ENCLOSURE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Diablo Canyon Power Plant Units 1 and 2	Docket No. 50-275 Facility Operating License No. DPR-80 Docket No. 50-323 Facility Operating License No. DPR-82 License Amendment Request No. 89-06

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) hereby applies to amend its Diablo Canyon Power Plant (DCPP) Facility Operating License No. DPR-80 (License).

The proposed changes amend the Technical Specifications (Appendix A of the Licenses) as regards to the deletion of Technical Specifications 3/4.5.4 and the associated Bases and the revision of Technical Specification Tables 3.3-5 and 3.8-1. Information on the proposed changes is provided in Attachments A and B.

These changes have been reviewed and are considered not to involve a significant hazards consideration as defined in 10 CFR 50.92 or require an environmental assessment in accordance with 10 CFR 51.22(b). Further, there is reasonable assurance that the health and safety of the public will not be endangered by the proposed changes.

Subscribed to in San Francisco, California this 15th day of May 1989.

Respectfully submitted,

Pacific Gas and Electric Company

By _____
J. D. Shiffer
Vice President
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MILDRED J. WILLIAMS
NOTARY PUBLIC - CALIFORNIA
CITY & COUNTY OF SAN FRANCISCO
My Commission Expires Aug. 7, 1989

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By _____
Richard F. Locke

Subscribed and sworn to before me
this 15th day of May 1989

_____
Mildred J. Williams, Notary Public in
and for the City and County of
San Francisco, State of California

My commission expires August 7, 1989.

2702S/0069K

Attachment A

DELETION OF TECHNICAL SPECIFICATIONS 3/4.5.4 "BORON INJECTION SYSTEM" AND ASSOCIATED BASES, AND REVISION OF TABLE 3.3-5, "ENGINEERED SAFETY FEATURES RESPONSE TIMES"

A. DESCRIPTION OF AMENDMENT REQUEST

This license amendment request (LAR) proposes the following Technical Specification (TS) changes:

1. Revision of the TS index to reflect proposed TS changes.

2. Deletion of TS 3.5.4.1, "Boron Injection Tank", TS 3.5.4.2, "Heat Tracing", and the associated Bases. The proposed changes will allow for bypassing or removing the Boron Injection Tank (BIT) and associated piping and components.

3. Revision of TS Table 3.3-5, "Engineered Safety Features Response Times", to make the safety injection response times consistent with BIT removal. The following is a summary of the proposed changes to Table 3.3-5:

   a. The original Table Notation 1 no longer applies to any components in the SI flow path and it has therefore been deleted. All references to the original Notation 1 have been changed to Notation 3 or Notation 7.

   b. The original Notation 5 is applied to Phase A Isolation, Component Cooling Water, and Auxiliary Saltwater Pumps in Items 2, 4, 5, and 6. It is more appropriate to apply Notation 3 to these components. As these were the only instances where Notation 5 was used, the references are changed to Notation 3 and the original Notation 5 is deleted.

   c. The original Notation 4 does not apply to any components in the SI flow path and so it has been modified by removing the last sentence. This information has then been moved to Notation 1 and all references in the table to the original Notation 4 have been changed to Notation 1.

   d. Three new notations, Notation 4, Notation 5, and Notation 7 are added to clarify the SI response times and to explain the VCT/RWST switchover sequence.

   e. Items 2a and 3a have been changed to reflect both the LOCA and Non-LOCA SI time response delay requirements.

   f. Items 4a, 5a, and 6a SI time response values only reflect the more limiting Non-LOCA requirements.

4. Revision of TS Table 3.8-1 to change the function of the BIT inlet and outlet valves to charging injection valves.

Changes to the Technical Specifications are noted in the marked-up copy of the applicable Specifications (Attachment B).

B. BACKGROUND

1. Current System Design Basis

The BIT holds 900 gallons of water with a boron concentration of between 20,000 and 22,500 ppm. Tank heaters and pipe heat tracing are provided to maintain a minimum solution temperature of greater than or equal to 145°F. Recirculation from the boric acid tanks (BAT) to the BIT is maintained continuously via boric acid transfer pumps to ensure that the BIT is full of concentrated boric acid at all times and to prevent boric acid stratification. The BIT is isolated from the reactor coolant system (RCS) during normal plant operation. During a safety injection, the suction of the charging pumps is diverted from the normal suction at the volume control tank (VCT) to the refueling water storage tank (RWST) and discharged through the BIT to the RCS. Concurrently, isolation valves in the recirculation line to the BAT close.

The operability of the boron injection system currently ensures that sufficient negative reactivity is injected into the core to offset the increase in positive reactivity caused by RCS cooldown due to a steam line break.

2. Design Changes to the Boron Injection System

The following system design changes will be made to the boron injection system:

a. The BIT will be bypassed by a piping modification. This will eliminate the need for tank heaters and temperature indicators. The associated temperature indicators and annunciator will be removed from the control room.

b. The heat tracing equipment associated with the BIT piping may be abandoned in place or may be removed.

c. The BIT recirculation isolation valves and their associated air supply will be locked closed. The corresponding hand switches in the control room will be removed.

d. The BIT recirculation path flow instrument may be abandoned in place or may be removed.

C. JUSTIFICATION

As discussed in NRC Generic Letter 85-16, there have been incidents at operating plants in which boric acid has crystallized in the internals of vital safety related pumps and piping, thereby rendering those systems inoperable. Heat tracing is presently necessary to maintain the BIT and associated piping at a sufficiently high temperature to prevent precipitation of the boric acid solution. Maintenance of the heat tracing equipment is common and can lead to plant unavailability due to technical specification shutdown requirements if heat tracing becomes inoperable.

Improvements to the analysis methods for calculating the consequences of a steam line break have provided results that demonstrate a reduced need for the highly concentrated boron (20,000 ppm) injection. Based on the inherent safety risks in the present system and the results from the improved calculation methods, PG&E is submitting this LAR for removal of the BIT.

