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December 29, 2009

PG&E Letter DCL-09-093

10 CFR 50.90

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 09-07, Delayed Access Offsite Power Circuit Conformance with GDC 17

Reference: 1. NRC Integrated Inspection Report 05000275/2009003 and 05000323/2009003, dated August 5, 2009.

In accordance with 10 CFR 50.90, enclosed is an application to amend Facility Operating License Nos. DPR-80 and DPR-82 for Diablo Canyon Power Plant (DCPP) Units 1 and 2, respectively.

The proposed amendments would revise the licensing basis as described in the Final Safety Analysis Report Update (FSARU) to discuss the conformance of the delayed access offsite power circuit (the 500-kV delayed access circuit) to the General Design Criterion 17 requirement that each of the offsite power circuits be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. This change is being made in response to the enforcement action discussed in Section 4OA5 of Reference (1).

The proposed amendment would also add information related to reactor coolant pump seal performance during and after (1) a loss of seal injection (with continued thermal barrier cooling); (2) a loss of thermal barrier cooling (with continued seal injection); and (3) a loss of all seal cooling (both thermal barrier cooling and seal injection).

The enclosure contains a description of the changes to the licensing basis, the supporting technical evaluation, and the no significant hazards consideration determination. Attachment 1 to the enclosure contains marked-up FSARU pages.

Pacific Gas and Electric (PG&E) has determined that this license amendment request (LAR) does not involve a significant hazard consideration as determined per

A001

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 changes in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR no later than December 29, 2010. PG&E requests the license amendments be made effective upon NRC issuance, to be implemented within 180 days from the date of issuance.

PG&E makes no regulatory commitments (as defined by NEI 99-04) in this letter. This letter includes no revisions to existing regulatory commitments.

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

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

Executed on December 29, 2009.

Sincerely,

James R. Becker
Site Vice President

tcg5/4231 DN#50109826
Enclosure
cc: Gary W. Butner, California Department of Public Health
Elmo E. Collins, NRC Region IV Regional Administrator
Diablo Distribution
cc/enc: Michael S. Peck, NRC, Senior Resident Inspector
Alan B. Wang, NRC Project Manager, Office of Nuclear Reactor Regulation

EVALUATION OF THE PROPOSED CHANGE

License Amendment Request 09-07, "Delayed Access Offsite Power Circuit Conformance with GDC 17"

ATTACHMENT:

1. Final Safety Analysis Report Update Page Markups

1. SUMMARY DESCRIPTION

This letter is a License Amendment Request (LAR) to amend Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively.

The proposed amendments would revise the licensing basis as described in the Final Safety Analysis Report Update (FSARU) to discuss the conformance of the delayed access offsite power circuit (the 500-kV delayed access circuit) to the General Design Criterion (GDC) 17 requirement that each of the offsite power circuits be designed to be available in sufficient time following a loss of all onsite alternating current (ac) power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. This change is being made in response to the enforcement action discussed in Section 4OA5 of Reference (1).

The proposed amendment would also add information to the FSARU related to reactor coolant pump (RCP) seal performance during and after (1) a loss of seal injection (with continued thermal barrier cooling); (2) a loss of thermal barrier cooling (with continued seal injection); and (3) a loss of all seal cooling (both thermal barrier cooling and seal injection).

2. DETAILED DESCRIPTION

Proposed Amendment

The proposed change would revise FSARU Section 8.2.1.2, "500-kV System," to add the following: "Plant procedures contain actions for operators to complete the 500-kV backfeed, isolation of RCP seal cooling, and restoration of RCS makeup flow within approximately 54 minutes upon loss of 230-kV and all onsite ac power. Completion of these actions within this time period assures that specified acceptable fuel design limits and design considerations of the reactor coolant pressure boundary are not exceeded. (Also see Sections 5.5.1.3.1 and 8.3.1.6)." This change would also delete the phrase "after about 30 minutes" from the current statement in FSARU Section 8.2.1.2 that reads "In the event of a loss of main generator output, the 500-kV backup source of auxiliary power could be placed in service after about 30 minutes."

The proposed change would also revise FSARU Section 5.5.1, "Reactor Coolant Pumps," to include a discussion of RCP seal performance during and after (1) a loss of seal injection (with continued thermal barrier cooling); (2) a loss of thermal barrier cooling (with continued seal injection); and (3) a loss of all seal cooling (both thermal barrier cooling and seal injection).

<u>Purpose for Proposed Amendment</u>

NRC Integrated Inspection Report 05000275/2009003 AND 05000323/2009003, dated August 5, 2009, identified the following issue:

> *The inspectors identified a noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a thermal hydraulic analysis to determine if prior NRC approval was required for a 30-minute delay time to align offsite power. This analysis, Calculation STA 274, "RETRAN Evaluation of GDC-17 Loss of AC Scenario," Revision 0, demonstrated that the 30-minute delayed offsite power source was acceptable. On December 31, 2008, a Pacific Gas and Electric 10 CFR 50.59 screen concluded that Calculation STA-274 was not required to be evaluated to determine if prior NRC approval was required for the delay time. On March 31, 2009, the inspectors concluded that the licensee was required to evaluate Calculation STA-274 to determine if prior NRC approval was needed. On May 27, 2009, Pacific Gas and Electric completed the 50.59 evaluation and concluded that prior NRC approval was required for the 30-minute delay time to align offsite power.*

The original FSAR stated that, "In the event of a loss of main generator output, this backup source of auxiliary power could be placed in service in approximately 30 seconds." This statement was clarified in amendment 34 to the FSAR, submitted during the licensing review, that added the phrase "plus operator time, from the time that the 500-kV breaker is tripped, which will permit manual control opening of the link." The link is the motor operated disconnect in the generator's main leads. The operator actions to align the 500-kV delayed access circuit for backfeed were further clarified in revisions to the FSARU made subsequent to issuance of the DCPP operating licenses. The 30 minute time to align the 500-kV delayed access circuit for back feed was added in FSARU Revision 13. That change was made in part to reflect the procedures for backfeeding from the 500-kV system.

This proposed amendment is being submitted to obtain NRC approval of the time to align the 500-kV delayed access offsite power circuit for backfeeding, isolate RCP seal cooling, and restore reactor coolant system (RCS) makeup flow upon loss of 230-kV and all onsite emergency ac power sources (i.e. the emergency diesel generators) to meet GDC 17 requirements.

3.0 TECHNICAL EVALUATION

Offsite Power System Description

DCPP has two offsite sources of power:

- 500-kV Switchyard supplied from three transmission lines
- 230-kV Switchyard supplied from two transmission lines

The function of the 500-kV system is to transmit power generated by the main generator to the 500-kV transmission system. The 500-kV system also functions as the delayed source of offsite power. A motor operated disconnect switch on each unit's 25 kV isolated phase bus must be opened to back feed power to a unit's auxiliaries.

The function of the 230-kV system is to provide offsite power to the 12 kV underground distribution system and to the plant's electrical system required for safe shutdown and startup of the plant. The 230-kV system is the immediately available offsite source of power. On an accident or a unit trip, unit loads necessary for continued plant safety are automatically transferred to the 230-kV source if available.

