NRR-DMPSPEm Resource

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Sent:	Wednesday, April 4, 2018 8:49 AM
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Subject:	Palo Verde 1, 2, and 3 - Final RAIs for TSTF-505 (Initiative 4b) LAR (CAC Nos. MF6576, MF6577, and MF6578; EPID L-2015-LLA-0001)
Attachments:	Palo Verde 4b RAIs (APLA and APLB).docx

Attached please find the **official** requests for additional information (RAIs) from the U.S. Nuclear Regulatory Commission (NRC) staff for the subject license amendment request (LAR), and provide your responses within 45 days from the date of this e-mail, as mutually agreed during the clarification call held on March 29, 2018. Per my telephone conversation with you on April 3, 2018, RAI 23 has been deleted without renumbering the rest of the RAIs. Further, based on the clarification call held on March 29, 2018, final responses to RAIs 11, 16, 21, and 29 will not be provided within 45 days, however, best estimate response dates/times/plan will be provided in your 45 days response for these RAIs. Your timely responses will allow the NRC staff to complete its review on schedule.

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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO ADOPT TSTF-505-A, REVISON 1 (4b)

RISK-INFORMED COMPLETION TIMES

ARIZONA PUBLIC SERVICE COMPANY

PALO VERDE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3

DOCKET NOS. 50-528, 50-529, AND 50-530

By letter dated July 31, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15218A300), as supplemented by letters dated April 11, 2016 and November 3, 2017 (ADAMS Accession Nos. ML16102A463 and ML17307A188, respectively), Arizona Public Service Company [APS, the licensee] submitted a license amendment request (LAR) to modify the Palo Verde Nuclear Generating Station (PVNGS) Technical Specification [TS] requirements to permit the use of Risk Informed Completion Times (RICT) in accordance with the Technical Specifications Task Force (TSTF) Traveler TSTF-505-A, Revision 1, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk Informed Technical Specification Task Force] Initiative 4b." To complete its review, the U.S. Nuclear Regulatory Commission (NRC) staff from Probabilistic Risk Assessment (PRA) Licensing Branch (APLA), PRA Licensing Branch 2 (APLB), and Mechanical Engineering and Inservice Testing Branch (EMIB) requests a response to the questions below.

Request for Additional Information (RAI) 1 APLA - Internal Events PRA Findings and Observations

Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities," Revision 2 (ADAMS Accession No. ML090410014), provides guidance for addressing PRA acceptability including addressing the need for the PRA model to represent the as-designed or as-built, as-operated plant; identifying permanent plant changes that have an impact on those things modeled in the PRA but have not been incorporated in the baseline PRA model; documenting that the parts of the PRA required to produce the results used in the decision are performed consistently with the standard as endorsed in the appendices of RG 1.200; a summary of the risk assessment methodology used to assess the risk of the application, including how the base PRA model was modified to appropriately model the risk impact of the application and results; Identifying the key assumptions and approximations relevant to the results used in the decision-making process; a discussion of the resolution of the peer review (or self-assessment, for peer reviews performed using the criteria in Nuclear Energy Institute (NEI) 00-02) findings and observations (F&O) that are applicable to the parts of the PRA required for the application; and, documenting the use of the parts of the PRA that conform to capability categories or grades lower than deemed required for the given application. Based on its review, the staff needs additional information to complete its review concerning the resolution of the peer F&O applicable to this application. Please provide the following:

a. internal events PRA (IEPRA) F&O AS-03 in LAR dated November 3, 2017 (hereafter called as LAR) Attachment 6 asks about the rationale for why the plant response to small loss-of-coolant accidents (LOCAs) and induced small LOCAs were modelled differently. The disposition to the F&O states that the finding has been resolved and closed by an update of the PRA model and documentation. Please explain and justify why the plant responses are different for these LOCAs and describe the update of the PRA model that was made to resolve the F&O. Please confirm that the success criteria for the plant responses to these LOCAs have received appropriate reviews and are documented.

b. The disposition to F&O IE-07 in LAR Supplement Attachment 6, Table A6-1 states that "leakage, spurious operation, and catastrophic failure modes of valves will be considered" when addressing the Closure Review Team recommendation. The NRC staff found that common cause failure (CCF) modes of valves is not identified. Please clarify if CCF modes of valves will be considered in the evaluation to close this F&O, consistent with the guidelines in NUREG/CR-5485, "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment," published in November 1998 or provide justification for not considering CCF of valves.

<u>RAI 2 APLA - Internal Flooding</u>

a. LAR Attachment 13 Table A13-1 explains that because no procedures exist for how to isolate a flooding event associated with the safety injection or chemical and volume control system piping it was conservatively assumed that operators would isolate the leak at the least advantageous point (i.e., which results in loss of one train of emergency core cooling system). The explanation then concludes that the impact of this assumption on the RICT program is insignificant.

