UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 E. LAMAR BLVD
ARLINGTON, TX 76011-4511

February 28, 2017

EA-16-168

Mr. Edward D. Halpin, Senior Vice President,
Generation and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56, Mail Code 104/6
Avila Beach, CA 93424

SUBJECT: NRC DENIAL OF PACIFIC GAS AND ELECTRIC COMPANY APPEAL OF DIABLO CANYON POWER PLANT - FINAL SIGNIFICANCE DETERMINATION OF A WHITE FINDING; NRC INSPECTION REPORT 05000275/2016010 AND 05000323/2016010

Dear Mr. Halpin:

On December 28, 2016, the U.S. Nuclear Regulatory Commission (NRC) provided Pacific Gas and Electric (PG&E) Company the final significance determination for a finding of low-to-moderate significance (White) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16363A429). This finding involved the failure of the Diablo Canyon Unit 2 residual heat removal pump 2-2 suction valve from the containment recirculation sump to open from the main control room. In that same letter, the NRC stated that PG&E had 30 calendar days to appeal the NRC's final significance determination. The letter also stated that such an appeal would be considered to have merit only if it met the criteria given in Inspection Manual Chapter (IMC) 0609, Attachment 2, "Process for Appealing NRC Characterization of Inspection Findings (SDP Appeal Process)" (ADAMS Accession No. ML101400502).

By letter dated January 26, 2017, PG&E agreed with the NRC's characterization of the matter as a violation, but appealed the NRC's characterization of the final significance determination as White (ADAMS Accession No. ML17026A485). PG&E submitted six individual appeal bases related to the NRC's final significance determination. Specifically, your staff stated that the NRC's final significance determination was inconsistent with applicable significance determination process (SDP) guidance or lacked justification. Your appeal also stated that actual (verifiable) plant hardware, procedures, or equipment configurations, identified by your staff to the NRC at the Regulatory Conference or in writing prior to the final significance determination, was not considered by the NRC staff. In your letter, you state that using the best available information, the finding results in an increase in core damage frequency of less than 6.0E-07 per year. Based on these results, your staff believes the finding should be characterized as having very low safety significance (Green).

This letter provides you the NRC's response to your January 26, 2017, appeal. The NRC concluded that PG&E's appeal did not have sufficient merit for review by the appeal process specified in IMC 0609, Attachment 2. Specifically, the NRC found that your contentions do not meet the limitations specified in IMC 0609.02, Section 03, "Limitations." Our evaluation of each individual appeal basis submitted by PG&E is provided in the attachment to this letter.

Accordingly, the NRC concluded that the final significance determination for the finding of low-to-moderate safety significance, documented in NRC Inspection Report 05000275/2016010 and 05000323/2016010, is unchanged and, based on your previous response to the Notice of Violation, no further response is required. If you have any questions on our conclusions on the merit of your appeal, please contact Mr. Troy W. Pruett, Director, Division of Reactor Projects, at (817) 200-1248, or Mr. Jeremy R. Groom, Chief, Reactor Projects Branch A, at (817) 200-1148.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room and in ADAMS, accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Sincerely,

/RA/

Kriss M. Kennedy
Regional Administrator

Docket Nos. 50-275 and 50-323
License Nos. DPR-80 and DPR-82

Attachment:
NRC Response to Final Significance Determination Appeal

NRC RESPONSE TO PACIFIC GAS AND ELECTRIC COMPANY APPEAL OF DIABLO CANYON POWER PLANT - FINAL SIGNIFICANCE DETERMINATION OF A WHITE FINDING; NRC INSPECTION REPORT 05000275/2016010 AND 05000323/2016010 – DATED FEBRUARY 28, 2017

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ADAMS ACCESSION NUMBER: ML17059D758

☒ SUNSI Review By: JRG/dll	ADAMS ☒ Yes ☐ No	☒ Non-Sensitive ☐ Sensitive	☒ Publicly Available ☐ Non-Publicly Available	Keyword: NRC-002

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NRC RESPONSE TO FINAL SIGNIFANCE DETERMINATION APPEAL (PG&E LETTER DCL-17-009 DATED JANUARY 26, 2017)

On January 26, 2017, PG&E submitted an appeal (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17026A485) of the NRC staff's final significance determination (FSD) associated with NRC Inspection Report 05000275/2016010 and 05000323/2016010, "Diablo Canyon Power Plant - Final Significance Determination of a White Finding, Notice of Violation and Follow-up Assessment Letter," dated December 28, 2016 (ADAMS Accession No. ML16363A429).