A schematic of the offsite power system is included in Figure 1.

GDC 17 Requirements Regarding Offsite Power Circuits

GDC 17 states, in part regarding offsite power circuits, that, "each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded."

Loss of the 230-kV immediate access circuit and the loss of all onsite emergency ac power sources result in the loss of RCP seal cooling (loss of both thermal barrier cooling and seal injection).

In response to questions from the Senior Resident Inspector regarding statements in the FSARU related to the time required to align the 500-kV delayed access circuit, Pacific Gas and Electric (PG&E) performed design calculation STA-274 to establish that the operators have at least one hour to complete the necessary actions associated with establishing the 500-kV backfeed, isolating RCP seal cooling, and restoring RCS makeup flow. PG&E enhanced the applicable operating procedures and performed simulator demonstrations with operating crews to evaluate the time required to complete these necessary actions and demonstrated that these actions can be reliably completed within approximately 54 minutes.

RCP Seal Performance

The RCP seal assembly consists of three seals located in series along the pump shaft just above the main flange. Cooling of the pump components above the impeller, including the bearings and seals, is provided by the thermal barrier heat exchanger and a seal water injection system. The thermal barrier heat exchanger is located between the impeller and the lower radial pump bearing, while seal injection is introduced between the thermal barrier heat exchanger and the lower radial bearing. Either cooling mode is capable of maintaining the pump components in an acceptable temperature range to prevent damage during operation or beyond design basis events. An overview of RCP seal design and operation is provided in WCAP-16396-NP, Revision 0, dated January 2005 (ADAMS Accession Number ML050320187).

FSARU Section 5.5.1.3.1 summarizes the RCP seal performance criteria for the No. 1 and No. 2 seals as follows: "Testing of pumps with the No. 1 seal entirely bypassed (full reactor pressure on the No. 2 seal) shows that relatively small leakage rates would be maintained for long periods of time. The plant operator is warned of this condition by the increase in No. 1 seal leakoff, and has time to close this line and to conduct a safe plant shutdown without significant leakage of reactor coolant to the containment. Thus, it may be concluded that gross leakage from the pump does not occur, even if seals were to suffer physical damage." Normal RCP seal leakage is in the range of three to five gpm, with the limit being five gpm per Technical Specification 3.4.13.

Westinghouse Technical Bulletin TB-04-22, "Reactor Coolant Pump Seal Performance – Appendix R Compliance and Loss of All Seal Cooling", Revision 1, dated August 9, 2005, was issued after NRC Information Notice (IN) 2005-14, which also addresses the loss of RCP seal cooling. The Technical Bulletin summarizes and provides direction on the response to IN 2005-14 for Westinghouse plants with high-temperature O-rings in their RCP seals, and is applicable to DCPP. The Technical Bulletin specifically addresses the consequences and actions for loss of all RCP cooling events, including the loss of all ac power, fires, loss of component cooling water (CCW), and loss of service water (auxiliary salt water for DCPP). The Technical Bulletin examines three conditions:

1. Loss of thermal barrier cooling to the RCP seals (CCW flow)
2. Loss of seal injection (charging flow)
3. Loss of all RCP seal cooling.

The Technical Bulletin establishes that plants may assume a maximum RCP seal leakage of 21 gpm per pump for any of these conditions. This increased RCP seal leakage occurs when <u>all</u> RCP seal cooling is lost and can not be restored before all of the existing cold liquid volume in the RCP seal chamber has leaked through the seals, and the seal surfaces heat up on exposure to the hot RCS liquid which then enters. The RCP seal leakage increases due to thermal expansion of the seal surfaces as they heat up. If the RCPs are still operating, they must be secured at this time. To prevent thermal shock damage and potentially even greater RCP seal leakage due to restoring either cold seal injection water or thermal barrier cooling to the RCP seals after they have heated up, seal cooling (both seal injection and thermal barrier cooling) must be isolated prior to restoring charging and CCW flow. There is no resulting damage to the seal and it will function acceptably as long as it is cooled down at a controlled rate as recommended by Westinghouse. This RCP seal coping strategy to limit RCP seal leakage and maintain RCP seal integrity upon a loss of all seal cooling is implemented in plant procedures.

RETRAN Evaluation of GDC 17 Loss of All AC Event

The purpose of design calculation STA-274 was to perform a thermal hydraulic evaluation of the plant response to a GDC 17 loss of all ac event to determine the maximum time allowable for implementing the 500-kV backfeed process, performing the RCP seal coping strategy actions, and restoring RCS makeup flow before any GDC 17 acceptance criteria are exceeded. The GDC 17 acceptance criterion of not exceeding any RCS pressure boundary design limit is met by demonstrating that the RCS pressure never exceeds 110 percent of the RCS ASME design value. The fuel design limit is met by demonstrating that peak clad temperature does not exceed 2200°F and the core is maintained in a coolable geometry since the GDC 17 loss of all ac event is most appropriately considered a Condition IV event as defined in the FSARU.

Based on the NUREG-0800 Standard Review Plan (SRP), the DCPP FSARU establishes more restrictive acceptance criteria for those events that are classified as more likely to occur during the life of the plant. Condition II events are defined as a "faults of moderate frequency" and are considered more likely to occur than Condition III events defined as "Infrequent Faults" and which are more likely to occur than Condition IV events defined as "Limiting Faults". While the GDC 17 loss of all ac event is not specifically analyzed or classified in the DCPP FSARU, it is considered a highly improbable beyond design basis event since it involves a loss of all onsite ac power (including emergency diesel generators) and offsite ac power which is not assumed in any other accident analyses. Therefore, the GDC 17 loss of all ac event is most appropriately categorized as a Condition IV event with a fuel design acceptance criteria based on 10CFR 50.46 for limiting peak clad temperature to less than 2200°F and maintaining a coolable core geometry. However, for this evaluation PG&E used a more restrictive Condition II acceptance criterion by demonstrating that there is adequate RCS subcooling margin to ensure the fuel cladding never approaches a departure from nucleate boiling (DNB) condition. This more restrictive Condition II fuel acceptance criterion reduces the complexity of the evaluation required and maintains significant conservative margin to the GDC 17 criteria.

Calculation STA-274 evaluates DCPP response to a loss of all ac power assuming a maximum RCP seal leakage flow of 21 gpm per pump as directed by Westinghouse Technical Bulletin TB-04-22, Revision 1. Since calculation STA-274 evaluates a total assumed RCP seal leakage for impacts on the RCS inventory and pressure, whether the leakage is from the No. 1 or No. 2 seal is not considered a factor and does not impact the evaluation results.

Description of the Analysis Performed

Design calculation STA-274 uses the RETRAN computer code to perform a thermal hydraulic evaluation of the DCPP plant response during a loss of all onsite and offsite ac power to provide a technical basis for conformance to the GDC 17 requirements. Since the event involves a loss of all seal cooling, the evaluation includes a maximum RCP seal leakage flow of 21 gpm per pump as directed by Westinghouse Technical Bulletin TB-04-22, Revision 1.

