UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 2 of the San Onofre Nuclear Generating Station	)	Docket No. 50-361 Amendment Application No. 69

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 69 to Facility Operating License NPF-10.

This amendment application consists of Proposed License Change No. NPF-10-275 to Facility Operating License No. NPF-10. Proposed License Change NPF-10-275 is a request to revise Technical Specification 3/4.3.1, "Reactor Protective Instrumentation," to increase the interval for surveillance tests, which are currently performed every 18 months, to each refueling, nominally 24 months and maximum 30 months. As the result of modifying the surveillance interval, changes are proposed to the Reactor Protective instrumentation setpoints in Technical Specification 2.2.1, Table 2.2-1; the High Logarithmic Power Level response time in Technical Specification 3/4.3.1, Table 3.3-2; and the Linear Power Level calibration tolerance in Technical Specification 3/4.3.1, Table 4.3-1.

Subscribed on this 8th day of January, 1990.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: _____
Harold B. Ray
Vice President

Subscribed and sworn to before me this

8th day of January 1990.

OFFICIAL SEAL
MICHELE R. KAHAL
NOTARY PUBLIC - CALIFORNIA
PRINCIPAL OFFICE IN
ORANGE COUNTY
My Commission Exp. Dec. 28, 1990

Michele R. Kahal
Notary Public in and for the
State of California

Charles R. Kocher
James A. Beoletto
Attorneys for Southern
California Edison Company

By: _____

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 3 of the San Onofre Nuclear Generating Station	)	Docket No. 50-362 Amendment Application No. 55

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit Amendment Application No. 55 to Facility Operating License NPF-15.

This amendment application consists of Proposed License Change No. NPF-15-275 to Facility Operating License No. NPF-15. Proposed License Change NPF-15-275 is a request to revise Technical Specification 3/4.3.1, "Reactor Protective Instrumentation," to increase the interval for surveillance tests, which are currently performed every 18 months, to each refueling, nominally 24 months and maximum 30 months. As the result of modifying the surveillance interval, changes are proposed to the Reactor Protective instrumentation setpoints in Technical Specification 2.2.1, Table 2.2-1; the High Logarithmic Power Level response time in Technical Specification 3/4.3.1, Table 3.3-2; and the Linear Power Level calibration tolerance in Technical Specification 3/4.3.1, Table 4.3-1.

Subscribed on this 8th day of January, 1990.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: Harold B. Ray
Harold B. Ray
Vice President

Subscribed and sworn to before me this

8th day of January 1990.

OFFICIAL SEAL
MICHELE R. KAHAL
NOTARY PUBLIC - CALIFORNIA
PRINCIPAL OFFICE IN
ORANGE COUNTY
My Commission Exp. Dec. 28, 1990

Michele R. Kahal
Notary Public in and for the
State of California

Charles R. Kocher
James A. Beoletto
Attorneys for Southern
California Edison Company

By: James A. Beoletto

DESCRIPTION AND SAFETY ANALYSIS
OF
PROPOSED CHANGE NPF-10/15-275

This is a request to revise Technical Specification 2.2.1, "Safety Limits and Limiting Safety System Settings - Reactor Trip Setpoints," and 3/4.3.1, "Reactor Protective Instrumentation."

Existing Specifications:

Unit 2: See Attachment "A"

Unit 3: See Attachment "B"

Proposed Specifications:

Unit 2: See Attachment "C"

Unit 3: See Attachment "D"

Supporting Documentation:

Attachment "E" - Figures

Attachment "F" - Surveillance History Review

Attachment "G" - Instrument Drift Study

Attachment "H" - Plant Protection System Setpoint Evaluation

Attachment "I" - Functional Analysis of Functional Units Not Impacted by Instrument Drift

Attachment "J" - Redundant Instrument Monitoring System (RIMS) Description

Description:

The proposed change would revise Technical Specification 3/4.3.1, "Reactor Protective System (RPS) Instrumentation" to increase the interval for surveillance tests, which are currently performed every 18 months, to each refueling, nominally 24 months and maximum 30 months. As the result of modifying the surveillance interval and changing certain calibration tolerances, the following changes are proposed:

	Affected Technical Specification	Description
1.	2.2.1, Table 2.2-1, "Safety Limits Safety System Settings -Reactor Trip Setpoints"	Modification of certain and Limiting Reactor Protective Instrumentation trip setpoints and allowable values.
2.	3/4.3.1, Table 3.3-2, "Reactor Protective Instrumentation Response Times"	Change in Log Power Level response time.

3.	3/4.3.1, Table 4.3-1, "Reactor Protective Instrumentation Surveillance Requirements"	Change in nuclear instrumentation calorimetric calibration tolerance increase the operating margin. This change will be administratively controlled pending approval of Proposed Change No. NPF-10/15-302 (PCN 302).