The NRC determined that for the finding documented in NRC Inspection Report 05000275/2016010 and 05000323/2016010, the prerequisites specified in IMC 0609, Attachment 2, "Process for Appealing NRC Characterization of Inspection Findings (SDP Appeal Process)", dated June 8, 2011, Section 0609.02-02, were met. Accordingly, the NRC reviewed the licensee's contentions to determine if they met one or more of the appealable categories specified in IMC 0609.02-03. The following provide the results of the NRC's evaluation of the licensee's appeal.

1. **PG&E Appeal Basis No. 1: Use of Averaged NUREG-1829 Data.**

   In Appeal Basis No. 1, PG&E asserts that the use of averaged 25-year and 40-year loss-of-coolant-accident (LOCA) frequency data from NUREG-1829, "Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process," April 2008 is not supported by applicable guidance.

   NRC Response to Basis No. 1:

   The NRC concluded that PG&E's Appeal Basis No. 1 does not have sufficient merit for review by the appeal process specified in IMC 0609, Attachment 2, because the contention does not meet the limitations specified in IMC 0609.02, Section 03.a. Specifically, the NRC followed the applicable guidance in IMC 0308, Appendix A, "Technical Basis for the At-Power Significance Determination Process," dated July 1, 2012, and IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, when using averaged 25-year and 40-year LOCA frequency data from NUREG-1829 in the FSD result. IMC 0308, Appendix A states that:

   > "The SPAR models consist of a set of plant-specific Probabilistic Risk Assessment (PRA) models that employ a standard approach for event-tree and fault-tree development as well as a standard approach for input data for initiating event frequencies, equipment performance, and human performance. These input data can be modified to be more plant- and event-specific when needed."

   Additionally, IMC 0609, Appendix A, states that, "To support the SDP timeliness goal, a senior reactor analyst (SRA) may make changes to the Standardized Plant Analysis Risk (SPAR) model of record, as appropriate, based on information from the inspectors and/or the licensee, to accurately reflect the risk significance of the finding. These changes must be documented in the associated inspection report and/or Significance and Enforcement Review Panel (SERP) package."

The NRC did make an appropriate change to the SPAR model, as documented in the FSD, Element 3, Pages A-1 and A-2,

> "When evaluating the lower range of the increase in core damage frequency for the FSD, the NRC updated the method of deconstructing medium LOCAs using a logarithmic-linear function method.
>
> Through the process of reviewing NUREG-1829, the NRC identified that initiating event frequencies for LOCAs in the NRC SPAR model were derived from the 25-year fleet average operations life when NUREG-1829 was published in 2008. The NRC averaged the 25-year and 40-year LOCA frequencies to obtain frequencies with a more accurate reflection of fleet average operations. The NRC discussed the fact that the LOCA initiating event frequencies in the SPAR model were dated at the regulatory conference.
>
> The application of these methods resulted in a reduction in the break frequency for medium LOCAs between 4.5 to 6 inches to 5.21E-6 per year."

While it is correct that NUREG-1829 does recommend the use of the 25-year data for 15 years (until average fleet life reaches 40 years), the NUREG recommendation is based on a projection that LOCA "mitigation procedures are in place, or will be implemented in a timely manner, to alleviate significant increases in future LOCA frequencies from existing degradation mechanisms." During development of the FSD, the NRC contacted a subject matter expert who was an original preparer for NUREG-1829, who provided insights into how the LOCA frequency data in the NUREG was developed and applied. This individual indicated that the SPAR model could be updated with 40-year values and that those LOCA frequencies were appropriate for use in the significance determination process. Personnel from the NRC's Office of Research and staff from Idaho National Lab also provided insights that NUREG-1829 was created based on work performed in 2004 and generally should be used as the basis for LOCA frequencies but updated with operational experience for the years following issuance of the report.

During development of the FSD, the 2016 fleet age was approximately 37 years. With the fleet age approaching 40 years and uncertainty as to whether the mitigation implementation met the expectations of the expert panel from NUREG-1829, the analyst applied the averaging assumption used in the final detailed risk evaluation rather than strictly applying the 40-year data. The analyst also noted that a pending update to the SPAR model, scheduled for early 2017, will use the 40-year data for LOCA frequencies. By averaging the 25-year and 40-year LOCA frequencies, the calculated increase in core damage frequency (CDF) used in the FSD was less than what the NRC could have calculated using 40-year LOCA frequencies.

