Exelon Generation®

200 Exelon Way
Kennett Square, PA 19348
www.exeloncorp.com

10 CFR 50.90

June 21, 2018

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

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Subject: Response to Request for Additional Information
License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

References:

1. Letter from David Helker (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," dated February 25, 2016 (ADAMS Accession No. ML16060A223).

2. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Supplement - License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated April 3, 2017 (ADAMS Accession No. ML17094A591).

3. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information, License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated January 11, 2018 (ADAMS Accession No. ML18011A665).

4. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information, License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated January 18, 2018 (ADAMS Accession No. ML18018B340).

5. Letter from Michael Marshall, U.S. Nuclear Regulatory Commission, to Bryan Hanson, Calvert Cliffs Nuclear Power Plant, Units 1 and 2- Request for Additional Information Regarding Risk-Informed Technical Specification Completion Times (CAC Nos. MF7415 AND MF7416; EPID L-2016-LLA-0001)," dated May 22, 2018 (ADAMS Accession No. ML18138A137).

By letter dated February 25, 2016 (ADAMS Accession No. ML16060A223) (Reference 1), as supplemented by letters dated April 3, 2017 (ADAMS Accession No. ML17094A591) (Reference 2), January 11, 2018 (ADAMS Accession No. ML18011A665) (Reference 3), and January 18, 2018 (ADAMS Accession No. ML18018B340) (Reference 4), Exelon Generation Company, LLC (Exelon) submitted a License Amendment Request (LAR) proposing to modify the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2 Technical Specification (TS) requirements to permit the use of risk-informed completion times (RICTs) in accordance with Technical Specification Task Force (TSTF) Traveler - 505, Revision 1, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b.”

By letter dated May 22, 2018 (Reference 5), the NRC staff determined that additional information is needed to complete their review of the LAR. The NRC letter provided 30 days for the response to request for additional information (RAI).

Attachment 1 to this letter provides a restatement of the RAI questions followed by our responses. Attachment 2 provides a revised Insert 2 TS markup in response to the RAI questions which supersedes the previous Insert 2 TS markup provided in Exelon's RAI response letter dated January 11, 2018 (Reference 3). All other TS markups provided in Attachment 2 of Exelon's Reference 3 letter remain valid.

Exelon has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Attachment 1 of the Reference 1 letter. Exelon has concluded that the information provided in this response does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92. In addition, Exelon has concluded that the information in this response does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

There are no regulatory commitments in this response.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the State of Maryland of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 21st day of June 2018.

Respectfully,

James Barstow
Director - Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments:

1. Response to Request for Additional Information Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."
2. Proposed Technical Specifications Changes (Markups)

cc: USNRC Region I, Regional Administrator
USNRC Project Manager, CCNPP
USNRC Senior Resident Inspector, CCNPP
S. T. Gray, State of Maryland

ATTACHMENT 1

License Amendment Request

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Response to Request for Additional Information
Revise Technical Specifications to Adopt Risk Informed
Completion Times TSTF-505, Revision 1, "Provide Risk-Informed
Extended Completion Times - RITSTF Initiative 4b."

By letter dated February 25, 2016 (ADAMS Accession No. ML16060A223) (Reference 1), as supplemented by letters dated April 3, 2017 (ADAMS Accession No. ML17094A591) (Reference 2), January 11, 2018 (ADAMS Accession No. ML18011A665) (Reference 3), and January 18, 2018 (ADAMS Accession No. ML18018B340) (Reference 4), Exelon Generation Company, LLC (Exelon) submitted a License Amendment Request (LAR) proposing to modify the Calvert Cliffs Nuclear Power Plant (Calvert Cliffs), Units 1 and 2 Technical Specification (TS) requirements to permit the use of risk-informed completion times (RICTs) in accordance with Technical Specification Task Force (TSTF) Traveler - 505, Revision 1, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b.”

By letter dated May 22, 2018 (Reference 5), the NRC staff determined that additional information is needed to complete their review of the LAR. Below is a restatement of the questions followed by our responses.

RAIs

Section 36(c)(2) of Title 10 of Code of Federal Regulation requires in part that limiting conditions of operations be included in TSs and that licensees shall follow any remedial action permitted by the TS until the condition can be met. The TSs for Calvert Cliffs, Units 1 and 2, contain limiting conditions of operations that prescribe completion times for remedial actions. The licensee has proposed using its probabilistic risk assessment (PRA) to determine risk-informed completion times that may be used in lieu of the prescribed completion times. Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (ADAMS Accession No. ML100910006) describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations. Revision 1 of RG 1.177, "An Approach for Plant-Specific, Risk Informed Decision-making: Technical Specifications," May 2011 (ADAMS Accession No. ML100910008) describes an acceptable risk-informed approach specifically for assessing proposed TS changes. To ensure that any remedial actions are completed in a timely manner, consistent with RG 1.174 and RG 1.177, the PRA models used in the calculation of the risk informed completion times need to be based on the as-built, as-operated and maintained plant, and reflect operating experience at the plant.