Analysis Assumptions

The following assumptions are made to conservatively evaluate the RCS pressure and fuel heatup (DNB) response to a GDC 17 loss of all ac event using the RETRAN plant model (Figure 2).

1. The plant is assumed to be operating at nominal hot full power initial conditions. Since the GDC 17 loss of all ac event results in an immediate reactor trip, the long term plant response is most affected by the initial core power and resultant decay heat. Therefore, the assumption of 100 percent steady state core power and the conservative 1973 ANS decay heat standard with actinides provides appropriate conservatism for this evaluation.
2. The turbine-driven auxiliary feedwater pump flow is assumed to be the minimum performance requirement of 205 gpm per steam generator. There is no credit for the motor-driven auxiliary feedwater pumps since this loss of all ac event assumes a loss of all offsite and emergency onsite ac power sources.
3. There is no credit for any emergency core cooling system (ECCS) injection flow even if a safety injection signal is generated, since no ac power is available.
4. The minimum time that the RCS hot fluid would reach the RCP seal surfaces following a loss of all seal cooling is 8.33 minutes. This value is based on the expected RCP seal leakage of 3 gpm and a conservative minimum seal purge volume per the evaluation guidelines in Westinghouse Technical Bulletin TB-04-22, Revision 1.

5. At 8.33 minutes when the RCP seal surfaces are exposed to hot RCS fluid, the RCP seal leakage is expected to increase and stabilize in the range of 21 gpm per seal as established in Westinghouse Technical Bulletin TB-04-22, Revision 1.
6. DCPP Unit 1 and Unit 2 are considered to have insignificant differences for the purposes of evaluating this GDC 17 loss of all ac event. The model uses the most limiting conditions of either unit.
7. The maximum design steam generator tube plugging of ten percent was used to conservatively minimize the steam generator heat transfer area for this evaluation.
8. The evaluation assumes that the turbine-driven auxiliary feedwater pump flow is throttled to maintain steam generator (SG) level in an acceptable range since reduced auxiliary feedwater (AFW) flow results in higher RCS temperatures.

Inputs

The RETRAN input file for this calculation is a consolidation of the input from the model used in the replacement steam generator FSAR 15.2.7 loss of load analysis and the model enhancements developed to perform a RETRAN validation of the plant simulator. The key initial plant operating parameters assumed in the evaluation are listed in Table 1.

Methodology

This calculation uses the DCPP RETRAN computer model to perform the thermal hydraulic evaluations of the GDC 17 loss of all ac event with increased RCP seal leakage. The RETRAN code is a versatile thermal hydraulic computer code developed by Electric Power Research Institute for the purpose of analyzing various Pressurized Water Reactor and Boiling Water Reactor transients. The NRC has reviewed the RETRAN code and issued a Safety Evaluation Report approving it for analyzing certain transients as delineated in NRC regulations. The DCPP RETRAN model was developed and used for the safety analysis of the loss of load event as documented in license amendment request LAR 95-06 and License Amendments 108 and 107, issued October 1, 1995. This calculation implements additional changes to the RETRAN input file to model the specific plant conditions and functions which would occur for the GDC 17 loss of all ac event.

Results

The sequence of events for the GDC 17 loss of all ac event is listed in Table 2. Figures 3 and 4 plot the key plant parameters of RCS T_{hot}, pressurizer pressure, RCS saturation temperature, and pressurizer level. Figure 5 plots the RCS flow and shows that after the RCPs trip, the RCS flow rapidly coasts down to natural circulation conditions. The SG pressure rapidly increases to the SG ten percent atmospheric dump valve (ADV) lift setpoint and the secondary steam release begins to remove the RCS residual and core decay heat. Since the turbine driven auxiliary feedwater (TDAFW) pump begins supplying cool AFW at 60 seconds, there is adequate secondary inventory to maintain natural circulation heat removal conditions for the duration of the event. Figure 6 plots the SG level and shows that as discussed in assumption 8, the AFW flow is throttled starting at about 3300 seconds which results in reduced secondary heat transfer and slightly higher RCS temperatures at this time.

The RETRAN results show that the pressurizer narrow range level remains on scale for almost 54 minutes and the RCS subcooling (T_{hot} minus T_{sat}) remains more than adequate such that the 500-kV backfeed and associated actions to stabilize the plant could be delayed up to one hour without any adverse consequences.

Summary

PG&E Calculation STA-274 demonstrates that the GDC 17 design requirements for a delayed offsite ac power source are met for up to a one hour time period for the operators to complete the necessary actions associated with establishing the 500-kV back feed, implementing the RCP seal coping strategy, and restoring RCS makeup flow. Therefore, these results demonstrate PG&E maintains margin to the GDC 17 acceptance criteria for the 500-kV backfeed as a delayed offsite ac power source.

Once the 500-kV backfeed is completed and RCS makeup is restored, the RCP seal leakage remains well within the makeup capability of a single charging pump such that no ECCS flow is required. There is adequate RCS inventory control, RCS natural recirculation flow, and secondary heat removal capability such that the operators can perform a controlled RCS cooldown to cold shutdown conditions as required by the applicable emergency operating procedures. This ensures the RCP seal integrity is maintained and the RCP seal leakage can be restored to normal operational limits per Westinghouse Technical Bulletin TB-04-22, Revision 1.

Table 1 – Initial Operating Parameters

Input Parameter /Assumption	Value
Power Level (%)	100
Core Power (MWt)	3411
Decay Heat	1973 ANS
RCS Pressure (psia)	2250
Pressurizer Level (%)	56.0
RCS Flow (gpm)	366,000
Core Bypass Flow Fraction (%)	4.5
RCS Tavg (F)	573.6
SG Pressure (psia)	835
SG Level (%)	65
Steam Flow (Mlb/hr)	14.86
Steam Generator Tube Plugging (%)	10

Table 2 – Sequence of Events

Event	Time (sec)
LOAC occurs	0.0
Turbine trip and reactor trip	0.0
RCP seal leakage = 3 gpm/RCP	0.0
RCPs trip	1.0
MSIVs close	10.0
RCS charging and letdown flow isolated	10.0
Pressurizer sprays and heaters unavailable	10.0
TDAFW pump starts on trip of both MFW pumps	60.0
RCP seal leakage = 21 gpm/RCP	500.0
Low Pressure SI (No ECCS Flow credited)	1865
Pressurizer Level = 0%	3235
TDAFW pump flow fully throttled	3555
Pressurizer empties	4465