Similar license amendments have been approved by the NRC for the North Anna Power Station, H. B. Robinson Steam Electric Plant, Indian Point Unit No. 2, Surry Power Station, and the V.C. Summer Nuclear Station.

D. SAFETY EVALUATION

Removal of the BIT required reevaluation of the steamline break accidents. The other FSAR Update Chapter 15 accident analyses are not adversely affected by the elimination of the BIT. The evaluation, performed by Westinghouse, is documented in WCAP-11938, "BIT Elimination Study for Diablo Canyon Units 1 and 2", Volumes 1 and 2. In addition, the proposed elimination of the BIT boron concentration required a change in the safety injection response times. The evaluation for the change in the safety injection response times included consideration of both LOCA and non-LOCA accident analyses. The following safety evaluation addresses accident analyses, containment analysis, environmental qualification of equipment for steamline break accidents, and the effect of BIT removal on the safety injection response times.

1. Accident Analyses

An analysis was performed for the "Accidental Depressurization of the Main Steam System (Depressurization of MSS)" (FSAR Update Section 15.2.13) and "Major Secondary Steam System Pipe Rupture (Pipe Rupture)" (FSAR Update Section 15.4.2) accidents for a system with the BIT removed. As in the present Diablo Canyon FSAR Update steamline break analysis, the system transient parameters for the new analyses (i.e. RCS pressure, temperatures, steam flow, core boron concentration, and core power) were

calculated using the LOFTRAN computer code. This computer code includes models of the reactor core, steam generators, pressurizer, primary piping, protection systems, and engineered safeguards systems. The changes in system volumes, initial conditions, and other design information corresponding to DCPP Units 1 and 2 were conservatively accounted for in the analysis. The analysis also conservatively accounted for a low steamline pressure setpoint of no less than 15 psia. The current low steamline pressure TS allowable setpoint is greater than or equal to 580 psig (595 psia).

VANTAGE 5 fuel and the associated transition cores were accounted for in the analyses by using consistent and conservative parameters between standard fuel and VANTAGE 5 fuel. The 1979 ANS decay heat curve was used in this analysis. Limiting coefficients for the Overpower Delta-T trip setpoint were used. The analysis uncertainties for temperature and pressure were also limiting values. The BIT and associated piping were conservatively modeled as being in place and filled with pure water. The modeling of delay times and volumes allows for the BIT to be removed completely, bypassed, or remain in place.

a. Pipe Rupture

The major secondary steam pipe rupture analysis was reanalyzed assuming BIT removal. This analysis, with and without offsite power available, analyzed for the largest double ended rupture of a main steam pipe. For the pipe rupture analysis, the same criterion was applied to the BIT removal analysis as is applied in the FSAR Update, Section 15.5.18. That is, for the most severe Condition IV break, radiation releases are required to be within the guideline values of 10 CFR 100 by demonstration that the DNB design basis is met. The steamline break DBA dose calculations performed for the FSAR Update assume that the plant has been operating with a fuel failure level of 1 percent.

Westinghouse initially analyzed a double ended rupture upstream of the flow restrictor with offsite power available as the most limiting DNB transient of the FSAR Update cases analyzed. The plant was initially assumed to be at hot zero power at the minimum required shutdown margin. Following the break, the RCS temperatures and pressures decreased rapidly, and in the presence of a large end of life (EOL) moderator coefficient of reactivity, the reactor returns to criticality with the rods inserted, assuming the most reactive RCCA in the fully withdrawn position. The reactor power was shown to increase at a decreasing rate until boron from the safety injection system reached the core and began to offset the positive reactivity insertion caused by the cooldown. The

core was subsequently brought subcritical with boron injection, aided by the abatement and eventual termination of steam flow from the faulted steam generator.

Westinghouse also analyzed for similar transients assuming no offsite power available. The reactor coolant pump coastdown reduces the thermal coupling between the secondary side and the primary side, reducing the cooldown experienced by the core. The additional delay for starting the emergency A/C diesel generators also accounts for a slightly longer time delay to deliver the high concentration boron to the core.

Although prevention of cladding damage is not a requirement for Condition IV events, the analysis demonstrated that the DNB design basis is met, i.e., the DNB Ratio remained greater than the limiting value and therefore no cladding damage occurs. The dose evaluation, which was performed assuming 1 percent failed fuel, therefore continued to demonstrate that the Condition IV accident criteria were satisfied.

b. Depressurization of the MSS

A summary of the Westinghouse analysis for the limiting Condition II event, Depressurization of the MSS, assuming removal of the BIT follows. For the current FSAR Update case, the analysis criterion is that the reactor remains subcritical. This Westinghouse criterion assures that the DNB design basis is met in a very conservative manner. Westinghouse has performed the analysis using a new criterion, whereby the plant may return to criticality but no damage may occur to the fuel. Criticality is attained and sustained with the BIT eliminated. DNB analyses for this case shows that the DNB design basis is met and no fuel failures are predicted. This constitutes a relaxation of the conservative internal Westinghouse criterion for Condition II events used previously for Diablo Canyon, but is a Westinghouse criterion which has been used in more recently approved analyses. This new criterion is in compliance with the criteria used by the NRC and ANS, which require that releases during depressurization of the MSS remain within the guideline values set forth in 10 CFR 20. These values can be met with a return to criticality if it is assured that there is no consequential fuel damage. This conclusion is also consistent with the conclusion for the Condition IV breaks, since no violation of the DNB design basis was calculated for the more extreme Condition IV, double ended ruptures.

The analysis provided by Westinghouse shows for the three cases; Major Secondary System Pipe Rupture, with and without a loss of offsite power, and Accidental Depressurization of the Main Steam System, that the DNB design basis is met and no fuel failure will occur.