   It is unclear to the NRC Staff whether the internal flood PRA (IFPRA) adequately considers the risk associated with this event because there are no proceduralized operator actions. In addition, if operator action is to isolate as described, it has not been demonstrated that this flood event may not be a significant risk contributor given the plant configuration at the time of the postulated flood event. Please provide adequate technical justification for the conclusion that this would not have a significant impact on the RICT program, or discuss how proper operator actions will be addressed in the IFPRA model and the RICT evaluations.

b. LAR Attachment 13, Table A13-1 explains that "floods are assumed to fail all equipment in the initiating room and then propagate out of the room to surrounding flood areas;" except for cases in which equipment was sufficiently high or flood barriers are not expected to retain water to sufficient flood levels are treated on an individual basis. Though the explanation concludes that this assumption would not have a significant impact on the RICT program, the discussion states that "the top cutsets are not impacted, however if very specific isolation actions were taken this assumption could be significant." Please explain whether these isolation actions are modeled in the IFPRA, or explain why they are excluded. Please confirm that supporting flood analyses for these potentially significant actions are reviewed and documented.

c. Attachment 13, Table A13-1, states:

   "The flood [human reliability analysis] HRA dependency analysis did not include large early release specific [human failure events] HFEs. HFEs specific to large early releases (i.e., post-core damage operator actions) are generally performed several hours after the initiating event occurs. No dependency between early and late operator actions. There is no impact on the model."

Although operator actions may be performed several hours after the initiating event occurs, that does not necessary imply that there is no dependency between early and late operator actions for an internal flood event. Please explain further your conclusion regarding these assumed HRA dependencies.

RAI 3 APLA - Evaluation of Common Cause Failure for Planned Maintenance

NEI 06-09, Revision 0, "Risk-informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (ADAMS Accession No. ML063390639), includes guidance that states that CCF adjustment is required for planned maintenance. In a letter dated May 17, 2007 from Jennifer M. Golder, NRC, to Biff Bradley, NEI, (ADAMS Accession No. ML071200238), the NRC provided its safety evaluation for NEI Topical Report NEI 06-09. The NRC staff based its acceptance of NEI 06-09 on RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998, (ADAMS Accession No. ML003740176). Specifically, SE Section 2.2 states that, "specific methods and guidelines acceptable to the NRC staff are also outlined in RG 1.177 for assessing risk-informed TS changes." Further, SE Section 3.2 of the NRC safety evaluation states that compliance with the guidance of RG 1.174, Revision 1, ["An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002, (ADAMS Accession No. ML023240437),] and RG 1.177, for ΔCDF [core damage frequency] and ΔLERF [large early release frequency,] is achieved by evaluation using a comprehensive risk analysis, which assesses the configuration-specific risk by including contributions from human errors and common cause failures."

The guidance in RG 1.177, Section 2.3.3.1, states that, "CCF modeling of components is not only dependent on the number of remaining inservice components, but is also dependent on the reason components were removed from service, i.e., whether for preventative or corrective maintenance." In relation to CCF for preventive maintenance, the guidance in RG 1.177, Appendix A, Section A-1.3.1.1, states:

> If the component is down because it is being brought down for maintenance, the CCF contributions involving the component should be modified to remove the component and to only include failures of the remaining components (also see Regulatory Position 2.3.1 of Regulatory Guide 1.177).

According to RG 1.177, if a component from a CCF group of three or more components is declared inoperable, the CCF of the remaining components should be modified to reflect the reduced number of available components in order to properly model the as-operated plant.

a. Please explain how CCFs are included in the PRA model (e.g., with all combinations in the logic models as different basic events or with identification of multiple basic events in the cut sets);

b. Please explain how the quantification and /or models will be changed when, for example, one train of a 3X100 percent train system is removed for preventative maintenance and describe how the treatment of CCF meets the guidance in RG 1.177 or meets the intent of this guidance when quantifying a RICT.

RAI 4 APLA - Instrumentation and Controls

The proposed TS limiting conditions for operations (LCOs) include one related to instrumentation and controls, specifically engineered safety features actuation system (ESFAS): TS 3.3.6 ESFAS Logic and Manual Trip. PRA technical adequacy attributes are provided in NEI 06-09-A Section 2.3.4 and the guidance provided in RG 1.200. An example of PRA modelling of instrumentation and controls is illustrated by guidance in TSTF-411, Revision 1, "Surveillance Test Interval Extension for Components of the Reactor Protection System (WCAP-15376)," and TSTF-418, Revision 2, "RPS and ESFAS Test Times and Completion Times (WCAP-14333). Please provide the following information:

a. Please explain how instrumentation is modelled in the PRA. This should include, but not be limited to, the scope of the instrumentation and controls (I&C) equipment (channels, relays, logic, etc.) and associated TS functions for which a RICT would be applied, and PRA modeling of the I&C and functions including how these are modeled in sufficient detail and based on plant-specific data, etc.

b. Please identify any digital instrumentation and/or controls, which are being proposed for equipment in the RICT program. Please indicate if the Units would apply the RICT to digital systems such as Common Q or hybrid (analog and digital). Please discuss how digital or hybrid I&C are modeled in the PRA.

RAI 5 APLA - Configuration Risk Management Program

a. Please describe how the constraints identified in the program (e.g. CCF, Loss of Function, and short CTs) will be addressed. Please describe the process to maintain the accuracy of pre-solved cutsets.

b. Please explain how the configuration risk management program (CRMP) tool will keep track of PRA functional/non-functional status.