20. Enclosure 4 of the letter dated February 25, 2016, states that a total seismic core damage frequency (CDF) contribution of 1.1E-6/year and a seismic large early release frequency (LERF) contribution of 1.1E-7/year will be added to the configuration specific delta CDF and delta LERF from the internal events and fire initiating event contributions to estimate the risk-informed completion time. The LAR states that these seismic estimates are based on the re-evaluated seismic hazard for Calvert Cliffs performed in response to the Near-Term Task Force 2.1 (ADAMS Accession No. ML14099A196) and an estimated plant level high confidence of low probability of failure (HCLPF) of 0.27 the acceleration due to Earth's gravity (g) peak ground acceleration (PGA) as used in the 2003 LAR entitled "Extension of Diesel Generator Required Action Completion Time" (ADAMS Accession No. ML031360410).

a. The staff notes that the Calvert Cliffs Expedited Seismic Evaluation Process Report (ESEP), submitted to the NRC on December 17, 2014 (ADAMS Accession No. ML14365A138), indicates that certain components, such as safety injection tanks, motor control centers, electrical buses, and main control room panels, would fail in a seismic event through interaction with nearby block walls, and those components were assigned a lower HCLPF of 0.175g due to the block wall lower capacity. This lower HCLPF could increase the seismic CDF and LERF estimates provided in the LAR.

   Justify the plant level HCLPF of 0.27g PGA, given the noted block walls failures at 0.175g indicated in the ESEP, or provide, with justification, updated seismic CDF and LERF estimates.

   Response

   The limiting HCLPF of 0.27g PGA that was used in determining the RICT seismic penalty was based on the evaluation performed for the 2003 EDG completion time extension request (Reference 6), which in turn was based on a seismic PRA performed for the IPEEE. It therefore accounts for the integrated plant response to seismic events and is a reasonably realistic representation of plant capability to mitigate seismic risk. The Expedited Seismic Evaluation Process (ESEP) evaluation does reflect a more recent examination of specific plant structures, systems, and components (SSCs), but was performed in a conservative manner that was intended to demonstrate minimum existing plant capability relative to the design basis for a specified safe shutdown path defined for the Fukushima response and FLEX implementation. The ESEP calculations were therefore conservative, based on a prescribed rather than realistic definition of the site hazard, and included simplifying assumptions regarding interaction-related failures. The limiting HCLPF approach applied for the RICT seismic penalty calculations assumes that a failure of a component that can be represented as having seismic capacity at that HCLPF level leads directly to core damage (CD). However, there are few SSCs whose failure would lead to seismic CD with any significant probability. Even common failure of all emergency diesel generators (EDGs) would not lead directly to CD, especially in light of the post-Fukushima FLEX mitigating strategies now in place. The IPEEE seismic PRA did not identify similar low HCLPF component issues leading to CD. Thus, the HCLPF values estimated for the ESEP report have significant conservatisms and are likely significantly underestimated.

   However, given that the Calvert Cliffs seismic PRA has not been maintained current with the as-operated plant, a review has been performed of the SSCs listed in Attachment C of the ESEP report (Reference 7) (for Unit 1; the list is similar for Unit 2) for which the estimated HCLPF is less than the 0.27g PGA value used for the RICT seismic penalty calculation. One point of clarification, relative to RAI 20, is that the safety injection tanks (SIT) are identified in the ESEP report as having an anchorage failure mode, not a block wall interaction failure mode as noted in the question, and were assigned a HCLPF of 0.21g. Of the 26 components with ESEP HCLPF values less than 0.27g, 15 are either not modeled in the Calvert Cliffs seismic or internal events PRAs (e.g., components such as Spent Fuel Pool instrumentation) or would not directly lead to CD if failed during a seismic event. Thus, the ESEP HCLPF values for these 15 components would not be an appropriate limiting HCLPF for use in the RICT seismic CDF impact calculation.

For the remaining components with an estimated HCLPF less than 0.27g, the review indicated that in most cases there would be additional mitigation equipment or actions that could be modeled in a true seismic risk model (as opposed to the simplified ESEP success path approach) such that the ESEP HCLPF would not be limiting. However, it is not possible to make this determination definitively without substantial new analysis. Because the seismic hazard for the Calvert Cliffs site is relatively low, the impact of applying a limiting HCLPF lower than the current 0.27g for the seismic RICT calculation will not have an unacceptable impact on RICT calculations. Therefore, until such time as a more refined analysis of seismic capacities of relevant plant SSCs may be undertaken, Exelon will use the ESEP block wall estimated seismic capacity of 0.175g as the limiting HCLPF for the seismic RICT impact calculation. The convolution of the Calvert Cliffs seismic hazard curve with a 0.175g HCLPF results in a seismic CDF adjustment of 3.7E-6/yr. rather than the adjustment of 1.1E-6/yr. previously reported in the LAR. This change does not affect the example calculated RICTs provided in Enclosure 1 of the LAR. The seismic LERF adjustment based on this new seismic CDF adjustment and the 0.1 conditional large early release probability (CLERP) value discussed in the response to Part c. of this question, will be 3.7E-7/yr.

Note that use of the Calvert Cliffs ESEP HCLPFs as described above for the initial implementation of the RICT program does not imply that Exelon intends to use ESEP information for any other risk-informed applications.

b. The 2003 LAR for extension of emergency diesel generator (EDG) completion times shows differences in estimated seismic CDF and LERF between Unit 1 and Unit 2. According to the NRC Technical Evaluation Report for the Individual Plant Examination for External Events, because the EDGs dedicated to Unit 2 are more dependent on service water cooling, which has a low fragility, the CDF value is higher for Unit 2 than for Unit 1.

Explain and justify how the seismic CDF and LERF estimates apply to both units.