2. Containment Analysis

The impact of BIT elimination on the containment mass and energy release and pressure/temperature response was analyzed by Westinghouse to determine if the containment pressure remained below its 47 psig design limit. Based on the new temperature response provided by Westinghouse, PG&E evaluated equipment qualification on components inside the containment.

The LOFTRAN computer code was used to generate the mass and energy release to the containment for a full spectrum of postulated accidents, which include large double-ended guillotine breaks at several locations in the steamline, small double-ended breaks in the steamline, and split breaks in the steamline. Each break was analyzed for four plant power levels, 102, 70, 30, and 0 percent power. In addition, a spectrum of single failure criteria was imposed on each postulated break scenario. The single failures assumed are: (a) failure of the MSIV in the faulted loop, (b) failure of the feedwater regulating valve in the faulted loop, and (c) auxiliary feedwater pump runout to the faulted loop.

The mass and energy releases to containment were used as input to the COCO computer code, which determines the containment pressure and temperature response. Each LOFTRAN mass and energy release was analyzed to envelope two different plant configurations as initial conditions. The two configurations were determined by TS 3/4.7.12, "Ultimate Heat Sink". This TS requires two component cooling water (CCW) heat exchangers in-service whenever ocean water temperature exceeds 64°F, and only 1 CCW heat exchanger in-service when ocean temperature is less than or equal to 64°F. These configurations impact the cooling capabilities of the containment fan coolers which are cooled by CCW flow. For all LOFTRAN mass and energy releases, where one of the above three failures is assumed, no additional failure is assumed as input to COCO. For LOFTRAN mass and energy releases where one of the three failures are not assumed, the single failure which is input to COCO is the failure of one diesel generator. This impacts containment response by decreasing containment spray flow rate, decreasing the number of operating fan coolers, and decreasing the available ultimate heat sink heat transfer capability by decreasing the CCW and auxiliary saltwater (ASW) systems flow rate. Additionally, each event is analyzed for a coincident loss of offsite power.

The Westinghouse results reflect the more limiting CCW heat exchanger configuration. The containment pressure transient response for the limiting case is given in Figure 4-1 of WCAP-11938, Volume 1. There is no single limiting run for temperature responses. The peak containment pressure is 46.55 psig, which compares with the current Chapter 3 FSAR Update value of 44 psig. However, 46.55 psig is still less than the maximum pressure of 46.65 psig for a LOCA. Figure 4-6 of WCAP-11938 Volume 1 is the aggregate temperature response data that envelopes every break scenario. The temperature transient of a given break will be enveloped by the combined curve. The peak containment temperature using the low steamline pressure setpoint is 345°F. This value compares with the current peak containment temperature with the BIT in place of 339°F.

The analysis shows that the containment design pressure of 47 psig will not be reached if a main steam line break should occur without the BIT. The most limiting accident for a containment pressure transient continues to be a LOCA; removal of the BIT does not impact the LOCA pressure transient. Containment temperature does increase without the BIT boric acid concentration requirement. PG&E has evaluated the peak temperature data provided by Westinghouse. The impact of the increased containment temperature and the new mass and energy release data on environmental qualification for equipment inside and outside of containment is presented in the next section.

3. Environmental Qualification

a. Inside containment

Based on the evaluation of containment temperature and pressure data, PG&E has determined that equipment inside containment remains environmentally qualified to provide emergency system functions following either the Depressurization of the MSS or Pipe Rupture accidents.

b. Outside containment

Westinghouse performed analyses assuming removal of the BIT to determine the mass and energy release due to steamline breaks outside of containment. The analyses consider superheating of the steam being released.

A spectrum of steamline break accidents was analyzed by Westinghouse. Break sizes from 5.6 to 0.1 square feet were modeled. These breaks were assumed to be split breaks and covered the full range of possible break sizes that could occur at the plant. Break locations both upstream and downstream of the main steam check valves were analyzed to

cover all possible outside containment steamline break locations. Initial power levels of 100 and 70 percent were evaluated. Steamline breaks initiated from lower power levels result in lower levels of steam superheating than analyses initiated from full power.

The LOFTRAN computer code was used in the analyses to calculate the mass and energy release data. Conservative assumptions on initial conditions and reactivity coefficients were made to bound operation of both Units 1 and 2. The analyses assume minimum auxiliary feedwater flow with the failure of one train of auxiliary feedwater to maximize the superheat calculated.

Specific data provided by PG&E to Westinghouse on auxiliary feedwater flow performance as a function of steam generator pressure was used as input to LOFTRAN. For these analyses, the 1979 ANS decay heat curve was used. Offsite power was assumed to be available in all analyses since this results in continued operation of the reactor coolant pumps and helps to maximize the blowdown rate.

The selection of cases and methodology used was consistent with the methodology used in the past by Westinghouse to determine Outside Containment Steamline break mass and energy response. In all, 81 cases were analyzed by Westinghouse to represent the possible combinations of steamline break sizes, break locations, reactor power levels, and potential future protection system logic and steam line pressure setpoint changes.

The results of the 81 cases analyzed by Westinghouse were presented to PG&E in the form of mass (steam flow in lb/sec) and energy (enthalpy in btu/lbm) tables. These data were reviewed by PG&E for impact on the environmental qualification of equipment.

The worst case MSLB outside containment at DCPP that affects equipment qualification is a break in Area GW (Elevation 115') of the Auxiliary Building east of column line J (see FSAR Update Figure 1.2-3). Main steam lines 3 and 4 pass through this area, while lines 1 and 2 do not. FSAR Update Figures 1.2-5 and 1.2-6 show the location of the four steam lines in the plant.

