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PG&E Letter DCL-03-011

U.S. Nuclear Regulatory Commission
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Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 03-01
Revision to Technical Specification 3.3.1, "Reactor Trip System (RTS) Instrumentation"

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively. The enclosed license amendment request (LAR) proposes to revise Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation."

This LAR proposes to revise Surveillance Requirements (SRs) 3.3.1.2 and 3.3.1.3 of TS 3.3.1, "Reactor Trip System Instrumentation." The change to SR 3.3.1.2 responds to a concern raised by Westinghouse Technical Bulletin ESBU-TB-92-14-R1, "Decalibration Effects of Calorimetric Power Measurements on the NIS High Power Reactor Trip at Power Levels less than 70 % RTP," dated February 6, 1996, and is consistent with NRC approved Industry/Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-371. The change to SR 3.3.1.3 is editorial in nature.

These SRs are only applicable to Function 2.a and Function 6 of TS Table 3.3.1-1, "Reactor Trip System Instrumentation."

PG&E is submitting this LAR in conjunction with an industry consortium of six plants as a result of a mutual agreement known as Strategic Teaming and Resource Sharing (STARS). The STARS group consists of the six plants operated by TXU Generation Company LP, Ameren Union Electric Company, Wolf Creek Nuclear Operating Corporation, Pacific Gas and Electric Company, STP Nuclear Operating Company, and Arizona Public Service Company. A similar LAR has been approved for Ameren Union Electric Company's Callaway Plant and Wolf Creek Nuclear Operating Corporation's Wolf Creek Plant.

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Enclosure 1 contains a description of the proposed change, the supporting technical analyses, and the no significant hazards consideration determination. Enclosures 2 and 3 contain marked-up and revised (clean) TS pages, respectively. Enclosure 4 provides the marked-up TS Bases changes for information only. TS Bases changes are provided for information only and will be implemented pursuant to TS 5.5.14, "Technical Specifications Bases Control Program." Enclosure 5 provides a copy of the plant-specific evaluation based on the guidance in Westinghouse Technical Bulletin ESBU-TB-92-14-R1 to determine the power level below which power range channel adjustments in a decreasing power direction become a concern.

PG&E has determined that this LAR does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

The change in this LAR is not required to address an immediate safety concern. PG&E requests approval of this LAR no later than March 1, 2004. PG&E requests the LAR be made effective upon NRC issuance, to be implemented within 60 days from the date of issuance.

Sincerely,

Greg Rueger

Gregory M. Rueger
Senior Vice President – Generation and Chief Nuclear Officer

nih/4259
Enclosures

cc:	Edgar Bailey, DHS
	Ellis W. Merschoff
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	Diablo Distribution
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	Docket No. 50-275 Facility Operating License No. DPR-80
Diablo Canyon Power Plant Units 1 and 2	Docket No. 50-323 Facility Operating License No. DPR-82

AFFIDAVIT

Gregory M. Rueger, of lawful age, first being duly sworn upon oath states that he is Senior Vice President - Generation and Chief Nuclear Officer of Pacific Gas and Electric Company; that he has executed license amendment request LAR 03-01 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

Greg Rueger

Gregory M. Rueger
Senior Vice President - Generation and Chief Nuclear Officer

Subscribed and sworn to before me this 6th day of February 2003.

Elizabeth J. Diamond

Notary Public
County of San Francisco
State of California

ELIZABETH J. DIAMOND
COMM. #1352219
NOTARY PUBLIC-CALIFORNIA
SAN FRANCISCO COUNTY
My Comm. Expires May 16, 2006

1.0 Description

This License Amendment Request (LAR) would revise Surveillance Requirements (SRs) 3.3.1.2 and 3.3.1.3 of Technical Specification (TS) 3.3.1, "Reactor Trip System Instrumentation." The change to SR 3.3.1.2 responds to a concern raised by Westinghouse Technical Bulletin ESBU-TB-92-14-R1, "Decalibration Effects of Calorimetric Power Measurements on the NIS High Power Reactor Trip at Power Levels less than 70 % RTP," dated February 6, 1996, and is consistent with NRC approved Industry/Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-371. The change to SR 3.3.1.3 is editorial in nature. These SRs are only applicable to Function 2.a, "Power Range Neutron Flux - High," and Function 6, "Overtemperature ΔT," of TS Table 3.3.1-1, "Reactor Trip System Instrumentation."

The associated SR 3.3.1.2 Bases changes provide a summary justification for the TS surveillance change and clarify when channel adjustments must be made. Associated SR 3.3.1.3 Bases changes reflect the relocation of its Note 1.

A plant-specific evaluation based on the guidance in Westinghouse Technical Bulletin ESBU-TB-92-14-R1 was performed to determine the power level below which power range channel adjustments in a decreasing power direction become a concern. This evaluation is documented in Enclosure 5 of this LAR.

2.0 Proposed Change

SR 3.3.1.2 currently states: "Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output." Note 1 of SR 3.3.1.2 currently states: "Adjust NIS channel if absolute difference is > 2%." Note 2 of SR 3.3.1.2 currently states: "Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP, but prior to exceeding 30% RTP."

This LAR would revise SR 3.3.1.2 to move the content of Note 1 to the body of the SR. SR 3.3.1.2 would be revised to state:

> "Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculation results exceed power range channel output by more than + 2% RTP."

This LAR would also revise SR 3.3.1.3 to move the content of Note 1 to the body of the SR for consistency with the change made to SR 3.3.1.2. SR 3.3.1.3 would be revised to state:

"Compare results of the incore detector measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%."

Enclosures 2 and 3 contain marked-up and revised (clean) TS pages, respectively. Enclosure 4 provides the marked-up TS Bases changes for information only.

The Bases changes provide a summary justification for the surveillance changes and clarify when channel adjustments must be made. In addition, the basis for not requiring performance of a secondary power calorimetric measurement until reaching 15 percent rated thermal power (RTP) is clarified. The power level of 15 percent RTP was chosen as the minimum power level for the NIS power range daily surveillance based on the Westinghouse nuclear steam supply system design basis capability of being able to achieve stable control system operation in the automatic control mode.

Enclosure 5 provides a copy of the plant-specific evaluation based on the guidance in Westinghouse Technical Bulletin ESBU-TB-92-14-R1 to determine the power level below which power range channel adjustments in a decreasing power direction become a concern.

3.0 Background

3.1 System Description of NIS Channels

The primary purpose of the NIS is to measure the number of neutrons leaking from the core to:

- Provide indication of reactor power level and rate of change
- Provide indication of power distribution within the core
- Provide reactor power level indication following a design basis accident
- Supply nuclear power control signals to the Full-Length Rod Control System
- Supply nuclear power protection signals to the Reactor Protection System

The NIS power range channels provide indications of reactor power to the Reactor Trip System (RTS). This is for the RTS trip of the reactor on power range high neutron flux. The daily NIS power range surveillance of SR 3.3.1.2 is to ensure that the power range channels accurately reflect the reactor power based on the calorimetric heat balance equation. A low power indication in the NIS power range channels would nonconservatively affect the RTS, and therefore the protection of the reactor.

3.2 Decalibration Effects of Calorimetric Power Measurements on NIS

The changes to SR 3.3.1.2 and SR 3.3.1.3 address an issue identified in Westinghouse Technical Bulletin ESBU-TB-92-14-R1, "Decalibration Effects of Calorimetric Power Measurements on the NIS High Power Reactor Trip at Power Levels less than 70 % RTP," dated February 6, 1996.

Westinghouse Technical Bulletin ESBU-TB-92-14-R1 identified potential effects of decalibrating the NIS power range channels at part-power operation. The decalibration can occur due to the increased uncertainty of the secondary side power calorimetric when performed at power levels less than 45 percent RTP. This power level was determined on a plant-specific basis following the guidance of the Technical Bulletin, and the calculation is included in Enclosure 5 of this LAR. When NIS channel indication is reduced to match calculated power, the decalibration potentially results in a nonconservative bias. The proposed amendment to the TS removes the requirement to adjust the NIS power range channels in the decreasing power direction when the indicated power is greater than the calorimetric heat balance calculation by an absolute difference of more than 2 percent RTP.

The primary error contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement, which is determined by a ΔP measurement across a feedwater venturi. While the measurement uncertainty remains constant in ΔP span as power decreases, when translated into flow, the uncertainty increases as a square term. Therefore, a 1 percent flow error at 100 percent power can approach a 10 percent flow error at 30 percent RTP even though the ΔP error has not changed. Westinghouse Technical Bulletin ESBU-TB-92-14-R1 discusses how the potential effects of this error increase at lower power levels. In the example presented, for a 10 percent error in the secondary side power calorimetric, the NIS power range could be sufficiently biased in the nonconservative direction to preclude a reactor trip within the assumptions of the safety analyses. One affected safety analysis is the rod withdrawal at power, as discussed in the Updated Final Safety Analysis Report (UFSAR) Section 15.2.2.

Westinghouse Technical Bulletin ESBU-TB-92-14-R1 recommends that caution be exercised if the NIS power range channels are adjusted in the decreasing power direction when the power range channels indicate a higher power than the secondary side power calorimetric measurement at low power levels. This recommendation is in conflict with the power range daily SR 3.3.1.2, which currently requires channel adjustment whenever the absolute difference is greater than 2 percent and the plant is operating at greater than or equal to 15 percent RTP.

Diablo Canyon Power Plant (DCPP) initially evaluated Westinghouse Technical Bulletin ESBU-TB-92-14-R1 in 1996. Interim administrative controls were implemented at that time to address the conflict within the TS. The concerns expressed by the Westinghouse Technical Bulletin ESBU-TB-92-14-R1 regarding the issue of increased calorimetric uncertainties at reduced power levels were understood by reactor engineering personnel, and DCPP operating procedures were revised to account for the increased uncertainty. This was done by requiring reduced NIS trip setpoints for adjustments made below 45% RTP.