RAI 6 APLB - Seismic PRA (SPRA) RG 1.200, Revision 2, PRA Acceptability, NEI 12-13

LAR Attachment 6 states that the categorization process for seismic hazards will use a peer reviewed plant specific Seismic PRA model in accordance with RG 1.200, Revision 2. The NRC staff was unable to determine, which peer review guidance document, NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance," NEI 05-04, "Process for Performing Follow-On PRA Peer Reviews Using the American Society of Mechanical Engineers (ASME) PRA Standard," or NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," was used. Please indicate, which guidance document was used to perform the SPRA peer review. If the peer review was performed using guidance not described in RG 1.200, Revision 2, such as NEI 12-13, "External Hazards PRA Peer Review Process Guidelines" (ADAMS Package Accession No. ML122400044), please provide the following additional information to justify the use of NEI 12-13:

a. Please describe how the qualifications of the SPRA peer review team comply with the peer review requirements in ASME/American Nuclear Society (ANS) RA-Sa-2009, Sections 1-6.2 and 5-3.2, as endorsed in RG 1.200.

b. Please identify any unreviewed analysis methods (UAMs) used in the SPRA, as determined by the peer review team, and describe each UAM with a level of detail appropriate for the NRC staff to evaluate its acceptability.

c. Please describe if the SPRA relies on expert judgement to meet any supporting requirement (SR) and, if so, demonstrate conformance to the expert judgment requirements of ASME/ANS RA-Sa-2009, Section 1-4.3. Also, please cite any information from the peer review report related to the evaluation of the use of expert judgment by the peer review team and whether the peer review team found the use of expert judgment to be appropriate.

d. Please clarify whether the SPRA was reviewed against Capability Category (CC-I) for any SR. Please provide a list of all SRs that were reviewed against CC-I or found to meet only CC-I without an associated finding. For each such SR, please justify why not meeting the SR at CC-II does not impact this application.

e. Please clarify whether an "in-process" peer review was performed for the SPRA. If an "in-process" approach was utilized, please confirm that (i) the approach met the requirements for an independent peer review as stated in ASME/ANS RA-Sa-2009 and the process described in NEI 12-13; (ii) a final review by the entire peer review team occurred after the completion of the SPRA; and (iii) peer reviewers remained independent throughout the PRA development activity as discussed in the enclosure to the letter dated November 16, 2012 from Mr. Donald G. Harrison, NRC to Mr. Biff Bradley, NEI (ADAMS Accession No. ML12321A280).

RAI 7 APLB- RG 1.200, Revision 2, PRA Acceptability, FPIE As Basis for SPRA

ASME/ANS RA-Sa-2009, Section 5-2.3, Part 5, "Requirements for Seismic Events At-Power PRA," assumes that full-scope internal-events at-power Level 1 and Level 2 LERF PRAs exist, and that those PRAs are used as the basis for the SPRA systems analysis. ASME/ANS RA-Sa-2009, High Level Requirement [HLR]-SPR-B calls for the incorporation of seismic analysis aspects that are different from the at-power IEPRA systems model. Therefore, the technical adequacy of the IEPRA model used as the foundation for the SPRA needs to be established.

Please identify IEPRA finding-level F&Os that were not closed in accordance with an NRC-accepted process and any IEPRA upgrades that had not been peer-reviewed prior to the development of the SPRA. For each identified finding-level F&O, please describe the resolution and the impact of the F&O on the SPRA as it pertains to this application.

RAI 8 EMIB – [SPRA] RG 1.200, Revision 2, PRA Acceptability, F&Os Not Resolved by Closure Review

In LAR Attachment 6, the finding for F&O SFR-F3-01 included a recommendation to justify the use the Best Estimate (BE) In-Structure Response Spectra (ISRS) as the median. The recommendation further states that the soil-structure interaction analysis using BE soil properties, BE structure stiffness and a conservative estimate of BE structure damping results in a 84th percentile response. This recommendation indicates to the NRC staff that the ISRS input may not be correct.

Please discuss your plans to address this recommendation including how your plans to close this finding through a peer review or through the NRC accepted F&O closure process prior to implementing the RICT Program.

RAI 9 APLB – RG 1.200, Revision 2, PRA Acceptability, F&Os Not Resolved by Closure Review

In LAR Attachment 6, the finding for F&O SHA-E2-01 included a recommendation to demonstrate that the updated set of soil peak ground acceleration (PGA) hazard curves fractiles (mean, and 5th, 16th, 50th, 84th, 95th) is bounded by the soil PGA hazard curves used in the seismic PGA model. If the updated set of soil PGA hazard curves is greater than those used in the seismic PRA model, the impact on seismic risk quantification should be assessed.

Please discuss your plans to address this recommendation including how your plans to close this finding through a peer review or through the NRC accepted F&O closure process prior to implementing the RICT Program.

