Pacific Gas and Electric Company®

James R. Becker
Site Vice President

Diablo Canyon Power Plant
Mail Code 104/5/601
P. O. Box 56
Avila Beach, CA 93424

805.545.3462
Internal: 691.3462
Fax: 805.545.6445

February 24, 2009

PG&E Letter DCL-09-009

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80, and Docket No. 50-323, OL-DPR-82

Subject: License Amendment Request 09-01
Revision to Technical Specification 3.3.1, "Reactor Trip System (RTS) Instrumentation," Deletion of the High Negative Rate Trip Function

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.90, Pacific Gas and Electric (PG&E) hereby requests a License Amendment to revise the Diablo Canyon Power Plant (DCPP) Units 1 and 2 Technical Specification (TS) for 3.3.1, "Reactor Trip System (RTS) Instrumentation." The amendment will delete the requirement for the power range neutron flux rate-high negative rate trip function. This function is specified in TS Table 3.3.1-1, "Reactor Trip System Instrumentation," as Function 3.b, "Power Range Neutron Flux Rate-High Negative Rate." The proposed changes are consistent with the NRC-approved methodology presented in Westinghouse Topical Report, WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990.

The enclosure provides a technical and regulatory evaluation of the changes, and one regulatory commitment. Proposed TS and Bases page markups are included as attachments to the enclosure.

Due to a current Unit 2 rod alignment issue, approval of the proposed amendment is requested by May 2, 2009, so that PG&E could implement the amendment prior to the next quarterly surveillance test, which must be performed by May 25, 2009. PG&E requests the license amendment(s) be made effective upon NRC issuance, to be implemented no later than startup from the Unit 1 Sixteenth Refueling Outage and the Unit 2 Fifteenth Refueling Outage, currently scheduled to finish in November 2010, and November 2009, respectively.

If you have any questions or require additional information, please contact Stan Ketelsen at 805-545-4720.

I state under penalty of perjury that the foregoing is true and correct.

Executed on February 24, 2009.

Sincerely,

James R. Becker
Site Vice President

lmp1/3386/N50200784
Enclosure
cc: Gary W. Butner, California Department of Public Health
Elmo E. Collins, NRC Region IV
Diablo Distribution
cc/enc: Michael S. Peck, NRC, Senior Resident Inspector
Alan B. Wang, NRC Project Manager, Office of NRR

Document Components:
PG&E Letter DCL-09-009.pdf 1,054,000 Bytes (approximately)

Evaluation of the Proposed Change

Subject: License Amendment Request 09-01
Revision to Technical Specification 3.3.1, “Reactor Trip System (RTS) Instrumentation" Deletion of the High Negative Rate Trip Function

1. SUMMARY DESCRIPTION
2. DETAILED DESCRIPTION
3. TECHNICAL EVALUATION
4. REGULATORY EVALUATION
    - 4.1 Applicable Regulatory Requirements/Criteria
    - 4.2 Precedent
    - 4.3 Significant Hazards Consideration
    - 4.4 Conclusions
5. ENVIRONMENTAL CONSIDERATIONS
6. REFERENCES

ATTACHMENTS:

1. Technical Specification Page Markups
2. Bases Page Markups
3. Retyped Technical Specification (and/or Operating License) Pages
4. List of Regulatory Commitments

1. SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating Licenses DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed changes would revise the Operating Licenses to delete Function 3.b, "Power Range Neutron Flux Rate-High Negative Rate" (also referred to as the negative flux rate trip [NRFT] in this discussion), in Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation." These changes are based on the NRC approved methodology contained in Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event" (Reference 1).

Deleting Function 3.b, NFRT, has the following benefits:

- Avoids unnecessary automatic reactor trips in response to a single rod drop event.
- Removes the concern of an unnecessary automatic reactor trip resulting from a dropped rod while performing routine surveillance tests.

In summary, the proposed change to delete Function 3.b, "Power Range Neutron Flux Rate-High Negative Rate," in TS 3.3.1, "Reactor Trip System Instrumentation," is well within current analyses, and will reduce the potential for unnecessary trip risks.

2. DETAILED DESCRIPTION

This License Amendment Request (LAR) would delete Function 3.b, "Power Range Neutron Flux Rate-High Negative Rate," trip from TS 3.3.1 Table 3.3.1-1. This change would not affect Function 3.a, "Power Range Neutron Flux Rate-High Positive Rate."