Response

The plant level HCLPF values used to determine the RICT seismic CDF impact were calculated based on the individual unit seismic CDF estimates from the 2003 EDG completion time (CT) extension requests (Reference 6) and the IPEEE seismic hazard curve. A plant level HCLPF of 0.27g was calculated for Unit 2 based on the higher Unit 2 seismic CDF of 1.2E-5/yr. from the 2003 EDG CT extension LAR. The Unit 2 plant level HCLPF was calculated assuming a combined uncertainty (βc) term value of 0.4 (a standard assumption for such calculations) and iterating on the median seismic capacity (A_m) term until the fragility curve represented by the HCLPF integrated over the IPEEE hazard curve equals the core damage frequency (CDF). A plant level HCLPF of 0.3g was calculated from the lower Unit 1 Seismic CDF of 9.9E-6/yr. from the 2003 EDG CT extension LAR using the same approach. Thus, the plant level HCLPF of 0.27g represents a limiting value for both units and was applied to the RICT seismic

CDF impact for both units. Note that the difference in Seismic CDF between the two units in the IPEEE is a function of the capability of the DGs. The two most important DGs from a seismic perspective (for Unit 1) are the 1A and 0C DGs. These DGs are air cooled and do not need service water cooling. The 1A DG can provide Unit 1 support and limited Unit 2 support through the AFW and electrical cross ties. The 0C DG can support either Unit 1 or 2. This is the primary reason for the lower seismic risk of Unit 1 as compared to Unit 2.

The plant has been significantly improved from a seismic perspective since the IPEEE evaluation. Improvements have been implemented for the 0C Diesel Generator (DG) to ensure that the 0C DG has a minimum HCLPF of 0.3g. The resultant Unit 1 and Unit 2 CDFs reflecting these improvements are reflected in Table 18 of the DG completion time extension submittal. However, despite these improvements, the 0.27g HCLPF was retained as a conservative value for the LAR.

To estimate the seismic LERF, the LAR assumes a 0.1 conditional large early release probability (CLERP) for seismic events, based on the internal events LERF to CDF ratio. The staff notes that a seismic event could lead to seismic-specific failures of structures, systems, and components, resulting in additional LERF sequences that are not in the internal events probabilistic risk assessment (PRA) model or potentially converting non-LERF sequences in the internal events PRA model to seismic LERF sequences. The LAR does not provide sufficient justification for the selected CLERP being able to capture or bound such considerations.

c. Justify the assumed value of 0.1 for CLERP. In the justification, explain why the containment is not expected to fail and other containment failure or bypass scenarios are not expected to be impacted by seismic events and therefore, would not noticeably affect the assumed 0.1 CLERP.

<u>Response</u>

The conditional large early release probability (CLERP) calculation performed for the LAR derived an average of CLERP (i.e., large early release frequency divided by core damage frequency) over the non-flooding PRA internal initiating events other than direct containment bypass events. The direct bypass events (steam generator tube rupture, interfacing system LOCAs) were not included because the LERF contribution from these events is independent of risk informed Technical Specification completion times, i.e., the bypass occurs regardless of containment or containment isolation response. Further, from a seismic perspective, the involved components (i.e., steam generator tubes, RCS pressure boundary valves) would generally be treated as seismically rugged in a seismic PRA.

Seismic-induced containment structural failure is not included in the internal events PRA containment isolation model. However, the model does account for a fraction of time during operation when containment isolation may be failed due to pre-existing maintenance errors or mechanical failures which allow a direct release pathway outside containment, and this contribution is reflected in the internal events PRA-based CLERP evaluation.

Seismic-induced failure of containment or failure of containment isolation was evaluated in the Calvert Cliffs IPEEE. Section 3.1.6 of the Calvert Cliffs IPEEE (Reference 8) identifies a limiting HCLPF of 0.70g (A_m=2.31g) for the containment. Depending on the failure mode and location, such a failure could be considered to lead to a large early release given core damage. A containment HCLPF of 0.70g is much higher than the limiting plant level HCLPF of 0.27g, such that seismic containment failures do not challenge an assumed 0.1 CLERP.

Seismic-induced failure of Containment Isolation is discussed in Section 3.1.5.2 (for Top Event "LL") and Section 3.1.6 (first paragraph) of the Calvert Cliffs IPEEE. The IPEEE analysis determined that all containment penetrations and containment isolation valves whose failure could lead to a release were screened at a HCLPF of 0.5g based on the walkdowns conducted. A more realistic HCLPF for containment penetrations and containment isolation valves would likely be much higher and support lower conditional probabilities for containment isolation failure over the range of g-levels, such that seismic containment isolation failures also do not challenge an assumed 0.1 CLERP.

Regarding any other scenarios of risk significance, the plant is designed so that most systems related to containment isolation would fail in a desirable (isolated) state during an earthquake. Containment isolation air operated valves (AOV) fail closed on loss of support and are generally seismically rugged. Certain containment isolation motor operated valves (MOV) require power for closure, but most such valves are typically already closed. The Calvert Cliffs internal events PRA models five MOVs as being required to close for containment isolation (Reference 9, Calvert Cliffs PRA Primary Containment System Notebook). One MOV is in the containment air line such that failure of containment isolation requires a pipe rupture and a check valve failure in addition to failure of the MOV to close. Two additional MOVs are the isolation valves in the containment drain lines, which would be open only a very small fraction of the time, i.e., several hours per year. The remaining two MOVs are the hydrogen purge line isolation valves, which are normally closed and would also only be open for a very small fraction of the time. The Interfacing System LOCA (ISLOCA) analysis in Reference 10 (Calvert Cliffs PRA Initiating Event Notebook) was also reviewed. Each potential path outside of Containment for ISLOCAs was determined to require either failure of check valves and normally closed MOVs, or require failure of seismically rugged AOVs.

