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PG&E Letter DCL-20-068

U.S. Nuclear Regulatory Commission
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10 CFR 50.91

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

References:

1. PG&E Letter DCL-20-066, "License Amendment Request 20-01, Exigent Request for Revision to Technical Specification 3.7.5, 'Auxiliary Feedwater System,'" dated August 12, 2020, ADAMS Accession No. ML20225A303

2. E-mail from NRC Senior Project Manager, Samson Lee, "Diablo Canyon request for additional information: Exigent License Amendment Request for Application to provide a new Technical Specification 3.7.5, 'Auxiliary Feedwater System,' Condition G (EPID: L-2020-LLA-0176)," dated August 14, 2020

Dear Commissioners and Staff:

In Reference 1, Pacific Gas and Electric Company (PG&E) submitted an exigent license amendment request to revise Technical Specification 3.7.5, "Auxiliary Feedwater System." In Reference 2, the NRC Staff provided a request for additional information (RAI) via an e-mail, dated August 14, 2020. The Enclosure to this letter provides the PG&E responses to the RAI.

This letter includes two new regulatory commitments (as defined by NEI 99-04), which are identified in Attachment 1 of the Enclosure.

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 16, 2020.

Paula Gerfen
Site Vice President

rntt/51084143-13
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

PG&E Response to NRC Request for Additional Information Regarding "License Amendment Request 20-01, Exigent Request for Revision to Technical Specification 3.7.5, 'Auxiliary Feedwater System'"

References:
1. PG&E Letter DCL-20-066, "License Amendment Request 20-01, Exigent Request for Revision to Technical Specification 3.7.5, 'Auxiliary Feedwater System,'" dated August 12, 2020 [ML20225A303]

2. E-mail from NRC Senior Project Manager, Samson Lee, "Diablo Canyon request for additional information: Exigent License Amendment Request for Application to provide a new Technical Specification 3.7.5, 'Auxiliary Feedwater System,' Condition G (EPID: L-2020-LLA-0176)," dated August 14, 2020

NRC RAI #1:

Corrosion: Please discuss:

a. Material type of the auxiliary feedwater (AFW) piping (e.g., carbon steel)
b. Degradation mechanism identified at Unit 2 (e.g., general corrosion)
c. Corrosion at Unit 2 was on the internal or external surfaces?
d. Corrosion at Unit 2 was at welds or piping?
e. Are there any differences between the AFW piping at Units 1 and 2 in terms of material type, corrosion protection (i.e., coatings), or environment (e.g., time of wetness, potential for contaminants such as chlorides to accelerate corrosion)?
f. Describe the locations of corrosion found in Unit 2 and plan for inspection in Unit 1.

PG&E Response:

a) The material type for the subject portions of the AFW piping is carbon steel. Specifically, the pipe is seamless carbon steel 3-inch Schedule 80 (0.300 inch nominal wall thickness) and fittings (e.g., elbows, tees, etc.) are wrought carbon steel.

b) The degradation mechanism is determined to be external corrosion exhibited in general and/or local outside metal loss patterns.

c) Using ultrasonic testing (UT) methods, the Diablo Canyon Power Plant (DCPP) qualified Inservice Inspection team verified there were no signs of internal thinning during the Unit 2 inspections. Based on these observations and system operating conditions, these portions of the AFW system are not susceptible to flow-accelerated corrosion. Therefore, with high confidence, the degradation mechanism is concluded to be external corrosion.

d) The corrosion found on Unit 2 AFW piping was observed to varying degrees on all external surfaces of pipe sections and fittings, including welds. However, areas exhibiting heavy local corrosion were found on pipe and fitting surfaces only. Welds were found to be distinct from surrounding pipe. For example, weld crowns and toes were readily identifiable.

e) Unit 1 and Unit 2 have no differences in material type or corrosion protection. Differences in environment are notable. Both Unit 1 and Unit 2 have two trains of AFW piping routed outdoors (supplying Steam Generators [SGs] 1 and 2 for each unit) that are exposed to the Pacific Coast marine environment. Unit 2 has a higher corrosive environment compared to Unit 1 due to localized weather patterns concentrating on the Unit 2 side. In addition, Unit 2 has historically experienced more forced outages and consequently operated its 10 percent atmospheric steam dumps more frequently. The steam dump valve exhaust, being located above the AFW piping, results in a wet environment due to falling condensation. The other two trains (supplying SGs 3 and 4 for each unit) are located indoors.

f) All locations that required weld repair on Unit 2 were found on piping downstream of Level Control Valves (LCVs)-111 and LCV-107. The plan for Unit 1 is to inspect those equivalent locations during power operations. A comprehensive extent-of-condition (EOC) inspection is planned to be performed on the remaining insulated outdoor AFW piping during the upcoming Unit 1 Twenty-Second Refueling Outage (1R22) in October 2020.