Technical Specification 3/4.3.1 provides instrumentation operability and surveillance requirements for the RPS to assure that the functional capability is maintained within its safety analysis design. The Technical Specifications also assure that the integrated operation of each of these systems is consistent with the assumptions used in the accident analysis.

The RPS instrumentation consists of transmitters, calculators, logic, and other equipment necessary to monitor selected Nuclear Steam Supply System (NSSS) conditions and to effect reliable and rapid reactor shutdown (reactor trip) when monitored conditions approach specified safety system limits. The RPS instrumentation functions to protect the core fuel design limits and RCS pressure boundary from anticipated operational occurrences. The instrumentation also provides assistance in limiting conditions for certain design basis events.

Technical Specification surveillance requirement 4.3.1 1 and Table 4.3-1 specify the modes and required frequency for the performance of the Channel Check, Channel Functional Test, and Channel Calibration operations for each RPS instrument channel. The Technical Specification also defines the number of channels of instrumentation required to be operable for each RPS functional unit. For most instruments, Channel Checks are performed at shiftly intervals, Channel Functional Tests at a monthly interval, and Channel Calibrations at an 18 month interval. These surveillance requirements assure that the instrumentation shall be operable and define the action(s) to be taken if the operability requirements are not met.

1.0 Introduction

The Technical Specifications include requirements to calibrate instruments located in areas which are not readily accessible during power operation. San Onofre Units 2 and 3 have both entered a nominal 24 month fuel cycle. Therefore, absent approval of this change, a plant shutdown will be required to perform portions of these surveillances prior to the units' normal refueling date. To avoid the need for an otherwise unnecessary unit shutdown, an evaluation was initiated to determine if surveillances could be modified for consistency with the 24 month fuel cycle.

The evaluation showed that the extension of the Refueling Interval to 24 months could be safely accomplished. This conclusion was arrived based on the following:

- o Comparative Review of Surveillance Testing

  Channel Checks provide a reasonable level of assurance of instrument operability.

- o Surveillance and Corrective Maintenance History Review
  Most instrumentation failures are immediately self revealing.

- o Drift Analysis

  A plant specific instrument drift study provides a statistical analysis of drift characteristics over an extended interval (taken as 30 months).

- o Setpoint Analysis

  Setpoints were revised based on changes to the Safety Analysis setpoints and changes to the trip setpoint calculations.

The proposed change requests the modification of RPS instrumentation trip setpoints and allowable values defined in Table 2.2-1 of Technical Specification 2.2.1, "Safety Limits and Limiting Safety System Settings - Reactor Trip Setpoints." The proposed change also requests the modification of the response time for High Log Power Level. The evaluation performed in support of the surveillance interval extension for the RPS functional units provides a basis for revising the setpoints and response time.

Technical Specification 3/4.3.1, Table 4.3-1 lists the following functional units for RPS instrumentation:

1. Manual Reactor Trip
2. Linear Power - High
3. Logarithmic Power Level - High
4. Pressurizer Pressure - High
5. Pressurizer Pressure - Low
6. Containment Pressure - High
7. Steam Generator Pressure - Low
8. Steam Generator Level - Low
9. Local Power Density - High
10. DNBR - Low
11. Steam Generator Level - High
12. Reactor Protective System Logic
13. Reactor Trip Breakers
14. Core Protection Calculators
15 Control Element Assembly Calculators
16. Reactor Coolant System Flow - Low
17. Seismic - High
18. Loss of Load

Table 4.3-1 provides the required surveillance frequency for the RPS functional units. The Technical Specifications define the Channel Calibration interval, "R", as "at least once per 18 months" for Functional Units 2 through

11 and 14 through 17. This same frequency is specified for Channel Functional Tests for Functional Units 1, 9, 10, 14 and 15. The proposed change would revise the requirement for surveillances from the current 18 months interval to an interval "at least once per refueling," nominally 24 months, or maximum 30 months. The 30 month interval is the maximum 25% extension of the surveillance interval permitted by Technical Specification 4.0.2.

Functional Units 12, "Reactor Protective System Logic," 13, "Reactor Trip Breakers," and 18, "Loss of Load" do not require 18 month surveillances. Therefore, there are no changes requested for the surveillance intervals for these functional units.

2.0 Methodology

Attachment "E", Figure 1 provides a flowchart of the method used to evaluate extending the calibration interval of each of the functional units. As shown on the chart, the surveillance testing and the Surveillance and Maintenance history was reviewed. In the cases where drift did not impact the component, these were adequate to justify the surveillance extension. If transmitter drift was a factor in the trip setpoint calculation for the functional unit, then the trip setpoint was recalculated using SONGS historical drift values. For such cases, the new setpoints were evaluated against the current safety analysis limits to verify consistency. In some cases, trip setpoints were revised to reflect more realistic containment environmental conditions, for pressure and temperature, and revised assumptions. In several cases, safety analysis setpoints were revised. However, no changes to the safety analysis limits were made. This section provides an overview of the evaluation process.