Given the above, seismic failure of containment isolation is effectively independent of seismic failures leading to core damage. Convolution of the seismic hazard with the adjusted limiting CD HCLPF of 0.175g (per the response to Part a.) and the conservative limiting containment isolation failure HCLPF of 0.5g results in an estimated seismic LERF adjustment factor of 1.9E-7/yr., or approximately 5% of the seismic CDF adjustment factor. Even though these seismic failure values are believed to be very conservative, it is recognized that there are uncertainties associated with this evaluation. Therefore, Exelon will retain the 10% seismic CLERP assumption, i.e., use a value of 3.7E-7/yr. for the seismic LERF adjustment.

As stated in the response to Part a. of this question, Exelon may in the future perform refined evaluations of the ESEP report HCLPF values and adjust both the seismic CDF and seismic LERF adjustment factors.

21. In RAI 10 (see letter dated November 13, 2017), the staff asked the licensee to explain how common cause failures (CCFs) are included in the PRA model and how the treatment of CCF either meets the guidance in RG 1.177 or meets the intent of this guidance when quantifying a risk-informed completion time (RICT) for preventative maintenance for components from a CCF group of three or more components. The response to the RAI states "common cause failures are modeled as separate basic events, with common cause combinations in the fault tree as different basic events." The licensee further stated that the "common cause grouping is not dynamically changed when a component is removed from service for preventative maintenance" and that this is appropriate because "the component, though not out of service for a reason subject to common cause failure, remains a participant in the common cause events for the remainder of the component operation." It is unclear how the out of service component "remains a participant in the common cause events for the remainder of the component operation" and; therefore, it is unclear how the intent of RG 1.177 is met.

    Explain clearly how CCFs are modeled in the Calvert Cliffs PRA and justify why adjusting the common cause grouping is not necessary for preventative maintenance. In the explanation, include examples of fault trees for a CCF group of three components and the associated numerical results.

    <u>Response</u>

    Common cause failures (CCFs) are addressed in the Calvert Cliffs Probabilistic Risk Assessment (CCPRA) by incorporating appropriate common cause basic events in the integrated plant fault tree model.

    The alpha-factor method is used to quantify CCFs. An example of a 3-component system with staggered testing, as used in the CCPRA in the PRA model is as follows:

$$Q_1^3 = Q_t \qquad (independent\ failure\ probability\ of\ 1\ component\ in\ a\ group\ of\ 3)$$
$$Q_2^3 = \frac{1}{2}\alpha_2 Q_t \quad (dependent\ failure\ probability\ of\ 2\ components\ in\ a\ group\ of\ 3)$$
$$Q_3^3 = \alpha_3 Q_t \quad (dependent\ failure\ probability\ of\ 3\ components\ in\ a\ group\ of\ 3)$$

    Where:

    - Q_t is a component's total failure probability, consisting of failure from both independent and dependent (i.e., common) causes;
    - *α* is the alpha parameter from the INL CCF data; and
    - α_k denotes *k* failed components in the common cause group.

    This approach is consistent with NUREG/CR-5485 (Reference 11), except that, for the CCPRA independent failure basic events (Q_1), the entire Q_t is used, rather than multiplying by the factor of α_1. That is, the independent failure events conservatively include the contribution from CCF events.

The CCF Tool in CAFTA software is used in the CCPRA to automatically identify the CCF combinations for a given CCF group, calculate the CCF event probabilities from alpha factors and failure probability, and place the appropriate CCF basic events into the fault tree. In some cases, the CCF Tool is not used and manual calculations are performed, using similar methodologies.

Figure 1 is a typical CCPRA CAFTA fault tree representation of a 3-train system where failure of 3-of-3 components fails the system top event. In this representation, the common cause event CCW0CCMZS_1_2 would fail the start function of pumps 11 and 12. Similarly, common cause event CCW0CCMZS_1_2_3 would fail all three pumps.

Using the example in Figure 1, when one pump is out of service for preventative maintenance, the system top event failure criterion is now effectively 2-of-2 components because one of the three inputs to the AND gate is made true because the pump that is out of service for maintenance cannot start. This is shown in Figures 2 and 3. Note that Figure 3 retains the CCF and independent events related to the remaining two components.

Adjustments to the CCF grouping or CCF probabilities are not necessary when a component is taken out-of-service for preventative maintenance:

- The component is not out-of-service for reasons subject to a potential common cause failure, and so the in-service components are not subject to increases in common cause probabilities.

- CCF relationships are retained for the remaining in-service components. For example, see event CCW0CCMZS_2_3 in Figure 3.

- The net failure probability for the in-service components includes the CCF contribution of the out-of-service component. This CCF contribution from the out-of-service component is conservatively retained two ways:

    1. The independent failure event used in the model includes both the independent and dependent failure probabilities (i.e., $Q_1 = Q_t$).

    2. The CCF event probabilities that include the out-of-service component are retained. For example, see event CCW0CCMZS_1_2 in Figure 3.