2. **PG&E Appeal Basis No. 2: Proceduralized Actions to Assess Time to Core Damage. "The FSD uses 2.8 hours for the time between 4 percent RWST level and core damage."**

   In Appeal Basis No. 2, PG&E asserts that the FSD uses 2.8 hours for the time between 4 percent refueling water storage tank (RWST) level and core damage, and this timeline does not reflect implementation of actual plant procedures to makeup from the volume control tank (VCT) to the reactor coolant system (RCS).

NRC Response to Basis No. 2:

The NRC concluded that PG&E's Appeal Basis No. 2 does not have sufficient merit for review by the appeal process specified in IMC 0609, Attachment 2, because the contention does not meet the limitations specified in IMC 0609.02, Section 03.b. Specifically, the NRC staff did consider proceduralized actions to assess time to core damage including implementation of actual plant procedures to makeup from the VCT to the RCS.

During development of the FSD, the NRC recognized and considered that recovery of valve SI-2-8982 included at least four different strategies to either slow the drain rate on the RWST, or add additional inventory to the RWST or RCS to prolong the available recovery time. The NRC's evaluation included consideration of actual plant procedures to makeup from the VCT to the RCS, as specified in emergency operating Procedure (EOP) ECA-1.1, "Loss of Emergency Coolant Recirculation," Revision 21, Appendix W. The success of these strategies, coupled with additional information including LOCA size and location, dictates the available time for implementation of the various recovery options including the mechanical, electrical contactor, and electrical jumper recovery actions.

Because of the complexity of the multiple recovery actions considered and the specific influence that certain actions have on effective human error probabilities, the NRC simplified the analysis by dynamically evaluating the available time to core damage, taking into account the changing plant conditions expected as the post-LOCA sequence progressed. At the time a specific recovery action was assumed to be called for, the NRC compared the estimated recovery action time to the dynamically evaluated available time. The NRC considered available time to be based on remaining RWST inventory, effective RWST drain rate and time to fuel heat up following cessation of emergency core cooling system (ECCS) flow. This resulted in the NRC concluding that the available time would be less than five times that required for the human performance basic events. Consequently, the NRC judged the "Time Available" performance shaping factor (PSF) as "Nominal," consistent with the guidance in NUREG/CR-6883, "The SPAR-H Human Reliability Analysis Method," August 2005.

Additionally, as documented in the FSD, Elements 7 and 8, Pages A-4 and A-5 respectively, the NRC considered the guidance in INL/EXT-10-18533, "SPAR-H Step-by-Step Guidance," Revision 2, Section 3.1, to develop the assumptions related to the available time PSF for the mechanical and electrical contactor recovery actions. The NRC, when developing the final composite human error probability, recognized that extremely optimistic assumptions could yield extra time to recover valve SI-2-8982B, while assumptions that are more realistic would yield considerably less time. As discussed in Section 3.1 of INL/EXT-10-18533, the NRC was cautious against relying on overly precise estimates that lead to threshold effects for the Available Time PSF. However, the NRC concluded that even using the most optimistic of scenarios, the available time is very close to five times that required for the human performance basic events. As such, the NRC determined it was appropriate, as discussed in Section 3.1, "Available Time," of INL/EXT-10-18533, to assume the "Time Available" PSF is "Nominal" for the action portion of the mechanical and electrical recovery options with the remainder of time assigned to diagnosis. Based on the contention in your

January 26, 2017, letter, the NRC staff re-confirmed with staff in the NRC's Office of Research that this splitting of time between the action and diagnosis PSF is the correct implementation of the SPAR-H Step-by-Step guidance.

For those recoveries that occur following the initial failure of the mechanical and electrical contactor recovery actions, and as discussed in the FSD, Element 9, Page A-7, and our January 17, 2017, letter (ADAMS Accession Number ML17017A437), the NRC did consider normal makeup to the RCS from normal charging when evaluating additional technical support center and emergency response organization (TSC/ERO) directed recoveries. It is important to note that the NRC does recognize that the TSC will be involved with all elements of recovery. However, the term "TSC directed recoveries" (see FSD, Element 9, Page A-7) was not used to specifically assess the mechanical and electrical contactor local recovery actions because those recoveries are directed by Procedure OP O-9, "Manual 'Seating' of Motor Operated Valve," Revision 21A and OP O-22, "Emergency Operation of Motor Operated Valves," Revision 6. Instead, when evaluating the composite human error probability for recovery, the term "TSC directed recoveries," was used to refer to those recovery actions that would be developed by the TSC following the initial failure of the mechanical and electrical contactor recovery options. This includes the electrical jumper recovery method and any subsequent electrical and mechanical recoveries implemented after failures to recover valve SI-2-8982B through use of Procedures OP O-9 or OP O-22.