2.1 Comparative Analysis of Surveillance Testing

A comparative analysis between on-line and Refueling Interval surveillances was performed for RPS instruments. The on-line surveillance testing was reviewed for each of the functional units to identify to what extent it encompasses that performed at Refueling Intervals. Attachment "E", Figures 2 through 6 provide a representation of this testing. Table E-1 in Attachment "E" provides additional details regarding surveillance test requirements.

The on-line surveillance testing review was conducted to ensure that all instrument channels under consideration are being monitored on a routine basis to detect inoperable conditions. The objective is to ensure that all operability problems are being identified in a timely manner, and to determine the importance of the 18 month surveillances in maintaining operability.

2.2 Surveillance and Maintenance History Review

A review of the refueling surveillance test results and corrective maintenance history was conducted on the components impacted by the extension of the refueling surveillance interval.

The approach taken was to evaluate the contribution of all of the Preventive Maintenance (PM) and surveillance program elements. The basis for PM interval extension is derived from the Reliability Centered Maintenance (RCM) methodology. This methodology has been submitted by SCE to NRC letter dated September 5, 1989, in support for the request for calibration extension for the Containment Area Monitors (NPF-10/15-266) and Containment High Range Monitors (NPF-10/15-267).

Several reviews are necessary in order to evaluate a surveillance for interval extension. A comparative analysis is performed of the PM surveillances to identify the different aspects of the surveillance tests. A secondary review determines if existing surveillances are detecting equipment problems and if operability is affected. Finally, a review of Corrective Maintenance (CM) history ensures that all problems affecting operability were detected by a condition or time directed means. Condition or time directed means include surveillances, alarms, or indications to the operator. The surveillance becomes a candidate for interval extension if the CM review indicates failures affecting operability are detected and corrected prior to performing the surveillance and if the surveillance history is free of failures affecting operability.

The PM surveillances performed on the RPS instrumentation include the 18 month, quarterly, monthly, daily and shiftly Channel Calibrations, 18 month and monthly Channel Functional Tests, and shiftly Channel Checks. The comparative analysis of surveillances determined that the monthly Channel Functional Test demonstrates the operability of the functional unit loop, exclusive of the transmitter. The shiftly Channel Check provides reasonable assurance of loop and transmitter operability. Attachment "F" provides additional methodology details, for the surveillance and maintenance history review, and the PM surveillance data.

For 9 of these 15 functional units, the functional analysis was adequate to justify the surveillance extension. This is detailed in Attachment "I".

2.3 Drift and Setpoint Analysis

On-line surveillances and the equipment history review provide a high level of assurance that the equipment does not degrade significantly over a fuel cycle. However, minor changes in transmitter calibration may not be detected on-line. To address this aspect of extending the surveillance interval, an analysis of the drift characteristics of pressure, differential pressure, and temperature transmitters was conducted.

The results of this Instrument Drift Study have been provided, in part, in Southern California Edison (SCE) to NRC letter dated May 15, 1989. This letter was submitted in support of PCN 290. PCN 290 requested a one-time exception to the surveillance testing requirements of selected

instrumentation in use at San Onofre Nuclear Generating Station (SONGS) Unit 2. This proposed change has been approved by the Commission.

In lieu of using generic vendor drift data, a plant specific review was performed of the long term drift characteristics of pressure, differential pressure and temperature transmitters. For the Plant Protection System (PPS), this experienced long term drift was statistically adjusted to reflect the maximum drift expected over a fuel cycle (taken as 30 months) at a 95% probability and 95% confidence level. The PPS includes both the Reactor Protective System (RPS) and the Engineered Safety Features Actuation System (ESFAS). The values determined from this effort are more conservative than generic vendor data. For example, the allowable for High Pressurizer Pressure has been changed from 1.88 to 3.75%. Attachment "G" provides additional details of the Instrument Drift Study.

The drift allowables derived from the Instrument Drift Study were incorporated into setpoint calculations. The CPC Uncertainty Analysis was reviewed to determine the impact of the drift allowables. The trip setpoints which resulted from incorporating increased drift allowances were reviewed from an operational perspective. In instances where the revised trip setpoint was judged to result in a potential increase in the number of unnecessary reactor trips, a review of the SONGS Units 2 and 3 trip setpoint calculations and safety analysis setpoints was performed. The trip setpoint calculations assumptions were revised for certain trip functions. The pressurizer pressure trip setpoint calculations were also revised to reflect more realistic containment environmental conditions, for pressure and temperature. In several cases, Safety Analysis setpoints were revised. No changes to safety analysis limits were made. This evaluation is detailed in Attachment H.

3.0 Results

3.1 Comparative Analysis of Surveillance Testing

The review of surveillance testing confirmed that the performance of Channel Functional Tests, monthly, daily or shiftly Channel Calibrations, or daily Channel Checks provide a level of assurance of instrument operability.