NRC RAI #2:

Please identify the AFW piping design Code and the process that will be followed for the repair consistent with the Code requirements. Is the one-to-one repair/replacement method being utilize?

PG&E Response:

The AFW piping design Code is ANSI B31.1-1967 with 1971 Addenda, with stress equations from the 1973 Summer Addenda.

The five pipe segments found on Unit 2 with below minimum wall thickness (seven locations total) were repaired with base metal weld buildup and verified with UT (for thickness), magnetic particle, and radiography to meet code requirements. Repairs were made per the Diablo Canyon Nuclear Welding Control Manual. Should repairs be required on Unit 1, the same repair method and standards will be used.

NRC RAI #3:

License amendment request (LAR), p.7 of 22, states that:

On July 23, 2020, with Diablo Canyon 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.

Describe the configuration of the valves and pumps, for example, which valve is downstream of Pump 2-1.

PG&E Response:

As depicted in Figure 1 provided on page 6 of the Exigent LAR (Reference 1), the Unit 1 piping configuration is as follows: LCV-107 is downstream of turbine-driven AFW Pump 1-1, and LCV-111 is downstream of motor-driven AFW Pump 1-2. The downstream piping from Unit 1 LCV-107 and LCV-111 merge together to provide a common AFW supply for SG 1-2.

Unit 2 maintains a similar piping configuration where LCV-107 is downstream of turbine-driven AFW Pump 2-1, and LCV-111 is downstream of motor-driven AFW Pump 2-2. The downstream piping from Unit 2 LCV-107 and LCV-111 merge together to provide a common AFW supply for SG 2-2.

Table 3-1 summarizes AFW flow while in proposed Technical Specification (TS) 3.7.5 Condition G.

Table 3-1 – Unit 1 AFW System Status While in Proposed TS 3.7.5 Condition G

AFW Pump (pump status)	Flow to SG 1-1	Flow to SG 1-2	Flow to SG 1-3	Flow to SG 1-4
Turbine-Driven AFW Pump 1-1 (available to provide flow to SG 1-1, 1-3, and 1-4)	Yes	No (isolated by LCV-107)	Yes	Yes
Motor Driven-AFW Pump 1-2 (available to provide flow to SG 1-1)	Yes	No (isolated by LCV-111)	No (pump normally aligned to SG 1-1 and 1-2)	No (pump normally aligned to SG 1-1 and 1-2)
Motor-Driven AFW Pump 1-3 (OPERABLE to provide flow to SG 1-3 and SG 1-4)	No (pump normally aligned to SG 1-3 and 1-4)	No (pump normally aligned to SG 1-3 and 1-4)	Yes	Yes
Number of AFW pumps capable of providing flow to SG while Condition G Required Actions are in effect	2 (Turbine-Driven AFW Pump 1-1 and Motor-Driven AFW Pump 1-2)	0	2 (Turbine-Driven AFW Pump 1-1 and Motor-Driven AFW Pump 1-3)	2 (Turbine-Driven AFW Pump 1-1 and Motor-Driven AFW Pump 1-3)

NRC RAI #4:

LAR, p.8 of 22, states that:

While in Condition G the steam generator (SG) 1-2 related technical specification (TS) required equipment will continue to remain operable.

Please explain what are the related TS required equipment.

PG&E Response:

Operability of the following SG TS-related equipment will not be impacted by the proposed TS 3.7.5 Condition G and its associated Required Actions. If any of these equipment/functions were to become inoperable, their TS Conditions and Required Actions would apply.