Figure 1 – Offsite Power System

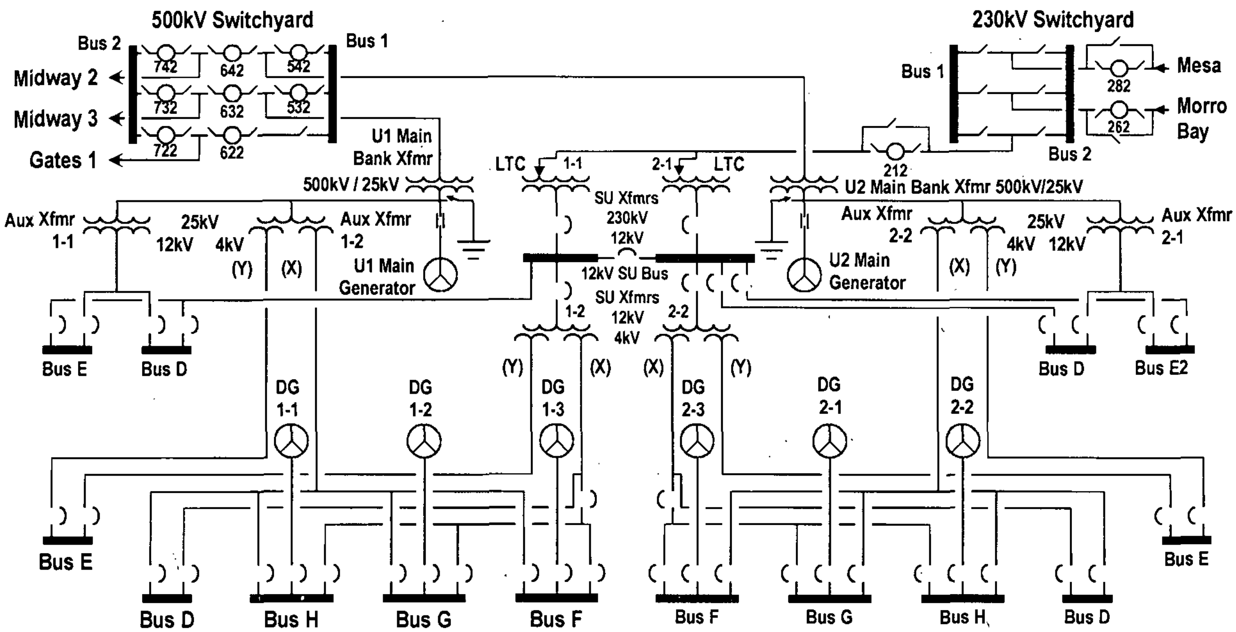


Figure 2 – RETRAN Model

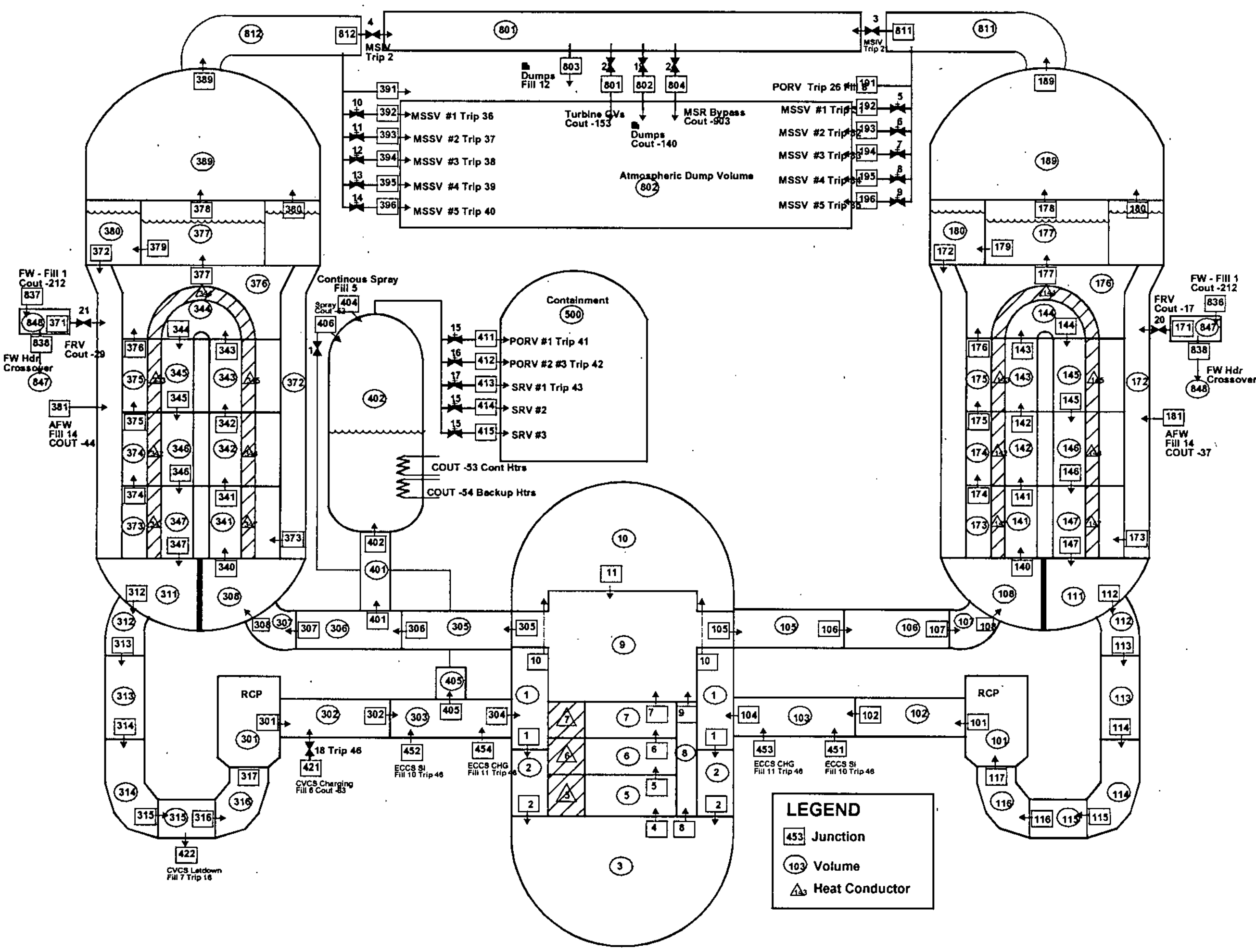


Figure 3

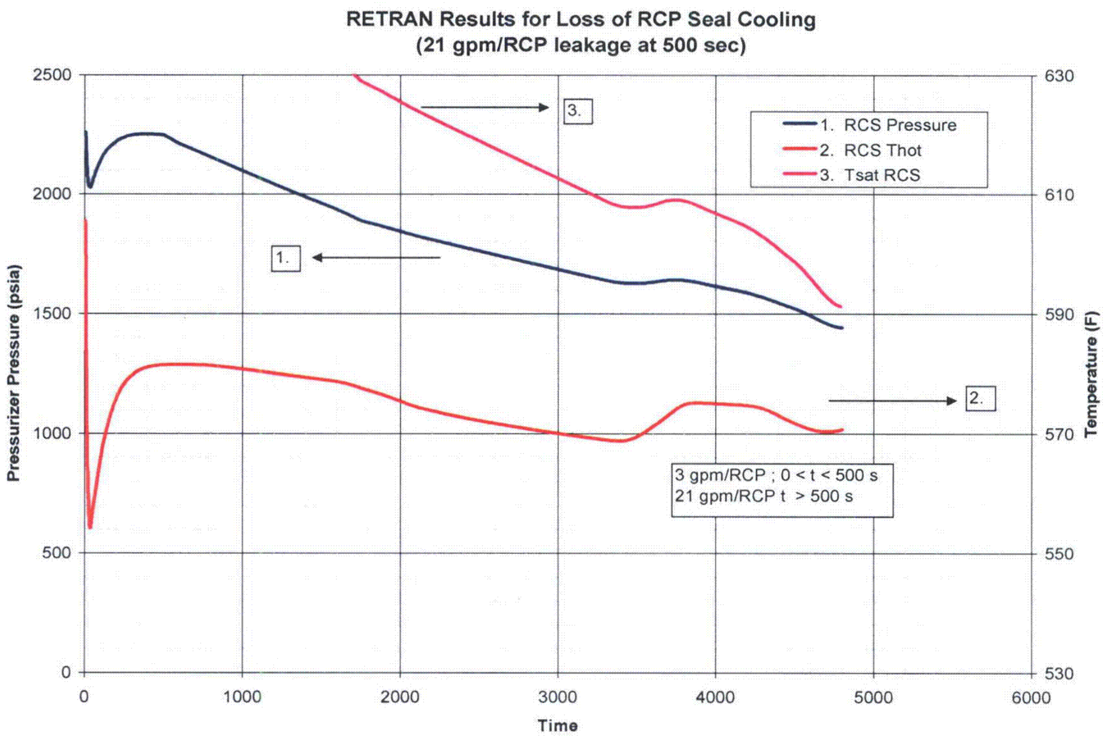

RETRAN Results for Loss of RCP Seal Cooling
(21 gpm/RCP leakage at 500 sec)
Pressurizer Pressure (psia)
Temperature (F)
Time
0
500
1000
1500
2000
2500
530
550
570
590
610
630
0