Currently, DCPP Unit 2 has a single control rod that is slightly misaligned. An operability determination has concluded the rod is tripable and still operable. Due to industry operating experience with misaligned rods, there is concern that manipulation of this rod may result in it dropping. Due to the location of this rod in the core, dropping this rod would likely result in a reactor trip. Deleting the TS trip function would allow operators to re-align this rod with the associated bank, without the increased threat of an unnecessary reactor trip. This rod must be manipulated in the next quarterly TS Surveillance, which is due May 2, 2009, and must be complete no later than May 25, 2009.

The LAR formally implements the methodology contained in NRC-approved Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event" (Reference 1). Westinghouse has been performing these conservative analyses for DCPP which do not credit the NFRT function, but until now, Pacific Gas and Electric Company (PG&E) has not pursued removing the NFRT from the plant TS.

3. TECHNICAL EVALUATION

3.1 System Description

The DCPP Final Safety Analysis Report Update (FSARU), Section 7.2.1, describes the power range high negative nuclear power rate trip function. This circuit trips the reactor when an abnormal rate of decrease in nuclear power occurs in two of the four power range channels. While no longer credited for the dropped rod analysis presented in Section 15.2.3, this trip was designed to provide protection against two or more dropped rods and is currently active. Protection against one dropped rod was never required to prevent occurrence of departure from nucleate boiling (DNB) at full power per the analysis of Section 15.2.3.

The DCPP FSARU, Section 15.2.3, describes rod cluster control assembly (RCCA) misoperation as a Condition II event, faults of moderate frequency. RCCA misoperation includes one or more dropped RCCAs, a dropped RCCA bank, or a statically misaligned RCCA.

Analysis results for one or more dropped RCCAs or a dropped RCCA bank follow:

(1) One or More Dropped RCCAs

The dropped rod accident is assumed to be initiated by a single electrical or mechanical failure which causes any number and combination of rods from the same group of a given bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to quickly decrease. The core is not adversely affected during this period since power is decreasing rapidly.

Power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern, and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Uncertainties in the initial conditions are included in the departure from nucleate boiling ratio (DNBR) evaluation as described in WCAP-11394-P-A. In all cases, the minimum DNBR remains above the safety analysis limit value.

Following plant stabilization, the operator may manually retrieve the RCCA(s) by following approved operating procedures.

(2) Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion of greater than 500 pcm. The core is not adversely affected during this period since power is decreasing rapidly. The transient will proceed as described in the previous section, except the return to power will be less severe due to the greater worth of the entire bank. The power transient for a dropped RCCA bank is symmetric. Following plant stabilization, normal procedures are followed.

WCAP-11394-P-A concluded that sufficient DNBR margin existed for Westinghouse plant designs and fuel types without crediting the power NFRT function, for a generically bounding reactivity worth of the dropped RCCA or RCCA bank, which is confirmed on a plant and cycle-specific basis. The NRC Safety Evaluation for WCAP-11394-P concluded that the analysis contained an acceptable analysis procedure for analyzing the dropped RCCA event for which no credit is taken for any direct reactor trip due to the dropped RCCA(s) or for an automatic power reduction due to the dropped RCCA(s). Westinghouse has performed the analysis of the DCPP dropped RCCA event in accordance with the methodology contained in WCAP-11394-P-A, and determined that the required DCPP plant specific comparisons ensure that favorable DNBR results are obtained. Since this analysis does not take credit for the NFRT, the NFRT function is not required to ensure that the design DNBR limits are met. Thus, this trip function can be deleted.

3.2 NFRT Design Basis

The original design basis for the NFRT function was to mitigate the consequences of two or more dropped RCCAs. The dropped RCCA event is an anticipated operational occurrence, and is caused by a single electrical or mechanical failure that results in one or more RCCAs from the same group of a given bank to drop to the bottom of the core. The resulting negative reactivity insertion causes nuclear power to quickly decrease and core radial peaking factors to increase. The reduced power and continued steam generation cause the reactor coolant temperature to decrease. In the manual control mode, the positive reactivity feedback due to dropping temperature causes the reactor power to rise to initial power level at a reduced reactor vessel inlet temperature with no power overshoot. In the automatic control mode, the plant control system detects the reduction in core power and initiates control bank withdrawal in order to restore core power. As a result, power overshoot occurs, resulting in a lower calculated DNBR. At higher power levels, in the event of a dropped RCCA event, the RTS will detect the rapidly decreasing neutron flux due to the dropped RCCAs and trip the reactor based on the NFRT function, thus ending the transient and assuring that DNBR design limits are maintained. Since the dropped RCCA event is an anticipated operational occurrence, it must be shown that to satisfy General Design Criterion (GDC) 10, "Reactor Design," requirements, the DNBR design limits are met for the combination of high nuclear power, high radial peaking factor, and other system conditions that exist following the dropped RCCA event.