TS 3.3.1, “Reactor Trip System (RTS) Instrumentation,” Table 3.3.1-1:

- Function 14.a, SG Water Level—Low Low
- Function 14.b, SG Water Level – Low Low Trip Time Delay (TTD)

TS 3.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” Table 3.3.2-1:

- Function 5.b, Feedwater Isolation, SG Water Level-High High
- Function 6.d.1, AFW, SG Water Level-Low Low
- Function 6.d.2, AFW, SG Water Level-Low Low TTD

TS 3.3.3, “Post Accident Monitoring (PAM) Instrumentation,” Table 3.3.3-1:

- Function 2, Steam Line Pressure
- Function 13.a) SG Water Level (Wide Range)
- Function 13.b) SG Water Level (Narrow Range)

TS 3.3.4, “Remote Shutdown System,” Table 3.3.4-1:

- Function 6, SG Pressure
- Function 7, SG Level
- Function 8, AFW Flow

TS 3.4.4, “RCS Loops-MODES 1 and 2”

TS 3.4.5, “RCS Loops-MODE 3”

TS 3.4.6, “RCS Loops-MODE 4”

TS 3.4.17, “Steam Generator (SG) Tube Integrity”

NRC RAI #5:

LAR, p.9 of 22, states that

> *Conditions B and GD are modified to add new Condition G as a Condition for which an inoperable Condition is applicable. Please verify the statement.*

PG&E Response:

Page 9 should have stated “Conditions B and D are modified to add new Condition G as a Condition for which an inoperable Condition is applicable” consistent with the TS 3.7.5 changes described on pages 3 and 4 of the submittal (Reference 1).

NRC RAI #6:

LAR p.10 states that:

Loss of Normal Feedwater Transient. The condition 2 event of loss of normal feedwater is addressed in the Final Safety Analysis Report Update (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.

FSARU 6.5.3.7 states that:

A better-estimate analysis is performed to address the reliability of the AFW system. This analysis is similar to that described above for the Chapter 15 analysis, but assuming that only a single motor-driven AFW system pump *supplies a minimum of 390 gpm to two of the four SGs. The cases considered in this additional analysis assume better-estimate conditions for several key parameters, including initial power level, decay heat, reactor coolant system (RCS) temperature, pressurizer pressure, and low-low SG level reactor trip setpoint. The results of this better-estimate analysis demonstrate that there is margin to pressurizer over-filling. While this analysis demonstrates that the AFW system remains highly reliable, the Diablo Canyon licensing basis requires that at least two AFW pumps delivering at 600 gpm to four SGs is required for this event.*

The staff requests the licensee to explain the bases that a better-estimate analysis (two DGs receiving 390 gpm) can be used to cover a licensing based event (four SGs receiving 600 gpm).

PG&E Response:

The Condition II licensing basis events of loss of normal feedwater (LONF) and loss of AC power to station auxiliaries (LOAC) are analyzed in Updated Final Safety Analysis Report (UFSAR) Chapter 15.2 according to typical accident analysis methodology, crediting the worst single active failure. There is no single active failure that would result in a reduction from four SGs to just two SGs, therefore there is no licensing need to analyze that within the Chapter 15 UFSAR events. The current LAR-proposed Condition G is three SGs available with two pumps per SG available to provide flow for LONF and LOAC events (see Table 3-1 above).

Historically, it was recognized that the critical need for AFW reliability is such that it is advisable to demonstrate accident mitigation with just one of the three AFW pumps (specifically, with just one motor-driven AFW pump feeding two SGs). Condition III and IV events are therefore analyzed with cooling to just two SGs. In the past, the LONF and LOAC UFSAR analyses also only credited one motor-driven AFW pump feeding two SGs. The current Chapter 15 LONF/LOAC analyses were entered into the UFSAR to justify a wider replacement steam generator water level range and lower minimum AFW flow rate. At the same time, the Chapter 6 better-estimate analysis was done to continue supporting the risk-reliability concerns and demonstrate the acceptability of accident mitigation with just two SGs. This dual approach has been used at other Westinghouse sites, such as the Callaway Plant, Unit 1 - Issuance of Amendment Regarding the Steam Generator Replacement Project (TAC NO. MC4437), dated September 29, 2005 [ML052570054].

As the AFW system cooling capability for the proposed Condition G is bounded by the better-estimate analysis—there are two pumps per SG providing flow instead of just one, and there are three SGs available instead of just two SGs—the better-estimate analysis is used as justification for adequate mitigation of LONF and LOAC events for the proposed, one-time-use Condition G.

NRC RAI #7:

The licensee proposed a completion time (CT) of 7 days for TS 3.7.5 Condition G, “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,” for Unit 1 during repair of AFW piping. The proposed Condition is modified by a note which identifies that the condition is only applicable to Unit 1 once during Unit 1 Cycle 22 during repair of AFW piping.