It has been determined a nominal 109 percent RTP high flux trip setpoint is adequately conservative for NIS adjustments performed at or above 45 percent RTP. For NIS adjustments at lower powers, the trip setpoints were set more conservatively to address the effects discussed in the Technical Bulletin. Originally, the power range high power reactor trip was reduced from 109 percent to 85 percent as recommended by the Technical Bulletin. The plant procedures have been revised to require setting trip setpoints to 76 percent prior to power ascension (this was determined to be adequate to accommodate an NIS adjustment at 15 percent RTP).

3.3 Purpose for Proposed Amendments

These changes are requested to help avoid the potential nonconservative effects of decalibrating the NIS power range channels at part-power operation, as discussed in Westinghouse Technical Bulletin ESBU-TB-92-14-R1. In addition, moving the content of SR 3.3.1.2 Note 1 to the body of the SR and using the "power range channel" terminology would provide consistency with NUREG-1431, “Standard Technical Specifications, Westinghouse Plants,” Revision 2. SR 3.3.1.3 would also be revised to move the content of its Note 1 to the body of the SR for consistency. Associated SR 3.3.1.2 Bases changes clarify when channel adjustments must be made. Associated SR 3.3.1.3 Bases changes reflect the relocation of its Note 1.

4.0 Technical Analysis

The following analysis assesses the impact of the proposed NIS power range daily surveillance change on the licensing and design basis and demonstrate that the change will not adversely affect the safe operation of the plant.

4.1 Power Range Instrumentation and RTS Design Functions

When operating at or above 15 percent RTP, each power range channel is checked on a daily basis to see if its indication matches the thermal power calculation results based on the secondary side heat balance (i.e., calorimetric). Currently, if the absolute difference exceeds 2 percent RTP, then a normalization or calibration is performed by adjusting the gain of each channel summing amplifier such that the NIS power range channel output matches the calorimetric heat balance calculated power. The amplifier output (0 percent to 120 percent RTP) provides the input signals to the associated channel reactor trip, permissive, and control interlock bistables as well as the associated power level indicators. Therefore, the proposed change to the NIS power range daily surveillance potentially impacts the following power range NIS functions:

- power range indications;
- power range RTS trip functions (high flux high setpoint, high flux low setpoint, and high positive rate) and permissive functions (P-8, P-9, and P-10);
- power range control system functions (control interlock C-2 power range high flux rod stop and automatic reactor control system nuclear power input); and
- miscellaneous alarm functions (power range channel deviation and Quadrant Power Tilt Ratio alarms).

Power Range Indications

Reactor power is monitored by the plant operators to assure that the unit is operated within the limits of the operating license and the safety analyses. The revision to the criteria for implementation of the daily surveillance will have a conservative effect on the power range channel indication (i.e., indicated power will be verified to be greater than or equal to actual power or will be adjusted in the increasing power direction). With regard to the core safety limits, reactor power is one of four operating parameters with uncertainties explicitly used in the Improved Thermal Design Procedure (ITDP). As discussed in UFSAR Section 4.4.1.1.2, plant parameter uncertainties, including a reactor power uncertainty, are used to statistically derive a design departure from nucleate boiling ratio (DNBR) limit which must be met in those plant safety analyses that use the ITDP. For those non-ITDP accidents listed in UFSAR Table 15.1-4, a ± 2 percent RTP power uncertainty is assumed in the analyses' initial conditions, as discussed in UFSAR Section 15.1.2.2. Plant-specific setpoint calculations demonstrate that the secondary side power calorimetric measurement uncertainty at full power conditions is less than this ± 2 percent RTP assumption. Since these plant-specific setpoint calculations are not invalidated by the proposed power range surveillance method change, the uncertainty assumption of ± 2 percent RTP continues to be a bounding allowance for the core safety limits and safety analyses. Therefore, the NIS power range indications are not adversely impacted by the proposed change.

Power Range RTS Trip Functions and Permissive Functions

The setpoint uncertainty assumptions associated with the methodology used to calculate the RTS trip setpoints account for the NIS power range daily surveillance specified by the TS. The setpoint uncertainty calculations demonstrate conservative margin between the associated nominal trip setpoints and the corresponding safety analysis limits (SAL). Since the daily surveillance will continue to be performed and the maximum allowed nonconservative deviation will continue to be less than or equal to 2 percent RTP, the power range high and low reactor trip setpoint calculations and applicable safety analysis limits are not affected by the surveillance change. With respect to the power range high positive rate reactor trip, this trip function is generated by time-delay relative comparison circuits. As such, the NIS power range high positive rate trip is not affected by the proposed change.

One potential nonconservative impact on the NIS RTS functions could occur when the power range channel indication is greater than the calorimetric heat balance calculated power during a unit shutdown. In this situation, the proposed change to no longer require adjustment of the power range channels in the decreasing power direction could delay the reset of permissive P-10. Reset of permissive P-10 (at $\approx$ 10 percent RTP) is required to enable the power range high neutron flux low setpoint reactor trip function which provides reactor protection for uncontrolled reactivity transients from subcritical and low power conditions (i.e., less than 10 percent RTP). There is no adverse impact on the P-10 reset function since the potential conditions with respect to actual and indicated reactor power remain bounded by the conditions assumed in the UFSAR Chapter 15 events. Diverse protection is also provided by the power range high positive rate, OTΔT, OPΔT, high pressurizer pressure, and high pressurizer water level reactor trip functions. In addition, administrative controls which require trip setpoint changes prior to channel adjustments in the decreasing power direction will continue to be imposed as detailed in the SR 3.3.1.2 Bases. Therefore, the NIS power range RTS trip functions and permissive functions are not adversely affected by the proposed change.

Power Range Control System Functions

The power range channels also provide input to the C-2 control interlock (i.e., power range high flux rod stop), which blocks automatic and manual control rod withdrawal, and provide the nuclear power input signal to the power mismatch circuits associated with automatic reactor control as shown in UFSAR Figure 7.7-1.

These control system functions are not required for plant safety, as discussed in UFSAR Section 7.7. Nevertheless, the proposed NIS power range daily surveillance change continues to limit the maximum allowed nonconservative calibration error; therefore, the change will not adversely impact the NIS power range control system functions.

Miscellaneous Alarm Functions

Miscellaneous alarm functions also use input signals from the NIS power range channels. These alarm functions are the power range channel deviation and Quadrant Power Tilt Ratio (QPTR) alarms. The channel deviation and QPTR alarms are generated by comparison of the power range channel output signals. Since these are relative comparisons between channels, these NIS power range alarm functions are not adversely affected by the proposed daily calibration change.

4.2 Power Range Instrumentation Safety Analysis Basis

LOCA and LOCA-Related Analyses

The following loss-of-coolant accident (LOCA) and LOCA-related analyses are not adversely affected by the proposed change to the NIS power range daily surveillance:

- large and small break LOCA;
- reactor vessel and loop LOCA blowdown forces;
- post-LOCA long term core cooling subcriticality;
- post-LOCA long term core cooling minimum flow; and
- hot leg switchover to prevent boron precipitation.

These analyses either do not credit any NIS functions or are limiting based on assuming full power conditions, which are not impacted since the full power calorimetric accuracy remains within ± 2 percent RTP. The proposed amendment does not affect the normal plant operating parameters, the safeguards systems actuation and accident mitigation capabilities important to LOCA mitigation, or the assumptions used in the LOCA-related analyses. The surveillance change does not create conditions more limiting than those assumed in these analyses. In addition, the proposed amendment does not affect the steam generator tube rupture (SGTR) analysis methodology or assumptions, and it does not alter the SGTR event analysis results.

Non-LOCA Related Analyses

The non-LOCA safety analyses presented in UFSAR Chapter 15 are not adversely affected by the proposed NIS power range daily surveillance change. As discussed in the UFSAR Section 15.1.2.2, the non-LOCA safety analyses assume a ± 2 percent uncertainty in reactor power to bound the calorimetric calculation. The proposed NIS surveillance requirements ensure that the potential range of uncertainty associated with a low power calorimetric NIS adjustment remain bounded by the UFSAR. Adjustment of the NIS power range channels in the upward direction could result in an NIS power channel output that is more than 2 percent greater than the actual power level. A plant transient from this condition would result in a less limiting power increase since the power range neutron flux reactor trip setpoint high setting (SAL = 118 percent RTP) would be reached earlier than what is assumed in the UFSAR.

As discussed above, the increased uncertainty in the calorimetric calculation at low power levels only affects the NIS power range channel output. There is no change in any plant operating parameter (e.g., RCS T_{avg} and pressurizer pressure) and no impact on the assumed initial thermal margin with respect to core power level, such that the UFSAR events which credit non-NIS reactor protection functions (e.g. OTΔT and high pressurizer pressure) are not impacted. Therefore, the conclusions presented in the UFSAR remain valid.

P-10 Reset Evaluation

The P-10 reset function and associated setpoint (10 percent RTP) establish which power range neutron flux reactor trip setpoint is credited for mitigation in the DCPP safety analysis. This function is bounded in the DCPP safety analysis based on crediting the power range neutron flux reactor trip low setting for transients which are assumed to initiate below 10 percent RTP, and crediting the power range neutron flux reactor trip high setting for transients which are assumed to initiate above 10 percent RTP. The safety analyses assume a nominal P-10 reset value of 10 percent RTP, since the uncertainty of ± 2 percent RTP is conservatively bounded by the assumed initial power level and the credited NIS reactor trip setpoint. There are two cases with respect to the P-10 reset function which need to be evaluated due to the increased uncertainty associated with a low power calorimetric NIS adjustment. The first case is that the actual reactor power is greater than the NIS power range channel output, such that the P-10 reset occurs with actual reactor power greater than 10 percent RTP. Conversely, the second case is that the actual reactor power is less than the NIS power range channel output, such that the actual reactor power has decreased below 10 percent RTP and the P-10 reset has not yet occurred. This evaluation assumes the case conditions are obtained during a power decrease associated with a plant shutdown. However, these case conditions could also be obtained during a subsequent plant power ascension. For the purposes of evaluating a transient from these two potential case conditions, only the initial conditions with respect to actual and indicated power matter, not how they were obtained. The following evaluation concludes that both of these potential cases remain bounded by the safety analysis.

