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10 CFR 50.91

August 12, 2020

PG&E Letter DCL-20-066

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

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
<u>License Amendment Request 20-01</u>
<u>Exigent Request for Revision to Technical Specification 3.7.5, "Auxiliary Feedwater System"</u>

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.91, Pacific Gas and Electric Company (PG&E) hereby requests approval of the enclosed proposed amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively. The enclosed license amendment request (LAR) proposes to revise Technical Specification (TS) 3.7.5, "Auxiliary Feedwater System."

The proposed changes would revise the Operating Licenses to provide a new TS 3.7.5, "Auxiliary Feedwater System" (AFW), Condition G to address a one-time planned Unit 1 Cycle 22 AFW system alignment for which current TS 3.7.5 would require shutdown. The new alignment proposed in Condition G is for one or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path(s) to one steam generator (SG). The new Condition G includes Required Actions to isolate AFW to the affected SG within 2 hours and to restore the AFW system to operable status within 7 days. TS 3.7.5 Conditions B and D are correspondingly revised to add reference to the new Condition G.

The TS 3.7.5 change requested in this exigent LAR provides a more appropriate TS 3.7.5 Condition and Completion Time that is commensurate with the online risk associated with a specific AFW system configuration which is conservatively anticipated could be identified during Unit 1 Cycle 22 planned upcoming inspections to the AFW system. The online risk for the proposed AFW system alignment, based on risk insights, is considered to not be risk significant as a result

of the substantial redundancy in the design of the AFW system. Additionally, the proposed Condition G actions assure that the plant remains within the bounds of existing design basis safety analyses. The TS 3.7.5 change will avoid an unnecessary plant shutdown during the expected time needed to perform the potential repairs, and associated post-maintenance inspections and testing to the Unit 1 AFW system piping.

Because of localized corrosion identified on Unit 2 AFW piping during a recent Unit 2 maintenance outage, PG&E plans to perform inspections of Unit 1 AFW piping in the near term to ensure that Unit 1 is not similarly affected. Accordingly, PG&E requests approval of this LAR on an exigent basis no later than August 25, 2020. PG&E requests the license amendment(s) be made effective upon NRC issuance, to be implemented within 48 hours 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.

In accordance with site administrative procedures and the Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91, PG&E is sending a copy of this proposed amendment to the California Department of Public Health.

If you have any questions or require additional information, please contact Mr. James Morris at 805-545-4720.

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

Executed on August 12, 2020.

Sincerely,

Paula Gerfen

Paula Gerfen
Site Vice President

kjse/4328/51084143

Enclosure

cc: Diablo Distribution

cc/enc: Samson S. Lee, NRR Senior Project Manager
Scott A. Morris, NRC Region IV Administrator
Christopher W. Newport, NRC Senior Resident Inspector
Gonzalo L. Perez, Branch Chief, California Department of Public Health

Evaluation of the Proposed Change

License Amendment Request 20-01
Exigent Request for Revision to Technical Specification 3.7.5,
“Auxiliary Feedwater System”

ATTACHMENTS:

1. Technical Specification Page Markups
2. Retyped Technical Specification Pages
3. Technical Specification Bases Page Markups

EVALUATION

1. SUMMARY DESCRIPTION

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

The proposed changes would revise the Operating Licenses to provide a new Technical Specification (TS) 3.7.5, "Auxiliary Feedwater System" (AFW), Condition G to address a one-time planned Unit 1 Cycle 22 AFW system alignment for which current TS 3.7.5 would require shutdown. The new alignment proposed in Condition G is for one or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path(s) to one steam generator (SG). The new Condition G includes Required Actions to isolate AFW to the affected SG within 2 hours and to restore the AFW system to operable status within 7 days. TS 3.7.5 Conditions B and D are correspondingly revised to include reference to the new Condition G.

The proposed TS 3.7.5 change requested in this exigent LAR provides a more appropriate TS 3.7.5 Condition and Completion Time that is commensurate with the online risk associated with a specific AFW system configuration which is conservatively anticipated could be identified during Unit 1 Cycle 22 during planned upcoming potential inspections to the AFW system. Additionally, the proposed Condition G actions assure that the plant remains within the bounds of existing design basis safety analyses. The TS 3.7.5 change will avoid an unnecessary plant shutdown during the expected time needed to perform potential repairs to the Unit 1 AFW system piping.

Because DCPP TS are combined for both Units 1 and 2, the changes are requested for both Operating Licenses DPR-80 and DPR-82 for DCPP Units 1 and 2, respectively. However, the proposed Condition G will include a Note specifying that the Condition is applicable to Unit 1 only. TS 3.7.5 Conditions B and D will need to be correspondingly revised to add reference to the new proposed Condition G such that the Condition G Completion Time of 7 days will apply for the Condition G AFW system alignment.

2. DETAILED DESCRIPTION

<u>Proposed Amendment</u>

A new Condition G is proposed to be added to TS 3.7.5.

Condition G states:

G. -----------NOTE------------

This Condition is only applicable to Unit 1 once during Unit 1 Cycle 22 during repair of AFW piping.

One, or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path(s) to one steam generator.

Condition G Required Action G.1 states:

"Isolate AFW flow path(s) to affected steam generator" with a Completion Time of "2 hours."

Condition G Required Action G.2 states:

"Restore AFW train(s) to operable status" with a Completion Time of "7 days."

Condition B is modified from:

"One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A."

to

"One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A or G."

In addition, existing Condition D is modified from:

"Required Action and associated Completion Time for Condition A, B, or C not met.

<u>OR</u>

Two AFW trains inoperable in MODE 1, 2 or 3 for reasons other than Condition C"

to

> "Required Action and associated Completion Time for Condition A, B, C, or G not met.
>
> OR
>
> Two AFW trains inoperable in MODE 1, 2 or 3 for reasons other than Condition C or G."