Figures 2 through 6, included in Attachment "E", provide Loop Functional Diagrams for all of the RPS functional units. These loop diagrams specifically identify the different aspects of the shiftly and Channel Functional Test and 18 month calibration surveillances. There are two types of on-line surveillances which are performed, the Channel Checks and the Channel Functional Tests.

The Channel Checks are cross channel checks which compare the output of the four independent channels measuring the same process parameter. It provides a reasonable level of assurance that the loop is intact and that the transmitter has not failed. For RPS loops, this cross-channel

check is performed every operating shift.

Channel Functional Tests are performed on portions of the RPS instrument loops. Within Attachment "I" is a more detailed discussion of these tests.

The comparative analysis determined that the importance of the 18 month calibration surveillance was to correct instrument drift, if it was occurring and to "tuneup" the instrument loop components adjusting component values to "nominal" conditions (mid-point at tolerance limits). A review of surveillance maintenance history discussed, in Section 3.2, summarizes the results of any problems found during calibrations. Section 3.3 discusses the results of the drift study that was undertaken to determine the sensitivity of instrument drift to their calibration interval.

Comprehensive surveillance testing of Nuclear Instrumentation, Core Protection Calculators, and Control Element Assembly Calculators is performed on-line. Channel Functional Tests or Channel Checks provide a reasonable level of assurance that the instrumentation has not significantly degraded. Attachment "I" provides a discussion of the results of this review.

3.2 Surveillance and Corrective Maintenance History Review

A confirmatory surveillance and CM history review was also performed for all RPS instruments. This review verified that instrument problems, associated with operability were detectable by operations personnel during the shiftly Channel Checks or during routine monitoring of plant parameters.

This review concluded that no repetitive failures have occurred. No instances were found involving redundant channels during the same time period. Therefore, the safety and operability impacts have been minimal. The data did not identify a correlation between the number of failures and the interval of calibration.

9 of the 15 functional units which required 18 month surveillances, were not impacted by instrument drift. Accordingly, the surveillance and corrective maintenance history review alone is adequate to justify the surveillance extension for the following Functional Units as listed in Technical Specification Table 4.3-1.

1 Manual Reactor Trip
2 Linear Power - High
3 Logarithmic Power Level - High
9 Local Power Density - High
10 DNBR - Low
14 Core Protection Calculators
15 Control Element Assembly Calculators
16 Reactor Coolant System Flow - Low
17 Seismic - High

Attachment "I" provides additional details for these functional units which are not impacted by drift.

The review consisted of an evaluation of instrument function, a review of preventive maintenance surveillances and corrective maintenances, and a comparison of on-line and Refueling Interval surveillances. The historical surveillance review included all surveillance records since the commencement of commercial operation to August 31, 1989. Attachment "F" provides the details of the surveillance review.

Additionally, verification of the assumptions used in the setpoint calculations identified a need to modify the tolerances used in the calibration of two functional units, Linear Power - High, and Logarithmic Power Level - High. The modification necessitated setpoint changes which have been provided in Attachment "H".

3.3 Drift and Setpoint Analysis

3.3.1 Drift Analysis Results:

Instruments for which drift was a factor were further evaluated. The Instrument Drift Study provides a conservative assessment of transmitter performance. A comparison of the results of analysis of the long term drift data is made to existing allowances for long term drift. The results are also compared to revised allowances for long term drift assuming 30 month intervals between calibrations. Use of the revised allowances for long term drift in setpoint and uncertainty calculations provides a sound basis for extending the calibrations interval of these transmitters. Additional details are provided in Attachment "G".

3.3.2 Setpoint Calculations:

Technical Specification 2.2.1, Table 2.2-1, lists Reactor Protective Instrumentation trip setpoints and allowable values. Table 1 provides revised RPS setpoints and allowable values. These were determined based on a comprehensive evaluation which included review of PM and surveillance history, review of the original setpoint calculation assumptions, and incorporation of the SCE experienced long term 95/95 drift values.

The drift study results were evaluated and setpoints determined which:

- o Meet all safety analysis limits
- o Minimize unnecessary actuations

The evaluation resulted in setpoint changes for the following Functional Units identified in Technical Specification Table 2.2.1:

2 Linear Power - High
3 Logarithmic Power Level - High
4 Pressurizer Pressure - High

5 Pressurizer Pressure - Low
6 Containment Pressure - High
7 Steam Generator Pressure - Low
8 Steam Generator Level - Low
11 Reactor Coolant Flow - Low
12 Steam Generator Level - High

These functional unit numbers are consistent with Technical Specification Table 2.2-1 and differ from the numbering system of Technical Specification 4.3-1. Use of the numbering system from Table 2.2-1 is restricted to this Section 3.4 and Table 1.