The NRC applied the additional time, assumed to be gained through implementation of makeup from the VCT to the RCS, when evaluating the human error probability of TSC/ERO directed recoveries. However, this additional time did not have a meaningful effect because high dependency dominated the human error evaluation. As documented in the FSD, the NRC based its assumption that high dependency was appropriate because the same crew would be used, the tasks would be close in time, and no additional cues would be present. Because the timeline of the event would be limited to a single ERO crew, the NRC assumed that the same electrical repair crew that performed the electrical contactor recovery actions would perform the electrical jumper recovery actions. Additionally, the NRC assumed the same control room crew that performed the actions to throttle ECCS, cooldown and depressurize the RCS and refill the RWST, would be used to initiate makeup from the VCT to the RCS. The NRC recognizes that this simplification results in a slight underestimation of the failure probability of these actions. However, because of the relatively insignificant contribution to the overall composite human error probability for recovery of valve SI-2-8982B (approximately 5E-3), this simplification has little effect on the outcome of the FSD.

3. PG&E Appeal Basis No. 3: Reference to Superseded Procedure When Assessing Operator Actions, Performance Shaping Factors, Timing of Recovery Actions, Sequence of Recovery Actions, and Dependency.

In Appeal Basis No. 3, PG&E asserts that the FSD references a superseded procedure and does not reflect actual plant procedures in effect during the period of interest. In Appeal Basis No. 3, PG&E also asserts that the dependency analysis used in the FSD was not performed in accordance with SPAR-H guidance.

NRC Response to Basis No. 3:

The NRC concluded that PG&E's Appeal Basis No. 3 does not have sufficient merit for review by the appeal process specified in IMC 0609, Attachment 2, because the contention does not meet the limitations specified in IMC 0609.02, Section 03.a and Section 03.b. Specifically, the NRC did consider actual plant procedures, including the correct revisions of emergency operating procedures, when assessing operator actions, PSFs, timing of recovery actions, sequence of recovery actions, and dependency. Additionally, the NRC staff did perform the dependency analysis used in the FSD in accordance with SDP guidance provided in NUREG/CR-6883 and INL/EXT-10-18533.

Your staff is correct that the FSD does reference a superseded version of procedure EOP E-1.3; this was an administrative error. However, as documented in our January 17, 2017, letter (ADAMS Accession Number ML17017A437), the NRC reviewed both EOP E-1.3, "Transfer to Cold Leg Recirculation," Revision 15 and Revision 22, to inform our understanding of what valves are manipulated prior to reaching the point at which operators enter Procedure ECA 1.1. The NRC also compared this information with the most recent revision of the Diablo Canyon Final Safety Analysis Report to understand the sequence of valve manipulations performed in the first few minutes of the transition to ECCS cold leg recirculation. The steps from EOP E-1.3, "Transfer to Cold Leg Recirculation," Revision 15, listed in the FSD, were not used to develop a principal assumption in the timing or sequence of recovery actions.

Contrary to the assertion in your January 26, 2017, letter, the NRC did consider parallel recovery actions. The NRC's timeline assumed that control room-directed EOPs and local recovery actions, consisting of electrical and mechanical repair teams, would proceed in parallel. The NRC independently developed several timelines of the likely sequence of events that occur following a LOCA with an assumed failure of valve SI-2-8982B. In these timelines, the NRC assumed that approximately 9 minutes after the RWST reached 33 percent level, operators discovered that valve SI-2-8982B would not open from the main control room. The NRC assumed that failure of this valve constituted the cue for the TSC to begin pursuing the mechanical recovery option. Approximately 6 minutes later, the NRC assumed that valve SI-2-8982A fails requiring exit from Procedure EOP E-1.3 and entry into procedure EOP ECA 1.1. The NRC assumed that entry into Procedure ECA 1.1 constituted the cue for the TSC to begin pursuing the electrical contactor recovery option in parallel with the mechanical recovery option.