The proposed TS changes are noted on the marked-up TS page provided in Attachment 1. The proposed retyped TS is provided in Attachment 2.

The TS Bases changes for new TS 3.7.5 Condition G are included in Attachment 3 for information only and will be implemented in accordance with the TS Bases Control Program.

In summary, a new TS 3.7.5 Condition G is proposed to address a one-time planned Unit 1 Cycle 22 AFW system alignment for which the current TS 3.7.5 would require shutdown to MODE 4. The new TS 3.7.5 Condition G is for one or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path(s) to one SG. The new Condition G includes Required Actions to isolate AFW to the affected SG within 2 hours and to restore the AFW system to operable status within 7 days. TS 3.7.5 Conditions B and D are revised to include new Condition G.

AFW System Description

The AFW system automatically supplies feedwater to the SGs to remove decay heat from the reactor coolant system (RCS) upon the loss of normal feedwater supply. The AFW pumps take normal suction from a single suction line from the condensate storage tank (CST) and deliver flow to the SG secondary side via separate and independent connections to the main feedwater piping outside containment. The SGs function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the SGs via the main steam safety valves (MSSVs) or atmospheric dump valves (ADVs). If the main condenser is available, steam may be released via the condenser steam dump valves and recirculated to the CST.

The AFW system consists of two motor-driven AFW pumps and one steam turbine-driven AFW pump configured into three trains. Each motor-driven pump provides 100 percent of the feedwater flow required for removal of decay heat from the reactor based on "better estimate" conditions. The better estimate evaluation provides a reliability basis for assuming availability of both motor-driven pumps for accident analyses. The turbine-driven pump provides 200 percent of the capacity of a motor-driven pump.

The turbine-driven AFW pump supplies a common header capable of feeding all SGs with vital alternating current (AC) powered control valves. Thus, the requirement for diversity in motive power sources for the AFW system is met.

The AFW System is capable of supplying feedwater to the SGs during normal unit startup, shutdown, hot standby, and hot shutdown conditions.

The AFW System supplies sufficient water to the SG(s) to remove decay heat with SG pressure at the lowest setpoint of the MSSVs. The AFW System supplies sufficient water to cool the unit to residual heat removal entry conditions, with steam released through the ADVs.

The AFW System (one turbine-driven and two motor-driven trains) actuates automatically upon actuation of the anticipated transient without scram ATWS mitigating system actuation circuitry (AMSAC).

The motor-driven pumps are additionally actuated by (1) safety injection signal; (2) an associated bus transfer to the diesel generator signal; (3) a trip of both main feedwater pumps; or (4) SG water level—low-low in one of four SGs.

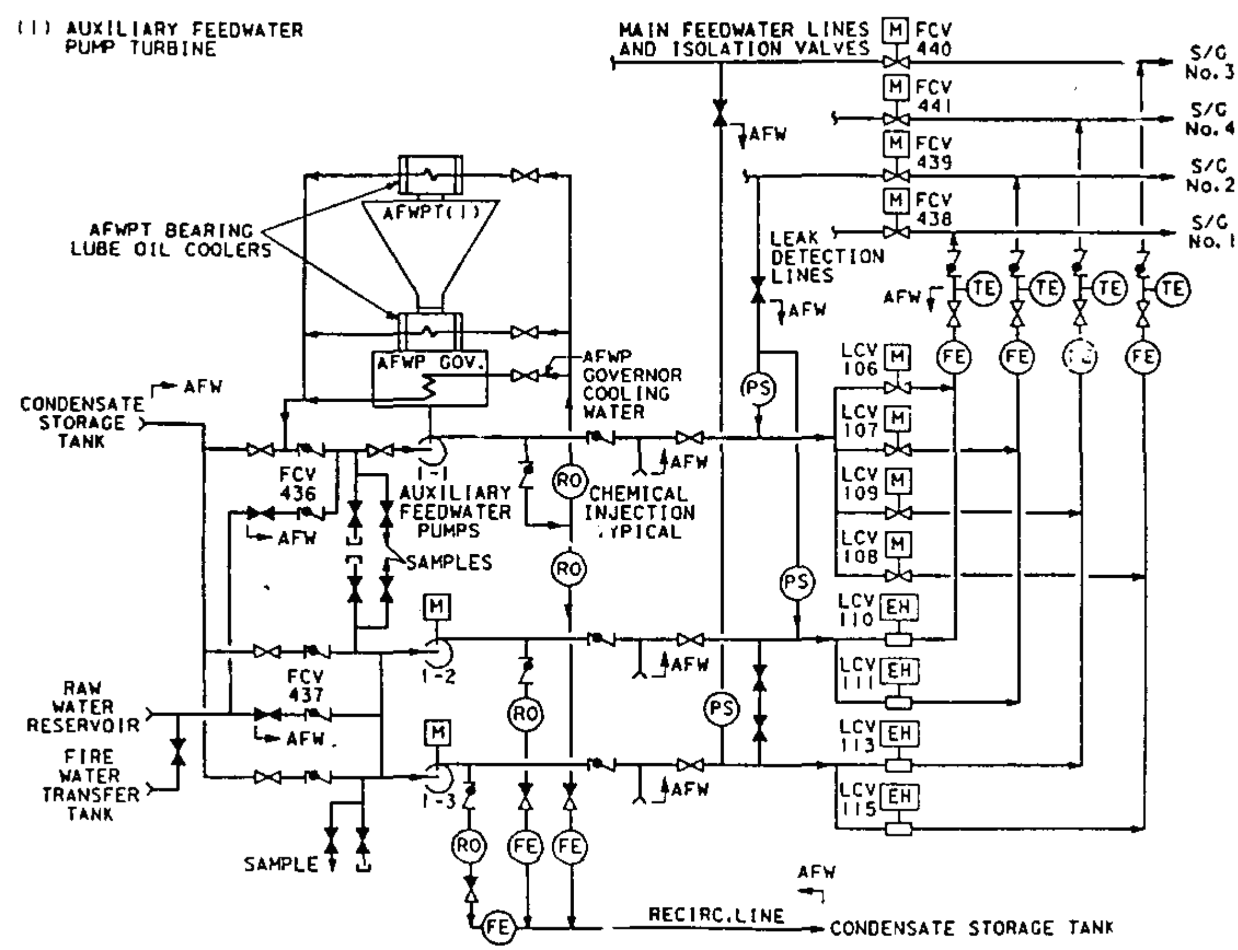
The turbine-driven pump is actuated by 12 kilovolt (kV) bus undervoltage or SG low-low level in two of four SGs via the Emergency Safety Feature Actuation System. A simplified diagram of the AFW is provided in Figure 1.