In 1982, an evaluation prepared by Westinghouse and documented in WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," determined that the NFRT function was only required when a dropped RCCA or RCCA bank exceeded a specific threshold value for reactivity worth. Any dropped RCCA or RCCA bank which had a reactivity worth below the threshold value would not require a reactor trip to maintain DNBR limits. An additional evaluation method, WCAP-11394-P, was developed by Westinghouse in 1987, which determined that sufficient DNBR margin existed for Westinghouse plant designs and fuel types without the NFRT function regardless of the reactivity worth of the dropped RCCA or RCCA bank, subject to a plant/cycle-specific analysis. The NRC subsequently reviewed and approved the Westinghouse analysis methodology and results, and concluded that the analysis contained an acceptable procedure for analyzing the dropped RCCA event for which no credit is taken for any direct reactor trip or automatic power reduction due to the dropped RCCA(s). Therefore, the NFRT function is not required to maintain existing DNBR limits and may be deleted.

3.3 Evaluation of Other Analyses

The following provides an evaluation of the proposed change with respect to other DCPP safety analyses and evaluations.

Loss of Coolant Accident (LOCA) and LOCA-Related Evaluations

The NFRT function is not modeled in the LOCA analyses. The following LOCA-related analyses are not affected by the proposed changes: large and small break LOCA, reactor vessel and RCS loop LOCA blowdown forces, post-LOCA long-term core cooling subcriticality, post-LOCA long-term core cooling minimum flow, and RCS hot leg switchover to prevent boron precipitation. The proposed changes do not affect the normal plant operating parameters, accident mitigation capabilities important to a LOCA, the assumptions used in the analysis of LOCA-related accidents, or create conditions more limiting than those assumed in these analyses.

Non-LOCA-Related Evaluations

The current non-LOCA safety analyses do not take credit for the NFRT function. Specifically, the dropped RCCA(s) analyses utilized for the current Unit 1 and Unit 2 cycles do not rely on actuation of the NFRT function to mitigate the consequences of the accident. These analyses were performed in accordance with the NRC-approved methodology for the analysis of dropped RCCA(s) events provided in WCAP-11394-P-A. The analysis assumptions and confirmation that the DNBR design basis is met are further confirmed as part of the reload safety analysis for each reactor core reload. The current reload safety analyses for DCPP Unit 1 Cycle 16, and Unit 2 Cycle 15, confirm that the DNBRs predicted for the dropped RCCA remain within safety analysis limits. Therefore, the conclusion presented in the FSARU, Chapter 15, that the DNBR design basis is met with respect to non-LOCA related evaluations remains valid for the proposed changes. These changes credit the application of WCAP-11394-P-A.

Mechanical Components and Systems Evaluation

Deletion of the NFRT function as described above does not affect RCS component integrity or the ability of the RCS to perform its intended safety function. The proposed changes do not affect the integrity of plant systems or their ability to perform intended safety functions.

Containment Integrity Evaluation

The NFRT function is not credited in the containment analyses. The proposed changes do not adversely affect the short-term or long-term LOCA mass and energy releases or the main steamline break mass and energy release inside containment assumed in the containment analyses. The proposed changes do not affect the normal plant operating parameters, system actuations, capabilities or assumptions important to the containment analyses, or create conditions more limiting than those assumed in these analyses. Therefore, the conclusions presented in the FSARU remain valid with respect to the containment analyses.

Main Steamline Break (MSLB) Outside Containment Evaluation

The NFRT function is not credited in the FSARU MSLB analyses. The proposed changes do not adversely affect the MSLB mass and energy releases outside containment for equipment qualification, and do not adversely affect the calculations for the steam mass release used as input to the radiological dose evaluation. The proposed changes do not affect the normal plant operating parameters, input assumptions, results or conclusions of the MSLB mass and energy release analyses, and steam release calculations. Also, conditions are not created which are more limiting than those evaluated in the current analyses. Therefore, the conclusions presented in the FSARU remain valid with respect to MSLB mass and energy release rates outside containment for equipment qualification and offsite dose calculations.