The SONGS Unit 2 & 3 trip setpoints were revised based on changes to the Safety Analysis Setpoints and changes to the trip setpoint calculations. The Safety Analysis Setpoints were revised for High Pressurizer Pressure, High Containment Pressure and Low Steam Generator Water Level trip functions. These evaluations demonstrate acceptable results when compared to the existing safety analysis limits. The trip setpoint calculations for Low Pressurizer Pressure, High and Low Steam Generator Level and Low Reactor Coolant Flow were revised to improve the operating margin while accounting for the increased transmitter drift and an increase in the allowed tolerance for trip bistable functional testing. The trips setpoint for High Linear Power was revised to incorporate an increase in the allowed tolerance for trip bistable functional testing and a decrease in the daily secondary calorimetric calibration tolerance to maintain the operating margin. The trip setpoint calculation for Low Steam Generator Pressure was revised to account for increased drift and the change in allowed tolerance for trip bistable functional test. The High Logarithmic Trip Setpoint calculation was revised to account for the increase in allowed tolerance for the trip bistable functional test. These changes to the trip setpoint calculations preserve the margin of safety while maintaining adequate operating margins. Operating margin to CPC generated trips has not been changed. Table 1 provides the revised setpoint values for RPS.

During this reanalysis, an inconsistency was noted between the safety analysis and the Technical Specification response time requirements for Logarithmic Power Level - High. Technical Specification Table 3.3-2 is being revised to resolve this inconsistency.

Setpoint calculations were reviewed in an effort to improve the operating margin for High Linear Power. Assuming a reduction in the allowed nuclear instrumentation calorimetric calibration tolerance from 2% to 1% accomplished the desired increase in operating margin. A request to change this tolerance value has been submitted to the Staff as Technical Specification Proposed Change No. NPF-10/15-302 (PCN 302). Administrative control of this change in tolerance, prior to staff approval of the proposed change, will ensure that the desired operating margin will be maintained.

Attachment "H" provides additional details.

Table 1
(Excerpt from Technical Specification Table 2.2-1)

Reactor Protective System Instrumentation Trip Setpoint
(30 Month Calibration Interval)

Functional Unit	Existing Trip Setpoint	Existing Allowable Value	Revised Trip Setpoint	Revised Allowable Value
2. Linear Power Level - High	≤ 110.0%	≤ 111.3%	≤ 110.0%	≤ 111.0%[1]
3. Logarithmic Power Level - High	≤ 0.89%	≤ 0.96%	≤ 0.83%[1]	≤ 0.93%[1]
4. Pressurizer Pressure-High	≤ 2382 psia	≤ 2389 psia	≤ 2375 psia[1,2]	≤ 2385 psia[1,2]
5. Pressurizer Pressure - Low	≥ 1806 psia	≥ 1763 psia	≥ 1740 psia[1,2]	≥ 1700 psia[1,2]
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig	≤ 3.1 psig[1,2]	≤ 3.4 psig[1,2]
7. Steam Generator Pressure-Low	≥ 729 psia	≥ 711 psia	≥ 741 psia[1,2]	≥ 729 psia[1,2]
8. Steam Generator Level-Low	≥ 25.0%	≥ 24.23%	≥ 21.0%[1,2]	≥ 20.0%[1,2]
11. Reactor Coolant Flow - Low c) Step	≤ 6.82 psid/sec	≤ 7.231 psid/sec	≤ 6.82 psid/sec	≤ 7.25 psid/sec[1]
12. Steam Generator Level High	≤ 90.0%	≤ 90.74%	≤ 89.0%[1,2]	≤ 89.7%[1,2]

NOTES:

(1) Revised Trip Setpoints and Allowable Values result from surveillance calibration requirements for the bistables associated with this functional unit.

(2) Revised Trip Setpoints and Allowable Values result from the analysis of long term drift to accommodate 30 month intervals between transmitter calibrations.

3.4 Similar supporting data is provided by the 12 month operating experience of the Redundant Instrument Monitoring System (RIMS). The RIMS was developed by SCE to improve the capability for on-line monitoring of instrumentation. It performs detailed cross-channel checks of monitored instruments and can be used to identify anomalous instrument behavior. The RIMS study evaluated transmitters similar to those analyzed by this proposed change. Trending of RIMS data over the past year has demonstrated the conservative values for drift calculated by SCE. Although not included as a primary basis for this proposed change, RIMS serves to provide an additional level of assurance of instrument operability. Attachment "J" provides a discussion of RIMS and typical plots of instrument performance.

4.0 Conclusions

The proposed change would increase the surveillance interval from 18 months to "Refueling Interval" for a nominal 24 month cycle, and maximum 30 month cycle. The actual time interval between surveillances will be a function of the plant capacity factor for that particular fuel cycle. The equilibrium fuel cycle length will be between 500 and 550 effective full power days (EFPD). Assuming a capacity factor between 70% and 80%, the actual cycle length, and the surveillance interval, would be between 21 and 26 months. Currently, Specification 4.0.2 allows a 25% extension of surveillance intervals which would accommodate uninterrupted operation for the equilibrium cycle length.