RAI 10 APLA - RG 1.200, Revision 2, PRA Acceptability, PRA Upgrades Identified in F&O Closure Review Report

LAR Attachment 6 states that all PRA upgrades (as defined by the ASME PRA Standard RA-Sa-2009) implemented since conduct of the Combustion Engineering Owners Group (CEOG) peer review in 1999 have been peer reviewed. The LAR indicated that one full-scope peer review was performed on the IEPRA model in July 1999, IFPRA (2010), SPRA (2013), FPRA (2012 and 2014). Please provide the following additional information to enable the NRC staff to evaluate whether the guidance provided in RG 1.200, Revision 2, regarding PRA upgrades was followed:

a. Please describe the changes made to the IEPRA since the full-scope peer review was conducted in 1999, including any changes that would impact the modeling framework for the PRA, such as converting the PRA to a one-top fault tree across all the PRA hazards. Provide the dates for when each change occurred. This description should be of sufficient detail for the NRC to determine whether the changes are considered PRA maintenance or PRA upgrades as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2. Include in your discussion: (1) any new methodologies (i.e., summarize the original method in the PRA and the new method); (2) changes in scope that impact the significant accident sequences or the significant accident progression sequences; (3) changes in capability that impact the significant accident sequences or the significant accident progression sequences.

b. For each change described in Part a. above, please indicate whether the determination for the change was PRA maintenance or a PRA upgrade as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2, along with justification for the determination.

c. Please discuss any focused-scope (or full-scope) peer reviews that have been performed for the PRA upgrades identified in Part b above, providing the timeline of when the peer reviews were performed and when the peer review reports were approved. For each upgrade identified, either:

i. Please provide the findings of the peer review(s) performed on the upgrade and the disposition of thefindings as it pertains to the impact on the 10 CFR 50.69 application. OR,
ii. Please confirm that the resulting F&Os from the peer review(s) on the upgrade were assessed in the F&O closure review in June 2017.

d. Please describe the changes that have been made to the IFPRA, SPRA, and FPRA since their respective peer reviews on November 2010, February 2013, December 2012 and December 2014. Provide information commensurate with that requested for the IEPRA in parts (a), (b), and (c), which indicates and justifies the determination of whether the changes were maintenance or an upgrade and, if an upgrade, provides information to support a technical acceptability determination.

RAI 11 APLA - F&O Closure

In a letter dated May 3, 2017, the NRC staff transmitted its review results of Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations" (F&Os) (ADAMS Accession No. ML17079A427). Based on the NRC staff review, the NRC found the process proposed by Appendix X acceptable, with conditions as specified in the letter, for use by licensee's to close F&Os that were generated during a peer review process.

LAR Supplement Attachment 6 states that the F&O closure process was used for the F&Os associated with Internal Events, Internal Flood, Fire, and Seismic PRA models. The NRC staff has identified three primary issues based on recent observations of industry's implementation of the closure process: 1) closure with respect to Capability Category CC-II for the SR; 2) written justification of basis for why closure is determined to be maintenance or upgrade; and 3) independence of reviewers.

Please summarize how the June 2017 F&O closure process fulfilled each of the guidelines below.

a. The documented licensee justification and associated F&O closure team assessment about whether each F&O finding resolution constitutes a PRA upgrade or maintenance update, as defined in ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2.

b. The review team's summary rationale for determining the adequacy for closure of each finding in relation to the affected portions of the associated SR for every SR and weakness identified in the F&O.

c. The description of remote reviewer's participation (if used) confirming web and teleconference connection between any remote reviewers and the on-site review team and host utility to support full participation of the remote reviewers.

d. The confirmation that every weakness in each F&O has been addressed, that a closed finding has been achieved (for applicable F&Os), and that the documentation has been formally incorporated in the PRA Model of Record before closure in the final F&O closure report.

RAI 12 APLA - RG 1.200, Revision 2, PRA Acceptability, Key Assumptions and Key Sources of Uncertainty

RG 1.200, Revision 2, Section 3.3.2, states “for each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making associated with the application.” Further, RG 1.200, Revision 2, Section 4.2 states that “these assessments provide information to the NRC staff in their determination of whether the use of these assumptions and approximations is appropriate for the application, or whether sensitivity studies performed to support the decision are appropriate.” RG 1.200, Revision 2, Section 3.3.2, defines the terms “key assumption” and “key source of uncertainty.”

LAR Attachment 13 states that “the list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of configuration-specific changes in risk. If the PVNGS model used a non-conservative treatment, or methods which are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine the impact on RICT Program calculations. Only those assumptions or sources of uncertainty that could significantly impact the configuration risk calculations were considered key for this application.

a. Please describe the approach used to identify and characterize the “key” assumptions and “key” sources of uncertainty in the SPRA for this application. Discuss (1) whether all assumptions and sources of uncertainty related to all aspects of the models (e.g., hazard, fragility, and plant response analysis for the SPRA) were evaluated to determine whether they were “key;” and (2) the criteria that were used to determine whether the modeling assumptions and sources of uncertainty were considered “key.”

b. Please describe each key assumption and key source of uncertainty identified in the SPRA. Provide this in sufficient detail to enable the NRC staff to identify whether the key assumptions used in the SPRA involve any changes to consensus approaches.

c. Please discuss how each key assumption and key source of uncertainty identified above was dispositioned for this application. If available, provide the results of any sensitivity studies that will be used to support the disposition for this application or use a qualitative discussion to justify why different reasonable alternative assumptions would not affect this application.