In the Enclosure of the LAR on pages 12-13, the licensee provides a list of risk management actions (RMAs) the licensee will implement during the TS 3.7.5 Condition G 7-day CT. It appears that part of the justification for the proposed temporary CTs relies on the RMAs listed in the Enclosure.

Provide further justification for the proposed note language, which does not currently mention the RMAs. Alternatively, consider rewording the proposed note language to indicate that the 7 day CT is contingent on implementation of the RMAs listed in the LAR.

In addition, clarify whether the identified risk management actions will be required to be in place for the duration of the extended completion time.

PG&E Response:

The proposed TS 3.7.5 Condition G, and the associated Required Actions and Completion Times do not rely on the RMAs discussed in the LAR.

The LAR is not a risk-informed submittal, as would be submitted using the guidance of Regulatory Guides 1.174 or 1.177. However, a risk evaluation was performed in support of the proposed LAR, and based on the associated risk insights, several RMAs were identified that PG&E intends to implement in order to manage the risk in accordance with station procedures. These RMAs were included in the submittal to better inform the staff reviewers of the actions DCPP intends to take when entering the required actions of Condition G.

Accordingly, PG&E is making the following regulatory commitments as part of the proposed LAR:

A. DCPP will implement the RMAs contained in Table 9-1 in the response to NRC RAI #9 while the Required Actions for TS 3.7.5 Condition G are in effect for Unit 1.

B. Any changes to plant configuration that affect the RMAs established as part of Condition G will be evaluated and the risk will be managed in accordance with DCPP Procedures AD7.ID14, “Assessment of Integrated Risk,” and

AD7.DC6, “On-Line Maintenance Risk Management,” which implement the provisions of 10 CFR 50.65.

NRC RAI #8:

In the description of risk management actions, the LAR describes “protecting” certain equipment during the extended completion time. Clarify whether this includes preventing the protected equipment from being taken out of service for testing and maintenance activities.

PG&E Response:

No scheduled testing or planned maintenance will occur on equipment while protected. Plant administrative procedures and controls prevent employees from taking protected equipment out of service for testing or maintenance. A requirement to perform shiftly verification that no work is being performed behind posted protected equipment boundaries also exists. Emergent conditions that require work behind physical barriers to support continued plant operations result in further station risk evaluation before approval to proceed can be granted.

NRC RAI #9:

The LAR proposes risk management actions (RMAs) based on insights from the probabilistic risk assessment (PRA) to be implemented during proposed TS 3.7.5 Condition G. These RMAs support the availability of AFW, feed and bleed, and main feedwater. The NRC staff identified additional actions that could potentially support the availability of these systems/functions, which are in alignment with the proposed RMAs in the LAR. Provide a justification for not including the following actions as RMAs in the LAR. Include in this justification, a discussion of relevant risk information associated with the proposed change (e.g., change in risk contribution or importance measures) and defense-in-depth related to system redundancy, independence and diversity. Alternatively, include these actions as RMAs in the LAR.

- *Protect the supporting equipment of AFW Pumps 1-1 and 1-2,*
- *Protect supporting equipment of AFW values (e.g., power, air/nitrogen),*
- *Ensure the Power-Operated Relief Valves (PORV) block valves remain open to ensure feed-and-bleed availability,*
- *Protect the residual heat removal (RHR) pumps and centrifugal charging/intermediate head pumps to ensure feed-and-bleed availability,*
- *Protect RHR sump recirculation valves (and support systems) to ensure feed-and-bleed availability,*
- *Protect the steam-driven main feedwater pumps and the turbine to ensure feedwater availability.*

PG&E Response:

The following RMAs, which include the actions listed in this RAI (NRC RAI #9), will be implemented during the proposed TS 3.7.5 Condition G 7-day Completion Time and have been accepted by Operations:

- Protect Turbine-Driven AFW Pump 1-1 and supporting equipment. This supporting equipment includes vital 4 kV and 480 V Bus G, vital DC Bus 2, Battery Charger 1-2, and Emergency Diesel Generator 1-2.

- Protect Motor-Driven AFW Pump 1-2 and supporting equipment. This supporting equipment includes vital 4 kV and 480 V Bus H, vital DC Bus 3, Battery Charger 1-32, and Emergency Diesel Generator 1-1.