1000
2000
3000
4000
5000
6000
1. RCS Pressure
2. RCS Thot
3. Tsat RCS
1.
2.
3.
3 gpm/RCP ; 0 < t < 500 s
21 gpm/RCP t > 500 s

Figure 4

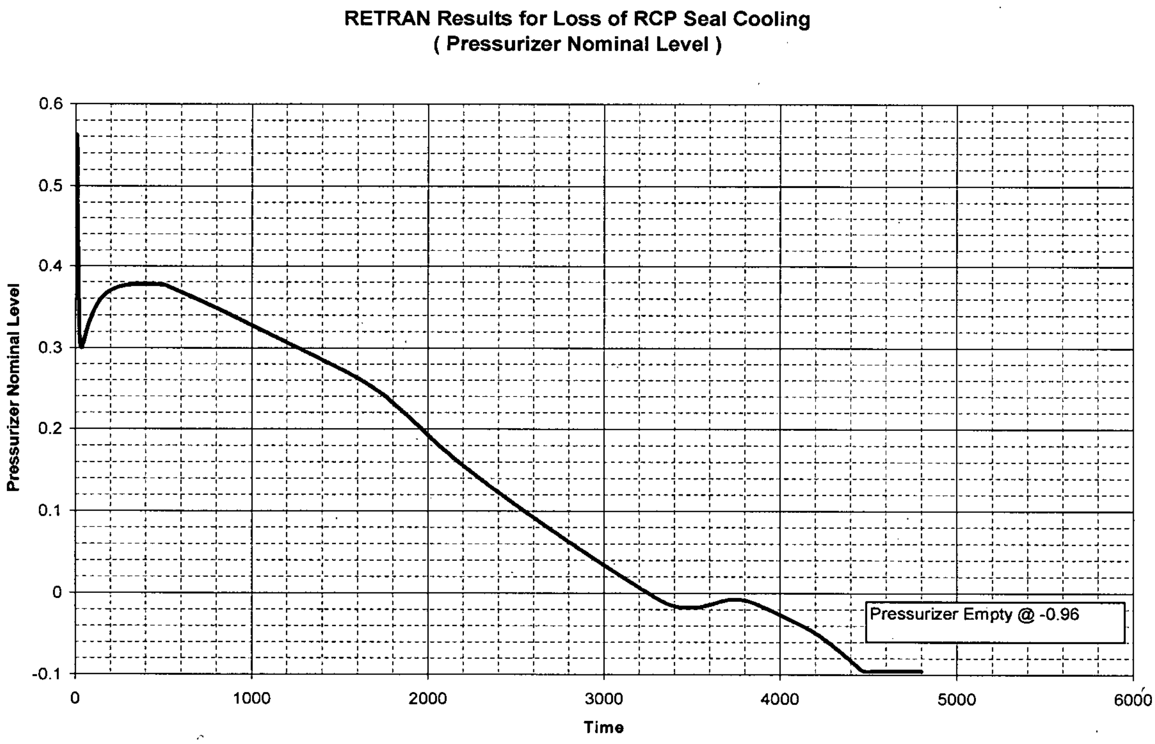

RETRAN Results for Loss of RCP Seal Cooling
(Pressurizer Nominal Level)
Pressurizer Nominal Level
0.6
0.5
0.4
0.3
0.2
0.1
0
-0.1
0
1000
2000
3000
4000
5000
6000
Time
Pressurizer Empty @ -0.96