The NRC also assumed that because of certain procedural limitations, completion of electrical recovery actions would be delayed until after the mechanical recovery option is complete. The basis for the delay is a procedural limitation associated with manual operation of motor operated valves. Station Operating Procedure OP O-9, "Manual 'Seating' of Motor Operated Valve," Revision 21A, provides several precautions and limitations related to personnel safety that would prevent electrical operation of a motor operated valve while the manual hand wheel is used. These precautions and limitations are consistent with standard industrial safety practices and vendor guidance that would prevent electrically operating a motor operated valve while the valve is de-clutched and manually manipulated. This industrial safety precaution was assumed to result in a delay in completing the electrical contactor recovery option until after the mechanical recovery actions are complete. The delay was not a result of any assumption related to Procedure EOP E-1.3.

Regarding the dependency analysis used in the FSD, your January 26, 2017, letter is incorrect in stating that the NRC applied dependency to the electrical and mechanical recovery actions, and to several control room-directed EOP actions. The human error probability of the mechanical recovery option and electrical contactor recovery option were evaluated in accordance with the SPAR-H method specified in NUREG/CR-6883 with no dependency applied.

When developing a composite recovery probability during development of the FSD, the NRC assumed that control room-directed EOP actions involving throttling of ECCS, depressurization and cooldown of the reactor coolant system, and refilling of the RWST are completed with 100 percent success with no dependency applied. The NRC recognizes that the assumption of 100 percent success for these EOP actions is a non-conservative input into the FSD. The NRC applied this assumption because in the FSD, the NRC was evaluating the realistic lower range of the increase in CDF. Application of SPAR-H guidance to these control room-directed EOP actions, including appropriate consideration of dependencies, would result in an increase in the CDF but not beyond the result documented in the FSD that produced a White result.

The NRC applied dependency to some recovery strategies and actions that occur after the initial mechanical and electrical contactor recovery actions were assumed to fail. These actions only included the electrical jumper recovery method and the EOP directed action to makeup from the VCT to the RCS, which occurs after RWST water inventory is depleted and all ECCS pumps are stopped. As documented in the FSD, Element 6, Page A-3, and Element 9, Page A-7, the NRC determined these tasks were subject to high dependency because the same crew would be used, the tasks would be close in time, and no additional cues would be present. The NRC reached this conclusion by applying the guidance for dependency in NUREG/CR-6883, Table 2-4, “SPAR-H Dependency Rating System.” Given the timeline of the event would be limited to a single ERO crew, the NRC assumed that the same electrical repair crew that performed the electrical contactor recovery actions would perform the electrical jumper recovery actions. Additionally, the same control room crew that performed the actions to throttle ECCS, cooldown and depressurize the RCS and refill the RWST, would be used to initiate makeup from the VCT to the RCS. These actions were assumed to be short in time, approximately 30 minutes, with no additional cues present. Note that the NRC’s documentation of no additional cues was included in the FSD for completeness but did not influence the assigned dependency for this action.

The NRC recognizes that the simplification of using only the dependency term for late term ERO/TSC directed recoveries results in a slight underestimation of the failure probability for these actions. However, because of the insignificant contribution to the overall composite human error probability for recovery of valve SI-2-8982B (approximately 5E-3), this simplification has little effect on the outcome of the FSD.

4. PG&E Appeal Basis No. 4: Application of SPAR-H Guidance for the Mechanical Recovery Procedure PSF

In Appeal Basis No. 4, PG&E asserts that the SPAR-H Step-by-Step guidance, Section 3.5, Procedures, directs the NRC to demonstrate that the procedure is a performance driver for opening the chamber guard cover as a prerequisite to evaluating the Procedure PSF quantitatively.

NRC Response to Basis No. 4:

The NRC concluded that PG&E's Appeal Basis No. 4 does not have sufficient merit for review by the appeal process specified in IMC 0609, Attachment 2, because the contention does not meet the limitations specified in IMC 0609.02, Section 03.a. Specifically, PG&E's appeal basis involving the selection of the procedures PSF is related to NRC staff's choice of probabilistic risk modeling assumptions used in the significance determination process, which is not appealable under IMC 0609, Attachment 2, because the NRC staff appropriately documented its justification. The NRC staff's quantitative evaluation of the human error probability for the mechanical recovery option did appropriately consider how the lack of site procedures was a performance driver. Additionally, the NRC appropriately applied INL/EXT-10-18533, Section 3.5.