RAI 13 APLA - PRA Peer Reviews

Please describe the peer reviews performed for the IEPRA), IFPRA, SPRA, and FPRA. Please confirm that these peer reviews were full-scope reviews meeting industry guidance for a peer review and were reviewed against capability category II (CC-II). In addition, please discuss which organization performed the review, and list the guidance documents followed for each review.

RAI 14 APLA - Other External Hazards Peer Review

LAR Attachment 6 states that a full-scope external hazards screening peer review was performed in December 2011 in accordance with RG 1.200, Revision 2. The LAR does not discuss the results from this external hazards screening peer review and does not state whether the F&O closure review in June 2017 addressed any findings from the external hazards screening peer review.

a. Please clarify whether the finding-level F&Os, if any, from the December 2011 peer review of the external hazards screening process was encompassed in the scope of the June 2017 F&O closure review.

b. If finding-level F&Os from the December 2011 peer review of the external hazards screening process were not addressed in the June 2017 F&O closure review, please provide these findings and the associated dispositions as it pertains to this application. For any open findings, please discuss your plans to close those findings through a peer review or through the NRC accepted F&O closure process prior to implementing the RICT Program.

<u>RAI 15 APLB - External Hazards</u>

NEI 06-09, Section 3.3.5, "External Events Consideration," clarifies that external hazards' impact on incremental configuration risk should be addressed for each RICT calculation. LAR Attachment 8, "Information Supporting Justification of Excluding Sources of Risk Not Addressed by the PVNGS PRA Models," addresses external events. This attachment summarizes the evaluation of the risk of external hazards that appears to be consistent with the ASME/ANS RA-Sa-2009, (i.e., screening associated with the baseline risk contribution). The results of the evaluation summarized in Table A8-1 seem to indicate, however, that the external hazards will be excluded from every configuration risk evaluation, (e.g. no unique PRA model for seismic events is required in order to assess configuration risk for the RICT Program). However, there may be situations where the hazard may be important in a configuration risk calculation even though the baseline risk can be screened out consistent with the ASME/ANS RA-Sa-2009. For example, external floods seem to be excluded because the plant design conforms to the standard review plan (SRP) criteria. Presumably smaller flood levels may fail plant equipment not required to be protected by the SRP criteria which could affect configuration risk, and sometimes the flood barriers themselves may be degraded or undergoing maintenance which could affect configuration risk. Similarly, extreme wind or tornado seem to be fully excluded because of low frequency of occurrence but those factors for the barriers may not have considered the plant configuration during a RICT.

a. Please clarify if the external hazard risks are excluded from the RICT program or if the program includes guidance to assure that the assumptions supporting the screening of the hazards remain applicable given the plant configuration during the RICT. If the hazards are fully excluded, please address the issue related to screening based on meeting the SRP criteria (e.g., design flood height and mitigating features) or based on low nominal risk values. If, instead guidance is provided, please describe the guidance, (e.g., in certain instances, hazards which were initially screened out from the RICT calculation may be considered quantitatively if the plant configuration could impact the RICT).

b. The LAR does not mention how design basis assumptions for external hazards are treated in the RICT program. A design basis assumption applicable to a hazard may temporarily be not applicable (e.g., barrier degradation), which may increase the likelihood of a plant challenge.

    i. Please discuss how structures, systems, and components (SSCs) important for the design basis assumptions for external events will be considered in RICT program guidance.

ii. Prior to entering a RICT or during a RICT, please discuss how the CDF and LERF are evaluated if design basis assumptions important for the screened-out hazard are or may be impacted by plant conditions.

RAI 16 APLA - RCP Seal Modeling

a. LAR Attachment 13, Table A13-1 states that reactor coolant pump (RCP) seal leakage is not modelled as a loss of reactor coolant system (RCS) inventory because the leakage is within the makeup ability of the charging pumps even if all four RCP seals fail. ASME/ANS RA-Sa-2009 provides for screening of initiating events and components as described in SRs IE-C6, SY-A15, and SY-B13. Please confirm that these screening criteria are met for screening out the RCP seal loss of coolant accident. If not met, please discuss your plans to resolve this issue prior to implementing the RICT program.

b. LAR Attachment 13, Table A13-1, indicated that WCAP-15749, "Guidance for the Implementation of the CEOG Model for Failure of RCP Seals Given Loss of Seal Cooling," Revision 0, December 2008, and pump seal vendor information was used to conclude that the leakage into the seal package from the RCS is limited to about 17 gallons per minute per pump. WCAP-15749 has not been endorsed by the NRC, however WCAP-16175-P-A, "Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS [Nuclear Steam Supply System] Plants," has been endorsed by the NRC (ADAMS Accession No. ML071130391). Please discuss the relationship between these two WCAP documents. Also please explain how the limitations and conditions of WCAP-16175-P-A have been addressed.