In the first case, the P-10 reset has occurred and the power range neutron flux reactor trip low setting is automatically enabled, however, the actual reactor power remains above 10 percent RTP. There are only two DCPP safety analyses that credit the power range neutron flux reactor trip low setting function. These are the Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from a Subcritical Condition, described in UFSAR Section 15.2.1, and the Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection), described in UFSAR Section 15.4.6. In both of these safety analyses, the core is assumed to be critical at an initial reactor power level (typically 10^{-5} percent RTP) that is well below the value at which nuclear heating effects are predicted to occur (typically 10^{-3} percent RTP). This allows a significant increase in the reactor power level before any nuclear heating and doppler reactivity feedback occurs and maximizes the net nuclear power increase rate for a given reactivity insertion. The rate at which nuclear power is increasing when the reactor trip occurs (along with the associated delays) determines the magnitude that the nuclear power increases above the trip setpoint before the negative trip reactivity terminates the increase. The safety analyses verify that the power range neutron flux reactor trip low setting (SAL = 35 percent RTP) occurs in time to maintain the peak reactor power and associated DNBR values within acceptable limits. For this P-10 reset case, the actual initial reactor power level would be greater than 10 percent RTP. Therefore, the reactor trip low setting (intermediate range) would be reached much earlier and the nuclear power overshoot for this case would remain bounded by the RCCA Bank Withdrawal from a Subcritical Condition or RCCA Ejection events analyzed in the DCPP UFSAR.

In the second case, the actual nuclear power would be less than 10 percent RTP, but the power range neutron flux reactor trip low setting has not been automatically enabled. The DCPP safety analyses bound transients initiated at a power level $\geq$ 10 percent RTP, based on crediting the power range neutron flux reactor trip high setting (SAL = 118 percent RTP). This safety analysis includes the uncertainties, which bound the $\pm$ 2 percent surveillance accuracy for adjusting the NIS power range channel output. The safety analyses verify that the power range neutron flux reactor trip high setting occurs in time to maintain the peak reactor power and associated DNBR values within acceptable limits. For this case, the NIS channel output is greater than the actual power. This offset between actual and indicated power levels represents a proportional gain bias which cannot become smaller as the actual power level increases. Therefore, during a power increase transient, the NIS power range channel output would reach the neutron flux reactor trip high setting of 118 percent RTP, well before the actual reactor power reaches this value. Therefore, the actual reactor power and associated DNBR values for an event from this condition would remain bounded by the DCPP UFSAR.

In conclusion, there is no adverse impact on the P-10 reset function, since the potential conditions with respect to actual and indicated power remain bounded by the conditions assumed in the DCPP UFSAR.

4.3 Related Design Considerations

Mechanical Components and Systems

The surveillance change as described does not affect the reactor coolant system component integrity or the ability of any system to perform its intended safety function. The proposed amendment does not affect the integrity of plant auxiliary fluid systems or the ability of those auxiliary systems to perform their design functions.

Instrumentation and Control Protection and Control Systems

With the specific exception of the NIS power range reactor trip and indication functions discussed above, the proposed NIS power range daily surveillance change does not involve other electrical systems, components, or instrumentation considerations. Direct effects as well as indirect effects on equipment important to safety have been considered. Indirect effects include conditions or activities which involve non-safety-related electrical equipment which may affect Class 1E, post-accident monitoring, or plant control systems. Consideration has been given to seismic and environmental qualification, design and performance criteria per Institute of Electrical and Electronics Engineers (IEEE) standards, functional requirements, and TS.

The proposed amendment does not affect systems that respond to design transients or maintain the margin to trip setpoints, nor does the proposed amendment affect the low temperature overpressure mitigation system.

RTS and ESFAS Setpoints

With the specific exception of the NIS power range reactor trip and indication functions discussed above, the proposed change to the NIS power range daily surveillance does not affect the RTS or the Engineered Safety Feature Actuation System (ESFAS) setpoints. This proposed change does not alter the current trip setpoints or instrument operability requirements identified in the TS. It will assure the continued operability of the NIS power range high neutron flux high setpoint reactor trip function at part-power conditions consistent with the safety analysis assumptions. Therefore, the proposed amendment has no effect on the RTS and ESFAS safety functions.

Other Safety-Related Areas and Analyses

The following safety-related areas and analyses are not affected by the proposed surveillance change:

- containment integrity analyses (short term/long term LOCA release);
- main steamline break mass and energy release;
- radiological analyses; and
- emergency response procedures.

4.4 Power Range Instrumentation Probabilistic Risk Assessment (PRA) Evaluation

The power range instrumentation is not explicitly modeled in the PRA model. The implicit assumption is that the instrumentation is designed and is operated per licensing and design requirements. Therefore, there is no impact on the

DCPP PRA model on the basis that functional capabilities of the instrumentation will continue to be demonstrated by the surveillance requirements listed in TS Table 3.3.1-1 for Function 2.a. Revising SR 3.3.1.2 for Function 2.a of TS Table 3.3.1-1 will continue to assure that the NIS power range channels perform their intended function. The editorial change to SR 3.3.1.3 has no PRA impact.

4.5 Power Range Instrumentation Summary/Conclusion

The proposed amendment changes the NIS power range daily surveillance requirement by only requiring a calibration adjustment when power range indicated power is less than the calculated secondary calorimetric power by greater than 2 percent RTP. The proposed surveillance change will assure the continued performance of the NIS power range high neutron flux high setpoint reactor trip function consistent with the safety analysis assumptions. Deletion of the requirement to adjust the power range channels in a decreasing power direction when their output is greater than calorimetric heat balance calculated power allows the channels to not be adjusted in a nonconservative direction at part-power conditions. This prevents the introduction of an error that has not been accounted for in the setpoint uncertainty calculations and the safety analyses associated with the power range high neutron flux high setpoint reactor trip function.

The analyses presented above assess the potential impact of the proposed daily surveillance change on applicable safety analyses and NIS power range indications, RTS trip functions and permissives, control system functions, and alarms. The assessments demonstrate that the change will not adversely affect the design basis, safety analyses, NIS power range functions, or the safe operation of the plant.

5.0 Regulatory Safety Analysis

5.1 No Significant Hazards Consideration

PG&E has evaluated whether or not a significant hazards consideration is involved for Diablo Canyon Power Plant (DCPP) with the proposed change based on the three standards set forth in 10 CFR 50.92(c) as discussed below:

(1) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change to Technical Specifications (TS) Surveillance Requirement (SR) 3.3.1.2 and SR 3.3.1.3 is consistent with the NRC approved Industry/Technical Specifications Task Force Standard Technical Specification

Change Traveler, TSTF-371, and NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 2.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The reactor trip system (RTS) instrumentation will be unaffected. Protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the request are maintained.

The probability and consequences of accidents previously evaluated in the Updated Final Safety Analysis Report (UFSAR) are not adversely affected because the change to the nuclear instrumentation system (NIS) power range channel daily surveillance assures the conservative response of the channel even at part-power levels.

The proposed change modifies the NIS power range channel daily surveillance requirement to help assure the NIS power range functions are tested in a manner consistent with the safety analysis and licensing basis.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the UFSAR.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

There is no hardware change or change in the method by which any safety-related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements or response time limits will be affected. The NIS power range high trip setpoint adjustment requirements, prior to adjusting indicated power in a decreasing power direction, will ensure the reactor power level is consistent with assumptions made in the safety analysis and licensing basis.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. There will be no adverse effect or challenges imposed on any safety-related system as a result of the change.

This amendment does not alter the design or performance of the Eagle 21 System, NIS, or Solid State Protection System used in the plant protection systems.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed change requires a revision to the criteria for implementation of NIS power range channel adjustments based on secondary power calorimetric calculations; however, the change does not eliminate any RTS surveillances or alter the frequency of surveillances required by the Technical Specifications. The revision to the criteria for implementation of the daily surveillance will have a conservative effect on the performance of the NIS power range channels, particularly at part-power conditions. The nominal trip setpoints specified in the Technical Specification Bases and the safety analysis limits assumed in the transient and accident analyses are unchanged. None of the acceptance criteria for any accident analysis is changed.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature, peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

The imposition of appropriate surveillance testing requirements will not reduce any margin of safety since the change will assure that safety analysis assumptions on reactor power are verified on a periodic frequency.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

<u>Conclusion</u>

Based on the considerations above, PG&E concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory bases and guidance documents associated with the systems discussed in this LAR are discussed in this section.

GDC-13 requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

GDC-20 requires that the protection system(s) shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC-21 requires that the protection system(s) shall be designed for high functional reliability and testability.

GDC-22 through GDC-25 and GDC-29 require various design attributes for the protection system(s), including independence, safe failure modes, separation from control systems, requirements for reactivity control malfunctions, and protection against anticipated operational occurrences.

Regulatory Guide 1.22 discusses an acceptable method of satisfying GDC-20 and GDC-21 regarding the periodic testing of protection system actuation functions. These periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident.

10 CFR 50.55a(h) requires that the protection systems meet IEEE 279-1971. Sections 4.9 - 4.11 of IEEE 279-1971 discuss testing provisions for protection systems.