The turbine-driven Pump 1-1 discharge lines include level control valves (LCV) to the SGs; LCV-106 to SG 1-1, LCV-107 to SG 1-2, LCV-108 to SG 1-3, and LCV-109 to SG 1-4 that can be operated from the Control Room. The Unit 1 piping between LCV-107 and SG 1-2 includes Line 570, which contains piping and elbows that potentially may not meet minimum code thickness requirements and could require repair.

The motor-driven Pump 1-2 discharge lines include LCVs LCV-110 to SG 1-1 and LCV-111 to SG 1-2 and motor-driven Pump 1-3 discharge lines include LCVs LCV-113 to SG 1-4 and LCV-115 to SG 1-3 that can be operated from the Control Room. The Unit 1 piping between LCV-111 and SG 1-2 includes Line 576, which contains piping and elbows that potentially may not meet minimum code thickness requirements and could require repair.

Figure 1 – Auxiliary Feedwater Simplified System Diagram

AUXILIARY FEEDWATER SIMPLIFIED SYSTEM DIAGRAM



The AFW System is discussed in the DCPP Final Safety Analysis Report Update (FSARU), Section 6.5.

Basis for Exigent Change

On July 17, 2020, with DCPP Unit 2 operating at 100 percent power, increasing hydrogen usage was observed in the Unit 2 Main Generator. In accordance with

plant procedures, the reactor was manually shutdown, and Unit 2 was transitioned to Mode 3.

On July 23, 2020, with DCPP Unit 2 still in Mode 3, a 3.9 gallons per minute calculated through-wall leak was observed coming out of the elbow just downstream of Valve LCV-111 in the discharge line for Unit 2 AFW Pumps 2-1 and 2-2 to SG 2-2. The unit was transitioned to Mode 4 in accordance with DCPP TS 3.7.5, Required Action D.2. Repairs were made to the AFW piping while shutdown prior to returning Unit 2 to power operation.

An Extent of Condition (EOC) Investigation performed for the Unit 2 AFW piping leak identified no additional leaks. However, six additional locations were identified in the Unit 2 AFW system where repairs were required because pipe wall thickness did not meet minimum ASME code requirements. The repairs were completed and inspected, and the affected AFW trains were returned to operable status on July 31, 2020.

In accordance with the DCPP Corrective Action Program, an EOC investigation is now being planned for DCPP Unit 1, and inspections and any corresponding necessary repair(s) are planned to begin at the end of August, 2020. Based on the conditions found in the Unit 2 AFW piping and walkdowns of the Unit 1 AFW piping, it is suspected there could be Unit 1 AFW piping and elbow locations that may not meet minimum code requirements and could require repair. Based on the experience with the Unit 2 AFW piping repairs, it is expected it could take up to 7 days to complete repairs to the Unit 1 AFW piping and elbow locations, including preparations, weld repairs, weld quality examination, removal of clearances, and return of AFW system to operable status.

The current AFW TS 3.7.5 does not contain a Condition that is applicable to the configuration the AFW may need to be placed in during current Mode 1 to perform the piping repairs at the Unit 1 AFW piping and elbow locations. While the planning and preparation for the Unit 1 EOC investigation are in progress, the decision was made to submit this LAR for NRC review on an exigent basis in accordance with 10 CFR 50.91(a)(6) in order to provide a more appropriate TS 3.7.5 Condition and Completion Time that is commensurate with the online risk associated with the specific AFW system configuration, which will exist during the planned repairs to the AFW system during Cycle 22. The TS 3.7.5 change will avoid an unnecessary plant shutdown from Mode 1 to perform potential repairs to the Unit 1 AFW system piping.

At this time, DCPP Unit 1 is in Mode 1 and the AFW system is operable.

An exigent LAR review is appropriate because:

(1) If similar below-minimum pipe wall thicknesses are found in the Unit 1 AFW system piping and elbows that were found in Unit 2, based on the

estimated time-to-repair gained from the Unit 2 repair, it is likely that the current TS 3.7.5 Required Actions B.1 or D.1 would result in the required shutdown of Unit 1 from current Mode 1;

(2) This LAR is timely submitted; that is, PG&E has assessed the potential extent of the upcoming needed Unit 1 AFW system piping repairs based on the required repairs for Unit 2 and is making its best efforts to make a timely application and has not created the exigency; and

(3) The background provided above describes the emergent potential need to perform repairs to the Unit 1 AFW system piping during Mode 1 and reasonably explains why this exigent situation occurred and why PG&E could not avoid this situation.

3. TECHNICAL EVALUATION

<u>Technical Specification Changes</u>

A new Condition G is proposed to be added to TS 3.7.5. The condition applies to an AFW system alignment, that is not included in the current TS Conditions; for one or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path(s) to one SG. The Condition includes MODES 1, 2, and 3 to permit AFW system repair during MODES that it is required to be operable. Since this TS change is applicable only for one-time use and only is applicable to Unit 1 Cycle 22, a note is included in Condition G that states the Condition is only applicable to Unit 1 once during Unit 1 Cycle 22 during any necessary repairs of AFW piping.