Annular plates that were installed at the containment penetrations on lines 3 and 4 to reduce jet impingement loads, and pipe sleeves installed around lines 3 and 4 in Area GW, limit the break size to an effective maximum break area of 0.8 square feet. For this analysis, the Westinghouse

blowdown energy time histories for break sizes of 0.9 square feet and less were considered. Therefore, out of the 81 cases analyzed by Westinghouse, 29 cases are not applicable for breaks in the GW area because of break size. The remaining cases were evaluated to determine which cases result in the worst temperature profiles in the break compartment and surrounding compartments.

It was determined that there were two blowdown cases which gave limiting temperatures, one for times less than 12 minutes after the break, and another for times greater than 12 minutes. The mass and energy release data for these two cases are for a 0.9 square foot break with the reactor at 100 percent power (large break case) and for a 0.1 square foot break at 100 percent power (small break case).

Based on these two cases, limiting temperature profiles for the break compartment and surrounding compartments affected by the break were calculated using the RELAP 4 Version 1.0 (Bechtel standard computer program NE 568) computer program. Temperature profiles were generated for the following areas in the plant:

- GW-115 (auxiliary building area GW, elevation 115 feet)
- GE-115 (auxiliary building area GE, elevation 115 feet)
- GW-100 (auxiliary building area GW, elevation 100 feet)
- GE-100 (auxiliary building area GE, elevation 100 feet)
- GE/GW-85 (auxiliary building areas GE/GW, elevation 85 feet) only for the 0.1 square foot break.

The temperature profile at elevation 85 feet for the 0.9 square foot break is assumed to be the same as the profile at the corresponding 100 foot elevation for conservatism in evaluating the environmental qualification (EQ) of the equipment at this elevation that must operate after the break.

The current DCPP EQ temperature for areas GE and GW with the BIT in service is 325°F. The new EQ temperature in the break area, with the BIT out of service and with superheated steam blowdown, is 475°F for equipment completing their safety actuation function within 12 minutes of the break, and 511°F for equipment required to operate after 12 minutes. For the large break case (limiting for the first 12 minutes), there is a relatively rapid cooldown after completion of blowdown with temperatures returning to the normal maximum design temperature in all compartments within 20 minutes after the

break. This is primarily due to inclusion of natural convection cooling in the model. This model includes the buoyancy force due to the temperature difference between the hot compartment temperature and the cooler outside air. The buoyancy force causes a chimney effect that draws in the cooler outside air and promotes the rapid cooldown.

For the small break case (limiting after 12 minutes), cooldown is not as rapid as for the large break case. After peaking at 511°F at about 14 minutes, the temperature drops below 200°F within 25 minutes after the break. This slower cooling is mainly the result of assuming that the blowout panels leading to the outside remain intact. The pressure buildup for the small break is lower than for the large break, and for conservatism all of the blowout panels are left in place in this model. Only the permanent openings to the outside provide vent paths for cooling.

A review of the safety related equipment in areas GE and GW was performed to determine those items that would be required to perform a safety function in shutting down the plant after a MSLB without the BIT. These items, plus selected others that have been included for clarity, are listed in Table 1. For each component, the area and elevation, and an evaluation of whether it must be qualified for the superheat environment is included. For those components that must be qualified, the post accident operability time, applicable EQ temperature profile, peak calculated compartment temperature, and EQ temperature margin are given.

In some cases, the requalification was accomplished by thermal lag analysis. This analysis takes into account the fact that the most limiting materials or items in any component may experience a much lower temperature than the compartment temperature due to heat transfer through the component itself. Since the peak compartment temperature lasts for only a short time, the actual maximum temperature that much of the internals of these components experience may be significantly less than the peak. Therefore, the qualification of the component is based on the maximum calculated temperatures that the individual materials in the component would experience based on the applicable compartment temperature profile.

Currently installed equipment that cannot be qualified will be replaced with qualified equipment prior to implementing the elimination of the BIT boric acid concentration requirement. This includes some limit switches, solenoid valves, valve operator motors, electrical tape splices, and electrical terminal blocks.

Electrical cable routed through areas GE and GW was reviewed for the effects of the higher temperatures. Qualification tests have been performed up to 540°F on new cable. Additional testing is currently being performed which includes irradiation and aging. Qualification of this cable to the superheat temperature will be demonstrated prior to elimination of the BIT.

The impact of the new 511°F superheat MSLB temperature profile on the concrete and steel in areas GE and GW has also been evaluated. Structural steel, miscellaneous steel, pipe support steel, pipe restraint steel, concrete walls and floors were included. No adverse safety effects were identified from this evaluation.

c. Results of environmental qualification review

Based on the above analysis, it is concluded that all of the safety related equipment and structural components in GE and GW that would be both subject to the new superheat accident environment and necessary to mitigate the consequences of an accident would either function as designed or would be requalified or replaced. Any testing or component replacement that PG&E determines is necessary to implement the BIT boron concentration requirement will be completed before implementation of the proposed change. Therefore, because plant safety would not be adversely affected, the BIT can safely be removed from service.

4. Analysis of Safety Injection Time Response Values

Westinghouse advised PG&E in 1987 that surveillance testing of the safety injection system components for time response requirements may be inconsistent with the FSAR Update safety analyses and Technical Specifications. Westinghouse noted that the interlock logic between the RWST and VCT outlet isolation valves ensures that the centrifugal charging pumps have a water source for the pump suction. Although this feature is of benefit for ensuring a water source to the charging pump, it will potentially delay the delivery of borated water to the RCS by the valve stroke time of the VCT outlet isolation valves. Since the VCT is pressurized by a hydrogen overpressure of 15-30 psig, the charging pumps may continue to preferentially draw their suction from the VCT until a VCT outlet isolation valve is closed. Westinghouse further stated that the effect of the inconsistent time response requirements on the LOCA and Non-LOCA analyses were negligible due to the BIT boron concentration requirement and the fact that the BIT is downstream of the charging pumps. Borated water from the BIT would be delivered to the RCS regardless of where the charging pump suction source was coming from. On this basis, PG&E concluded that no technical specification changes were required as long as the BIT boron concentration requirement was in effect.