There have been no changes to the RTS instrumentation design that could affect compliance with the above regulatory requirements and guidance. This LAR revises surveillance testing requirements on the NIS power range neutron flux channels consistent with those requirements and guidance documents. Therefore, the requirements and guidelines in GDC-13, GDC-20, GDC-21, GDC-22, GDC-23, GDC-24, GDC-25, GDC-29, Regulatory Guide 1.22, and 10 CFR 50.55a(h) will continue to be met.

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 Environmental Consideration

PG&E has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. PG&E has evaluated the proposed amendment and has determined that the amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed amendment is not required.

7.0 References

7.1 References

1. NRC approved Industry/Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-371, "NIS Power Range Channel Daily SR TS Change to Address Low Power Decalibration."
2. Westinghouse Technical Bulletin ESBU-TB-92-14-R1, "Decalibration Effects of Calorimetric Power Measurements on the NIS High Power Reactor Trip at Power Levels less than 70% RTP," dated February 6, 1996.
3. PG&E, NSSS Calculation N-212, Revision 1.
4. WCAP-11082, "Westinghouse Set point Methodology for Protection Systems: Diablo Canyon Units 1 & 2, 24 Month Fuel Cycle Evaluation," Revision 5.
5. Callaway License Amendment No. 148, issued February 5, 2002.
6. Westinghouse Technical Bulletin NSD-TB-92-14, "Instrumentation Calibration at Reduced Power," January 18, 1993.
7. ABB Combustion Engineering Infobulletin 94-01, "Potential Nonconservative Treatment of Power Measurement Uncertainty," June 21, 1994.

8. WCAP-8567, "Improved Thermal Design Procedure," July 1975.
9. WCAP 11594 Revision 2, "Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology," January 1997.
10. 10 CFR 50 Appendix A, January 1, 2001 Edition.
11. Regulatory Guideline 1.22, "Periodic Testing of Protection System Actuation Functions," February 17, 1972.
12. 10 CFR 50.55a(h), January 1, 2001 Edition.
13. Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," approved April 5, 1972.
14. Wolf Creek License Amendment No. 148, issued October 2, 2002.

7.2 Precedent

The changes to SR 3.3.1.2 and SR 3.3.1.3 were previously approved for the Callaway Plant in License Amendment Number 148, on February 5, 2002. They were also approved for Wolf Creek in License Amendment Number 148, on October 2, 2002. This License Amendment Request is consistent with NRC approved Industry/Technical Specifications Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-371, "NIS Power Range Channel Daily SR TS Change to Address Low Power Decalibration." It is also consistent with the amendments issued to the Callaway Plant and the Wolf Creek Plant. These changes were also approved for Farley Units 1 and 2, in Farley License Amendment Nos. 144 (Unit 1) and 135 (Unit 2) dated October 1, 1999.

MARKED-UP TECHNICAL SPECIFICATIONS

<u>**REMOVE PAGE**</u>	<u>**INSERT PAGE**</u>
3.3-8	3.3-8

SURVEILLANCE REQUIREMENTS

NOTE

Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	NOTE ~~1. Adjust NIS channel if absolute difference is > 2%.~~ ~~2.~~ Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP, but prior to exceeding 30% RTP. Compare results of calorimetric heat balance calculation to ~~Nuclear Instrumentation System (NIS) channel output~~. Insert A	24 hours
SR 3.3.1.3	NOTE ~~1. Adjust NIS channel if absolute difference is ≥ 3%.~~ ~~2.~~ Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. Compare results of the incore detector measurements to ~~NIS AFD~~. Insert B	31 effective full power days (EFPD)
SR 3.3.1.4	NOTE This Surveillance must be performed on the reactor trip bypass breaker, for the local manual shunt trip only, prior to placing the bypass breaker in service. Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS

(continued)

<u>Insert A:</u> power range channel output. Adjust power range channel output if calorimetric heat balance calculation results exceed power range channel output by more than + 2% RTP

<u>Insert B:</u> Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%

REVISED TECHNICAL SPECIFICATIONS

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	-----NOTE----- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP, but prior to exceeding 30% RTP. ----- Compare results of calorimetric heat balance calculation to power range channel output. Adjust power range channel output if calorimetric heat balance calculation results exceed power range channel output by more than + 2% RTP.	24 hours
SR 3.3.1.3	-----NOTE----- Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. ----- Compare results of the incore detector measurements to Nuclear Instrumentation System (NIS) AFD. Adjust NIS channel if absolute difference is ≥ 3%.	31 effective full power days (EFPD)
SR 3.3.1.4	-----NOTE----- This Surveillance must be performed on the reactor trip bypass breaker, for the local manual shunt trip only, prior to placing the bypass breaker in service. ----- Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS

(continued)

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

(for information only)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2

Insert A

~~SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS power indications every 24 hours. If the calorimetric exceeds the NIS power indications by > 2% RTP, the NIS is not declared inoperable, but the excore channel gains, must be adjusted consistent with the calorimetric power. If the NIS power indications cannot be properly adjusted, the channel is declared inoperable.~~

Insert B

~~Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS power indications shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS power indications and the calorimetric is > 2% RTP.~~

At lower power levels (<45% RTP), calorimetric data are inaccurate.

Discretion must be exercised if the power range channel output is adjusted in the decreasing power direction due to a part-power calorimetric (<45% RTP). This action could introduce a non-conservative bias at higher power levels which could delay an NIS reactor trip until power is above the power range high safety analysis limit (SAL) of 118% RTP. The cause of the non-conservative bias is the decreased accuracy of the calorimetric at reduced power conditions. The primary error contributor to the instrument uncertainty for a secondary side power calorimetric measurement is the feedwater flow measurement, which is determined by a ΔP measurement across a feedwater venturi. While the measurement uncertainty remains constant in ΔP span as power decreases, when translated into flow, the uncertainty increases as a square term. Thus, a 1% flow error at 100% power can approach a 10% flow error at 30% RTP even though the ΔP error has not changed.

To assure a reactor trip below the power range high SAL, the maximum allowable Power Range Neutron Flux-High trip Setpoint is limited according to DCPP surveillance procedures, prior to adjusting the power range channel output whenever the calorimetric power is between 15% and 45% RTP. **Insert C** **Insert C**

For example, to assure a reactor trip below the power range high SAL, the Power Range Neutron Flux-High trip Setpoint is reduced as necessary prior to adjusting the power range channel output whenever the calorimetric power is ≥ 15% RTP and <45% RTP. The maximum allowable Power Range Neutron Flux-High trip Setpoint may be increased with increasing RTP in accordance with surveillance procedures. Following a plant refueling outage, it is prudent to reduce the Power Range Neutron Flux–High trip Setpoint prior to startup.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

Before the Power Range Neutron Flux–High trip Setpoint is re-set to its nominal full power value (≤109% RTP), the power range channel calibration must be confirmed based on a calorimetric performed at ≥ 45% RTP.

Insert D

~~The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP but prior to exceeding 30% RTP.~~ The 24-hour allowance after increasing THERMAL POWER above 15% RTP provides a reasonable time to attain a scheduled power plateau, establish the requisite conditions, perform the required calorimetric measurement, and make any required adjustments in a controlled, orderly manner and without introducing the potential for extended operation at high power levels with instrumentation that has not been verified to be acceptable for subsequent use.

Insert E

~~The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.~~

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

<u>SR 3.3.1.3</u>

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. The comparison checks for differences due to changes in core power distribution since the last calibration.

Insert from next page

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

(continued)

SURVEILLANCE REQUIREMENTS

The Note to SR 3.3.1.3

This Note

SR 3.3.1.3 (continued) **Move to previous page**

~~Two Notes modify SR 3.3.1.3. Note 1 indicates that~~ tThe excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%. ~~Note 2~~ clarifies that the Surveillance is required only if reactor power is ≥ 50% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 50% RTP. ~~Not 2~~ allows power ascensions and associated testing to be conducted in a controlled and orderly manner, at conditions that provide acceptable results and without introducing the potential for extended operation at high power levels with instrumentation that has not be verified to be acceptable for subsequent use. Due to such effects as shadowing from the relatively deep control rod insertion and, to a lesser extent, the dependency of the axially-dependent radial leakage on the power level, the relationship between the incore and excore indications of axial flux difference (AFD) at lower power levels is variable. Thus, it is prudent to defer the calibration of the excore AFD against the incore AFD until more stable conditions are attained (i.e., withdrawn control rods and higher power level). The AFD is used as an input to the Overtemperature ΔT reactor trip function and for assessing compliance with ITS LCO 3.2.3, "AXIAL FLUX DIFFERENCE." Due to the DNB benefits gained by administratively restricting the power level to 50% RTP, no limits on AFD are imposed below 50% RTP by LCO 3.2.3; thus, the proposed change is consistent with LCO 3.2.3. requirements below 50% RTP. Similarly, sufficient DNB margins are realized through operation below 50% RTP that the intended function of the Overtemperature ΔT reactor trip function is maintained, even though the excore AFD indication may not exactly match the incore AFD indication. Based on plant operating experience, 24 hours is a reasonable time frame to limit operation above 50% RTP while completing the procedural steps associated with the surveillance in an orderly manner.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, since the changes in neutron flux are slow during the fuel cycle, the expected change in the absolute difference between the incore and excore AFD will be less than 3 percent AFD during this interval.

(continued)

Insert A

SR 3.3.1.2 compares the calorimetric heat balance calculation to the power range channel output every 24 hours. If the calorimetric heat balance calculation results exceed the power range channel output by more than + 2% RTP, the power range channel is not declared inoperable, but the excore channel gains must be adjusted. The power range channel output shall be adjusted consistent with the calorimetric heat balance calculation results if the calorimetric calculation exceeds the power range channel output by more than + 2% RTP. If the power range channel output cannot be properly adjusted, the channel is declared inoperable.

Insert B

To assure a reactor trip consistent with the safety analysis, adjustments to the power range channel in the decreasing direction are not required. This allowance does not preclude making indicated power adjustments, if desired, when the calorimetric heat balance calculation power is less than the power range channel output. To provide close agreement between indicated power and to preserve operating margin, the power range channels are normally adjusted when operating at or near full power during steady-state conditions.