Emergency Operating Procedure (EOPs) Evaluation

Deletion of the NFRT function will not adversely affect the EOPs. Responding to dropped or misaligned RCCA events are covered by abnormal operating procedures which instruct the operators to manually trip the reactor for multiple dropped RCCAs.

Safety System Allowable Values and Setpoints Evaluation

Deletion of the NFRT function does not change the current allowable value information for any other function shown in the TS, and does not change the current setpoint information for any other function shown in the Equipment Control Guidelines. Therefore, since no credit for the NFRT function is taken in the safety analyses, the NFRT deletion has no impact on the plant safety functions.

Steam Generator Tube Rupture (SGTR) Evaluation

The NFRT function is not credited in the SGTR analyses. The proposed changes do not adversely affect the normal plant operating parameters, results or conclusions of the SGTR thermal and hydraulic analyses. Also, conditions are not created which are more limiting than those evaluated by the current analyses for break flow and steam release. Therefore, the conclusions presented in the FSARU remain valid with respect to the SGTR event.

Control Systems Evaluation

The proposed changes have no adverse impact on the control systems evaluation. The deletion of the NFRT function could increase plant availability because the proposed changes eliminate a potential source of inadvertent reactor trips.

Physical Security Evaluation

Deletion of the NFRT function will not adversely affect the physical security plan.

3.4 Summary/Conclusion

For the past several fuel cycle designs, a dropped RCCA analysis has been performed in accordance with the methodology described in WCAP-11394-P-A. PG&E will use the NRC-approved methodology in WCAP-11394-P-A for each fuel cycle to ensure the minimum DNBR is maintained above the DNBR safety limit. The NFRT function is not credited in the current cycle-specific dropped RCCA analysis, the current analysis conforms to WCAP-11394-P-A, and DCPP continues to meet the applicable DNBR limits.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following regulatory requirements are applicable to the proposed TS changes discussed in the license amendment application.

GDC 10, requires that the reactor core and associated coolant, control and protection system be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Also, 10 CFR 50.36 (c) (2) (ii), stipulates that a TS limiting condition for operation (LCO) must be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

> Discussion: The NFRT is not used for detection and indication in the control room of any degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

> Discussion: The NFRT is not an initial condition of a design basis accident or transient analysis.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

> Discussion: No credit is taken for the NFRT in the DCPP accident analysis. The NFRT is not considered as part of the primary success path related to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

> Discussion: The NFRT is not relied upon as a signal to initiate a reactor trip for any events modeled in the scope of the probabilistic risk analysis model. The NFRT function is not significant to public health and safety in that no credit was taken for this trip in any accident analysis.

Therefore, the NFRT does not meet any of the four criteria of 10 CFR 50.36, and does not warrant inclusion in the technical specifications as an LCO.

4.2 Precedent

The following are three recent examples where similar plants have received license amendments to delete the negative flux rate trip function.

- PSE&G, Salem Nuclear Generating Station, Units Nos. 1 and 2, in Amendment Nos. 278 and 261 to Facility Operating License Nos. DPR-70 and DPR-75, “Salem Nuclear Generating Station, Unit Nos. 1 and 2, Issuance of Amendments re: Power Range Neutron Flux High Negative Rate Trip,” dated March 19, 2007.

- AEP, Donald C. Cook Nuclear Plant, Units 1 and 2, in Amendment Nos. 293 and 275 to Facility Operating License Nos. DPR-58 and DPR-74, "Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments to Delete the Power Range Neutron Flux High Negative Rate Trip Function," dated February 10, 2006.

- Southern Company, Joseph M. Farley Nuclear Plant, Units 1 and 2 in Amendment Nos. 171 and 164 to Facility Operating License Nos NPF-2 and NPF-8, "Joseph M. Farley Nuclear Plant, Units 1 and 2 re: Issuance of Amendments," dated February 27, 2006.

4.3 No Significant Hazards Consideration

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” 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 removal of the power range neutron flux rate-high negative rate trip function from the DCPP TS does not increase the probability or consequences of accidents resulting from dropped RCCA events previously analyzed. The safety functions of other safety-related systems and components, which are related to mitigation of these events, have not been altered. All other reactor trip system protection functions are not impacted by the deletion of the trip function. The dropped RCCA accident analysis does not rely on the negative flux rate trip to safely shut down the plant. The safety analysis of the plant is unaffected by the proposed change. Since the safety analysis is unaffected, the calculated radiological releases associated with the analysis are not affected.

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 accident from any accident previously evaluated?

