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PG&E Letter DCL-04-097

U.S. Nuclear Regulatory Commission
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Docket No. 50-323, OL-DPR-82
Diablo Canyon Unit 2
Response to NRC Request for Additional Information Regarding Emergency License Amendment Request 04-03, "Revision to Technical Specification 3.6.6, 'Containment Spray and Cooling Systems'"

Pacific Gas and Electric (PG&E) Letter DCL-04-096, dated July 30, 2004, submitted Emergency License Amendment Request (LAR) 04-03, "Revision to Technical Specification 3.6.6, 'Containment Spray and Cooling Systems.'" LAR 04-03 requested a one-time change to the Completion Time of Required Action A.1 of Technical Specification (TS) 3.6.6, "Containment Spray and Cooling Systems," to increase the Completion Time for the Unit 2 Containment Spray Pump 2-2 during Unit 2 cycle 12, from 72 hours to 14 days.

On July 30, 2004, the NRC staff requested additional information required to complete the review of LAR 04-03. PG&E's responses to the staff's questions are provided in Enclosure 1.

This information does not affect the results of the technical evaluation or the no significant hazards consideration determination previously transmitted in PG&E Letter DCL-04-096.

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

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

Executed on July 30, 2004.

James R. Becker
Vice President - Diablo Canyon Operations and Station Director

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Enclosures
cc: Edgar Bailey, DHS
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PG&E Response to Request for Additional Information Regarding Emergency License Amendment Request (LAR) 04-03, "Revision to Technical Specification 3.6.6, 'Containment Spray and Cooling Systems'"

NRC Question 1

Provide additional justification of the basis and the nature of the emergency, the basis for the need to extend the Completion Time (CT) for Condition A for Technical Specification (TS) 3.6.6, "Containment Spray and Cooling Systems," to 14 days, and discuss the schedule for completing the repairs.

PG&E Response

The original ground in the control circuit cable for Containment Spray Pump (CSP) 2-2 was identified on conductor 2-1 on June 4, 2004. This was the only problem noted at the time. Since this conductor provides indication only, and no breaker control functions, CSP 2-2 was determined to be operable. However, a temporary modification was designed to bypass the grounded conductor. On July 28, 2004 at 0525 PDT, the Unit 2 Shift Foreman declared CSP 2-2 inoperable to install the temporary modification to bypass conductor 2-1, so it could be removed from service. During post-maintenance testing of the temporary modification, additional conductors in the nine-conductor cable were identified as having grounds. The newly identified grounded conductors provide breaker control functions. As a result, the CSP 2-2 could not be returned to operable status. Thus, CSP 2-2 remains in Condition A 72-hour CT of TS 3.6.6.

Upon discovery of the additional grounds, a troubleshooting team was assigned to coordinate and conduct the investigation. The focus of the investigation has been to identify a specific location in the approximately 550 foot length of the conduit where cable degradation is evident. Late on July 29, 2004, a conduit pull box was found with water inside. Boroscope inspection of the affected conduit identified a section of the conduit full of water. The water was drained and the circuits measured for grounds. Removal of the water resulted in the ground readings improving.

The primary success path is to replace the entire 550 foot cable. The estimated completion time for installing the new cable is Wednesday, August 4, 2004, and completion of post-maintenance testing on Thursday, August 5, 2004. This worst-case schedule is considered realistic, and is to be worked on a 24 hours per day basis. This schedule requires a total completion time of approximately nine days. Additional margin is needed to allow for rework and retesting, if needed. Therefore, LAR 04-03 requests 14 days. Shorter repair paths may become available, e.g., splicing the original cable, but none are expected to allow for completion and post-maintenance testing within the current 72-hour CT required by TS 3.6.6, Condition A.

NRC Question 2

The probabilistic risk assessment (PRA) discussed in LAR 04-03 makes the assumption that only CSP 2-2 is inoperable. Discuss how PG&E has determined that no other safety-related systems, or components are inoperable.

PG&E Response

In the same conduit that houses the grounded CSP 2-2 125 vDC control circuits, there are other vital bus H equipment circuits associated with:

Auxiliary Feedwater Pump 2-2
Component Cooling Water Pump 2-3
Safety Injection Pump 2-2
Diesel Generator 2-2
Supply Fan 2S-67

The location inside the conduit where the ground fault of the CSP 2-2 circuit was determined likely to be was boroscope-inspected late on July, 29, 2004. Water drained out of the conduit when the conduit plug was removed. The inspection could not identify any obvious physical damage to the conduits or the circuits.

For the grounded CSP 2-2 circuits, all conductors were re-measured for grounds and 3 of the 9 conductors were found to have greater than a 4 meg-ohm reading which is acceptable (acceptance criteria is greater than 2 meg-ohm at 500v). This re-measurement took place after some water was drained out of the conduit and led to the conclusion that the water inside the conduit is likely a major factor in causing the ground to the CSP 2-2 circuit.