The new Condition G includes two Required Actions, G.1 and G.2. Required Action G.1 is to isolate AFW flow paths to the affected SG with a Completion Time of 2 hours. This action is accomplished by closing the LCV(s) for potentially inoperable AFW piping due to insufficient wall thickness from one turbine-driven AFW pump and/or motor -driven pump. Two hours provides adequate time for operators to isolate the AFW flow path(s) by closing the LCV(s) (i.e., LCV-107 and/or LCV-111 to SG 1-2) to the affected SG and verify effective isolation through plant instrumentation upon entry into Condition G.

Required Action G.2 is to restore AFW train(s) to operable status with a Completion Time of 7 days. The Completion Time of 7 days provides sufficient time to repair the Unit 1 AFW piping and is reasonable based on the capabilities of the two motor-driven AFW pumps and one turbine-driven AFW pump to provide adequate AFW cooling flow, the time needed for Unit 1 AFW repairs, and the low probability of a design basis accident occurring during this period. While in Condition G the SG 1-2 related TS required equipment will continue to remain operable. In addition, based on risk insights using the DCPP Probabilistic Risk

Assessment (PRA) model, the 7-day Completion Time is considered to not be risk significant. While in Condition G, the turbine-driven AFW pump and both motor-driven AFW pumps will remain available to supply AFW to three SGs. It is noted that current TS 3.7.5 Condition A allows a 7-day Completion Time for a turbine-driven AFW train inoperable due to one inoperable steam supply because there is a redundant steam supply line for the turbine-driven pump and the turbine-driven train is still capable of performing its specified function for most postulated events.

Conditions B and G are modified to add new Condition G as a Condition for which an inoperable Condition is applicable. This change ensures the new Condition G Completion Time of 7 days will be applicable when one or two AFW trains are inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path to one SG. The change also applies Required Actions D.1 and D.2 if new Condition G Required Actions G.1 or G.2 are not met within their Completion Times.

System Safety Analysis Basis

DCPP FSARU Section 6.5.2.2, "Design Conditions," identifies that the following reactor plant conditions impose safety-related performance requirements on the AFW System:

(1) Loss of normal feedwater [main feedwater] transient

  (a) Loss of normal feedwater with offsite power available

  (b) Loss of offsite power to the station auxiliaries

(2) [Major] Secondary system pipe ruptures

  (a) Feedline rupture

  (b) Steam line rupture (inside containment)

(3) Loss of all AC power

(4) Small break loss-of-coolant accident (LOCA)

(5) Cooldown

These reactor plant conditions are evaluated below for the new proposed TS 3.7.5 Condition G.

In the event that AFW System Lines 570 and/or 576 are discovered to be inoperable, the AFW piping to SG 1-2 may be isolated for a period to conduct

repairs while both motor-driven AFW pumps and the turbine-driven AFW pump are available. Motor-driven Pump 1-2 is available to deliver flow to SG 1-1 and motor-driven Pump 1-3 is available to deliver flow to SG 1-3 and SG 1-4. The turbine-driven AFW pump is available to provide flow to SGs 1-1, 1-3, and 1-4. Having flow available from three pumps to three SGs is a less limiting situation than what is assumed in the FSARU accident analyses for Condition 3 and 4 events, which assume flow from only one motor-driven pump to two SGs. Therefore, this alignment during the repairs is bounded by the more conservative assumptions for major accidents.

In general, repairs for identified out-of-service components is addressed by deterministic FSARU Chapter 15 accident analysis by consideration of the worst single failure. That is, the inoperable component is postulated, for the brief period of repair while in a TS Condition, to be the assumed single failure, and accident analyses will already address the impact or bound the impact.

For the accidents that require evaluation below for acceptable AFW function while in the proposed TS 3.7.5 Action due to inoperable AFW system piping, no additional equipment failures are required to be postulated when demonstrating the safety function is still maintained.

Loss of Normal Feedwater Transient

The condition 2 event of loss of normal feedwater is addressed in FSARU Section 15.2.8. This transient is modeled with an assumed single failure of the turbine-driven pump, resulting in the remaining two motor-driven pumps operable and feeding all four SGs with a total of 600 gallons per minute (gpm) flow.

The proposed possible isolation of AFW flow to SG 1-2 means that the AFW system will have three available AFW pumps, which can provide well above 600 gpm, but only to the three unaffected SGs.

There is an additional FSARU analysis of the loss of normal feedwater transient in Section 6.5.3.7, termed a “better-estimate” analysis, that is done for AFW reliability demonstration. FSARU Section 6.5.3.7 notes that the FSARU Section 15.2.8 analysis has considerable margin when 4 SGs are credited, and that the better-estimate analysis shows successful event mitigation with just two SGs receiving a total of 390 gpm. Therefore, the proposed SG 1-2 isolation case, with three available SGs, is bounded by the FSARU Section 6.5.3.7 better-estimate case which only credits AFW flow to 2 SGs.

Loss of Offsite Power to the Station Auxiliaries

The loss of offsite power evaluation in FSARU Section 15.2.9 assumes AFW flow to all four SGs, and isolation of SG 1-2 will result in AFW flow to only three SGs. This is judged to be similar to the loss of normal feedwater event, but less limiting

as the AFW heat removal is less by the 20 MWs of reactor coolant pump power that is conservatively assumed for the loss of normal feedwater event. Therefore, the loss of offsite power case is bounded by the better-estimate loss of normal feedwater event. The loss of normal feedwater case is more limiting because it assumes only two SGs and a higher RCS heat load.

Secondary Pipe Ruptures – Feedline Rupture

The Feedline Rupture analysis is concerned with minimum AFW flow. The current FSARU analysis assumes a single motor-driven pump supplies two SGs with a total flow of 390 gpm starting ten minutes after a unit trip. For most scenarios, isolating AFW flow to SG 1-2 but having no other single failure assures well above 390 gpm flow divided among three SGs. Therefore, the transient is bounded by the FSARU accident analysis.