Insert C

in the decreasing power direction

Insert D

The Note to SR 3.3.1.2 clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP, but prior to exceeding 30% RTP. A power level of 15% RTP is chosen based on plant stability, i.e., automatic rod control capability and the turbine generator synchronized to the grid.

Insert E

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than + 2% RTP is not expected in any 24 hour period.

LOW POWER NIS ADJUSTMENT UNCERTAINTY CALCULATION

IDAP CF3.ID4
ATTACHMENT 7.1

REV. 1

TITLE: CALCULATION COVER SHEET

PACIFIC GAS AND ELECTRIC COMPANY
CALCULATION COVER SHEET

File No. : NSSS
Calculation No.: N-212

/_/ Preliminary /X/ Final

Project: NIS Trip Setpoint Uncertainty Date : 5-22-96
Department/ Group: NSSS Engineering / Reactor Engineering

Structure, System or Component: Power Range NIS

Type or Purpose of Calculation: Low Power NIS Adjustment Uncertainty

No. of Sheets: 35, plus 25 pages of Reviewer's Comments. (REV. 0) *

*REV. 1 REPLACES REV. 0 FOR 24 MONTH FUEL CYCLES. REV. 0 IS FOR 18 MONTHS.

	Signature	Discipline/Dept.	Date
PREPARER:	Douglas N. Poland	NSSS/Rx Engr.	5-22-96
CHECKER :	William Bogdi	NSSS/Rx Engr.	5-22-96
APPROVAL:	Larry Cossette Supervisor	NSSS/Rx Engr	5-22-96

For civil calculation, enter the registered engineer's stamp or seal and expiration date of certificate or authority in this space.

For other calculation, enter the registered engineer's full name and registration number, or stamp, or seal in this space.

Larry Lester Cossette
#2327

RECORDS OF REVISIONS

Revision Number	Date	Reasons for Revision	Prepared By	Checked By	Approval Regis. Engr.	Approval Supvr.
1	10/8/97	24 MONTH FUEL CYCLES	W. Bogdi	[illegible]	Larry Cossette	Larry Cossette

I. Purpose

The purpose of this calculation file is to determine the required NIS high-flux trip setpoint as a function of the power at which the NI calibration is performed for 24 month fuel cycles (see AR A0439874). This calculation was originally performed in Revision 0 of this calculation file for 18 month fuel cycles. Revision 1 replaces Revision 0 in its entirety.

II. Summary Of Results

The results of this calculation file show that the values previously obtained for 18 month fuel cycles will bound 24 month fuel cycles. This is because the 24 month uncertainty analysis (WCAP-11594, Rev. 2) took credit for two operable feedwater flow transmitters per loop, whereas the old 18 month analysis assumed only one operable flow transmitter per loop. In STP R-2B1, the operator is instructed to check the quality code of the computer points used in the PPC heat balance calculation. If the summary page outputs from the calorimetric calculation are not either "GOOD" or "DALM", then STP R-2B2 must be used. STP R-2B2 uses the Barton ΔP gauges, whose uncertainty is bounded by the assumption of two operable flow transmitters per loop.

Attachment 8.6 of STP R-2B2 (Rev. 7, OTSC dated 9/10/97 is the latest) can be changed as follows, if desired. Since the new numbers are less restrictive, the change is not required.

TABLE 1

Calorimetric Power	OLD NIS Trip Setpoint	NEW NIS Trip Setpoint
≥45% RTP	≤109% RTP	≤109% RTP
≥40% RTP	≤107% RTP	≤107% RTP
≥35% RTP	≤105% RTP	≤106% RTP
≥30% RTP	≤102% RTP	≤103% RTP
≥25% RTP	≤97% RTP	≤99% RTP
≥20% RTP	≤88% RTP	≤91% RTP
≥15% RTP	≤72% RTP	≤76% RTP

III. NIS Trip Setpoint Calculation

A. Trip Setpoint Uncertainty Equation

We wish to calculate the highest allowable high flux trip setpoint for the power range excore Nuclear Instrumentation System (NIS) adjustments/calibrations done at various reactor power levels. This is given by setting the total allowance for uncertainty (Safety Analysis Limit - Nominal Trip Setpoint) equal to the total setpoint uncertainty (i.e., the Channel Statistical Allowance defined in Table 3-1 of WCAP-11082, Rev. 5). We obtain the following quadratic equation where the unknown is NTS, the Nominal Trip Setpoint in %RTP.

$$SAL - NTS = 1.2\,[PMA^2 + SCA^2 + SD^2 + (RCA + RCSA + RD)^2 + RTE^2]^{1/2} \qquad \text{(EQN. 1)}$$

where the factor of 1.2 is used to convert the % span errors into % RTP errors for the 0-120% RTP range of the NIS % RTP indication (i.e., 1.2 x (%span error)=(%RTP error)).

SAL = Safety Analysis Limit = 118% RTP [Table 3-25 of WCAP-11082, Rev. 5]

NTS = Nominal Trip Setpoint = the unknown being solved for (units of %RTP)

PMA = Process Measurement Accuracy = 4.20% span [Table 3-1 of WCAP-11082, Rev. 5]

SCA = Sensor Calibration Accuracy (units of % span) = $(E_{cal}/1.2) \times (NTS/P_{cal})$

where E_{cal} = heat balance error (at power, P_{cal}, where the heat balance is performed) in units of %RTP
P_{cal} = power level of NI calibration (%RTP)

SD = Sensor Drift (units of % span) = $(D/1.2) \times (NTS/P_{cal})$

where D = maximum allowable NI drift (%RTP) = 2.00% RTP

RCA = Rack Calibration Accuracy = 0.5% span [Table 3-1 of WCAP-11082, Rev. 5]

RCSA = Rack Comparator Setting Accuracy = 0.25% span [Table 3-1 of WCAP-11082, Rev. 5]

RD = Rack Drift = 1.0% span [Table 3-1 of WCAP-11082, Rev. 5]

RTE = Rack Temperature Effects = 0.5% span [Table 3-1 of WCAP-11082, Rev. 5]

The above PMA term covers all transient effects which can cause the NI calibration error to increase, e.g., xenon effects, temperature effects, control rod effects, etc. Also, note that both the SCA and SD terms include the effect of "gaining" these uncertainties as power increases to the trip setpoint during a power transient.

Rearranging Equation 1 gives the following quadratic equation:

$$a \times NTS^2 + b \times NTS + c = 0 \qquad \text{(EQN. 2)}$$

The constants in this equation can be expressed as:

$$a = 1 - (E_{cal}^2 + D^2)/P_{cal}^2 = 1 - (E_{cal}^2 + 4)/P_{cal}^2 \qquad \text{(EQN.3)}$$

$$b = -2 \times SAL = -236$$

$$c = SAL^2 - (1.2)^2[PMA^2 + (RCA + RCSA + RD)^2 + RTE^2] = 13893.8284$$

The power calorimetric uncertainty in units of %RTP, E_{cal}, is a function of power. It will be determined in the next section.

B. Solution Of NIS Trip Setpoint Equation

Before Equation 2 can be solved, E_{cal} has to be determined as a function of power. This was previously determined for 18 month fuel cycles on Page 6 of N-212, Revision 0. The general approach is to begin with the full power heat balance uncertainty, subtract the component due to feedwater ΔP uncertainty and then add the power-dependent component of the feedwater ΔP uncertainty. The equation becomes:

$$E_{cal}(\%RTP) = [((\varepsilon_{cal}^{100\%})^2 - (\varepsilon_{\Delta P}^{100\%})^2 / N_{loops})(P_{cal}/100) + (S_{\Delta P} \times E_{\Delta P}(inches))^2/N_{loops}]^{1/2}$$

(EQN. 4)

where

$\varepsilon_{cal}^{100\%}$ = the relative error of the full power heat balance (%)
= 1.72% [Calc File N-228, Rev. 0]

$\varepsilon_{\Delta P}^{100\%}$ = the loop feedwater ΔP component of the total full power relative heat balance error (%)
= 1.227% [Page 27 of WCAP-11594, Rev. 2]

$S_{\Delta P}$ = the feedwater ΔP uncertainty sensitivity factor (%RTP/inches ΔP)
= $(\partial P/\partial(\Delta P))$
= $57.850/P^{1.3346}$ [Pages 5-6 of N-212, Rev. 0]

$E_{\Delta P}$(inches) = uncertainty in loop feedwater flow transmitter differential pressure reading (inches of water)
= (.0186)(546.2") = 10.2" [Page 5 of Calc File J-121, Rev. 1, and Page 26 of WCAP-11594, Rev. 2]

N_{loops} = number of feedwater loops = 4

The solution of Equation 2 is just a standard quadratic equation solution.

$$NTS = (-b - (b^2 - 4ac)^{1/2})/(2a)$$

Using Equation 4 in Equation 3 gives the following results:

TABLE 2

P_{cal} (%RTP)	$S_{\Delta P}$ (%RTP/inch)	E_{cal} (%RTP)	NTS(%RTP)
100	0.1239	1.726	111.764
90	0.1426	1.688	111.620
80	0.1669	1.669	111.417
70	0.1995	1.683	111.116
60	0.2450	1.760	110.631
50	0.3125	1.952	109.765
45	0.3597	2.121	109.051
40	0.4209	2.367	107.991
35	0.5030	2.726	106.350
30	0.6179	3.260	103.686
25	0.7882	4.083	99.149
20	1.0616	5.440	91.048
15	1.5584	7.941	76.096

IV. Conclusion

The current Nominal Trip Setpoint at full power is 109% RTP. Table 2 shows that this setpoint is valid when the power calorimetric is done at ≥ 45% RTP. Otherwise, at lower power levels, the required setpoint becomes more restrictive (lower) to accommodate the additional power uncertainties associated with the heat balance and with the NI gain effect on the calibration uncertainties.