This evaluation consisted of a comprehensive review of all of the RPS instrumentation for which a calibration extension was requested. The evaluation consisted of a comparative analysis of all PM surveillances, a PM history review for all instruments, a statistical evaluation of instruments impacted by drift, review of safety analysis, and review of trip setpoint calculations. The study findings support the extension of the calibration interval.

This PM history review verified that instrument problems, associated with operability were detectable by operations personnel during the shiftly Channel Checks or during routine monitoring of plant parameters by operations. This review concluded that no repetitive failures have occurred. No instances were found involving redundant channels during the same time period. Therefore, the safety and operability impacts have been minimal. The data did not identify a correlation between the number of failures and the interval of calibration.

The trip setpoints were revised based on the SCE experienced long term drift for the Reactor Protective System transmitters, changes to the Safety Analysis setpoints and changes to the trip setpoints. These changes to the trip setpoint calculations preserve the margin of safety while maintaining adequate operating margins. Operating margin to CPC generated trips has not been changed.

This evaluation verifies that the revised setpoints, revised response time, and revised assumptions will maintain safety, operability and minimize unnecessary reactor trips by increasing certain operating margins.

The RIMS can be used to identify anomalous instrument behavior. The RIMS study evaluated transmitters similar to those analyzed by this proposed change. Although not included as a primary basis for this proposed change, RIMS serves to provide an additional level of assurance of instrument operability.

Safety Analysis

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

   Response: No.

   The proposed change would revise the Technical Specification to increase the interval for surveillances currently performed every 18 months, to each refueling, nominally 24 months and maximum 30 months.

   SCE has performed a comprehensive evaluation of the effect of extending the calibration interval for Reactor Protective System (RPS) instrumentation. This evaluation was performed for all RPS functional units which required 18 month surveillances. The review consisted of an evaluation of instrument function, comparative analysis of all PM surveillances, a review of preventive maintenance surveillances and corrective maintenance history, a statistical evaluation of instruments impacted by drift, review of the safety analysis, and review of trip setpoint calculations.

   The confirmatory surveillance and CM history review served to identify problems experienced by the RPS instrumentation. This instrumentation included pressure sensors, differential pressure sensors, temperature elements, speed sensors, logic channels, and nuclear power range equipment. This PM history review verified that instrument problems, associated with operability were detectable by operations personnel during the shiftly Channel Checks or during routine monitoring of plant parameters. This review concluded that no repetitive failures have occurred. No instances were found involving redundant channels during the same time period. Therefore, the safety and operability impacts have been minimal. No correlation was found to exist between the number of failures and the interval of calibration.

   Instruments for which drift was a factor were further evaluated. This evaluation consisted of an analysis of plant specific transmitter calibration data for San Onofre Nuclear Generating Station, Units 2&3.

   The long term drift characteristics of pressure, differential pressure and temperature transmitters were determined. This Instrument Drift

Study provides an analysis of the calibration history of instruments impacted by drift which were used in the Reactor Protective System (RPS). The experienced long term drift was statistically adjusted to reflect the maximum drift expected over a fuel cycle, taken as 30 months. The 30 month interval was used in order to account for the 25% extension, to the existing surveillance interval, which is allowed by Technical Specification 4.0.2. The 30 month interval is based on a nominal 24 month fuel cycle. For instrumentation related to RPS, the statistically adjusted drift was determined on a 95/95 basis, i.e., 95% probability and 95% confidence level.

Drift allowances were determined based on this study. The drift allowances were determined by inspecting the 30 month drift values and selecting a value, for each transmitter model, which would bound the 95/95 values.

The impact of the larger drift allowances of RPS instrumentation on setpoint calculations and instrument response times was evaluated and new setpoints calculated, where required. The CPC Uncertainty Analysis was reviewed and it was determined that existing uncertainty allowances remain conservative considering the revised drift allowances. This review verified that the new values of drift were bounded by the existing uncertainty analysis.