However, a feedline break may create a low resistance path which could divert AFW flow out of the break and away from intact SGs. For example, a break on the SG 1-4 line could prevent motor-driven Pump 1-3 from supplying flow to SG 1-3. This is a recognized concern for the feedline rupture event response, and operators are currently trained on the time-critical operator action of isolating any ruptured SG within ten minutes as part of the DCPP Time Critical Operator Action Program. Plant Procedure OP1.ID2, “Time Critical/Sensitive Operator Action”, Action 20 is the operator action to isolate the faulted SG within 10 minutes of the break initiation for the AFW, Main Steam, and Main Feedwater systems. Establishing appropriate AFW flow to the unfaulted SGs occurs as part of isolating the faulted SG. No AFW flow is credited in the accident analysis until ten minutes. The worst identified scenario would be a break in the line to SG 1-3 and subsequent isolation of SG 1-3. This is worse than an isolation of SG 1-4, since the turbine-driven pump requires steam from either SG 1-2 or 1-3, and both would be isolated. In this scenario, both motor-driven pumps would be operating. The two motor-driven pumps would be feeding one SG each. This is bounded by the accident analysis assumption of one motor-driven pump feeding two SGs, and is therefore acceptable.

[Major] Secondary Pipe Ruptures – Main Steam Line Break (MSLB)

The MSLB analysis is limited by concerns of maximum AFW flow. The concern is that cooling of the RCS provides positive reactivity and a return to power. A loss of one SG through its isolation would be a benefit for this concern. The transient is therefore not negatively impacted by isolating the AFW to SG 1-2.

Other Events

The AFW functions for the Loss of all AC power, LOCA, and RCS Cooldown can be achieved with SG 1-2 isolated and no additional postulated equipment failures. The Loss of all AC power cooldown evaluation is similar in terms of

AFW requirements to the loss of offsite power to station auxiliary systems. Based on the successful evaluation of the better-estimate loss of normal feedwater, the loss of all AC would involve less heat load and one additional SG. Therefore, it would be bounded. The LOCA and RCS Cooldown functions only credit long term AFW decay heat removal with flow requirements that are bounded by the FSARU design basis accident analysis. Therefore, these AFW functions do not require any further detailed evaluation for this change.

In conclusion, the redundancy of the AFW system makes the loss of AFW to SG 1-2, while in proposed TS 3.7.5 Condition G, bounded by FSARU accident analyses. This evaluation identifies that for a main feedline break on SG 1-3 or SG 1-4, operator action is required to assure that the existing analysis is bounding. That operator action is an existing time critical operator action in DCPP Plant Procedure OP1.ID2 and is tested to be completed within 10 minutes prior to accident analysis assumed AFW initiation time. Therefore, the main feedline break accident is also bounded by the FSARU Chapter 15 analysis.

Risk Insights

This LAR is not a risk-informed LAR. However, for additional information, the risk insights determined with the DCPP PRA model are summarized here.

The isolation of AFW supply to SG 1-2 has been assessed using the DCPP PRA model, which includes internal events, internal flooding, fire, and seismic. This assessment assumes that during the extended Completion Time: (1) the unavailable SG 1-2 is isolated from AFW only, thus AFW cooling to SGs 1-1, 1-3 and 1-4 is not impacted; (2) motor-driven AFW Pump 1-2 is still available to supply cooling to SG 1-1; (3) turbine-driven AFW Pump 1-1 is still available to supply cooling to SGs 1-1, 1-3 and 1-4; (4) all three AFW trains are available; and (5) no other PRA credited components are assumed to be out of service. Average maintenance model of record “DC04” is used as the baseline model for the application model “LARSG12.”

The proposed 7-day Completion Time is used as the exposure period for AFW flow isolated to SG 1-2 to calculate the incremental conditional core damage probability (ICCDP) and an incremental conditional large early release probability (ICLERP) using the Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) increase from the model above. This results in ICCDP and ICLERP below 1.00E-06 and 1.00E-07, respectively. Since the proposed ICCDP for 7 days is below 1.00E-06 and the proposed ICLERP for 7 days is below 1.00E-07 the risk increase due to this extended Completion Time is considered to not be risk significant.

The following risk management actions have been determined based on insights from the PRA evaluation above and have been accepted by operations to be implemented during the proposed TS 3.7.5 Condition G 7-day Completion Time:

- Protect AFW Pump 1-3 and supporting equipment. This supporting equipment includes vital 4 kV and 480 V Bus F, vital DC Bus 1, Battery Charger 1-1 and Emergency Diesel Generator 1-3.

- Protect AFW Pump 1-1 and 1-2

- Protect the remaining AFW LCVs (LCV-106, LCV-108, LCV-109, LCV-110, LCV-112, LCV-113) locally

- Protect Pressurizer Power-Operated Relief Valves PCV-455C and PCV-456

- Shiftly tailboard procedure Emergency Operating Procedure FR-H.1, “Response to Loss of Secondary Heat Sink,” on providing Main Feedwater to the SGs on a loss of AFW

- Shiftly tailboard on feed and bleed cooling (including Residual Heat Removal sump recirculation) on a loss of AFW and Main Feedwater

AFW System Summary/Conclusion

The DCPP FSARU Section 6.5.2.2, Design Conditions for the AFW system have been evaluated for the new proposed TS 3.7.5 Condition G.

The substantial redundancy of the AFW system makes the loss of AFW to SG 1-2, while in proposed TS 3.7.5 Condition G, bounded by the FSARU accident analyses. The evaluation identified that for a main feedline break on SG 1-3 or SG 1-4, operator action is required to assure that the existing analysis is bounding, and that operator action is an existing time critical operator action in DCPP Plant Procedure OP1.ID2 and is tested to be completed within the accident analysis assumed AFW initiation time. Therefore, the main feedline break accident is bounded by the FSARU Chapter 15 analysis.