However, the proposed elimination of the BIT boron concentration requirement now necessitates revision of TS Table 3.3-5, "Engineered Safety Features Response Times." Westinghouse has evaluated the change in safety injection response times needed to accommodate the removal of the BIT. This evaluation included consideration of both LOCA and Non-LOCA accident analyses.

a. Non-LOCA SI Response Analyses Evaluation

The Chapter 15 and Chapter 6 Non-LOCA analyses in the FSAR Update have been reviewed to determine the impact of increasing the time response values in TS Table 3.3-5. Also considered in this evaluation was the impact on the outside containment steam line break mass and energy release calculation. The primary transients affected by the change are the steam line break accidents. Core response, inside containment mass and energy release, and outside containment mass and energy release scenarios due to the steam line break all assume safety injection flow that delivers borated water from the RWST in a specified interval. The feedline break and spurious safety injection transients include safety injection flow, but these transients are not sensitive to the delay time. They are primarily based on safety injection flow delivery and are not sensitive to when boron is delivered to the core.

When the BIT removal analyses were performed, it was realized that an extra safety injection response time delay should be included to account for the sequential opening of the RWST valve and closing of the VCT valve. The analyses accounted for at least 22 seconds delay in all three types of steamline break studies (core response, inside containment, and outside containment) for the with-power cases (that is, with no diesel generator starting times included). For the loss of offsite power situation, which is only analyzed for the core response cases, the delay time accounted for in the recent BIT removal analyses was 37 seconds. After the analyses were completed, an extra second of delay was evaluated and found to be acceptable for both the with-power and without-power scenarios. Therefore 23 seconds and 38 seconds are accounted for in the BIT removal analyses.

Based on time response testing results and the additional sequential opening time of the RWST valve and the closing time of the VCT valve, PG&E has now determined that a minimum of 25 seconds should be allowed in the with-power scenarios and 35 seconds in the without-power scenarios. This means that it is necessary to evaluate a 3 second delay in addition to the 22 second delay, which was used in the with-power BIT removal analyses.

This evaluation was performed and it was determined that the extra 3 second delay could be accommodated. Part of this determination was based on the assumption that the BIT will be completely removed (or bypassed) and not left in place and filled with unborated water. The analyses had conservatively assumed the BIT to be left in place. The BIT will be removed (or bypassed) and a much smaller piping volume than the BIT tank volume will take its place. It is judged that the time gained by not having to purge the full BIT volume will offset the extra 3 seconds of delay not originally accounted for in the analyses.

In addition to the BIT not being in place, the evaluation is also based on other arguments regarding the insensitivity of the transients to such a small safety injection delay, and conservatisms in the analyses that will definitely offset a 3 second additional delay. It was judged that even if the BIT were left in place, enough conservatism exists (while still maintaining bounding assumptions) to offset the 3 second additional delay.

b. LOCA SI Response Analyses Evaluation

The VCT to RWST sequential transfer delay without the BIT boron concentration requirement has a potential effect on several LOCA-related accident analyses. Westinghouse evaluated the large break LOCA, small break LOCA, reactor vessel and loop LOCA blowdown forces, and post-LOCA long term core cooling minimum flow requirements. The above LOCA related accident analyses are not adversely affected since the proposed changes do not affect the normal plant operating parameters, the safeguards systems actuations, or accident mitigation capabilities important to a LOCA, or the assumptions used in the LOCA-related analyses. The proposed change does not create conditions more limiting than those assumed in the LOCA analyses. In addition, the above mentioned accident analyses do not consider the effect of the boron concentration in the ECCS water delivered to the RCS to mitigate the effects of a LOCA. However, the following accident analyses are potentially affected and examined in more detail for the proposed change to the safety injection response times.

The effect of prolonged addition of zero ppm borated water from the VCT was evaluated with respect to the post-LOCA long term core cooling subcriticality requirement. It was determined that the current methods and models used to perform this calculation account for the effects of uncertainties in mass inventories due to such things as the sequential stroking of the VCT and RWST valves and are therefore unaffected by the proposed change to TS Table

3.3-5. This calculation is checked on a cycle by cycle basis and the ultimate assurance that the core remains subcritical during post-LOCA conditions is provided as part of the reload evaluation process.

Westinghouse also evaluated the hot leg switchover to prevent potential boron precipitation for any potential changes due to the VCT to RWST sequential valve transfer delay without the BIT boron concentration requirement. It was determined that consideration of VCT water as an additional dilution source for this analysis would act to lengthen the 13.5 hours reported in the FSAR Update. Therefore, the proposed change to relax the SI delay times in the technical specifications will have no adverse effect on the post-LOCA hot leg switchover time.

Based upon the information provided above, PG&E believes that there is reasonable assurance that the health and safety of the public will not be adversely affected by elimination of the BIT and the revised safety injection response times.

E. NO SIGNIFICANT HAZARDS

PG&E has evaluated the no significant hazard considerations involved with the proposed amendment, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards considerations, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

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

   Analysis was performed for an "Accidental Depressurization of the Main Steam System" (FSAR Update Section 15.2.13) and "Major Secondary Steam System Pipe Rupture" (FSAR Update Section 15.4.2)

with the BIT removed. For both cases after the reactor trip, the analysis determined that criticality is reattained due to plant cooldown, but the DNB design basis is met and no fuel failure will occur. Further analysis was performed to determine the impact of BIT removal on the containment mass and energy release and containment pressure and temperature response. It was shown that the containment pressure remained below its 47 psig design limit. The containment temperature response increased from the presently reported peak temperature value of 339°F to 345°F. PG&E has determined that the components inside containment critical to safety are not adversely affected by this small increase in temperature. Therefore, analysis results determined that the containment pressure transient response for the most limiting case assured pressure below design and the aggregate temperature response would not affect the current equipment qualification inside containment. Finally, analysis was performed assuming removal of the BIT to determine the mass and energy release due to steamline breaks outside containment assuming superheated steam release. Analysis results demonstrate that for the worst case main steamline break outside containment, all safety-related equipment required to mitigate the steamline break accident outside containment and structural components that would be both subject to the new superheat accident environment and necessary to mitigate the consequences of an accident would either function as designed or would be requalified or replaced.