RAI 17 APLA - Fire PRA Methods

a. LAR Attachment 13, Table A13-1 explains that fire areas defined by the Fire Hazards Analysis (FHA) will substantially contain the adverse effects of fires originating from any currently installed fixed ignition source or reasonably expected transient ignition source and that fire zone boundaries are similarly assumed adequate or combined. The description explains that because fire zones have a lesser pedigree than fire areas, their boundaries are verified adequately in this notebook by a FHA review and plant walkdowns and that fire zones boundaries that appear unable to withstand the fire hazards within the zone are combined. It's not clear to the NRC staff from this description what criteria were used to determine when a fire zone was unable to contain the impact of a fire in that zone. Please describe the criteria used to determine when a fire zone was unable to contain the impact of a fire in that zone relative to passive barriers (or spatial separation if used), and active fire barriers, and the basis for those criteria. Justify the basis for this credit consistent with the NRC-accepted guidance.

b. LAR Attachment 13, Table A13-1 indicates that breaker fuse coordination was evaluated for the plant and states "when selective tripping cannot be demonstrated, the internal fire PRA model will credit recovery procedures planned to correct the coordination." The explanation also states that this assumption is considered to have low consequence to RICT evaluations because "electrical coordination will either be established or recovery procedures will be implemented to correct the coordination." These statements infer that one or more circuits with inadequate breaker fuse coordination were determined to exist at the plant. From the description in the LAR, it is not clear to the NRC staff whether recovery procedures are used to preclude the possibility of failures associated with inadequate circuits (such as loss of a power supply) or whether the inadequacies were modelled in the PRA along with HFEs associated with potential recovery of lost equipment. It is also not clear to the NRC staff how inadequate breaker fuse coordination and recovery procedures are modelled and what the bases is for that modelling. Please address the following:

If inadequate breaker fuse coordination and/or recovery procedures have been modelled in the fire PRA, please provide the following information:

i. Please explain how inadequate breaker fuse coordination is modelled in the fire PRA and justify that this treatment addresses the failures that could occur as a result of the identified circuit inadequacies.

ii. Please include a description of the circuit failure modes addressed and how associated component failures are modelled in the fire PRA. Also, please describe and justify assumptions made in the fire PRA about how fire-induced faults associated with inadequately coordinated/protected circuits impact upstream and downstream components from the fault.

iii. Given that the sizing and coordination of electrical protective devices appear to be in question, please include an explanation of how the potential for secondary fires is addressed in the fire PRA. If secondary fires are not modelled and fire-induced faults in inadequately protected circuits could lead to secondary fires, then please justify this modeling exclusion.

iv. Please explain how recovery procedures are modelled and how the feasibility of recovering components is established when after the power supply for the associated circuits is lost.

If inadequate breaker fuse coordination and recovery procedures have not been modelled in the fire PRA, then please provide justification for this modeling exclusion or incorporate modeling that addresses the concerns discussed) above. Alternatively, please correct the coordination issues which are relevant for the RICT program.

c. The NRC staff has formally accepted fire PRA method refinements during resolution of UAMs during individual plant reviews, as well as through the frequently asked question (FAQ) program for National Fire Protection association (NFPA) 805 in accordance with Regulatory Issue Summary (RIS) 2007-19, "Process For Communicating Clarifications Of Staff Positions Provided In Regulatory Guide 1.205 Concerning Issues Identified During The Pilot Application Of National Fire Protection Association Standard 805.". FAQ guidance relevant to the RICT program are those associated with fire PRA methods and not those specific to the NFPA 805 program.

i. Please identify and provide technical justification for any fire PRA methodology that has not been formally accepted by the NRC staff. Evaluate the significance of the proposed use of any unaccepted fire PRA method on the RICT program.

ii. If a method has been used for the fire PRA, please confirm that the accepted version of the method is used per the NRC position. If not, then please provide justification for the difference in the method or NRC position, or incorporate an NRC accepted method or position into the Fire PRA.

d. For purposes of Risk-informed Technical Specifications Initiative 4b, the fire PRA should be updated with the current studies prior to implementation of the RICT program in order to ensure it is consistent with currently NRC-accepted fire PRA methods. Please discuss the plans to incorporate acceptable methods in the fire PRA prior to implementing the RICT program. Some example acceptable methods include:

> "NUREG/CR-7150," Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE FIRE)," Volume 2, which is supported by a letter from the NRC to NEI, "Supplemental Interim Technical Guidance on Fire-induced Circuit Failure Mode Likelihood Analysis," (ADAMS Accession Nos. ML14086A165 and ML14017A135).
>
> "NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009" (ADAMS Accession No. ML15016A069).

e. Please describe how the evaluation includes the possible increase in heat release rates (HRR) caused by the spread of a fire from the ignition source to other combustibles, and summarize how suppression is included in the evaluation. Please discuss how this is consistent with, or any differences from, NRC accepted guidance.

f. Please confirm that the manual suppression probabilities are consistent with NUREG/CR-6850, EPRI[Electric Power Research Institute]/NRC-RES[Office of Nuclear Regulatory Research] Fire PRA Methodology for Nuclear Power Facilities," Volume 2, Appendix P (ADAMS Accession No. ML050940189), in that these are not less than the 0.001 floor.