In addition, based on risk insights using the DCPP PRA model, the 7-day Completion Time for proposed TS 3.7.5 Condition G is considered to not be risk significant. Risk management actions have been determined based on insights from the PRA evaluation and have been accepted by operations to be implemented during the proposed TS 3.7.5 Condition G 7-day Completion Time.

4. REGULATORY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

The proposed change to the requirements in Technical Specification (TS) 3.7.5, "Auxiliary Feedwater (AFW) System," provides a new TS 3.7.5 Condition G which is applicable for one or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path to one SG. The new Condition G includes Required Actions to isolate AFW to the affected SG within 2 hours and to restore the AFW system to operable status within 7 days. TS 3.7.5 Conditions B and D are revised to include new Condition G. This change does not involve a permanent change to the design or performance capability of any DCPP systems, structures, and components.

General Design Criteria

DCPP Units 1 and 2 were designed to comply with the Atomic Energy Commission (AEC) (now the Nuclear Regulatory Commission, or NRC) General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, published in July 1967. PG&E has made subsequent commitments to GDCs issued later (e.g., 1971 GDC 11, 1967 is supplemented by GDC 19, 1999 for Dose) that are discussed in Section 3.1 of the DCPP FSARU. The applicable criterion listed below related to this change are individually addressed.

It is noted that 1971 GDC 34, Residual Heat Removal, and 1971 GDC 44, Cooling Water, have no direct correlation with the 1967 GDC and therefore, DCPP was not designed to these 1971 criteria.

The following 1967 GDC and discussion following are applicable to the changes in this license amendment request:

Criterion 11 (1967) - Control Room

The facility shall be provided with a Control Room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the Control Room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the Control Room is lost due to fire or other cause.
Criterion 19 (1999) - Control Room

A Control Room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection shall be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Holders of operating licenses using an alternative source term under 10 CFR 50.67 shall meet the requirements of this criterion, except that with regard to Control Room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effect dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident.

AFW system instruments and controls are located in the Control Room, as well as the hot shutdown panel and the AFW system supports maintaining radiation exposures below 5 rem TEDE.

Criterion 12 (1967) - Instrumentation and Control Systems

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

AFW system instruments and controls are located in the Control Room, as well as the hot shutdown panel.

Criterion 15 (1967) - Engineered Safety Features Protection Systems

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

An important safety function of the reactor protection system is that of processing signals used for engineered safety feature *(*ESF) actuation and generation of the actuation demand.

The AFW system automatically supplies feedwater to the SGs to remove decay heat from the RCS upon the loss of normal feedwater supply.

Criterion 20 (1967) - Protection Systems Redundancy and Independence

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different

principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Sufficient redundancy and independence is designed into the protection systems to ensure that no single failure nor removal from service of any component or channel of a system will result in loss of the protection function. The minimum redundancy is exceeded in each protection function that is active with the reactor at power.

Functional diversity and consequential location diversity are designed into the systems. DCPP uses the Westinghouse Eagle 21 Process Protection System.

Criterion 21 (1967) - Single Failure Definition

Multiple failures resulting from a single event shall be treated as a single failure.

When evaluating the protection systems, the ESF, and their support systems, multiple failures resulting from a single event are treated as a single failure.

Criterion 22 (1967) - Separation of Protection and Control Instrumentation Systems

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

The protection systems comply with the requirements of IEEE-279, 1971, Criteria for Protection Systems for Nuclear Power Generating Stations, although construction permits for the DCPP units were issued prior to issuance of the 1971 version of the standard.

Each protection system is separate and distinct from the respective control systems. The control system is dependent on the protection system in that control signals are derived from protection system measurements, where applicable. These signals are transferred to the control system by isolation amplifiers that are classified as protection system components. The adequacy of system isolation has been verified by testing or analysis under conditions of all postulated credible faults. Isolation devices that serve to protect Instrument Class IA instrument loops have all been tested. For certain applications where the isolator is protecting an Instrument Class IB instrument loop, and the isolation device is a simple

linear device with no complex failure modes, the analysis was used to verify the adequacy of the isolation device. The failure or removal of any single control instrumentation system component or channel, or of those common to the control instrumentation system component or channel and protection circuitry, leaves intact a system that satisfies the requirements of the protection system.

Criterion 23 (1967) - Protection Against Multiple Disability of Protection Systems

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Physical separation and electrical isolation of redundant channels and subsystems, functional diversity of subsystems, and safe failure modes are employed in design of the reactors as defenses against functional failure through exposure to common causative factors. The redundant logic trains, reactor trip breakers, and ESF actuation devices are physically separated and electrically isolated. Physically separate channel trays, conduits, and penetrations are maintained upstream from the logic elements of each train.

The protection system components have been qualified by testing under extremes of the normal environment. In addition, components are tested and qualified according to individual requirements for the adverse environment specific to their location that might result from postulated accident conditions.

Criterion 26 (1967) - Protection Systems Fail-Safe Design

The reactor protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

The protection systems are designed with due consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip channel is designed on the de-energize-to-trip principle, so loss of power, disconnection, open channel faults, and the majority of internal channel short circuit faults cause the channel to go into its tripped mode. Additional defenses against loss of function are discussed under Criterion 23.

Criterion 37 (1967) - Engineered Safety Features Basis for Design

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Engineered safety features are provided to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends, and to cope with any steam or feedwater line break up to and including the main steam or feedwater headers.

Limiting the release of fission products from the reactor fuel is accomplished by the emergency core cooling system (ECCS) which, by cooling the core, keeps the fuel in place and substantially intact and limits the metal-water reaction to an acceptable amount. A reinforced concrete, steel-lined containment structure is provided and encloses the entire RCS. It is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the rupture of the largest pipe in the RCS.