The results of the safety injection response time evaluation demonstrated that delivery of borated water to the RCS meets all accident acceptance criteria.

The results of the above analyses demonstrate that consequences of previously evaluated events are not significantly increased. The results of the above analyses further demonstrate an increase in the probability of a return to criticality during a Condition II event (depressurization of the main steam system). However, there is no increase in the probability of fuel failure and releases remain within the guideline values of 10 CFR 20. Therefore, the equipment inside and outside containment necessary to mitigate the consequences of an accident would function as designed after modification and releases during depressurization of the main steam system remain within the guideline values of 10 CFR 20.

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

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

As discussed above, environmentally qualified equipment to provide emergency system functions inside and outside containment during a steamline break has been evaluated for the new environment that could result during accidents with the BIT removed. The analysis results demonstrated that this equipment will either still respond during accidents or will be requalified or replaced.

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

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

For both the "Accident Depressurization of the Main Steam System" (FSAR Update Section 15.2.13) and "Major Secondary Steam System Pipe Rupture" (FSAR Update Section 15.4.2), the Westinghouse analysis shows that the DNB design basis is met and no core damage results. Therefore, for the depressurization of the main steam system, releases associated with this accident will remain within the guideline values set forth in 10 CFR 20 and for the major steam line break the radiation releases are within the guideline values set by 10 CFR 100. The safety injection response times continue to mitigate the consequences of LOCA and non-LOCA accidents with sufficient safety margin.

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

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, PG&E concludes that the activities associated with this LAR satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL EVALUATION

PG&E has evaluated the proposed changes and determined that the changes do 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 changes meet 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 changes is not required.

TABLE 1

MAIN STEAM LINE BREAK (MSLB) COMPONENT LIST - GE/GW AREA

COMPONENT NUMBER	DESCRIPTION	BUILDING LOCATION	EQ TIME REQUIRED	EQ TEMP PROFILE	PEAK TEMP.	TEMP. MARGIN	NOTES	POST MSLB FUNCTION
1-FCV-43	SG-3 MSIV (AO)	GW-115	N/A	-	-	-	1,4,7	Close on operator action 10 minutes after
1-FCV-44	SG-4 MSIV (AO)	GW-115	N/A	-	-	-	1,4,7	accident.
1-SV-303	MSIV-3 Safeguard Tr. A Sol.	GW-115	0	-	-	-	2,6	Remain de-energized, or de-energize. OK to
1-SV-306	MSIV-4 Safeguard Tr. A Sol.	GW-115	0	-	-	-	2,6	fail.
1-SV-304	MSIV-3 Safeguard Tr. B Sol.	GW-115	10 min.	1	475	+15	1,6	Energize on operator action 10 minutes
1-SV-307	MSIV-4 Safeguard Tr. B Sol.	GW-115	10 min.	1	475	+15	1,6	after accident, to close MSIV.
1-POS-381	MSIV-3 Closed Position Swit	GW-115	10 min.	1	475	+19	1	Indicate MSIV status until valve is closed.
1-POS-823	MSIV-3 Closed Position Swit	GW-115	10 min.	1	475	+19	1	"
1-POS-383	MSIV-4 Closed Position Swit	GW-115	10 min.	1	475	+19	1	"
1-POS-825	MSIV-4 Closed Position Swit	GW-115	10 min.	1	475	+19	1	"
1-POS-380	MSIV-3 Open Position Switch	GW-115	10 min.	1	475	+19	1	"
1-POS-382	MSIV-4 Open Position Switch	GW-115	10 min.	1	475	+19	1	"
1-FCV-22	MSIV-4 Bypass Isolation AO	GW-115	N/A	-	-	-	3,7	Remain closed (close or fail closed if open).
1-FCV-23	MSIV-3 Bypass Isolation AO	GW-115	N/A	-	-	-	3,7	"
1-SV-182	FCV-22 Solenoid Valve	GW-115	10 min.	1	475	+15	1	Remain de-energized, or de-energize to close
1-SV-186	FCV-23 Solenoid Valve	GW-115	10 min.	1	475	+15	1	valve.
1-POS-183	FCV-22 Closed Position Swit	GW-115	10 min.	1	475	+19	1	Indicate MSIV bypass status.
1-POS-185	FCV-23 Closed Position Swit	GW-115	10 min.	1	475	+19	1	"
1-POS-182	FCV-22 Open Position Switch	GW-115	10 min.	1	475	+19	1	"
1-POS-184	FCV-23 Open Position Switch	GW-115	10 min.	1	475	+19	1	"
1-FCV-530	SG-3 Feedwater Control AO V	GW-115	N/A	-	-	-	4,7	Close on coincident reactor trip and low
1-FCV-540	SG-4 Feedwater Control AO V	GW-115	N/A	-	-	-	4,7	T(avg.) FW signal within 10 minutes.
1-SV-530A	FCV-530 Safeguard Tr. B Sol	GW-115	10 min.	1	475	+15	1	De-energize to close valve.
1-SV-530B	FCV-530 Safeguard Tr. A Sol	GW-115	10 min.	1	475	+15	1	"
1-SV-540A	FCV-540 Safeguard Tr. B Sol	GW-115	10 min.	1	475	+15	1	"
1-SV-540B	FCV-540 Safeguard Tr. A Sol	GW-115	10 min.	1	475	+15	1	"
1-POS-530	FCV-530 Open Position Switc	GW-115	10 min.	1	475	+19	1,10	Remain lit until FCV closure, to convey
1-POS-540	FCV-540 Open Position Switc	GW-115	10 min.	1	475	+19	1,10	valve status.
1-FCV-1530	SG-3 Feedwater Bypass AO Va	GW-115	N/A	-	-	-	3,7	Remain closed (close or fail closed if open).
1-FCV-1540	SG-4 Feedwater Bypass AO Va	GW-115	N/A	-	-	-	3,7	"
1-SV-1530A	FCV-1530 Safeguard Tr. B So	GW-115	10 min.	1	475	+15	1	Remain de-energized, or de-energize to close
1-SV-1530B	FCV-1530 Safeguard Tr. A So	GW-115	10 min.	1	475	+15	1	valve.
1-SV-1540A	FCV-1540 Safeguard Tr. B So	GW-115	10 min.	1	475	+15	1	"
1-SV-1540B	FCV-1540 Safeguard Tr. A So	GW-115	10 min.	1	475	+15	1	"
1-POS-1530	FCV-1530 Open Position Swit	GW-115	10 min.	1	475	+19	1,10	Remain lit until FCV closure, to convey
1-POS-1540	FCV-1540 Open Position Swit	GW-115	10 min.	1	475	+19	1,10	valve status.
1-FCV-440	SG-3 Feedwater Isolation MO	GW-115	10 min.	1	475	+17	1	Close on feedwater isolation signal within
1-FCV-441	SG-4 Feedwater Isolation MO	GW-115	10 min.	1	475	+17	1	10 min. after accident.
1-PT-534	MSL-3 P3<P4 High dP	GW-115	10 min.	1	475	+166	1,11	Initiates SI for certain breaks.
1-PT-535	MSL-3 P3<P1 High dP	GW-115	10 min.	1	475	+166	1,11	"
1-PT-536	MSL-3 P3<P2 High dP	GW-115	10 min.	1	475	+166	1,11	"
1-PT-544	MSL-4 P4<P3 High dP	GW-115	10 min.	1	475	+166	1,11	"
1-PT-545	MSL-4 P4<P2 High dP	GW-115	10 min.	1	475	+166	1,11	"
1-PT-546	MSL-4 P4<P1 High dP	GW-115	10 min.	1	475	+166	1,11	"