Figure 5

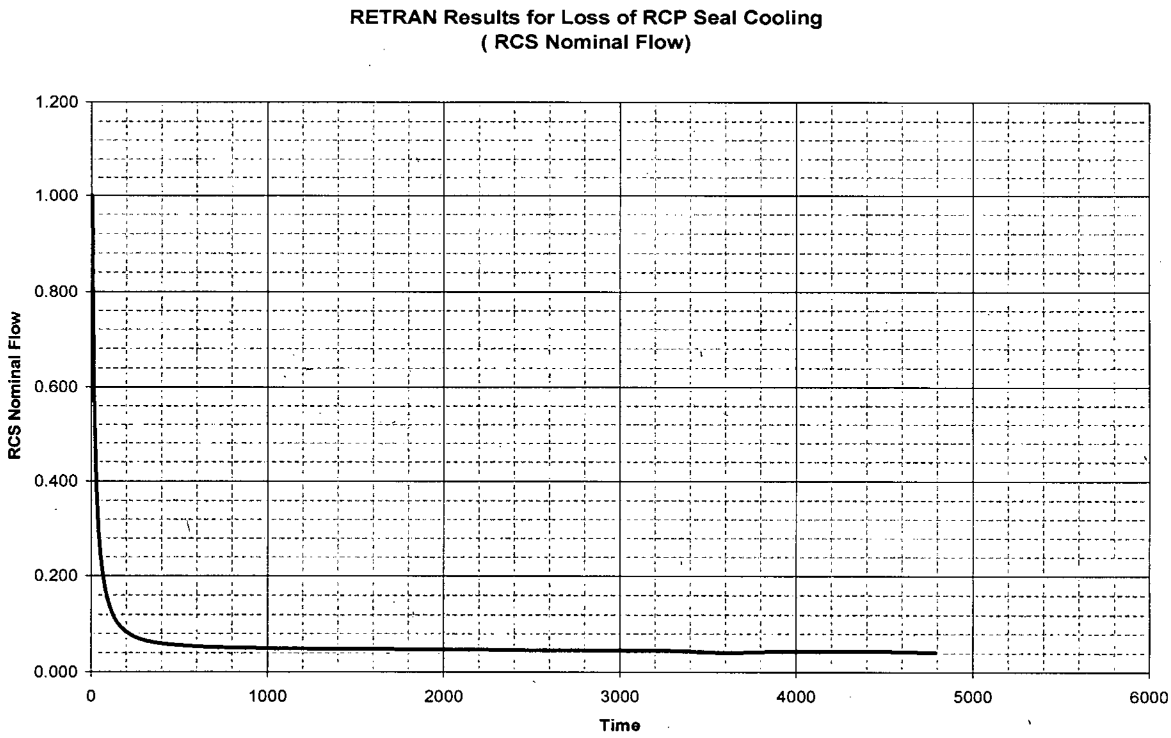
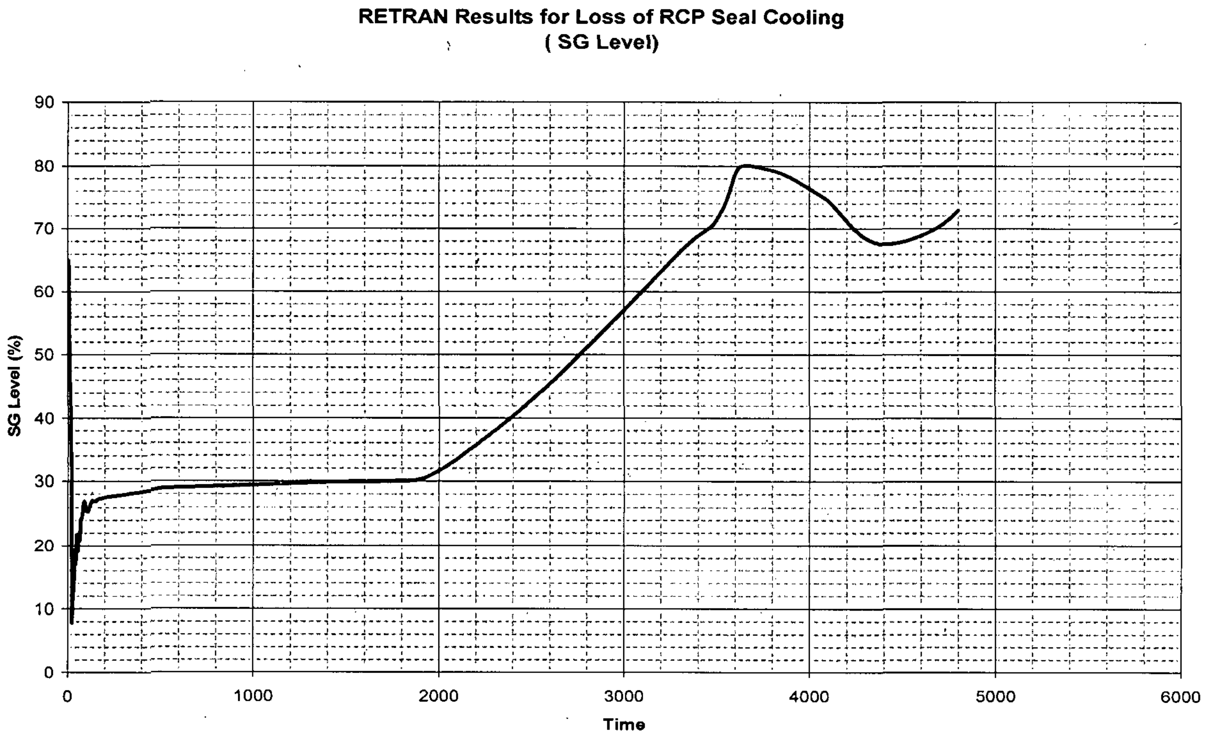

RETRAN Results for Loss of RCP Seal Cooling
(RCS Nominal Flow)
RCS Nominal Flow
1.200
1.000
0.800
0.600
0.400
0.200
0.000
0
1000
2000
3000
4000
5000
6000
Time

Figure 6


RETRAN Results for Loss of RCP Seal Cooling
(SG Level)
SG Level (%)
Time
0
10
20
30
40
50
60
70
80
90
0
1000
2000
3000
4000
5000
6000

4.0 REGULATORY EVALUATION

4.1 No Significant Hazards Consideration

Pacific Gas and Electric (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 proposed amendments would revise the licensing basis as described in the Final Safety Analysis Report Update (FSARU) to discuss the conformance of the delayed access offsite alternating current (ac) power circuit (the 500-kV delayed access circuit) to the General Design Criterion (GDC) 17 requirement that "each of the offsite power circuits be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded." It would also add information related to reactor coolant pump (RCP) seal performance during and after (1) a loss of seal injection (with continued thermal barrier cooling); (2) a loss of thermal barrier cooling (with continued seal injection); and (3) a loss of all seal cooling (both thermal barrier cooling and seal injection).

PG&E Calculation STA-274 demonstrates that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded following a loss of the 230-kV immediate access offsite power circuit and all onsite emergency ac power supplies until the 500-kV delayed access circuit can be aligned for backfeed. Alignment of the 500kV delayed offsite circuit to backfeed, implementing RCP seal coping strategy actions to limit maximum RCP seal leakage to 21 gpm per pump, and restoring reactor coolant system (RCS) makeup flow to stabilize the plant can be completed within approximately 54 minutes to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded.

The proposed changes will not add any accident initiators, or adversely affect how the plant safety-related structures, systems, or components (SSCs) are operated, maintained, modified, tested, or inspected. There is no increase in the probability of a GDC 17 loss of all ac event occurring, and since the same applicable GDC 17 acceptance criteria continue to be met with the increased RCP seal leakage, there is no change in the consequences associated with this event.

Therefore, the proposed amendment 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 RCP Seal coping strategy implemented in response to Westinghouse Technical Bulletin TB-04-22, Revision 1, ensures that RCP seal integrity is maintained following a loss of all seal cooling associated with the GDC 17 loss of all ac event. PG&E Calculation STA-274 demonstrates that the GDC 17 requirements for a delayed offsite ac power source are met for up to a one hour time period for the operators to complete the necessary actions associated with establishing the 500-kV backfeed, implementing the RCP seal coping strategy to limit maximum RCS seal leakage to 21 gpm per pump, and restoring RCS makeup flow. This proposed change provides assurance that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. The proposed change does not introduce new equipment that could create a new or different kind of accident, and no new equipment failure modes are created. As a result, no new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of this proposed amendment.