Criterion 38 (1967) - Reliability and Testability of Engineered Safety Features

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

A comprehensive program of testing has been formulated for all equipment and instrumentation vital to the functioning of ESFs. The program consists of startup tests of system components and integrated tests of the system. Periodic tests of the activation circuitry and system components, throughout the station lifetime, with maintenance performed as necessary, ensure that the initially high reliability will be maintained and that the system will perform on demand. Details of the test program are provided in the Technical Specifications.

Criterion 41 (1967) - Engineered Safety Features Performance Capability

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

The overall capacity of the ESF meets the requirements of 10 CFR 100 for the occurrence of any rupture of a reactor coolant or steam system pipe, including the double-ended rupture of a reactor coolant pipe, known as the design basis accident (DBA).

Criterion 42 (1967) - Engineered Safety Features Components Capability

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss of coolant accident.

Instrumentation, motors, cables, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

The ECCS pipes serving each loop are anchored at the missile barrier in each loop area to restrict potential accident damage to the portion of piping beyond this point. The anchorage is designed to withstand, without failure, the thrust force exerted by any branch line severed from the reactor coolant pipe and discharging fluid to the atmosphere, and to withstand a bending moment equivalent to that producing failure of the piping under the action of a free-end discharge to atmosphere or motion of the broken reactor coolant pipe to which the emergency core cooling pipes are connected. This prevents possible failure at any point upstream from the support point, including the branch line connection into the piping header.

10 CFR 50.62

10 CFR 50.62 requires that pressurized water reactors have equipment diverse from the reactor protection system to initiate the AFW system under conditions indicative of an ATWS. The AFW system is required to assure adequate removal of heat from the RCS during an ATWS.

The worst common mode failure which is postulated to occur is the failure to scram the reactor after an anticipated transient has occurred. The effects of Anticipated Transient Without Scram (ATWS) event are not considered as part of the design basis for transients analyzed in FSARU Chapter 15. The final NRC ATWS rule requires that Westinghouse designed plants install Anticipated transient without scram mitigating system actuation circuitry (AMSAC) to initiate a turbine trip and actuate AFW flow independent of the Reactor Protection System. DCPP has AMSAC installed and the design is described in FSARU Section7.6.1.4.

There are no changes being proposed such that compliance with any of the regulatory requirements above would come into question. The evaluations documented above confirm that PG&E will continue to comply with all applicable regulatory requirements.

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.

4.2 Precedent

None.

4.3 Significant Hazards Consideration

PG&E has evaluated whether 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 change revises the requirements in Technical Specification (TS) 3.7.5, “Auxiliary Feedwater (AFW) System,” to provide a new Condition G to address a one-time planned Unit 1 Cycle 22 AFW system alignment for which one or two AFW trains are inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path to one

steam generator. The AFW System is not an initiator of any design basis accident or event, and therefore, the proposed change does not increase the probability of any accident previously evaluated. The AFW System is used to respond to accidents previously evaluated. The proposed change affects only the actions taken when portions of the AFW System are inoperable and does not affect the design of the AFW System. With the change to TS 3.7.5, adequate AFW cooling flow continues to be provided for accidents previously evaluated and there is no significant impact on accident consequences. No physical changes are made to the plant. The proposed change does not significantly change how the plant would mitigate an accident previously evaluated.

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

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

Response: No.

The proposed change does not result in a change in the manner in which the AFW System provides plant protection. The AFW System will continue to remain available to supply water to three of the four the steam generators while in the proposed TS 3.7.5 Condition G to remove decay heat and other residual heat by delivering at least the minimum required flow rate to provide adequate cooling. There are no design changes associated with the proposed changes. The changes to the Conditions and Required Actions do not change any existing accident scenarios, nor create any new or different accident scenarios. The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed). The required manual control of one or two AFW level control valves is not an accident initiator.

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

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

Response: No.

The proposed change does not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by this change. The proposed change will not result in plant operation in a

configuration outside the design basis since AFW cooling flow to two steam generators can provide adequate core cooling.

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

Based on the above evaluation, PG&E concludes that the proposed change does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

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

5. ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendment and has determined that the proposed amendment does not involve (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

None.

Proposed Technical Specification Changes (marked-up)

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

---NOTE---
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

--NOTE---
LCO 3.0.4b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Turbine driven AFW train inoperable due to one inoperable steam supply <u>OR</u> -----------NOTE--------------- Only applicable if MODE 2 has not been entered following refueling. ---------------------------------- Turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A or G.	B.1 Restore AFW train to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----------NOTE--------------- Only applicable when the remaining OPERABLE motor driven AFW train provides feedwater to the steam generator with the inoperable steam supply. ---------------------------------- Turbine driven AFW train inoperable due to one inoperable steam supply. <u>AND</u> One motor driven AFW train inoperable.	C.1 Restore the steam supply to the turbine driven train to OPERABLE status. <u>OR</u> C.2 Restore the motor driven AFW train to OPERABLE status.	48 hours 48 hours
D. Required Action and associated Completion Time for Condition A, B, ~~or~~ C, or G not met. <u>OR</u> Two AFW trains inoperable in MODE 1, 2 or 3 for reasons other than Condition C or G.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 18 hours
E. Three AFW trains inoperable in MODE 1, 2, or 3.	E.1 -----------NOTE--------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. ---------------------------------- Initiate action to restore one AFW train to OPERABLE status	Immediately
F. Required AFW train inoperable in MODE 4.	F.1 Initiate action to restore AFW train to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. -----------NOTE--------------- This Condition is only applicable to Unit 1 once during Unit 1 Cycle 22 during repair of AFW piping. --------------------------------- One or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path(s) to one steam generator.	G.1 Isolate AFW flow path(s) to affected steam generator.	2 hours
	AND	
	G.2 Restore AFW train(s) to OPERABLE status.	7 days