Truncating the setpoint values in Table 2 gives the results summarized above in Table 1. These results are valid for both 24 month and 18 month fuel cycles. These setpoints are bounded by the values previously obtained for 18 month fuel cycles which are currently in STP R-2B2. Therefore, STP R-2B2 need not be revised at this time.

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
IDAP CF3.ID4
ATTACHMENT 7.1

TITLE: CALCULATION COVER SHEET

PACIFIC GAS AND ELECTRIC COMPANY CALCULATION COVER SHEET

Page 1 of 1
FOR REFERENCE ONLY
File No. : NSSS
Calculation No.: N-212
Check DCI for Current Rev

/_/ Preliminary /X/ Final

Project: NIS Trip Setpoint Uncertainty Date : 5-22-96
Department/ Group: NSSS Engineering / Reactor Engineering

Structure, System or Component: Power Range NIS

Type or Purpose of Calculation: Low Power NIS Adjustment Uncertainty

No. of Sheets: 35, plus 25 pages of Reviewer's Comments.

	Signature	Discipline/Dept.	Date
PREPARER:	Douglas N. Poland	NSSS/Rx Engr.	5-22-96
CHECKER :	William [illegible]	NSSS/Rx Engr.	5-22-96
APPROVAL:	Larry [illegible] Supervisor	NSSS/Rx Engr	5-22-96

For civil calculation, enter the registered engineer's stamp or seal and expiration date of certificate or authority in this space.

For other calculation, enter the registered engineer's full name and registration number, or stamp, or seal in this space.

Larry Lester Cossette
#2327

RECORDS OF REVISIONS

Revision Number	Date	Reasons for Revision	Prepared By	Checked By	Approval Regis. Engr.	Approval Supvr.

Summary

This calculation is in response to Westinghouse Technical Bulletin ESBU-TB-92-14-R1, dated March 21, 1996 (Reference 1), which discusses increased NIS high flux trip setpoint uncertainties which arise as a result of performing NIS adjustments at less than full power. The purpose of this calculation is to evaluate the effect of low power NIS adjustments on the acceptability of the Power Range NIS channel high flux trip setpoints (see Reference 2).

The bulletin raises two concerns which must be addressed for impact on DCPP procedures and setpoints:

1. *Performing NIS normalizations via gain adjustments, as opposed to offset adjustments, at less than full power.* DCPP performs NIS normalizations by adjusting the fine adjust gain pot in the NIS summation circuit (this is labeled "R303" in Figure 1). The concern with this practice is that the effect of a given gain uncertainty on NIS power is proportional to the power. Therefore, errors or uncertainties present in the NIS gain at low power produce increasingly larger NIS power uncertainties as power is increased toward the trip setpoint. This effect is illustrated in Figures 3 and 4 of Attachment 3 to Reference 1.

2. *Feedwater flow uncertainty increases significantly as power decreases.* . DCPP uses feedwater ΔP measurements to determine feedwater flow and calorimetric reactor power (References 3 and 4). Because flow is proportional to the square root of the measured ΔP, the flow uncertainty is inversely proportional to the measured flow. As a result, the calorimetric power uncertainty due to the uncertainty in ΔP increases as power decreases.

During a postulated transient from low power, these effects combine to dramatically increase the uncertainty in the NIS power at the high flux trip setpoint from its nominal value based on full power calorimetrics and NIS adjustments. During performance of an NIS adjustment at low power, the calorimetric uncertainty is increased due to the increased feedwater DP uncertainty. Then as power approaches the trip setpoint, the effect of this uncertainty grows in term of % RTP. Also, if additional calorimetrics are performed at low power following the low power NIS adjustment, the NIS power may be left reading as much as 2% RTP below calorimetric power. The effect of this difference also multiplies as power is increased. There is no explicit allowance for this difference in the current nominal setpoint, though there is margin, under nominal conditions, to cover this difference (see Section 6.1.1.1 of Reference 5).

Evaluation of these factors for impact on DCPP procedures and setpoints results in the following limitation which must be incorporated into Reference 4: *Whenever NIS adjustments are performed at less than 45% RTP (calorimetric power), the NIS trip setpoints must be set according to the following table:*

Calorimetric Power	Required NIS Trip Setpoint
≥ 40% RTP	≤ 107% RTP
≥ 35% RTP	≤ 105% RTP
≥ 30% RTP	≤ 102% RTP
≥ 25% RTP	≤ 97% RTP
≥ 20% RTP	≤ 88% RTP
≥ 15% RTP	≤ 72% RTP

Table 1: Required NIS High Flux Trip Setpoints as a function of Calorimetric Power

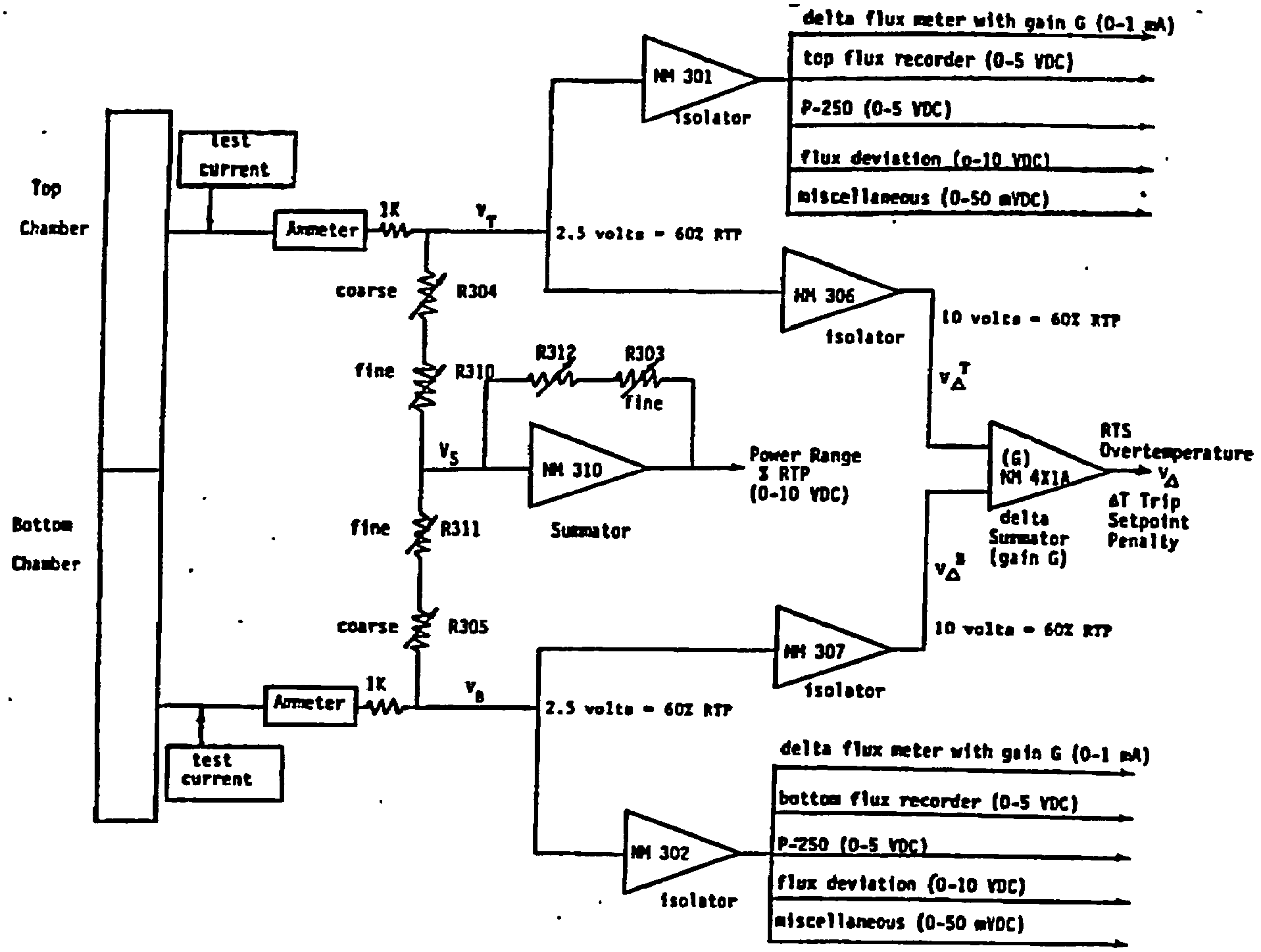

Figure 1: Simplified Circuit Diagram for Power Range NIS Channel

NIS Gain Adjustment

Discussion

A simplified circuit diagram for the Power Range NIS channels is presented in Figure 1. The voltage V_S represents the summator input voltage; this is proportional to the sum of the raw NIS detector currents. The summator circuit multiplies V_S by a gain, g, which is adjusted as necessary to normalize the indicated NIS power. The gained voltage is the indicated NIS power which is in volts as seen by the NIS trip circuit and in % RTP on the NIS drawer meter. For indicated NIS power and its uncertainty, we have the following equations:

$$P_{NIS} = g\, V_S \quad \text{Eqn. 1}$$

$$\begin{aligned} \delta P_{NIS} &= \delta g\, (\partial P_{NIS}/\partial g) \\ &= \delta g\, (V_S) \\ &= \delta g/g\, (P_{NIS}) \end{aligned} \quad \text{Eqn. 2}$$

It is assumed that NIS response is linear with respect to reactor power, meaning that g and δg are is constant with respect to power. The power uncertainty, in units of % RTP, is thus proportional to the NIS power and therefore to reactor power. Thus a gain error which results in a 2% RTP error in NIS power at 50% RTP results in a 4% RTP error at 100% RTP.