Response: No.

The proposed change does not adversely alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related systems or components. NRC-approved Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990, has demonstrated that the negative flux rate trip function can be deleted.

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

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

Response: No.

The margin of safety associated with the acceptance criteria of any accident is unchanged. It has been demonstrated that the negative flux rate trip function can be deleted by the NRC-approved methodology described in WCAP-11394-P-A. DCPP cycle-specific analyses have confirmed that for dropped RCCA events, limits on DNB are not exceeded by deleting the negative flux rate trip. The proposed change will have no effect on the availability, operability, or performance of safety-related systems and components.

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

Based on the above, PG&E concludes that the proposed amendment does not involve a 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.

4.4 Conclusions

In conclusion, 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) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendment and has determined that the proposed amendment does not involve: (1) a significant hazards consideration, (2) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) a significant increase in 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), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6 REFERENCES

6.1 References

1. Westinghouse Electric Company WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990 (proprietary).
2. Westinghouse Electric Company WCAP-11395-NP, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990 (nonproprietary)
3. Westinghouse Electric Company WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," dated June 1983 (proprietary).
4. NRC Letter from A. C. Thadani (U.S. NRC) to R. A. Newton (Westinghouse Owners Group), "Acceptance for Referencing of Licensing Topical Reports WCAP-11394(P) and WCAP-11395(NP), 'Methodology for the Analysis of the Dropped Rod Event,'" dated October 23, 1989.

Proposed Technical Specification Changes (marked-up)

Page:
3.3-3
3.3-12

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------------NOTE------------------ For functions 6, 7, and 8.b, the inoperable channel and/or one additional channel may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours, or both the inoperable and the additional channel may be surveillance tested in bypass for up to 12 hours. For functions 2.b~~,~~ *and* 3~~.a~~, ~~and 3.b,~~ only the inoperable channel may be bypassed for surveillance testing of other channels. For function 14.a, the inoperable channel and/or one additional channel may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours. This note is not intended to allow simultaneous testing of coincident channels on a routine basis ---	
	E.1 Place channel in trip.	72 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	78 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	24 hours
	<u>OR</u>	
	F.2 Increase THERMAL POWER to > P-10.	24 hours

(continued)

Table 3.3.1-1 (page 1 of 7)

Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[(a)] TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	3[(b)], 4[(b)], 5[(b)]	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110.2% RTP	109% RTP
b. Low	1[(c)],2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 26.2% RTP	25% RTP
3. Power Range Neutron Flux Rate						
~~a.~~ High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 5.6% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
~~b. High Negative Rate~~	~~1,2~~	~~4~~	~~E~~	~~SR 3.3.1.7~~ ~~SR 3.3.1.11~~ ~~SR 3.3.1.16~~	~~≤ 5.6% RTP with time constant ≥ 2 sec~~	~~5% RTP with time constant ≥ 2 sec~~
4. Intermediate Range Neutron Flux	1[(c)], 2[(d)]	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.6% RTP	25% RTP

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlocks.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

Proposed Technical Specification Changes (retyped)

Remove Page	**Insert Page**
3.3-3	**3.3-3**
3.3-12	**3.3-12**

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	------------------NOTE------------------ For functions 6, 7, and 8.b, the inoperable channel and/or one additional channel may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours, or both the inoperable and the additional channel may be surveillance tested in bypass for up to 12 hours. For functions 2.b and 3, only the inoperable channel may be bypassed for surveillance testing of other channels. For function 14.a, the inoperable channel and/or one additional channel may be surveillance tested with one channel in bypass and one channel in trip for up to 12 hours. This note is not intended to allow simultaneous testing of coincident channels on a routine basis ---	
	E.1 Place channel in trip.	72 hours
	<u>OR</u>	
	E.2 Be in MODE 3.	78 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	24 hours
	<u>OR</u>	
	F.2 Increase THERMAL POWER to > P-10.	24 hours

(continued)

Table 3.3.1-1 (page 1 of 7)

Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL(a) TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	3(b), 4(b), 5(b)	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110.2% RTP	109% RTP
b. Low	1(c),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 26.2% RTP	25% RTP
3. Power Range Neutron Flux Rate						
High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 5.6% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(c), 2(d)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.6% RTP	25% RTP

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlocks.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

Changes to Technical Specification Bases Pages
(For information only)

B 3.3.1 Pages:

10
11
38
65

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Power Range Neutron Flux—Low (continued)

The LCO requires all four of the Power Range Neutron Flux—Low channels to be OPERABLE (2-out-of-4 coincidence).