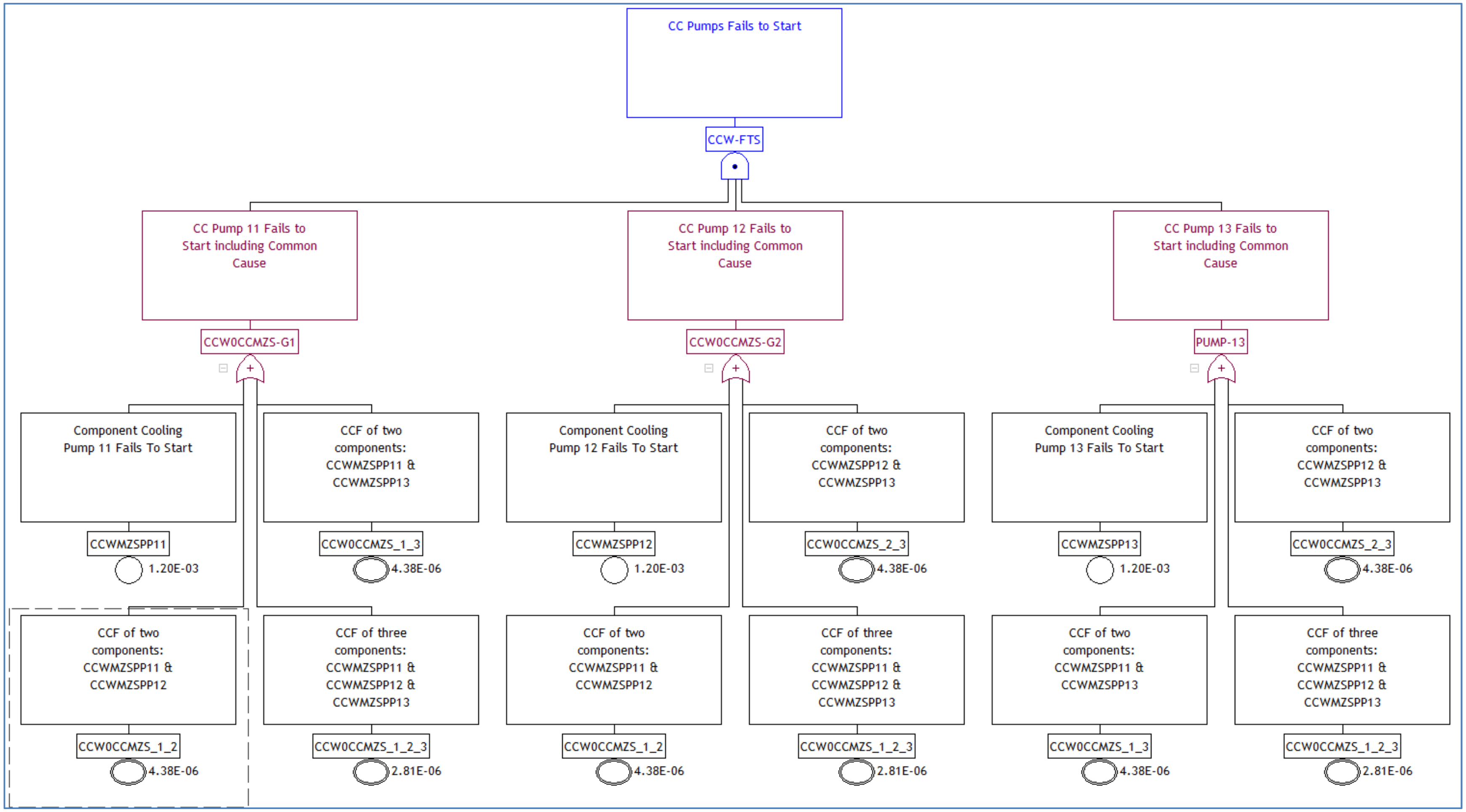

Figure 1 – Example 3-Train System Logic in the Model

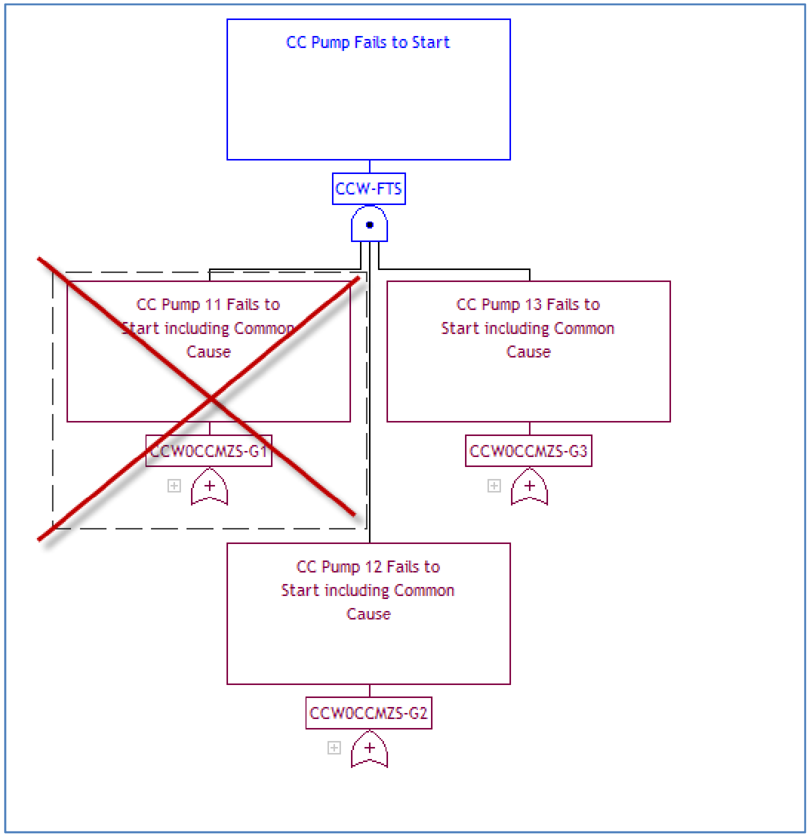

Figure 2 – 3-Train System Representation of Taking One Train Out-of-Service

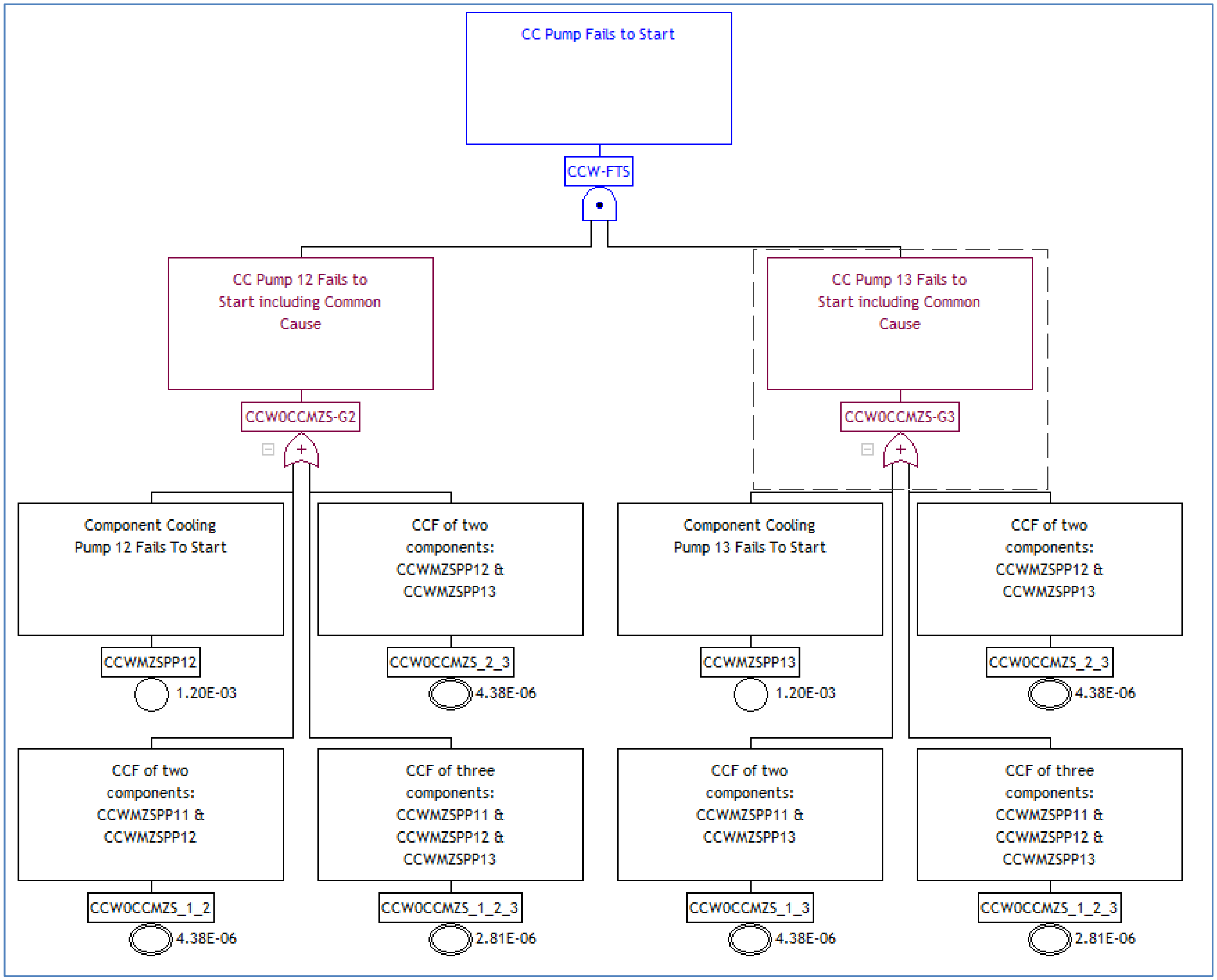

Figure 3 – 3-Train System Effective Logic with One Train Out-of-Service

As described in RG 1.177, Section A-1.3.2.2, the CCF term should be treated differently when a component is taken down for preventive maintenance (PM) than as described for failure of a component. For PMs, the common cause factor is changed so that the model represents the unavailability of the remaining component. In the example provided in the RG for a 2-train system, the CCF event can be set to zero for PMs. This is done so that the model represents the unavailability of the remaining component, and not the common cause multiplier. The Calvert Cliffs approach is conservative in that for a 2-train system, the CCF event is retained for the component removed from service. Likewise, for systems with three or more trains, the CCF events that are related to the out-of-service component are retained.

The Vogtle RICT Amendment Safety Evaluation (Reference 12) describes the Vogtle approach for modeling common cause events with planned inoperability:

> “For planned inoperability, the licensee sets the appropriate independent failure to “true” and makes no other changes while calculating a RICT.”

The Calvert Cliffs approach described above is the same as this Vogtle approach.