Proposed Technical Specification Changes (retyped)

Remove Page	Insert Page
3.7-10	**3.7-10**
3.7-11	**3.7-11**
	3.7-11a

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

---NOTE---
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

---NOTE---
LCO 3.0.4b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Turbine driven AFW train inoperable due to one inoperable steam supply <u>OR</u> -----------NOTE--------------- Only applicable if MODE 2 has not been entered following refueling. --------------------------------- Turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A or G.	B.1 Restore AFW train to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----------NOTE--------------- Only applicable when the remaining OPERABLE motor driven AFW train provides feedwater to the steam generator with the inoperable steam supply. --------------------------------- Turbine driven AFW train inoperable due to one inoperable steam supply. AND One motor driven AFW train inoperable.	C.1 Restore the steam supply to the turbine driven train to OPERABLE status. OR C.2 Restore the motor driven AFW train to OPERABLE status.	48 hours 48 hours
D. Required Action and associated Completion Time for Condition A, B, C, or G not met. OR Two AFW trains inoperable in MODE 1, 2 or 3 for reasons other than Condition C or G.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 18 hours
E. Three AFW trains inoperable in MODE 1, 2, or 3.	E.1 -----------NOTE--------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. --------------------------------- Initiate action to restore one AFW train to OPERABLE status	Immediately
F. Required AFW train inoperable in MODE 4.	F.1 Initiate action to restore AFW train to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. -----------NOTE--------------- This Condition is only applicable to Unit 1 once during Unit 1 Cycle 22 during repair of AFW piping. ---------------------------------- One or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable AFW piping affecting the AFW flow path(s) to one steam generator.	G.1 Isolate AFW flow path(s) to affected steam generator.	2 hours
	AND	
	G.2 Restore AFW train(s) to OPERABLE status.	7 days

Technical Specification Bases Page Markups
(For information only)

BASES

ACTIONS

A.1 (continued)

b. For the inoperability of a turbine driven AFW pump while in MODE 3 immediately subsequent to a refueling, the 7-day Completion Time is reasonable due to the minimal decay heat levels in this situation; and

c. For both the inoperability of a steam supply line to the turbine driven pump due to one inoperable steam supply and an inoperable turbine driven AFW pump while in MODE 3 immediately following a refueling outage, the 7-day Completion Time is reasonable due to the availability of redundant OPERABLE motor driven AFW pumps, and due to the low probability of an event requiring the use of the turbine driven AFW pump.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A or G, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

C.1 and C.2

With one of the required motor-driven AFW trains (pump or flow path) inoperable and the turbine-driven AFW train inoperable due to one inoperable steam supply, action must be taken to restore the affected equipment to OPERABLE status within 48 hours. Assuming no single active failures when in this condition, the accident (a FWLB or MSLB) could result in the loss of the remaining steam supply to the turbine-driven AFW pump due to the faulted SG.

A note in Condition C limits applicability to only when the remaining OPERABLE motor-driven AFW train provides feedwater to the SG with the inoperable steam supply. This Condition will only apply during the following two scenarios:

1) Motor-driven AFW pump 2 OPERABLE, motor-driven AFW pump 3 inoperable, and steam supply from SG 2 inoperable, or

2) Motor-driven AFW pump 2 inoperable, motor-driven AFW pump 3 OPERABLE, and steam supply from SG 3 inoperable.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2 (continued)

This ensures that if a FWLB were to occur affecting the OPERABLE motor driven AFW pump, the turbine-driven AFW pump would still be capable of providing AFW to two intact SGs. If a MSLB were to occur on the SG feeding the remaining OPERABLE steam supply to the turbine-driven AFW pump, the OPERABLE motor-driven AFW pump would still be capable of providing AFW to two intact SGs.

If motor-driven AFW pump 2 and the steam supply from SG 3 are inoperable, or if motor-driven AFW pump 3 and the steam supply from SG 2 are inoperable, then Condition D for two inoperable AFW pumps applies.

The 48 hour Completion Time is reasonable based on the fact that the remaining motor-driven AFW train is capable of providing 100 percent of the AFW flow requirements, and the low probability of an event occurring that would challenge the AFW system.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1 and D.2

When Required Action A.1, B.1, C.1, ~~or~~ C.2, G.1, or G.2 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3 for reasons other than Condition C or G, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

(continued)

BASES

ACTIONS (continued)

E.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no PG&E Design Class I means for conducting a cooldown, and only limited means for conducting a cooldown with non-PG&E Design Class I equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

F.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops-MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

G.1 and G.2

With one or two AFW trains inoperable in MODE 1, 2, or 3 due to inoperable Unit 1 AFW piping affecting the AFW flow path(s) to one steam generator, action must be taken to restore the affected train(s) to OPERABLE status within 7 days. Assuming no single active failures when in this condition, a FWLB could result in the loss of SG cooling in an additional SG due to operator action to isolate flow to the faulted SG.

A note in Condition G limits applicability only to Unit 1 once during Unit 1 Cycle 22 during repair of AFW piping. This condition is not applicable to Unit 2 or to Unit 1 at any other time.

Required Action G.1 to isolate AFW flow path(s) to the affected steam generator is performed by the operator by closing LCV-107 from the turbine driven pump to steam generator 1-2 and/or closing LCV-111 from the motor driven pump to steam generator 1-2 to isolate inoperable AFW piping. The 2 hour Completion time is reasonable based on the time required to complete the Required Action from the control room.

The Required Action G.2 7-day Completion Time to restore AFW train(s) to OPERABLE status is reasonable, based on the capabilities of the two motor driven AFW pumps and one turbine AFW pump to provide adequate AFW cooling flow, the time needed for Unit 1 AFW

repairs, and the low probability of a design basis accident occurring during this period.

When Required Action G.1 or G.2 cannot be completed within the required Completion Time, Condition D is entered to place the unit in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

(continued)