Uncertainty in the NIS Gain

At the time of NIS normalization, the gain is defined and set such that P_{NIS} is equal to P_{cal} (calorimetric power), which means (using this equality and Equation 1):

$$g = P_{cal}/\dot{V}_S \qquad \text{Eqn. 3}$$

At the time of NIS normalization, the only error present in g is that due to the error in calorimetric power. Following the same steps as in Equation 2, we arrive at the following for the fractional uncertainty in gain:

$$\delta g/g = \delta P_{cal}/P_{cal} \qquad \text{Eqn. 4}$$

At subsequent calorimetrics, P_{NIS} will not be exactly equal to P_{cal} and g may be left with P_{NIS} differing from P_{cal} by as much as 2% RTP. To obtain an expression for this gain error, we may substitute 2% RTP for the δP_{NIS} term in Equation 2 and rearrange:

$$\delta g/g = 2\%\ RTP/P_{NIS} \qquad \text{Eqn. 5}$$

It is important to note that in these expressions for the gain errors, the powers are those which exist at the time of the calorimetric power calculation. The total gain error is an SRSS combination of these two independent effects:

$$\delta g/g = [(\delta P_{cal}/P_{cal})^2 + (2\%\ RTP/P_{NIS})^2]^{1/2} \qquad \text{Eqn. 6}$$

Effect on the Accuracy of the NIS High Flux Trip Setpoints

The uncertainty in indicated NIS power is presented in Equation 2. It is important to note that the P_{NIS} term in that equation refers to current indicated power. For the uncertainty in P_{NIS} at the time of trip, we substitute P_{TS} (the trip setpoint in units of % RTP) for P_{NIS} in Equation 2 and combine Equations 2 and 6 to obtain an expression for the uncertainty in the NIS trip setpoint as a function of the powers at the time of the calorimetric:

$$\delta P_{TS} = [(\delta P_{cal}(P_{TS}/P_{cal}))^2 + (2\%\ RTP(P_{TS}/P_{NIS})^2]^{1/2} \qquad \text{Eqn. 7}$$

We see that the individual trip setpoint error terms are proportional to the ratio of the trip setpoint to the power at the time of the calorimetric. Thus, the setpoint uncertainty is inversely proportional to the power at the time of the calorimetric.

Feedwater Flow Uncertainty as a Function of Reactor Power

Discussion

For ΔP flow measurements, the measured flow is primarily a function of the square root of the measured ΔP. The uncertainties associated with the ΔP channel are constant, in units of ΔP, across the ΔP scale (e.g., an uncertainty of ±5" of water at 500" ΔP is still an uncertainty of ±5" of water at 50" ΔP). Since the conversion from ΔP to flow is nonlinear, these uncertainties are not constant after conversion to units of flow. The conversion from ΔP uncertainty (typically reported in units of % FS ΔP, where FS denotes Full Scale) to flow uncertainty is derived in Section 4.10 of Reference 5 and is:

$$\delta F_{meas} = F_{meas}\ [\ \delta\Delta P\ /\ (2)(100)\ ]\ [F_{max}\ /\ F_{meas}]^2 \qquad \text{Eqn. 8}$$

where δF_{meas} = uncertainty in measured flow, in flow units
F_{meas} = measured flow
$\delta\Delta P$ = uncertainty in ΔP, in units of % FS ΔP
F_{max} = Full Scale indicated flow

If we combine all of the constant terms, we may rewrite this equation as:

$$\delta F_{meas} = C_1 / F_{meas} \qquad \text{Eqn. 9}$$

Thus the flow uncertainty, in flow units, is not constant across the flow scale but is inversely proportional to the measured flow. Therefore as flow is decreased from nominal, the flow uncertainty is increased from its nominal value by a multilplier of $(F_{nominal}/F_{meas})$.

Calculational Methodology

In order to quantify the effect of the increased flow uncertainty on the NIS trip setpoint uncertainties, we employ the methodology which was used to calculate the calorimetric power uncertainty in Reference 6. This consists of performing heat balance calculations at representative conditions in order to arrive at the appropriate sensitivities. These sensitivities are then used in combining the input parameter uncertainties and arriving at the calorimetric power uncertainty (see Section 3 of Reference 6). For the power calorimetric uncertainty in % RTP due to ΔP uncertainties, we have the following:

$$\delta P_{cal,\Delta P} = \delta\Delta P \ (\partial P_{cal}/\partial\Delta P) \qquad \text{Eqn. 10}$$

where $\delta\Delta P$ = uncertainty in ΔP in units of % FS ΔP
$\partial P_{cal}/\partial\Delta P$ = sensitivity of calorimetric power to changes in ΔP

The sensitivity then includes conversion of uncertainty from ΔP units to flow units, as we see by rewriting the sensitivity in terms of the measured flow sensitivity:

$$\delta P_{cal,\Delta P} = \delta\Delta P \ [(\partial P_{cal}/\partial F_{meas})\ (\partial F_{meas}/\partial\Delta P)] \qquad \text{Eqn. 11}$$

We may rearrange Equation 8, combining all the constant terms except for $\delta\Delta P$, to express the last term as

$$\partial F_{meas}/\partial\Delta P = \delta F_{meas}/\delta\Delta P = C_2 / F_{meas} \qquad \text{Eqn. 12}$$

Equation 11 may then be written as

$$\delta P_{cal,\Delta P} = \delta\Delta P \ [(\partial P_{cal}/\partial F_{meas})\ (C_2 / F_{meas})] \qquad \text{Eqn. 13}$$

As power and measured feedwater flow decrease from nominal, the second term in the sensitivity increases by a factor of $(F_{nominal}/F_{meas})$. If no other input parameters to the calorimetric heat balance calculation were to change, then the first term would remain constant (i.e., power would be directly proportional to feedwater flow and $\partial P_{cal}/\partial F_{meas}$ would be constant). However as power is decreased, thermodynamic conditions are changed (feedwater temperature and pressure, etc.), slightly changing the proportionality of calorimetric power to feedwater flow and thus making $\partial P_{cal}/\partial F_{meas}$ somewhat dependent on power. Therefore we expect the sensitivity to be approximately but not exactly inversely proportional to feedwater flow

Calculation of Calorimetric Power Sensitivity to changes in ΔP as a Function of Power

Heat balance calculations were performed using the OPHB2B code and representative plant data to provide empirical values for $\partial P_{cal}/\partial\Delta P$ (see Equation 10). The input parameters need only be approximate since the effect of minor input parameter variations on the sensitivities would be negligible - the results of these calculations are used only in a relative sense, not absolute. The plant data was taken from Unit 1 Cycle 8 power ascension STP R-2B1 data (Appendix 2); this ensures that the data represents reasonably stable plant conditions and has the added advantage of already having been reviewed for reasonableness via the normal STP review process.

Calorimetric calculations were performed at each R-2B1 statepoint for each of three feedwater ΔP values: nominal, -10" and +10" (Appendix 1). The change in power was divided by the change in ΔP for each perturbation and the maximum value at each plateau was taken as the sensitivity. This is equivalent to the loop power sensitivity calculations of Reference 6, except that by perturbing all 4 loops it yields a loop averaged sensitivity. From the preceding derivations, we expect this fit to be close to an inverse linear relationship. Figure 2 presents a logarithmic plot of calorimetric power sensitivity to ΔP as a function of calorimetric power. (Note that if a function y is of the form $y=ax^b$, then by taking logarithms of both sides we may write $\log y = b(\log x) + \log a$. The slope of a linear least squares fit to log y vs. log x is therefore equal to the exponent b, and the log of the intercept of the fit is equal to the coefficient a.) The powers and sensitivities are presented below and the result, as seen in Figure 2, is that the sensitivity is equal to $57.850/\text{Power}^{1.3346}$.

Calculated Calorimetric Power (% RTP)			Sensitivity (% RTP/inch ΔP)	
ΔP=nominal	ΔP=nom-10"	ΔP=nom+10"	Actual	Fit
100.02	98.78	101.24	0.124	0.124
95.77	94.45	97.06	0.132	0.131
86.63	85.13	88.11	0.150	0.150
71.14	69.19	73.03	0.195	0.195
48.99	45.81	51.97	0.318	0.321
28.82	22.26	34.17	0.656	0.652

Table 2: Calorimetric Power Sensitivity to ΔP as a Function of Calorimetric Power

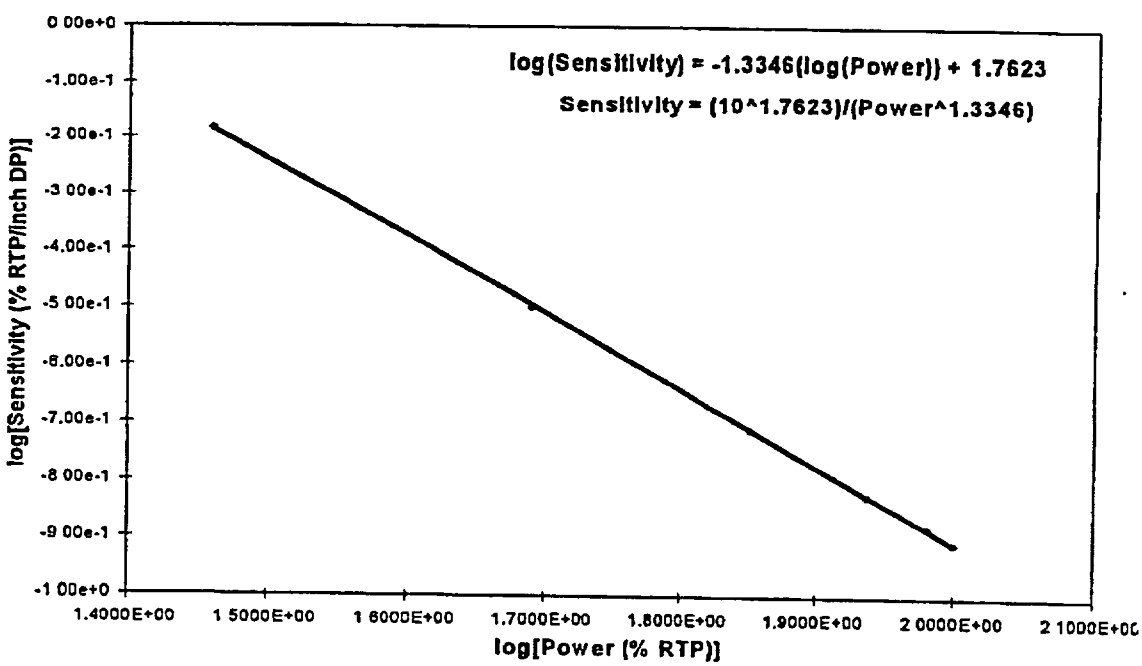

Figure 2: Calorimetric Power Sensitivity to ΔP vs. Calorimetric Power

Calculation of Calorimetric Power Uncertainty as a Function of Power

The next step in our evaluation is to recalculate the calorimetric power uncertainty as a function of power level. The nominal calorimetric power calculations from Reference 7 provide the starting point for this calculation and are summarized below:

Parameter	Sensitivity	Uncertainty	Product	Wt	Product^2/Wt
FW Temp	1.84E-01	4.20E+00	0.774	3	0.1997
FW Pressure	4.10E-04	8.00E+01	0.033	1	0.0011
K (AMAG)	4.11E-03	3.44E+02	1.414	1	1.9989
FW D/P	1.24E-01	1.20E+01	1.488	4	0.5539
STM Press	3.40E-03	3.70E+01	0.126	4	0.0040
BD Flow	9.24E-03	1.50E+01	0.139	4	0.0048
Alpha	7.28E+04	4.00E-07	0.029	1	0.0008
NPHA	3.52E-08	2.40E+06	0.085	1	0.0071
X	8.70E+01	2.50E-03	0.217	1	0.0473
		Sum of Weighted Squares (SWS):			2.8177
		δP_{cal} = Square Root of SWS:			1.68

Table 3: Breakdown of Nominal (100% RTP) Calorimetric Power Uncertainty

Table 3 was generated at nominal 100% RTP conditions, so we must scale the sensitivity and uncertainty values as a function of calorimetric power. The dependence of calorimetric power sensitivity to feedwater ΔP on calorimetric power has been addressed in the previous section. For the remainder of the terms, we make the following assumptions:

1) The uncertainty terms, which are shown in engineering units, remain constant with power. Any deviations from this assumption are expected to be of small magnitude and would result in negligible effects on δP_{cal} since it is heavily dominated by the feedwater ΔP error as power decreases.

2) The remainder of the sensitivity terms are scaled as a function of calorimetric power. This is based on the assumption that the effect in percent of calculated power is constant for a given input error in engineering units. The translation from % of calculated power to % RTP requires multiplying by (P_{meas}/100% RTP), hence the assumed proportionality. As above, any deviations from this assumption are expected to be of small magnitude and would result in negligible effects on δP_{cal} since it is heavily dominated by the feedwater ΔP error as power decreases.

Given these assumptions, we calculate the revised SWS as a function of calorimetric power by subtracting the old ΔP contribution to SWS, rescaling the remainder (i.e., the non-ΔP sensitivities), and then adding in the revised ΔP SWS term which uses the power dependent sensitivity equation from Figure 2:

$$\begin{aligned} SWS' &= [SWS - (S_{\Delta P,old}(\delta\Delta P))^2/W](P_{TS}/P_{cal}) + [S_{\Delta P,new}(\delta\Delta P)]^2/W \\ &= [SWS - 0.5539](P_{TS}/P_{cal}) + [57.850/(Pcal^{1.3346})(12.0)]^2/4 \qquad \text{Eqn. 14} \end{aligned}$$

The revised calorimetric power uncertainty as a function of power is simply the square root of the revised SWS. The calculated $S_{\Delta P,new}$ terms and the revised SWS and calorimetric uncertainties are presented below vs. calorimetric power:

Calorimetric Power	Sensitivity	SWS	δP_{cal} (% RTP)
100%	1.24E-01	2.82E+00	1.68
90%	1.43E-01	2.77E+00	1.66
80%	1.67E-01	2.81E+00	1.68
70%	1.99E-01	3.02E+00	1.74
60%	2.45E-01	3.52E+00	1.88
50%	3.13E-01	4.65E+00	2.16
45%	3.60E-01	5.68E+00	2.38
40%	4.21E-01	7.28E+00	2.70
35%	5.03E-01	9.90E+00	3.15
30%	6.18E-01	1.44E+01	3.80
25%	7.88E-01	2.29E+01	4.79
20%	1.06E+00	4.10E+01	6.40
15%	1.56E+00	8.78E+01	9.37

Table 4: Revised Calorimetric Power Uncertainty Terms

Calculation of Required NIS Trip Setpoint

The NIS high flux trip setpoint is relied upon for overpower protection. The safety analysis limit for the high flux trip is 118% RTP (References 8 and 9). The actual setpoint must be such that at the setpoint, the sum of the indicated NIS power (i.e., the setpoint) and the NIS Channel Statistical Allowance (CSA) is less than or equal to the safety analysis limit. The CSA is presented in Reference 9 and in terms of % span (which corresponds to 120% RTP for the NIS channels) is

$$\begin{aligned} CSA &= [(1.7)^2 + (4.2)^2 + (0.5 + 0.3 + 1.0)^2 + (0.5)^2]^{1/2} \\ &= 4.9\% \text{ span} \end{aligned} \qquad \text{Eqn. 15}$$

The first term (1.7% span) is the PMA term for the calorimetric power uncertainty applicable to the most recent NIS adjustment. See Reference 9 for discussion of the other terms; none of these terms are affected by Reference 1 nor do they change as a result of any of the assumptions of this analysis. We will modify the CSA equation as follows:

1) Combine all terms other than the calorimetric error and treat the combination as a constant based on the above justification.

2) Convert those constant terms from % span to % RTP by multiplying the sum of their squares by $(\%RTP/\%span)^2 = (120/100)^2 = 1.44$.

3) Replace the calorimetric power uncertainty PMA term $((1.7)^2)$ with the square of the combined calorimetric and difference uncertainty term (δP_{TS}) from Equation 7.

The resulting CSA equation is:

$$CSA = [(\delta P_{cal}(P_{TS}/P_{cal}))^2 + (2\%\ RTP(P_{TS}/P_{NIS}))^2 + 1.44(21.13)]^{1/2} \qquad \text{Eqn. 16}$$

In order to protect the safety analysis limit, the allowable NIS high flux trip setpoint, P_{TS}, must be set such that:

$$P_{TS} \le 118 - CSA \qquad \text{Eqn. 17}$$

Substituting the above expression in for the CSA, isolating the radical and squaring, we obtain:

$$P_{TS}^2 - 236\,P_{TS} + 118^2 \le -[(\delta P_{cal}(P_{TS}/P_{cal}))^2 + (2\%\ RTP(P_{TS}/P_{NIS}))^2 + 30.427] \qquad \text{Eqn. 18}$$

Putting this into standard quadratic form ($ax^2+bx+c=0$):

$$[1-(\delta P_{cal}/P_{cal})^2 + (2\%\ RTP/P_{NIS})^2]P_{TS}^2 - 236\,P_{TS} + (118^2 - 30.427) \le 0 \qquad \text{Eqn. 19}$$

We may now solve for PTS as a function of Pcal (we assume that PNIS = Pcal, which is acceptable since we are explicitly accounting for a 2% RTP random difference) using the quadratic formula, replacing "=" with "≤":

$$x \le (1/2a)\,(-b - (b^2 - 4ac)^{1/2}) \qquad \text{Eqn. 20}$$

where a, b and c are the appropriate coefficients from Equation 19. The resulting trip setpoint requirements as a function of calorimetric power are presented below (and in summary form in Table 1):

Calorimetric Power	Required NIS Trip Setpoint
100 % RTP	≤ 112 % RTP
≥ 90 % RTP	≤ 112 % RTP
≥ 80 % RTP	≤ 111 % RTP
≥ 70 % RTP	≤ 111 % RTP
≥ 60 % RTP	≤ 111 % RTP
≥ 50 % RTP	≤ 110 % RTP
≥ 45 % RTP	≤ 109 % RTP
≥ 40 % RTP	≤ 107 % RTP
≥ 35 % RTP	≤ 105 % RTP
≥ 30 % RTP	≤ 102 % RTP
≥ 25 % RTP	≤ 97 % RTP
≥ 20 % RTP	≤ 88 % RTP
≥ 15 % RTP	≤ 72 % RTP

Table 5: Required NIS High Flux Trip Setpoints as a function of Calorimetric Power

References

1 Westinghouse Technical Bulletin ESBU-TB-92-14-R1, "Decalibration Effects of Calorimetric Power Measurements on the NIS High Power Reactor Trip at Power Levels Less Than 70% RTP," 2/6/96.

2 Action Request #A0403054.

3 DCPP Procedure STP R-2B1, "PPC Operator Heat Balance," Rev. 6.

4 DCPP Procedure STP R-2B2, "Manual Operator Heat Balance," Rev. 4.

5 WCAP 13556, "Bases Document for Westinghouse Setpoint Methodology for Protection Systems," Rev. 0.

6 NSSS Calculation N-184, "Power Calorimetric Uncertainty."

7 NSSS Calculation N-191, "2C7 Uncertainties for AMAG UFM."

8 DCPP FSAR Update, Section 15.1.

9 WCAP 11082, "Westinghouse Setpoint Methodology for Protection Systems; Diablo Canyon Stations Eagle 21 Version," Rev. 2.