RAI 18 APLA - Planned Modifications and Recovery Actions

Attachment 13, Table A13-1 includes the following assumption/uncertainty:

> "Planned plant modifications and recovery actions are assumed in the base case model. These modeled modifications are assumed to correct the fire vulnerability and not introduce any new failure modes."

a. "Recovery action" is a term used for plants adopting NFPA 805, which PVNGS has not identified as adopting in the LAR. Therefore, please explain the use of the term "recovery action."

b. Please describe the process used to identify these recovery actions, the method used to quantify them, and whether these actions have been evaluated for feasibility and reliability. If these recovery actions are necessary to meet RG 1.174 guidelines, please provide them.

RAI 19 APLA - Fire PRA Modeling

a. The original LAR Attachment 4, "List of Regulatory Commitments," included an action to validate that the Unit 1 internal fire PRA model is bounding for Units 2 and 3 to reflect field-routed cabling or create unit-specific internal fire models for Units 2 and 3 prior to use of the RICT program at Units 2 and 3. If separate fire PRAs need to be completed for Units 2 and 3, please discuss your plans to complete the peer reviews and submit F&Os for these fire PRA models as required by ASME/ANS RS-Sa-2009 as clarified by RG 1.200, Revision 2.

b. Equipment relied upon for safe shutdown may or may not be included in the RICT program. Please explain whether or not this equipment is in the Fire PRA and how the fire PRA is capable of evaluating the risk contribution from this equipment when it is out-of-service.

c. Please confirm that there are no fire events in the fire PRA model, which have a conditional core damage probability [CCDP] = 1.0. If there are, this would mask the risk contribution from equipment out-of-service. In such cases, please discuss your plans to resolve this issue prior to implementing the RICT program.

RAI 20 APLA - Maintenance Rule

NEI 06-09, Revision 0-A references NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 3, (ADAMS Accession No. ML031500684) as endorsed by RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants" (ADAMS Accession No. ML003740117). These references have been updated such that the latest guidance is NUMARC 93-01, Revision 4A (ADAMS Accession No. ML11116A198), as endorsed by RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3 (ADAMS Accession No. ML113610098). Please discuss whether these updated references are being used for the RICT program and if not, please discuss your plans for using them prior to implementation of the RICT program.

RAI 21 APLA - Total CDF and LERF

LAR Attachment 9, Table A9-1 provides the total baseline CDF and LERF. Completing the actions in the NRC staff's RAIs may impact the CDF and LERF values. After completing the actions in the other RAIs, please identify changes made to the PRA in response to RAIs, and provide the estimated CDF and LERF values that incorporate all changes and modifications. Also, please propose a plan to incorporate those changes that affect total baseline CDF and LERF values prior to implementation of the RICT program.

RAI 22 APLA - Reported Baseline Risk Values

LAR Attachment 9, Table A9-1, provides the CDF and LERF values for internal events, internal flooding, internal fire, and seismic events for PVNGS, Units 1, 2, and 3. The CDF and LERF values of each hazard presented in LAR Attachment 2 are identical for each unit. Typically, differences in CDF and LERF results exist for multiple-unit plants, even if the differences are not significant. Also, the LAR states numerous times that the risk management process ensures the PRA model used in the application reflects the as-built and as-operated plant for Units 1, 2, and 3. It is not clear to the NRC staff whether the risk values reported in LAR Attachment 2 are the results of separate PRAs performed for each unit or whether PRAs were performed only for a given unit and assumed to represent all three units.

a. If the PRAs were performed only for a given unit and assumed to represent all three units, then for each hazard please justify that the PRA model is an adequate representation of all three units. Please include a discussion of SSCs that are shared between units and how these were implicitly or explicitly modeled.

b. If the PRAs were performed for each unit separately, please explain why the risk results are identical.

RAI 23 APLA - Plant Modifications

DELETED

RAI 24 APLA - Modeling Assumptions and Uncertainty – Planned Plant Modifications

Regulatory Position 2.3.4 of RG 1.174 states that the PRA results used to support an application are derived from a PRA model that represents the as-built and as-operated plant to the extent needed to support the application. Consistent with this regulatory position, the PRA should realistically reflect the risk associated with the plant at the time of the application. The discussion for the key assumption/uncertainty identified in LAR Supplement Attachment 13 (page 13-24) implies that the risk results presented in LAR Supplement Attachment 9 reflects a future plant configuration with additional plant modifications to address fire risk, including installation of an additional Steam Generator Makeup Pump. The NRC staff did not find these modifications listed in LAR Attachment 4. Please provide the following:

a. A description of the planned plant modifications along with the status/schedule for implementation of these modifications to reduce fire risk.

b. Results of a sensitivity analysis if these modifications will not be completed before implementation of the RICT program that includes CDF and LERF for each hazard with these modifications removed from the PRA models.

c. Please discuss your plans to complete these modifications prior to implementation of the RICT program.

RAI 25 APLA - Dependencies Between Units

Regulatory Position 2.3.4 of RG 1.174 states that the PRA results used to support an application are derived from a PRA model that represents the as-built and as-operated plant to the extent needed to support the application. Consistent with this regulatory position, the PRA should realistically reflect the risk associated with the plant at the time of the application.