- Protect the supporting equipment for the AFW LCVs, which include the vital 480 V buses F, G, and H, 125 VDC buses 1, 2, and 3, and vital AC instrument channels I, II, III, and IV.

- Protect the 10 percent atmospheric dump valves and supporting equipment, including vital AC instrument channels I, II, III, and IV, instrument air supply system, nitrogen supply system, and the 10 percent atmospheric dump valve backup air bottles.

- Protect pressure-operated relief valves (PORV) block valves to ensure they remain open to ensure feed and bleed availability.

- Protect Residual Heat Removal (RHR) Pump 1-1, RHR Pump 1-2, Charging Pump 1-1, Charging Pump 1-2, Safety Injection Pump 1-1, and Safety Injection Pump 1-2.

- Protect the RHR sump recirculation valves, which include: CCW-1-FCV-364, CCW-1-FCV-365, RHR-1-8700A, RHR-1-8700B, RHR-1-8982A, and RHR-1-8982B. Also protect the support systems for these valves, which include vital 480 V buses G and H, 125 VDC buses 2 and 3, and instrument air.

- Protect the Feedwater Pumps 1-1, 1-2, and the turbine to ensure feedwater availability.

These RMAs are in addition to those documented in the Risk Insights section of the LAR submittal.

Table 9-1 documents the RMAs that will be implemented.

Table 9-1 – Risk Management Actions (RMAs)

Risk Management Actions (RMAs)	
Equipment-Related RMAs	
1	Protect Turbine-Driven AFW Pump 1-1 and supporting equipment. This supporting equipment includes vital 4 kV and 480 V Bus G, vital DC Bus 2, Battery Charger 1-2, and Emergency Diesel Generator 1-2.
2	Protect Motor-Driven AFW Pump 1-2 and supporting equipment. This supporting equipment includes vital 4 kV and 480 V Bus H, vital DC Bus 3, Battery Charger 1-32, and Emergency Diesel Generator 1-1.
3	Protect Motor-Driven 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.
4	Protect the remaining AFW LCVs (LCV-106, LCV-108, LCV-109, LCV-110, LCV-112, and LCV-113) locally.
5	Protect Pressurizer PORV PCV-455C and PCV-456.
6	Protect the supporting equipment for the AFW valves, which include the vital 480 V buses F, G, and H, 125 VDC buses 1, 2, and 3, and vital AC instrument channels I, II, III, and IV.
7	Protect the 10 percent atmospheric dump valves and supporting equipment, including vital AC instrument channels I, II, III, and IV, instrument air supply system, nitrogen supply system, and the 10 percent atmospheric dump valve backup air bottles.
8	Protect PORV block valves to ensure they remain open to ensure feed and bleed availability.
9	Protect RHR Pump 1-1, RHR Pump 1-2, Charging Pump 1-1, Charging Pump 1-2, Safety Injection Pump 1-1, and Safety Injection Pump 1-2.
10	Protect the RHR sump recirculation valves, which include: CCW-1-FCV-364, CCW-1-FCV-365, RHR-1-8700A, RHR-1-8700B, RHR-1-8982A, and RHR-1-8982B. Also protect the support systems for these valves, which include vital 480 V buses G and H, 125 VDC buses 2 and 3, and instrument air.
11	Protect the Main Feedwater Pumps 1-1 and 1-2 and the turbine to ensure feedwater availability.
Procedure-Related RMAs	
12	Shiftly tailboard of 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.
13	Shiftly tailboard on feed-and-bleed cooling (including RHR sump recirculation) on a loss of AFW and Main Feedwater.

NRC RAI #10:

The leak and degraded conditions on the Unit 2 AFW system were discovered on July 23rd. The staff is aware that the licensee has performed initial walkdown of Unit 1 AFW piping. The licensee had originally scheduled the Unit 1 AFW extent of condition inspection for the week of August 10th. The staff is aware that the current plan is to perform inspection the week of August 24th. Please discuss, from a safety perspective, why it is acceptable to wait two additional weeks, approximately one month after the Unit 2 leak, was identified?

PG&E Response:

Operators walked down the Unit 1 AFW piping on the pipe rack, outside of Containment on the evening of July 25, 2020. The same piping was subsequently walked down the next day, on July 26, 2020, by engineers from the Engineering Fix It Now group who are trained and proficient in recognizing and evaluating degraded piping. Neither walkdown identified any leaks or areas of immediate concern.