It is recognized that other modifications could be made to CCF factors for planned maintenance, particularly for common cause groups of three or more components. For example, in the Safety Evaluation in the Vogtle RICT Amendment, the NRC identifies a possible planned maintenance CCF modification to “modify all the remaining basic event probabilities to reflect the reduced number of redundant components.”

Like Vogtle, the Calvert Cliffs CCF approach is a straightforward simplification that has inherent uncertainties. In the context of modifying CCF basic events for PMs, the Vogtle SE states the following:

> “The NRC staff also notes that common cause failure probability estimates are very uncertain and retaining precision in calculations using these probabilities will not necessarily improve the accuracy of the results. Therefore, the NRC staff concludes that the licensee's method is acceptable because it does not systematically and purposefully produce non-conservative results and because the calculations reasonably include common cause failures consistent with the accuracy of the estimates.”

The Calvert Cliffs approach for CCF during PMs is the same as the Vogtle approach. Therefore, the Calvert Cliffs CCF approach is acceptable for RICT calculations, and adjusting the common cause grouping is not necessary for PMs.

22. In RAI 11.a (see letter dated November 13, 2017), the staff requested the licensee to confirm and describe how the treatment of CCF in the case of emergent failures either meets the guidance in RG 1.177 or meets the intent of this guidance when quantifying a RICT. In response to RAI 11.a the licensee stated that risk management actions will be implemented. However, in the response to RAI 11.b the licensee added the option to "numerically account for the increased possibility of CCF in the RICT calculation" to the text for the Calvert Cliffs, Units 1 and 2, TS Administrative Section 5.5.18, without further justification on how it meets the intent of the guidance in RG 1.177 requested in RAI 11.a.

    a. Explain how the task to "numerically account for the increased possibility of CCF in the RICT calculation" will be performed for emergent failures.

    <u>Response</u>

    If a numeric adjustment is performed, the RICT calculation shall be adjusted to numerically account for the increased possibility of CCF in accordance with RG 1.177, as specified in Section A-1.3.2.1 of Appendix A of the RG. Specifically, when a component fails, the CCF probability for the remaining redundant components shall be increased to represent the conditional failure probability due to CCF of these components in order to account for the possibility the first failure was caused by a common cause mechanism.

    b. Justify how the treatment of CCF meets the intent of the guidance in RG 1.177.

    <u>Response</u>

    See response to Item a. above.

23. In RAI 14 (see letter dated November 13, 2017), the staff requested explanation on how the containment spray and the containment cooling systems are modeled, and how a RICT based on CDF and LERF can be quantitatively determined for these systems. In response to RAI 14 the licensee stated that both systems are explicitly modeled in the PRA and that the PRA modeling "includes system components, such as pumps, valves and heat exchangers, and system dependencies, such as electrical and cooling water systems." The licensee further explained that the PRA success criteria is one of the two headers for the containment spray system, and two out of four air coolers for the containment air recirculation and cooling system. The licensee stated that these systems "can be numerally quantified for impact on CDF and LERF," however the licensee did not explain how these systems impact core damage or large early release. Since the containment spray and containment cooling systems are generally related to the long-term release sequences (not large early release), it is not immediately clear to the NRC staff the impact that these systems have on the core damage and large early release in the licensee's PRA model. Further, the iodine removal function of the containment spray system is not usually captured in the PRA.

    Explain and justify how these systems impact CDF and LERF.

<u>Response</u>

The containment spray system (CSS) is actuated by the containment spray actuation signal upon high containment pressure. After the recirculation actuation signal (RAS), CSS is aligned through the shut-down heat exchangers.

The containment air coolers (CAC) are also actuated by the containment spray actuation signal on high containment pressure. This system is cooled by the service water system.

In the core damage accident sequence analysis, containment temperature and pressure control are required when the reactor coolant system boundary is breached or once-through-core-cooling is required. In either case, containment temperature and pressure control are accomplished by CSS or CAC. In the PRA accident sequence event tree, this function is required to provide the ultimate heat rejection outside of containment.

Success of CSS or CAC can directly impact LERF, as the Level 1 (core damage) accident sequence leads directly into the Level 2 (containment) accident sequences. Therefore, changes in CDF contribution are reflected in the Level 2 results including LERF. In the Level 2 PRA accident sequences, CSS or CAC is also explicitly questioned for sequences that lead to late containment failure, not large early release sequences.

If the design basis success criteria parameter values can be met for CSS, all the different functions are satisfied with sufficient margins because the safety margins are included in the design basis parameter selection.

24. Section 36(c)(5) of 10 CFR Part 50 requires, in part, that TSs contain administrative controls related to procedures and reporting necessary to assure operation of the facility in a safe manner. The licensee is proposing that a new program called Risk-Informed Completion Time Program be added to TS Section 5, "Administrative Controls," that describes the controls on the calculated risk-informed completion time that may be used in lieu of the prescribed completion time. Appropriate controls are needed to ensure that any changes to the PRA models used in the calculation of the risk-informed completion time be based on methods approved by the NRC, and be based on the as-built, as-operated and maintained plant, and reflect the operating experience at the plant, consistent with the guidance in RG 1.174, Revision 2.