After the water was removed, no ground indications were identified in subsequent measurement of any of the safety-related circuits in the affected conduit.

Based on the extent of condition investigation to date, PG&E is confident that the other equipment whose circuits are routed in the conduit with the CSP 2-2 grounded cable remain capable of performing their safety functions.

NRC Question 3

What truncation values are used in the PRA model? Why is the PRA-calculated impact on risk so small?

PG&E Response

The Diablo Canyon Power Plant (DCPP) PRA model uses a large event tree/small fault tree approach. No truncation values or very low truncation values (in the 1E-15 range) are used for the fault tree quantification. The event tree quantification uses truncation values between 1.0E-11 to 1.0E-15.

The PRA uses the core damage frequency (CDF) and large early release on frequency (LERF) figures of merit to characterize the risk significance. The PRA results indicate the Containment Spray (CS) system has minimal impact on these figures of merit. The basis for this assertion is provided below. Additionally, it should be noted that the current delta CDF reported by DCPP is conservative because it shows an increase in CDF due to the unavailability of one train of the CS system. In a more detailed analysis, the containment spray system unavailability would be expected to reduce the CDF. The CDF for events with CS operation is higher due to the following:

- For large and medium loss-of-coolant accident (LOCA) events, actuation of containment spray would decrease the time at which recirculation would be required. This shorter time window increases the human error probability.
- For accident sequences involving feed and bleed cooling, containment spray actuation and recirculation, the recovery actions become more complicated and the time to recirculation would decrease, As above, the reduction in time to recirculation would result in a higher human error probability.

The design purposes of the CS system are:

- To remove heat from the containment atmosphere following a LOCA or main steam line break (MSLB).
- To remove fission products from the containment atmosphere following a LOCA.
- To deliver sufficient sodium hydroxide solution to ensure a proper Containment sump pH.

Isolating one of the CS trains will reduce the ability of the system to perform the above-mentioned functions. However, this decrease in the reliability of the CS system does not have a direct impact on the CDF or the LERF figures of merit on the basis that:

- None of the above functions are credited for mitigating the consequences of an accident in the PRA model.
- The PRA model does not credit any of the above functions for the success of its accident mitigating functions (e.g., containment sump pH is not a factor in the effectiveness of the recirculation function)

- The LERF model does not credit the fission-product-removal function of the CS system.
- The LERF model does not credit the CS system for reducing the LERF figure of merit contribution from potential severe accident phenomena such as high pressure melt ejection or hydrogen burns. These events are not expected to be a significant threat to containment because:

    - The containment internal structure design and the large total volume.
    - The pressure generated by the hydrogen burn phenomenon is unlikely to challenge DCPP containment, which has a large total volume, until additional hydrogen is generated from core-concrete interaction. Significant core-concrete interaction does not occur within four hours of reactor vessel beach. Therefore, hydrogen burn as a contributor to LERF can be ignored.

- The DCPP LERF figure of merit, consistent with the other industry PRA studies, is dominated by the steam generator tube rupture event (as an initiator or as a consequential event) and the interfacing system LOCA event. Neither of these major initiators are affected by the decrease in the reliability of the CS system.

NRC Question 4

Discuss how the proposed change impacts defense-in-depth and adequacy of safety margins following the proposed change.

PG&E Response

The technical evaluation performed in support of this LAR is not intended to demonstrate that DCPP can meet the design basis requirements without the CS function. This is neither possible nor necessary since by definition not having the CS function available is outside the DCPP design basis. This LAR addresses the CT associated with having one of two CS pumps inoperable. DCPP still meets the design basis requirements for CS, since a single failure is not required to be postulated for design basis events during the Technical Specification CT, and the DCPP design basis only requires one CS pump operating to meet the applicable analysis acceptance criteria.

This evaluation only provides technical input in support of the PRA assessment of the relative increase in risk associated with extending the CS pump CT. The evaluation focused on whether not having the CS system available could cause a significant increase in the key analysis parameters with respect to the current design basis limits, and whether this represents a credible source for significant degradation of design basis equipment, including containment. This evaluation assessed the potential for degradation to design basis equipment was done with

respect to whether additional equipment failure scenarios should be included in the PRA risk assessment model for the CT extension.

The LOCA design basis frequency is insignificant with respect to this CT time extension, so the technical evaluation focused on the MSLB event inside containment. A GOTHIC computer model was used to evaluate the relative increase in the containment pressure and temperature for a limiting MSLB event without CS available. The MSLB case was performed with the same conservative assumptions and methodology as the FSAR Chapter 6 analyses, except there was no credit taken for any CS flow. These results were compared to the current design basis results and analysis limits for containment integrity and equipment qualification. The results showed that the magnitude and duration of the peak pressure and temperature inside containment which exceeded the current design basis limits were relatively minor with respect to the safety margins inherent in the design of containment and other safety equipment. Similarly, there was no significant increase in the MSLB long term temperature profile when compared to the design basis thermal aging profile for environmentally qualified equipment.