Engineering documented the basis for the timing of the Unit 1 EOC activities for the Unit 1 AFW system piping in the corrective action program (CAP) on July 26, 2020. This basis credits the lower environmentally-induced corrosion impacts that exist on Unit 1 and the lower corrosive impact from localized wetting from 10 percent atmospheric steam dump operation, combined with the results of the above discussed walkdowns, the recognition that corrosion rates are slow (2 to 8 mils per year), and subsequent detailed analysis that demonstrates a further reduced minimum acceptable wall thickness over that applied to the Unit 2 pipe conditions. Together, these form the basis of confidence in potential Unit 1 AFW corrosion being less extensive and support continued operation of Unit 1 in the short time frame (based on corrosion rates) to develop and implement an inspection plan.

Although the basis for the timing of the Unit 1 EOC can be applied to potentially extend the detailed EOC inspection to the 1R22 outage, DCPP plans to inspect the same five pipe segments on Unit 1 that were below minimum wall thickness acceptance limits on Unit 2 to validate the basis, which is consistent with our CAP methodologies for EOC. These five pipe segments are viewed to represent the most vulnerable Unit 1 conditions (three elbows and two straight pipe segments). These inspections are planned with established contingencies, including pipe base metal repair, and will be implemented as soon as practicable after approval of this exigent LAR when plant conditions allow, and all the risk management controls can be implemented.

Regulatory Commitments

A. Diablo Canyon Power Plant (DCPP) will implement the risk management actions (RMAs) contained in Table 9-1 in the response to NRC RAI #9 while the Required Actions for TS 3.7.5 Condition G are in effect for Unit 1.

B. Any changes to plant configuration that affect the RMAs established as part of Condition G will be evaluated and the risk will be managed in accordance with DCPP procedures AD7.ID14, "Assessment of Integrated Risk," and AD7.DC6, "On-Line Maintenance Risk Management," which implement the provisions of 10 CFR 50.65.

Table 9-1 – Risk Management Actions

Risk Management Actions (RMAs)	
Equipment-Related RMAs	
1	Protect Turbine-Driven Auxiliary Feedwater (AFW) Pump 1-1 and supporting equipment. This supporting equipment includes vital 4 kV and 480 V Bus G, vital DC Bus 2, Battery Charger 1-2, and Emergency Diesel Generator 1-2.
2	Protect Motor-Driven AFW Pump 1-2 and supporting equipment. This supporting equipment includes vital 4 kV and 480 V Bus H, vital DC Bus 3, Battery Charger 1-32, and Emergency Diesel Generator 1-1.
3	Protect Motor-Driven 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.
4	Protect the remaining AFW level control valves (LCVs) (LCV-106, LCV-108, LCV-109, LCV-110, LCV-112, and LCV-113) locally.
5	Protect Pressurizer Power-Operated Relief Valves (PORVs) PCV-455C and PCV-456.
6	Protect the supporting equipment for the AFW valves, which include the vital 480 V buses F, G, and H, 125 VDC buses 1, 2, and 3, and vital AC instrument channels I, II, III, and IV.
7	Protect the 10 percent atmospheric dump valves and supporting equipment, including vital AC instrument channels I, II, III, and IV, instrument air supply system, nitrogen supply system, and the 10 percent atmospheric dump valve backup air bottles.
8	Protect PORV block valves to ensure they remain open to ensure feed and bleed availability.
9	Protect Residual Heat Removal (RHR) Pump 1-1, RHR Pump 1-2, Charging Pump 1-1, Charging Pump 1-2, Safety Injection Pump 1-1, and Safety Injection Pump 1-2.
10	Protect the RHR sump recirculation valves, which include: CCW-1-FCV-364, CCW-1-FCV-365, RHR-1-8700A, RHR-1-8700B, RHR-1-8982A, and RHR-1-8982B. Also protect the support systems for these valves which include vital 480 V buses G and H, 125 VDC buses 2 and 3 and instrument air.
11	Protect the Main Feedwater Pumps 1-1 and 1-2 and the turbine to ensure feedwater availability.
Procedure-Related RMAs	
12	Shiftly tailboard of Emergency Operating Procedure FR-H.1, "Response to Loss of Secondary Heat Sink," on providing Main Feedwater to the steam generators on a loss of AFW.
13	Shiftly tailboard on feed-and-bleed cooling (including RHR sump recirculation) on a loss of AFW and Main Feedwater.

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