Adjustments were made to Safety Analysis setpoints as deemed necessary to address operating concerns. Affected Safety Analyses were reevaluated or reanalyzed. The SONGS Unit 2 & 3 trip setpoints were revised based on changes to the Safety Analysis Setpoints and changes to the trip setpoint calculations. The Safety Analysis Setpoints were revised for High Pressurizer Pressure and High Containment Pressure trip functions. These evaluations demonstrate acceptable results when compared to the existing safety analysis limits. The trip setpoint calculations for Low Pressurizer Pressure, High and Low Steam Generator Level and Low Reactor Coolant Flow were revised to improve the operating margin while accounting for the increased transmitter drift and an increase in the allowed tolerance for trip bistable functional testing. The trip setpoint for High Linear Power was revised to incorporate an increase in the allowed tolerance for trip bistable functional testing and a decrease in the daily secondary calorimetric calibration tolerance to maintain the operating margin. The trip setpoint calculation for Low Steam Generator Pressure and High Steam Generator Level were revised to account for increased drift and the change in allowed tolerance for trip bistable functional test. The High Logarithmic Trip Setpoint calculation was revised to account for the increase in allowed tolerance for the trip bistable functional test. These changes to the trip setpoint calculations preserve the margin of safety while maintaining adequate operating margins. Operating margin to CPC generated trips has not been changed.

During this reanalysis, an inconsistency was noted between the safety analysis and the Technical Specification response time requirements for

Logarithmic Power Level - High. Technical Specification Table 3.3-2 is being revised to resolve this inconsistency.

In order to improve the operating margin for Linear Power Level, the calorimetric calibration tolerance was reduced from 2% to 1%.

The study findings support the extension of the calibration interval. Based on the impact of these factors and the adjustments to setpoints, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of any previously evaluated accident.

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 change does not modify the configuration of the facility or its mode of operation. The proposed change extends the calibration interval for the RPS instrumentation from 18 to 24 months, nominally, and from 22 1/2 to 30 months maximum. Revised setpoints are within the existing safety analysis assumptions. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

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

   Response: No

   The proposed change increases the calibration interval to a maximum of 30 months and revises certain Limiting Safety System Settings.

   The surveillance and corrective maintenance history review has confirmed that problems are identified as the result of shiftly Channel Checks and Channel Functional Tests. This review confirmed that there have been no operability failures for these functional units.

   Revised drift allowables and calibration review results were evaluated and increased allowables were incorporated into setpoint calculations to accommodate experienced values over the increased interval. This review verified that the new values of drift were bounded by the existing uncertainty analysis.

   The impact of larger drift allowances was assessed. Adjustments were made to Safety Analysis setpoints as deemed necessary to address operating concerns. Affected Safety Analyses were reevaluated or reanalyzed. The SONGS Unit 2 & 3 trip setpoints were revised based on changes to the Safety Analysis Setpoints and changes to the trip setpoint calculations. The Safety Analysis Setpoints were revised for High Pressurizer Pressure and High Containment Pressure trip functions.

These evaluations demonstrate acceptable results when compared to the existing safety analysis limits. The trip setpoint calculations for Low Pressurizer Pressure, High and Low Steam Generator Level and Low Reactor Coolant Flow were revised to improve the operating margin while accounting for the increased transmitter drift and an increase in the allowed tolerance for trip bistable functional testing. The trips setpoint for High Linear Power was revised to incorporate an increase in the allowed tolerance for trip bistable functional testing and a decrease in the daily secondary calorimetric calibration tolerance to maintain the operating margin. The trip setpoint calculation for Low Steam Generator Pressure and High Steam Generator Level were revised to account for increased drift and the change in allowed tolerance for trip bistable functional test. The High Logarithmic Trip Setpoint calculation was revised to account for the increase in allowed tolerance for the trip bistable functional test. These changes to the trip setpoint calculations preserve the margin of safety while maintaining adequate operating margins. Operating margin to CPC generated trips has not been changed. The conclusions of the accident analysis were not revised as a result of these setpoint changes.

During this reanalysis, an inconsistency was noted between the safety analysis and the Technical Specification response time requirements for Logarithmic Power Level - High. Technical Specification Table 3.3-2 is being revised to resolve this inconsistency.

In order to improve the operating margin for Linear Power Level, the calorimetric calibration tolerance was reduced from 2% to 1%.

This evaluation verifies that the revised setpoints and response time will maintain safety and minimize unnecessary reactor trips by increasing certain operating margins.

The proposed change will not involve a result in a significant reduction in the accident and transient analysis margin of safety for RPS instrumentation.

SAFETY AND SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above 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; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT A

SONGS UNITS 2

EXISTING TECHNICAL SPECIFICATIONS

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	≤ 110.0% of RATED THERMAL POWER	≤ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia	≥ 1763 psia
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia	≥ 711 psia
8. Steam Generator Level - Low	≥ 25% (4)	≥ 24.23% (4)
9. Local Power Density - High (5)	≤ 21.0 kw/ft	≤ 21.0 kw/ft
10. DNBR - Low	≥ 1.31 (5)	≥ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	≤ 0.22 psid/sec (6)(8)	≤ 0.231 psid/sec (6)(8)
b) Floor	≥ 13.2 psid (6)(8)	≥ 12.1 psid (6)(8)
c) Step	≤ 6.82 psid (6)(8)	≤ 7.231 psid (6)(8)
12. Steam Generator Level - High	≤ 90% (4)	≤ 90.74% (4)
13. Seismic - High	≤ 0.48/0.60 (7)	≤ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA, or certain intermediate steam line breaks. This trip initiates a reactor trip at a linear power level of less than or equal to 111.3% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of less than or equal to 0.96% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2389 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to 1763 psia. This trip's setpoint may be manually decreased, to a minimum value of 300 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before emergency feedwater is required.