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux—Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than or equal to 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux—High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux—Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.

~~a.~~ Power Range Neutron Flux—High Positive Rate

The Power Range Neutron Flux—High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function complements the Power Range Neutron Flux—High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux—High Positive Rate channels to be OPERABLE (2-out-of-4 coincidence).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~a.~~ Power Range Neutron Flux—High Positive Rate (continued)

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux—High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux—High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

~~b. Power Range Neutron Flux—High Negative Rate~~

~~The Power Range Neutron Flux—High Negative Rate trip Function ensures that protection is provided for multiple rod drop accidents. At high power levels, a multiple rod drop accident could cause local flux peaking that would result in an unconservative local DNBR. DNBR is defined as the ratio of the heat flux required to cause a DNB at a particular location in the core to the local heat flux. The DNBR is indicative of the margin to DNB. No credit is taken for the operation of this Function for those rod drop accidents in which the local DNBRs will be greater than the limit.~~

~~The LCO requires all four Power Range Neutron Flux—High Negative Rate channels to be OPERABLE (2-out-of-4 coincidence).~~

~~In MODE 1 or 2, when there is potential for a multiple rod drop accident to occur, the Power Range Neutron Flux—High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux—High Negative Rate trip Function does not have to be OPERABLE because the core is not critical and DNB is not a concern. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the required SDM is increased during refueling operations. In addition, the NIS power range detectors cannot detect neutron levels present in this MODE.~~

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux—Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Power Range Neutron Flux—High Positive Rate;
- ~~Power Range Neutron Flux—High Negative Rate;~~
- Pressurizer Pressure—High; and
- SG Water Level—Low Low.

A known inoperable channel must be placed in the tripped condition within 72 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 72 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 28.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note for Functions 6, 7 and 8.b, that allows an inoperable channel and/or one additional channel to be tested with one channel in bypass and the other channel in trip for up to 12 hours for performing surveillance testing. Additionally, for Function 6, 7 and 8b, both the inoperable and the additional channel maybe placed in bypass for up to 12 hours for surveillance testing. The Note allows only the inoperable channel for Functions 2.b~~,~~ *and* 3~~.a~~, ~~and 3.b,~~ to be bypassed for up to 12 hours for surveillance testing of other channels. This note is not intended to allow simultaneous testing of coincident channels on a routine basis. In accordance with WCAP 10271, very specific circumstances are related to the use of this bypass condition for RTS Functions 2.b~~,~~ *and* 3~~.a, and 3.b~~. Since these channels are not designed with Bypass-capable logic that meets the requirements of IEEE 279, the provisions for bypass only apply to a specific type of channel failure. To apply, the channel must fail in such a way that it does not trip the bistables. With this type of failure, the channel may be returned to service and considered "bypassed" under this Note. Specifically, the bypass condition is the state when a failed channel is taken out of the forced "tripped" state and placed in operation.

(continued)

BASES

REFERENCES (continued)

17. WCAP-11082, Rev. 6, "Westinghouse Setpoint Methodology for Protection Systems, Diablo Canyon Units 1 and 2, 24 Month Fuel Cycle Evaluation," February 2003.

18. NSP-1-20-13F Unit 1 "Turbine Auto Stop Low Oil Pressure."

19. NSP-2-20-13F Unit 2 "Turbine Auto Stop Low Oil Pressure."

20. J-110 "24 Month Fuel Cycle Allowable Value Determination / Documentation and ITDP Uncertainty Sensitivity."

21. IEEE Std. 338-1977.

22. License Amendment 61/60, May 23, 1991.

23. 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.

24. DCPP NSSS Calculation N-212, Revision 1.

25. License Amendments 157/157, June 2, 2003.

26. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994.

27. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

28. WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.

29. WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," March 2003.

30. WCAP-11394-P-A, "Methodology For The Analysis of the Dropped Rod Event," January, 1990.

List of Regulatory Commitments

The following table identifies the regulatory commitments in this document. Any other statements in this submittal represent intended or planned actions, are provided for information purposes, and are not considered to be regulatory commitments.

COMMITMENT	TYPE	DUE DATE
Pacific Gas and Electric Company will use the NRC-approved methodology in WCAP-11394-P-A for each fuel cycle to ensure the minimum departure from nucleate boiling ratio (DNBR) is maintained above the DNBR safety limit.	Continuing	Concurrent with implementation of the amendment