    In RAI 15 (see letter dated November 13, 2017), the staff provided wording for a proposed license condition, consistent with the license condition included in NRC-approved Amendment Nos. 188 and 171 for the pilot risk-informed completion time LAR (ADAMS Accession No. ML15127A669). In response, the licensee proposed the following text to be added to the Calvert Cliffs, Units 1 and 2, TS Administrative Section 5.5.18:

    a. *A RICT must be calculated using the PRA and non-PRA methods approved by the NRC, including internal events, internal floods, and fire PRA. Changes to these PRA and non-PRA methods require prior NRC approval. The PRA maintenance and upgrade process will validate that changes to the PRA models used in the RICT program follow the guidance in Appendix 1-A of ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications."*

b. *A report shall be submitted following each PRA upgrade and associated peer review, and prior to using the upgraded PRA to calculate* a *RICT. The report shall describe the scope of the upgrade.*

The license condition approved for the pilot contains both "methods" and "approaches." The proposed TS 5.5.18 text does not appear to be consistent with the approved precedent. Propose TS 5.5.18 text consistent with the approved precedent or the draft TSTF-505, Revision 2 (ADAMS Accession No. ML17290A003), or provide detailed technical justification for your proposal. This justification should describe, with examples, what constitutes a PRA and non-PRA methods and approaches that if changed, would require prior NRC approval.

Response

To address this issue, Exelon proposes to replace the previous proposed wording for Item e. in TS Section 5.5.18 with the following wording which is consistent with the approved precedent:

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the completion times must be PRA methods used to support Amendment Nos. XXX/XXX, or other methods approved by the NRC for generic use. Any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

Note: Item f. previously proposed in TS Section 5.5.18 was associated with the original proposed Item e. Since the original Item e. is being replaced with the above, there is no longer a need for Item f. Therefore, Item f. is being deleted from TS Section 5.5.18 as reflected in the proposed TS markup. In addition, the proposed TS markup also corrects a typographical error in TS Section 5.5.18 which changes NEI 06-09, Revision 0 (first paragraph) and NEI 06-09-A (Item c.) to NEI 06-09, Revision 0-A.

See Attachment 2 for the proposed TS markup.

References

1. Letter from David Helker (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," dated February 25, 2016 (ADAMS Accession No. ML16060A223).

2. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Supplement - License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated April 3, 2017 (ADAMS Accession No. ML17094A591).

3. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information, License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated January 11, 2018 (ADAMS Accession No. ML18011A665).

4. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information, License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated January 18, 2018 (ADAMS Accession No. ML18018B340).

5. Letter from Michael Marshall, U.S. Nuclear Regulatory Commission, to Bryan Hanson, Calvert Cliffs Nuclear Power Plant, Units 1 and 2- Request for Additional Information Regarding Risk-Informed Technical Specification Completion Times (CAC Nos. MF7415 AND MF7416; EPID L-2016-LLA-0001)," dated May 22, 2018 (ADAMS Accession No. ML18138A137).

6. “Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318 License Amendment Request: Extension of Diesel Generator Required Action Completion Time,” Constellation Energy Group, May 12, 2003 (ADAMS ML031360410).

7. Calvert Cliffs Expedited Seismic Evaluation Process Report (ESEP), December 17, 2014, ADAMS Accession No. ML14365A138.

8. RAN 97-031, “Calvert Cliffs Individual Plant Examination External Events” (IPEEE), Baltimore Gas & Electric, Sections 3.1.5.2 and 3.1.6.

9. Calvert Cliffs Nuclear Power Plant Unit 1 and 2 Primary Containment System Notebook, 2015 PRA Update, CA-PRA-005.059 Revision 1, November 2015.

10. Calvert Cliffs Nuclear Power Plant Unit 1 and 2 Initiating Events Notebook, 2015 PRA Update, CA-IE-001 Revision 3, November 2015.

11. Idaho National Engineering and Environmental Laboratory, NUREG/CR-5485, “Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment,” November 1998.

12. Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Implementation of Topical Report Nuclear Energy Institute NEI 06-09, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specification (RMTS) Guidelines”, Revision 0-A (CAC Nos. ME9555 And ME9556), ML15127A669.

ATTACHMENT 2

License Amendment Request Supplement

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Docket Nos. 50-317 and 50-318

Response to Request for Additional Information
Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b."

Proposed Technical Specification Changes (Markups)

5.5 Programs and Manuals

inleakage, and assessing the CRE boundary as required by paragraphs c and d respectively.

5.5.18 ~~Not Used~~ Insert 2

5.5.19 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

Insert 2

5.5.18 Risk Informed Completion Time Program

This program provides controls to calculate a Risk Informed Completion Time (RICT) and must be implemented in accordance with NEI 06-09, Revision 0-A, "Risk-Managed Technical Specifications (RMTS) Guidelines." The program shall include the following:

a. The RICT may not exceed 30 days;

b. A RICT may only be utilized in MODE 1, and 2;

c. When a RICT is being used, any change to the plant configuration, as defined in NEI 06-09, Revision 0-A, Appendix A, must be considered for the effect on the RICT.

   1. For planned changes, the revised RICT must be determined prior to implementation of the change in configuration.

   2. For emergent conditions, the revised RICT must be determined within the time limits of the Required Action Completion Time (i.e., not the RICT) or 12 hours after the plant configuration change, whichever is less.

   3. Revising the RICT is not required if the plant configuration change would lower plant risk and would result in a longer RICT.

d. If the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the Completion Time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:

   1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or

   2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

e. The risk assessment approaches and methods shall be acceptable to the NRC. The plant PRA shall be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant, as specified in Regulatory Guide 1.200, Revision 2. Methods to assess the risk from extending the completion times must be PRA methods used to support Amendment Nos. XXX/XXX, or other methods approved by the NRC for generic use. Any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.