During development of the FSD, the NRC considered all of the procedure PSFs provided in NUREG/CR-6883 when evaluating the mechanical recovery option. The NRC judged that procedures were a performance driver for the mechanical recovery action and therefore quantitatively evaluated the impact of this PSF on the proposed recovery action.

The NRC determined that the more risk significant "Procedure – Not Available" PSF closely aligned to the mechanical recovery because procedures needed to open the valve SI-2-8982B chamber guard and access the component were not available. However, because the NRC was performing an evaluation to determine the lower range of the increase in CDF and because some information was available related to opening the valve SI-2-8982B chamber guard (albeit in unrelated outage procedures), the NRC determined that the less risk significant "Procedures – Incomplete" PSF was an appropriate assumption.

The NRC's assumption was based on statements by your staff, both during inspection interviews and during the regulatory conference, that there is no existing emergency procedure to open the valve SI-2-8982B chamber guard. As documented in the FSD, Element 7, Page A-4, the NRC concluded that, "Information needed to complete the mechanical recovery method was not contained in standard operating procedures. In particular, [the licensee] stated that there is not an existing emergency procedure to open the valve SI-2-8982B chamber guard and that during the postulated event, existing outage related work instructions would be used to develop the emergency instructions to open the chamber guard to allow for the mechanical recovery method."

In your January 26, 2017, letter, you state that the "simplicity of these actions substantiates that procedures would be of minimal significance to successful performance." Your staff also suggested that because of this simplicity, the NRC should assign the "Procedures – Available, But Poor," PSF when evaluating the human error probability of the mechanical recovery option.

The NRC reviewed the "Procedures – Available, But Poor," PSF when developing the FSD but determined this PSF was inappropriate since it refers to existing procedures that are ambiguous, inadequate or have formatting errors. The NRC concluded that procedures were incomplete or missing. The NRC determined that this lack of guidance more closely aligned with the definition of an incomplete procedure rather than a procedure that is available but poor, and used this assumption when developing the FSD.

5. Appeal Basis No. 5: Recovery Time Available for SLOCAs.

In Appeal Basis No. 5, PG&E asserts that the NRC changed the Time Available PSF from "extra" in the preliminary significance determination to "nominal" in the FSD without separately accounting for small loss-of-coolant-accidents (SLOCA) recovery timeliness, as discussed at the Regulatory Conference.

NRC Response to Basis No. 5:

The NRC concluded that PG&E's Appeal Basis No. 5 does not have sufficient merit for review by the appeal process specified in IMC 0609, Attachment 2, because the contention does not meet the limitations specified in IMC 0609.02, Section 03.a. Specifically, the NRC staff's evaluation of the Time Available PSF for the electrical contactor recovery was made based on the best available information, including consideration of information presented by your staff at the regulatory conference. While your staff did state that new information relative to SLOCA recovery timelines was under development at the regulatory conference, no analysis was provided to the NRC to justify a longer SLOCA recovery timeline. The NRC staff did qualitatively evaluate the impact of SLOCAs on recovery and concluded that the "Time Available - Nominal" PSF was an appropriate assumption for both the mechanical and electrical contactor recovery actions.

The decision to change the Time Available PSF from "Extra-Time" to "Nominal" was an assumption based on the best information available. After the regulatory conference, the NRC evaluated parallel actions to complete the mechanical and electrical contactor recovery options. However, because of certain procedural limitations, the NRC assumed that completion of electrical recovery option would be delayed until after the mechanical recovery option has failed. The basis for the delay is procedural limitations associated with manual operation of motor operated valves. Specifically, Procedure OP O-9 provides several precautions and limitations related to personnel safety that would prevent electrical operation of a motor operated valve while the manual hand wheel is used. These precautions and limitations are consistent with standard industrial safety practices and vendor guidance that would prevent electrically operating a motor operated valve while the valve is de-clutched and manually manipulated. Using this sequence as an assumption, the timing for the electrical contactor recovery changed such that there was no longer sufficient time to justify an "Extra-Time" PSF. Accordingly, the analyst changed the Time Available PSF to "Nominal."

