117 12/13/88

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. 83 11/2/84

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.

However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE. 117 12/13/88

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes. 83 11/2/84

ACTION 5 - With the number of OPERABLE channels one less than the Minimum 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. 83 11/2/84

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.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, within one hour reduce THERMAL POWER such that T_{ave} is less than or equal to 551.5°F, and place the rod control system in manual mode.

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ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirements, be in at least HOT STANDBY within 6 hours; however, one channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

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.

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TABLE 4.1.1 (Continued)

TABLE NOTATION

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent. 117 12/13/88

(3) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(4) - The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(5) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Setpoint verification is not applicable.