Therefore, the proposed changes do 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 implementation of the RCP seal coping strategy ensures that RCP seal leakage is limited to 21 gpm per pump following a loss of all seal cooling such that there is no impact or reduction in the margin of safety associated with the GDC 17 loss of all ac event. The analysis associated with the change supports the ability to align the 500-kV delayed access circuit, implement the RCP seal coping strategy actions, and restore RCS makeup flow in sufficient time following a loss of all onsite ac power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. The proposed amendment would not alter the way any safety-related SSC functions and would not alter the way the plant is operated. The amendment demonstrates that the 500-kV backfeed, isolation of RCP seal cooling, and restoration of RCS makeup flow can be reliably completed within 54 minutes, and that there is considerable margin to the GDC 17 acceptance criteria for the 500-kV backfeed as a delayed offsite ac power source. The proposed amendment would not introduce any new uncertainties or change any existing uncertainties associated with any safety limit. Since the proposed amendment would have no impact on the structural integrity of the fuel cladding or reactor coolant pressure boundary, and maintains the RCP seal leakage within controllable limits, there is no impact on the containment structure. Based on the above considerations, the proposed amendment would not degrade the ability to safely shutdown the plant in the event of a loss of all ac power.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above evaluation, PG&E concludes that the proposed changes present 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.

4.2 Applicable Regulatory Requirements/Criteria

General Design Criterion (GDC) 17, "Electric Power Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, requires, in part, that each of the offsite power circuits be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded.

PG&E Calculation STA-274 demonstrates that the specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded following a loss of the 230-kV immediate access offsite power circuit and all onsite alternating current power supplies provided the 500-kV delayed access circuit is aligned for backfeed, RCP seal cooling is isolated, and RCS makeup flow is restored within approximately 54 minutes.

4.3 Precedent

None

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 (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) 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

1. NRC Integrated Inspection Report 05000275/2009003 and 05000323/2009003, dated August 5, 2009. (ADAMS Accession No. ML092170781)
2. Westinghouse Topical Report WCAP-16396-NP, "Westinghouse Owners Group Reactor Coolant Pump Seal Performance for Appendix R Assessments," Revision 0, dated January 2005 (ADAMS Accession No. ML050320187)
3. Westinghouse Technical Bulletin TB-04-22, "Reactor Coolant Pump Seal Performance – Appendix R Compliance and Loss of All Seal Cooling", Revision 1, dated August 9, 2005.

Proposed Final Safety Analysis Report Update Changes (Marked-up)

Section 5.5.1.3.1, (Reactor Coolant) Pump Performance
Section 8.2.1.2, 500-kV System

The instrumentation monitors are mounted in a common rack located on the operating deck in containment. Alarms in the control room are provided by the rack in containment. Vibration data from the instrument rack is collected and stored on a server in the TSC, and analyzed at a personal computer in the administration building. The computer is shared by both units. The computer may be turned off to support maintenance or power switching, as the vibration equipment will still provide alarms and indication. The RCP vibration monitoring system does not perform a safety function.

As shown in Table 5.2-13, all parts of the pump in contact with the reactor coolant are austenitic stainless steel except for seals, bearings, and special parts. Component cooling water is supplied to the two oil coolers on the pump motor and to the pump thermal barrier heat exchanger.

The pump shaft, seal housing, thermal barrier, bolting ring, and motor stand can be removed from the casing as a unit without disturbing the reactor coolant piping. The flywheel is available for inspection by removing the cover.

The performance characteristic, shown in Figure 5.5-2, is common to all of the fixed-speed mixed-flow pumps, and the "knee" at about 45 percent design flow introduces no operational restrictions since the pumps operate at full speed.

5.5.1.3 Design Evaluation

This section discusses RCP design features incorporated to ensure safe and reliable operation while maintaining RCS integrity.

5.5.1.3.1 Pump Performance

The RCPs are sized to equal or exceed the required flowrates. Initial RCS tests confirm the total delivery capability. Thus, assurance of adequate forced circulation coolant flow is provided prior to initial plant operation.

The reactor trip system (RTS) ensures that pump operation is within the assumptions used for loss-of-coolant flow analyses, which also ensures that adequate core cooling is provided to permit an orderly reduction in power if flow from an RCP is lost during operation.

An extensive test program was conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long-term tests were conducted on less than full-scale prototype seals as well as on full-size seals. Operating plants continue to demonstrate the satisfactory performance of the controlled leakage shaft seal pump design.

The support of the stationary member of the No. 1 seal (seal ring) is such as to allow large deflections, both axial and tilting, while still maintaining its controlled gap relative to the seal runner. Even if all the graphite were removed from the pump bearing, the shaft could not deflect far enough to cause opening of the controlled leakage gap. The "spring-rate" of the hydraulic forces associated with the maintenance of the gap is high enough to ensure that the ring follows the runner under very rapid shaft deflections.

Testing of pumps with the No. 1 seal entirely bypassed (full reactor pressure on the No. 2 seal) shows that relatively small leakage rates would be maintained for long periods of time. The plant operator is warned of this condition by the increase in No. 1 seal leakoff, and has time to close this line and to conduct a safe plant shutdown without significant leakage of reactor coolant to the containment. Thus, it may be concluded that gross leakage from the pump does not occur, even if seals were to suffer physical damage.

The effect of loss of offsite power on the pump itself is to cause an RCS pump trip, and temporary stoppage in the supply of injection water to the pump seals and component cooling water to the thermal barrier for seal and bearing cooling if a generator trip results. The emergency diesel generators are started automatically due to loss of offsite power, so that component cooling water flow is automatically restored to ensure cooling of the pump seals and bearings when the reactor coolant temperature is above 150°F. Seal water injection flow is subsequently restored by automatically restarting a charging pump on diesel generator electrical power.

The effect of loss of thermal barrier cooling water through a malfunction in the component cooling water system (without the loss of seal injection) would subject some of the pump components nearest to the thermal barrier heat exchanger to higher temperatures. However, the pump bearings and seal would still be cooled by the seal injection flow and would stabilize at a slightly higher temperature compared to that with the thermal barrier heat exchanger operable. In the event that the thermal barrier cooling is temporarily lost, the vendor guidelines should be followed for restoration of component cooling water flow.

The effect of loss of seal injection (with continued thermal barrier heat exchanger cooling) subjects the pump components to higher temperatures as RCS fluid, after being cooled at the thermal barrier heat exchanger, enters the pump annular cavity and flows to the seal and bearing region. However, the pump bearings and seals should still be sufficiently cooled to prevent any significant increase in the flow through the No. 1 seal. In the event that seal injection is temporarily lost, the vendor Instruction Book should be followed for restoration of seal injection.

The effect of a loss of all seal cooling (the loss of thermal barrier cooling and seal injection) subjects the pump components to higher temperatures as the hot RCS fluid enters the pump annular cavity and flows to the seal and bearing region. Initially, the pump bearings and seals will be sufficiently cooled by the cool water in the pump annular cavity. As this cooler water flows through the No. 1 seal, it is replaced in the pump annular volume by the hot RCS fluid. At this time, the seal leak rate undergoes a transient that settles at a long term leakage value of 21 gpm.