1-FCV-151	SG-1 Blowdown Isolation Val	GE-100	N/A	-	-	-	1,4,7	Close automatically on "T" signal.
1-FCV-154	SG-2 Blowdown Isolation Val	GE-100	N/A	-	-	-	1,4,7	"
1-FCV-157	SG-3 Blowdown Isolation Val	GE-100	N/A	-	-	-	1,4,7	"
1-FCV-160	SG-4 Blowdown Isolation Val	GE-100	N/A	-	-	-	1,4,7	"
1-SV-237	FCV-160 Blowdown Isol. Sol.	GE-100	10 min.	1	367	+123	1	De-energize to close valve.
1-SV-238	FCV-157 Blowdown Isol. Sol.	GE-100	10 min.	1	367	+123	1	"
1-SV-239	FCV-154 Blowdown Isol. Sol.	GE-100	10 min.	1	367	+123	1	"
1-SV-240	FCV-151 Blowdown Isol. Sol.	GE-100	10 min.	1	367	+123	1	"
1-POS-330	FCV-160 Closed Position Swi	GE-100	>10 min.	1,2	438	+56	1	Indicate valve status.
1-POS-332	FCV-157 Closed Position Swi	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-334	FCV-154 Closed Position Swi	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-336	FCV-151 Closed Position Swi	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-331	FCV-160 Open Position Switc	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-333	FCV-157 Open Position Switc	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-335	FCV-154 Open Position Switc	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-337	FCV-151 Open Position Switc	GE-100	>10 min.	1,2	438	+56	1	"
1-FCV-244	SG-4 Blowdown Sample Isol.	GE-100	N/A	-	-	-	1,4,7	Close automatically on "T" signal.
1-FCV-246	SG-3 Blowdown Sample Isol.	GE-100	N/A	-	-	-	1,4,7	"
1-FCV-248	SG-2 Blowdown Sample Isol.	GE-100	N/A	-	-	-	1,4,7	"
1-FCV-250	SG-1 Blowdown Sample Isol.	GE-100	N/A	-	-	-	1,4,7	"
1-SV-244	FCV-244 Blowdn. Sam. Isol.	GE-100	10 min.	1	367	+123	1	De-energize to close valve.
1-SV-246	FCV-246 Blowdn. Sam. Isol.	GE-100	10 min.	1	367	+123	1	"
1-SV-248	FCV-248 Blowdn. Sam. Isol.	GE-100	10 min.	1	367	+123	1	"
1-SV-250	FCV-250 Blowdn. Sam. Isol.	GE-100	10 min.	1	367	+123	1	"
1-POS-344	FCV-244 Closed Position Swi	GE-100	>10 min.	1,2	438	+56	1	Indicate valve status.
1-POS-348	FCV-246 Closed Position Swi	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-352	FCV-248 Closed Position Swi	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-356	FCV-250 Closed Position Swi	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-345	FCV-244 Open Position Switc	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-349	FCV-246 Open Position Switc	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-353	FCV-248 Open Position Switc	GE-100	>10 min.	1,2	438	+56	1	"
1-POS-357	FCV-250 Open Position Switc	GE-100	>10 min.	1,2	438	+56	1	"
1-FCV-37	Isol. MOV Fr. MSL-2 to AFWP	FW-124	0	-	-	-	1,8	Does not see MSLB environment.
1-FCV-38	Isol. MOV Fr. MSL-3 to AFWP	GE-115	0	-	-	-	2,5	Remain open.
1-FCV-95	Isol. MOV, MSL-2&3 to AFWP	GE-115	10 min.	1	355	+137	1	Open to supply steam to AFWP turbine.
1-FT-78	AFW Flow to MSL-3	GE-115	>10 min.	1,2	465	+65	1	Indicate AFW flowrate.
1-FT-79	AFW Flow to MSL-4	GW-115	>10 min.	1,2	511	+19	1	"
1-LCV-108	Control Valve TD AFWP to MS	GE-115	>10 min.	1,2	465	+27	1	Control AFW flow to MSL-3.
1-LCV-109	Control Valve TD AFWP to MS	GE-115	>10 min.	1,2	465	+27	1	Control AFW flow to MSL-4.
1-MOV-8801A	Charging Injection Valve	GE 104	10 min.	1	367	+125	1	Open on SI signal.
1-MOV-8801B	Charging Injection Valve	GE 104	10 min.	1	367	+125	1	"
1-MOV-8716A	RHR HX1 to RCS Loops 1&2 Ho	GW-105	0	-	-	-	2,5	Remain open.
1-MOV-8716B	RHR HX2 to RCS Loops 1&2 Ho	GE-100	0	-	-	-	2,5	"
1-MOV-8809A	RHR HX1 to RCS Loops 1&2 Co	GW-107	0	-	-	-	2,5	"
1-MOV-8809B	RHR HX2 to RCS Loops 3&4 Co	GE-100	0	-	-	-	2,5	"
1-MOV-8980	RHR Pumps Suction from RWST	GE-105	0	-	-	-	2,5	"
1-MOV-8700A	RHR Pump 1-1 Suction	GW-76	0	-	-	-	1,8	Does not see MSLB environment.
1-MOV-8700B	RHR Pump 1-2 Suction	GE-76	0	-	-	-	1,8	"
1-FT-970A	RHR Hx 1-1 out	GW-85	>10 min.	1,2	428	+102	1	Indicate RHR flow.
1-FT-970B	RHR Hx 1-1 out	GW-85	>10 min.	1,2	428	+102	1	"
1-FT-971A	RHR Hx 1-2 out	GE-90	>10 min.	1,2	428	+102	1	"
1-FT-971B	RHR Hx 1-2 out	GE-90	>10 min.	1,2	428	+102	1	"