Local Power Density-High

The Local Power Density-High trip is provided to prevent the linear heat rate (kw/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	≤ 0.40 seconds*
3. Logarithmic Power Level - High	≤ 0.45 seconds*
4. Pressurizer Pressure - High	≤ 0.90 seconds
5. Pressurizer Pressure - Low	≤ 0.90 seconds
6. Containment Pressure - High	≤ 0.90 seconds
7. Steam Generator Pressure - Low	≤ 0.90 seconds
8. Steam Generator Level - Low	≤ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.68 seconds*
b. CEA Positions	≤ 0.68 seconds**
c. CEA Positions: CEAC Penalty Factor	≤ 0.53 seconds
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.68 seconds*
b. CEA Positions	≤ 0.68 seconds**
c. Cold Leg Temperature	≤ 0.68 seconds##
d. Hot Leg Temperature	≤ 0.68 seconds##
e. Primary Coolant Pump Shaft Speed	≤ 0.68 seconds#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.68 seconds
g. CEA positions: CEAC Penalty Factor	≤ 0.53 seconds

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4), R(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M,(12)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4),S(7) R(4,5),M(8)	M(11),R(6)	1, 2
15. CEA Calculators	S	R	M,R(6)	1, 2
16. Reactor Coolant Flow-Low	S	R	M	1, 2
17. Seismic-High	S	R	M	1, 2
18. Loss of Load	S	N.A.	M	1 (9)

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) - Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC delta T power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistant with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

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

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - Above 55% of RATED THERMAL POWER.

(10) - Deleted.

(11) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(12) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

MAY 1 6 1983

ATTACHMENT B

SONGS UNITS 3

EXISTING TECHNICAL SPECIFICATIONS

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	≤ 110.0% of RATED THERMAL POWER	≤ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia	≥ 1763 psia
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia	≥ 711 psia
8. Steam Generator Level - Low	≥ 25% (4)	≥ 24.23% (4)
9. Local Power Density - High (5)	≤ 21.0 kw/ft	≤ 21.0 kw/ft
10. DNBR - Low	≥ 1.31 (5)	≥ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	≤ 0.22 psid/sec (6)(8)	≤ 0.231 psid/sec (6)(8)
b) Floor	≥ 13.2 psid (6)(8)	≥ 12.1 psid (6)(8)
c) Step	≤ 6.82 psid (6)(8)	≤ 7.231 psid (6)(8)
12. Steam Generator Level - High	≤ 90% (4)	≤ 90.74% (4)
13. Seismic - High	≤ 0.48/0.60 (7)	≤ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA, or certain intermediate steam line breaks. This trip initiates a reactor trip at a linear power level of less than or equal to 111.3% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of less than or equal to 0.96% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2389 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to 1763 psia. This trip's setpoint may be manually decreased, to a minimum value of 300 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor conlant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before emergency feedwater is required.

Local Power Density-High

The Local Power Density-High trip is provided to prevent the linear heat rate (kw/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	≤ 0.40 seconds*
3. Logarithmic Power Level - High	≤ 0.45 seconds*
4. Pressurizer Pressure - High	≤ 0.90 seconds
5. Pressurizer Pressure - Low	≤ 0.90 seconds
6. Containment Pressure - High	≤ 0.90 seconds
7. Steam Generator Pressure - Low	≤ 0.90 seconds
8. Steam Generator Level - Low	≤ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.68 seconds*
b. CEA Positions	≤ 0.68 seconds**
c. CEA Positions: CEAC Penalty Factor	≤ 0.53 seconds
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ 0.68 seconds*
b. CEA Positions	≤ 0.68 seconds**
c. Cold Leg Temperature	≤ 0.68 seconds##
d. Hot Leg Temperature	≤ 0.68 seconds##
e. Primary Coolant Pump Shaft Speed	≤ 0.68 seconds#
f. Reactor Coolant Pressure from Pressurizer	≤ 0.68 seconds
g. CEA positions: CEAC Penalty Factor	≤ 0.53 seconds

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4), R(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M,(12)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4), S(7), R(4,5), M(8)	M(11),R(6)	1, 2
15. CEA Calculators	S	R	M,R(6)	1, 2
16. Reactor Coolant Flow-Low	S	R	M	1, 2
17. Seismic-High	S	R	M	1, 2
18. Loss of Load	S	N.A.	M	1 (9)

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) - Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC delta T power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

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

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - Above 55% of RATED THERMAL POWER.

(10) - Deleted.

(11) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC per Specification 2.2.2.

(12) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.