The NRC, when developing the final composite human error probability, did qualitatively consider a slightly reduced drain rate due to SLOCAs. Using information related to break flows from NUREG-1829, the NRC determined that a 3-inch LOCA effective break size would yield greater than 5000 gallons per minute and a 1.625-inch effective break size would yield a break flow of greater than 1500 gallons per minute. Additionally, the NRC's timeline for recovery included actions to cooldown and depressurize the reactor coolant system. Using this information, the NRC concluded a nominal amount of ECCS flow, consisting of at least a single centrifugal charging pump, is needed for all LOCAs. As discussed in Section 3.1 of INL/EXT-10-18533, the NRC was cautious against relying on overly precise estimates that lead to threshold effects for the Available Time PSF.

Consequently, the NRC determined that for all LOCAs, the resulting drawdown rate of the RWST produced available recovery times sufficient to assume the time available PSF as “Nominal.”

Additionally, as discussed in Item 2 above, and as documented in the FSD, Elements 7 and 8, Pages A-4 and A-5 respectively, the NRC considered the guidance in INL/EXT-10-18533, Section 3.1, to develop the assumptions related to the available time PSF for the mechanical and electrical contactor recovery actions. As such, the NRC determined it was appropriate, as discussed in INL/EXT-10-18533, to judge the “Time Available” PSF as “Nominal” for the action portion of the mechanical and electrical recovery options with the remainder of time assigned to diagnosis.

6. **Appeal Basis No. 6: Use of a Range to Assess Overall Risk Significance.**

In Appeal Basis No. 6, PG&E asserts that the use of a range of values is inconsistent with applicable SDP guidance, including IMC 0308, Attachment 3 and IMC 0609, Attachment 1. The FSD does not specify the result as a single mean value, or point estimate, based on the “best available information.” In Appeal Basis No. 6, PG&E also asserts that the NRC’s “upper range” of the delta CDF is the FSD, which is the same as the delta CDF in the preliminary significance determination, does not account for actual plant hardware, procedures, and equipment configurations. The upper end of the range should be reassessed based on the new information presented at the Regulatory Conference.

NRC Response to Basis No. 6:

The NRC concluded that PG&E’s Appeal Basis No. 6 does not have sufficient merit for review by the appeal process specified in IMC 0609, Attachment 2, because the contention does not meet the limitations specified in IMC 0609.02, Section 03.a and Section 03.b. Specifically, the NRC staff’s use of a range to assess the significance of an inspection finding was in accordance with applicable SDP guidance.

During development of the FSD, the NRC reviewed and utilized both IMC 0308, Attachment 3, “Significance Determination Process Technical Basis Document,” dated June 16, 2016, and IMC 0609, Attachment 1, “Significance and Enforcement Review Panel (SERP) Process,” dated December 8, 2016. Neither of these guidance documents require development of a single mean value, or point estimate, to support issuance of a FSD. The NRC significance determination process is not focused on providing a specific point estimate as a result but instead in demonstrating how the significance of the issue falls relative to the thresholds of the color significance assigned to the finding. NRC technical staff and management, including during the NRC SERP panel, extensively discussed the use of a range to demonstrate the significance of the underlying performance deficiency. Because of the complexity of the many recovery methods provided by your staff and because of the uncertainty involving the sequence and timing of TSC and control room actions to recover valve SI-2-8982B, the NRC determined that the use of a range was appropriate in this case to support a risk-informed decision. The SERP and NRC management determined that the use of a range was appropriate because both the lower and upper estimates of increase in CDF associated with the performance deficiency were within the thresholds for a White finding. In accordance with IMC 0308, Attachment 3, Section 03.03.g, the NRC staff determined that further refinement of the SDP analysis would not result in a different outcome (i.e. a color threshold change), and proceeded to a risk-informed determination using the best available information and reasonable technical and probabilistic judgements.

Additionally, the NRC did include considerations relative to actual plant hardware, procedures, and equipment configurations, when developing the FSD. As documented in the FSD, the NRC considered many additional factors, which removed much of the conservativism from the assumptions used in the preliminary risk assessment, and predicted a high likelihood of success (96.4 percent success) for recovering valve SI-2-8982B. Using these assumptions, the NRC concluded the lower range of increase in CDF associated with the performance deficiency to be 1.3E-6 per year.

While you are correct that the NRC did not change the assumptions used to develop the preliminary significance determination result, it is important to note that the NRC's use of an "upper range" of the delta CDF in the FSD was not to demonstrate explicitly that the issue is of White significance. Instead, the NRC used an upper range to demonstrate that the significance of the issue, if evaluated using more conservative assumptions such as those found in the preliminary significance determination, is not of substantial safety significance (Yellow).