TABLE 1

1-FCV-360	CCW from Cont. Fan Cool. 1,	GE-85	N/A	-	-	-	1,9	Open on "S" signal. OK to fail.
1-FCV-366	CCW from Cont. Fan Coolers	GE-85	N/A	-	-	-	1,9	"
1-SV-397	FCV-360 Solenoid Valve	GE-85	10 min.	1	373	+115	1	De-energize to open FCV.
1-SV-400	FCV-366 Solenoid Valve	GE-85	10 min.	1	373	+115	1	"
1-FCV-356	CCW MOV to Reactor Coolant	GE-108	0	-	-	-	2,5	Remain open until hot shutdown condition is reached.
1-FCV-357	CCW MOV from Reactor Coolan	GE-105	0	-	-	-	2,5	
1-FCV-363	CCW MOV from RCP Oil Cooler	GE-108	0	-	-	-	2,5	"
1-POS-896	FCV-356 Open/Closed Positio	GE-108	0	-	-	-	2	Not required for MSLB.
1-POS-899	FCV-357 Open/Closed Positio	GE-105	0	-	-	-	2	"
1-POS-898	FCV-363 Open/Closed Positio	GE-108	0	-	-	-	2	"
1-FT-65	CCW Supply Header B	GW-85	>10 min.	1,2	428	+102	1	Indicate CCW flow.
1-FT-68	CCW Supply Header A	GW-85	>10 min.	1,2	428	+102	1	"
1-MOV-8100	RCP Seal Water Return Isola	GE-92	0	-	-	-	2,5	Remain open.
1-BTG-1E thru 1-BTG-37E	Electrical Penetrations	GW-115	>10 min.	1,2	511	+64	1	Maintain containment integrity & electrical operability.
	Electrical Cable	All Areas	>10 min.	1,2	511	later	12	Electrical operability.
	Electrical Termination (Raychem Splice)	All Areas	>10 min.	1,2	511	later	12	Electrical operability.

NOTES

1. Device must operate following a MSLB in area GE/GW.
2. Device is not required to operate following a MSLB in area GE/GW.
3. Valve is normally closed, air-to-open, fail closed design.
4. Valve is normally open, air-to-open, fail closed design.
5. This motor operated valve is normally open, fail-as-is.
6. The MSIV solenoids must be energized to close the MSIV's. The safeguard Train A solenoid valves (SV-303 and 306) are only in the auto-isolation circuits, while the Train B SV's (SV-304 and 307) are in both the auto and manual isolation circuits. Since the Train A SV's cannot be energized by an operator from the main control room switch they will not be involved in shutting down the plant for the MSLB (within area GW) without the BIT. Therefore, only the Train B SV's (SV-304 and 307) must be qualified for the superheat temperature.
7. Temperature is not a concern for the valve body. This air operated valve is designed to fail closed. There are no electrical components on this valve, and EQ is not applicable.
8. This item is located in an area that does not see the MSLB environment for a break in area GW.
9. Valve is normally closed, air-to-close, fail open design.
10. The design does not include a separate closed position switch. The signal from the switch to the control room status light is present when the valve is open, and absent when the valve is fully closed.
11. Deletion of the high steam line delta P safety function is pending. Analysis to support this deletion is currently underway by Westinghouse. For this accident main steam line isolation is assumed on operator action at 10 minutes after the accident.
12. Several new electrical cables and splice have been tested for 2 hours minimum at 540 degrees F. Additional testing on aged cable and splice is currently underway. Qualification will be demonstrated prior to removal of BIT.