LAR Supplement Attachment 9 indicates that PRA models were only developed for one unit because each of the plant units are "nearly identical". The degree of dependency between units is unclear, though LAR Supplement Attachment 13 (page 13-2) does state that the station blackout generators (SBOGs) is the only plant system modeled in the PRA that is shared between the three units, that simultaneous multiple unit station blackout conditions are screened based on low probability, and that risk management actions (RMAs) for use of Diverse and Flexible Mitigation Capability [FLEX] equipment will be considered when electric power systems to risk significant equipment are unavailable under a RICT. Given that PRAs were only performed for one unit, please explain how the risk associated with a common cause failure event at one unit is addressed for the other units. Please include discussion of the shared systems (e.g., fire water supply, ultimate heat sink, SBOGs, etc.) and resources needed to address events that would trip multiple units. If consideration of this risk was excluded, then please justify this exclusion. Also please address whether RMAs, similar to those being considered for station blackout events, should be considered for other common cause events.

RAI 26 APLA - T.S. 3.8.4

Regulatory Position 1.1.2 of RG 1.177, Revision 1, states that TS requirements can be changed to reflect improved design features in a plant or to reflect equipment reliability improvements that make a previous requirement unnecessarily stringent or ineffective.

In LAR Supplement Attachment 5, Table A5-2, both the high and low estimates of the calculated RICT are reported to be less than 1 hour for proposed Condition C of Technical Specification LCO 3.8.4 DC Sources – Operating. While proposed Condition C is for a loss of function and has a proposed backstop of 24 hours, the Condition is a high risk configuration that would appear to always exceed the acceptance guidelines for incremental CDP [ICDP] and incremental large early release probability [ILERP] in NEI 06-09-A. Please provide technical justification for inclusion of Condition C for LCO 3.8.4 given the reported high risk for this condition.

RAI 27 APLA - Risk Management Actions

a. LAR Attachment 12 states that PVNGS will use either the EPRI CRMP tool for RICT Program calculations. The EPRI tool provides insights such as the important equipment available during the RICT to help in identifying RMAs. However, other insights (e.g., important fire areas) may also be checked for potential RMAs. Please describe how the RICT program guidance considers insights, other than the CRMP tool-generated list of equipment, to identify RMAs.

b. In addition, given component(s) are PRA functional or not PRA functional, please discuss how the RICT program will ensure appropriate RMAs are established.

RAI 28 APLA – Joint Human Error Probability Floor

Guidance in NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA)”, (Table 2-1) April 2005, (ADAMS Accession No. ML051160213) recommends joint human error probability (HEP) values should not be below 1E-05. Table 4-3 of EPRI 1021081, “Establishing Minimum Acceptable Values for Probabilities of Human Failure Events,” provides a lower limiting value of 1E-06 for sequences with a very low level of dependence. The NRC staff notes that underestimation of minimum joint probabilities could result in non-conservative RICTs of varying degrees for different inoperable SSCs.

Given that it is not clear whether or to what extent a dependency analysis was performed as part of the HRA, and whether minimum joint probabilities were applied to combinations of HEPs appearing in the same cutset, please provide the following:

a. Please describe the HRA dependency analysis performed in the PRA and whether it is consistent with NRC accepted guidance. If the approach to performing HRA dependency analysis is not consistent with NRC guidelines, then please justify this departure.

b. Please confirm that each joint HEP value used in the internal events PRA below 1E-06 and each joint HEP used in the fire PRA below 1E-05 includes its own separate justification that demonstrates the inapplicability of the NUREG-1792 lower guideline values. Please provide an estimate of the number of joint HEPs below the guideline values, discuss the range of values, and provide at least two different examples where justification has been developed.

c. If the assessment described in item b) has not been performed or if minimum joint probability “floor” was not applied or the value of the “floor” cannot be justified, then please explain how underestimating joint HEPs impacts the RICT estimate.

RAI 29 APLA - Modeling Assumptions and Uncertainty – Battery Life

Regulatory Position 2.3.4 of RG 1.174 states that the PRA results used to support an application are derived from a PRA model that represents the as-built and as-operated plant to the extent needed to support the application. Consistent with this regulatory position, the PRA should realistically reflect the risk associated with the plant at the time of the application. The discussion and disposition for the key assumption/uncertainty identified in LAR Supplement Attachment 13 (page 13-5) states that a battery life of two hours was assumed in the PRA, but implies that procedures are available for load shedding to extend battery life for up to six hours. The disposition further states that this assumption is conservative for the RICT program. The NRC staff questions that the allowable RICT calculated assuming a two hour battery life is less than (conservative) the allowable RICT calculated assuming a six hour battery life. Please provide the following:

a. Please clarify if plant procedures are available for load shedding and, if so, please describe the conditions under which these procedures would be implemented.

b. If plant procedures are available for load shedding and extending battery life, please provide justification for the conclusion that the PRA assumption of a 2-hour battery life would result in a conservative RICT for a battery removed from service, including providing an example calculation to support this justification.