For DCPP RCPs which use Westinghouse shaft sealing systems with Westinghouse high temperature O-rings, field experience and testing has demonstrated that the O-rings will withstand the expected conditions for loss of all seal cooling. With the RCS at 550 F and 2250 psi, analyses documented in WCAP-10541, Rev. 2, "Reactor Coolant Pump Seal Performance Following a Loss of All AC Power" predict that the pressure drops across the No. 1 and No. 2 seals will be about 1400 and 800 psi respectively, which are 400 psi lower than the pressures used for testing. The tests, as reported in Supplement 1 to the WCAP, were performed at 550 F with eccentric extrusion gaps to maximize the potential for O-ring failure. The extrusion testing maintained the conditions for 18 hours, with some tests extended to 168 hours. The tests showed no O-ring failures when exposed at 550 F and 400 psi above the seal pressure drops expected for a loss of all seal cooling event.

Seal cooling (either by thermal barrier cooling or seal injection) should not be re-established to a hot seal. Westinghouse recommends that seal temperature be the parameter on which operator decisions for restoration of seal cooling is based. In the event that seal temperature is not available, the time required for hot RCS fluid to reach the seal may be used. If seal temperature or time requirements cannot be met, operator actions are required to isolate seal cooling prior to the restoration of RCS makeup or component cooling water flow.

5.5.1.3.2 Coastdown Capability

It is important to reactor operation that the reactor coolant continues to flow for a short time after reactor trip. To provide this flow after a reactor trip, each reactor coolant pump is provided with a flywheel. Thus, the rotating inertia of the pump, motor, and flywheel is employed during the coastdown period to continue the reactor coolant flow.

The pump is designed for the design earthquake (DE) at the site. Bearing integrity is maintained as discussed below. It is, therefore, concluded that the coastdown capability of the pumps is maintained even under the most adverse case of a pump trip coincident with the DE.

5.5.1.3.3 Flywheel Integrity

Integrity of the RCP flywheel is discussed in Section 5.2.6.

5.5.1.3.4 Bearing Integrity

The design requirements for the RCP bearings are primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over

(1) 230-kV oil circuit breaker. If the power loss is due to mechanical or electrical failure of the oil circuit breaker, the circuit breaker can be isolated and bypassed by means of manual switching operations. A physical disruption of the short section of 230-kV line from the switchyard to the plant is considered highly unlikely.

(2) Loss of either 230-kV/12-kV standby startup transformer 11 or 21 or the associated 12-kV breakers or buses. Standby startup transformers 11 and 21 are normally separated on the 12-kV side, with transformer 11 feeding Unit 1 and transformer 21 feeding Unit 2. In case of a failure of either transformer, the faulted transformer can be manually switched out of service, its bus can then be transferred to the other transformer by closing the 12-kV bus tie vacuum circuit breaker. This circuit breaker is common to the 12-kV standby startup buses of Units 1 and 2, and is normally kept open (i.e., procedurally controlled).

(3) Failure of 4.16-kV standby startup transformer 12 (22). By means of manual switching after a failure, the buses served from this transformer can be supplied from the 230-kV system by unit auxiliary transformer 12 (22) through unit auxiliary transformer 11 (21), fed from the 12-kV standby startup bus. This requires removal of links in the generator bus at the main transformer as well as opening of the disconnecting switch to the generator. This is an unusual configuration and is used only when better methods are not available.

While the above failure mechanisms are possibilities, the 230-kV transmission system and the 230-kV/12-kV standby startup power system are designed in a manner intended to obtain a high degree of service reliability and to minimize the time and extent of outage if failures do occur.

8.2.1.2 500-kV System

The 500-kV system provides for transmission of the plant's power output, and provides a delayed access source of offsite power to the plant auxiliary systems and ESF buses when the main generator is not in operation. The 500-kV system is available in sufficient time to safely shutdown the plant during non-accident conditions. Power is backfed via the main transformer and the unit auxiliary transformers. A dc motor-operated disconnecting switch in the generator's main leads is opened to use this source. This switch is a telescoping type that is an integral part of the generator isolated phase bus. This switch is operated under manual control from the control room and is interlocked to prevent opening under load. Upon actuation, the motor-operated disconnect switch takes approximately 30 seconds to isolate the main generator from the main and the unit auxiliary transformers. In the event of a loss of main generator output, the 500-kV backup source of auxiliary power could be placed in service. Plant procedures contain actions for operators to complete the 500-kV backfeed, isolation of RCP seal cooling, and restoration of RCS makeup flow within approximately 54 minutes upon loss of 230-kV and all onsite ac power. Completion of these actions within this time period assures that specified acceptable fuel design limits and design considerations of the reactor coolant pressure boundary are not exceeded. (Also see Sections 5.5.1.3.1 and 8.3.1.6). After the two 500-kV breakers are opened, operations personnel coordinate with PG&E's Transmission Operations Center to realign plant protective relaying; open the generator disconnect; and re-close the generator output breakers. The position of the motor-operated disconnect switch is verified prior to backfeeding from the 500-kV switchyard. Figure 8.1-1 (Plant Single Line Diagram) shows the three 500-kV outlet lines and the interconnections to the plant auxiliaries. Reference 5 shows the arrangement of the 500-kV switch, bus, and circuit breaker structures.

Each 500-kV transmission line to the 500-kV transmission system is provided with relay protection terminal equipment consisting of two line relay sets (directional comparison), each operating over physically separate channels, microwave and power line carrier, and each provided with a separate dc power circuit. Single-pole tripping is not enabled for any of the lines. High-speed automatic reclosing is not enabled for the circuit breakers at the DCPP end of the lines. Backup protection (provided by a distance relaying terminal, including distance and directional ground relays) is normally cut-out, and cut-in when either primary relay set is not operable. Each 500-kV line between the 500-kV switchyard and a generator step-up transformer bank is provided with redundant current differential protection channels. Directional over-current relays are available as backup.

The 500-kV switchyard dc control power is provided by a lead-acid battery and two battery chargers. Each charger is capable of supplying the normal dc load of the 500-kV switchyard and maintaining the battery in a fully charged condition. Normally, one charger is operating with the second charger available on standby. Both chargers may be operated in parallel, if desired. Each charger is equipped with an ac failure alarm that operates on loss of ac to the charger. The battery and chargers feed two 125-Vdc distribution panels, one of which is equipped with a dc undervoltage relay that initiates an alarm if the dc voltage should drop below a preset value. Separate dc control circuits are provided for each 500-kV power circuit breaker.

8.2.2 ANALYSIS

8.2.2.1 Load Flow and Dynamic Loading Analyses

The 230-kV system is the immediate source of offsite power following a design basis accident or unit trip. Operability is based on the ability to transfer to the 230-kV system following a design basis accident or unit trip without loading the emergency diesel generators, and provide adequate voltages to the safety related loads. Load flow and dynamic loading analyses are performed for anticipated configurations of the transmission network (e.g., generating units out of service, transmission line(s) out of service, or voltage control devices out of service) to ensure that the 230-kV system has sufficient capacity and capability to operate the engineered safety features for a design