UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 6, 2022

Mr. Robert Schuetz
Chief Executive Officer
Energy Northwest
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76 North Power Plant Loop
P.O. Box 968
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SUBJECT: COLUMBIA GENERATING STATION − REGULATORY AUDIT AGENDA AND QUESTIONS FOR LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4B" (EPID L-2022-LLA-0023)

Dear Mr. Schuetz:

By letter dated February 3, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22034A992), Energy Northwest (the licensee) submitted a license amendment request for Columbia Generating Station (Columbia). The proposed amendment would modify Columbia's Technical Specification requirements to permit the use of risk-informed completion times in accordance with Technical Specifications Task Force (TSTF) Traveler TSTF-505, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b," Revision 2.

On March 18, 2022 (ADAMS Accession No. ML22068A234), the U.S. Nuclear Regulatory Commission (NRC) staff issued an audit plan that conveyed intent to conduct a regulatory audit to support its review of the subject license amendment request. In the audit plan, the NRC staff requested an electronic portal setup and provided a list of documents to be added to the portal.

The NRC staff has performed an initial review of these documents and developed a list of audit questions. The proposed date for the audit is from Monday, August 1, 2022, through Thursday, August 4, 2022. The proposed agenda for the audit is provided as enclosure 1, and the list of Audit questions are provided as enclosure 2 to this letter.

If you have any questions, please contact me at (301) 415-8371 or by email at Mahesh.Chawla@nrc.gov.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Audit Agenda
2. Audit Questions

cc: Listserv

AUDIT AGENDA

COLUMBIA GENERATING STATION TSTF-505 LICENSE AMENDMENT REQUEST

AUGUST 1, 2022 – AUGUST 4, 2022

Day 1 – Monday, August 1, 2022

10:00 a.m. to 1:00 p.m. Eastern Time (ET)

- Entrance Meeting
    - Opening comments by the U.S. Nuclear Regulatory Commission (NRC) staff and Energy Northwest (the licensee)
    - Introductions and logistics

- Real-time risk (RTR) model demonstration, risk management action times and risk-informed completion times (RICT) sample calculations by Energy Northwest

- Discuss RTR model and calculation of RICT estimates
    - Dialog item – RTR model benchmarking, update, and interim evaluation process.
    - Impact of Seasonal Variations (Question (Q) Number 3 (Q3) - Probabilistic Risk Assessment (PRA) Licensing Branch A/C (APLA/APLC))
    - In-Scope LCOs [Limiting Conditions for Operation] and Corresponding PRA Modeling (Q5 - APLA/APLC)

2:00 p.m. to 5:00 p.m. ET[1][2]

- Success Criteria for TS [Technical Specification] 3.5.1.E (Q11 - APLA)
- Discussion of surrogate questions (Q7 through Q10 - APLA)
- Discuss disposition of assumptions and sources of uncertainty (Q1 - APLA/APLC)
- Proposed Administrative Controls for the RICT Program (Q13 - APLA)
- Summary of the day (4:30 p.m.)

Day 2 – Tuesday, August 2, 2022

10:00 a.m. to 1:00 p.m. ET

- Summary of previous day and review open items
- Missing Information from Table E1-1 (Q6 - APLA)
- Feasibility of Diverse and Flexible Coping Strategies (FLEX) Operator Actions (Q2 - APLA/APLC)
- PRA modeling and uncertainty of FLEX equipment and actions (Q12 - APLA/APLC)

[1] If discussion topics are completed early, additional discussions for the day may be on the next day's agenda items.
[2] If the afternoon session runs long, the remaining questions will be covered during the afternoon of Day 3 or morning of Day 4.

- Performance Monitoring (Q4 - APLA)

2:00 p.m. to 5:00 p.m. ET[1,2]

- Dispositions of FPRA [Fire PRA] Model Assumptions and Sources of Uncertainty (Q1 - PRA Licensing Branch B (APLB))
- Deviations from NRC Endorsed Guidance as Source of Modeling Uncertainty (Q2 - APLB)
- Summary of the day (4:30 p.m.)

Day 3 – Wednesday, August 3, 2022

10:00 a.m. to 1:00 p.m. ET

- Summary of previous day and review open items
- Discuss Technical Specifications Branch (STSB) audit questions (Q1 and Q2 - STSB)
- Discuss Instrumentation and Controls Branch (EICB) audit question (Q1 - EICB)
- Discuss Electrical Engineering Branch (EEEB) audit questions (Q1 through Q11 - EEEB)

2:00 p.m. to 5:00 p.m. EDT[3,4]

- Follow up on any remaining or new open action items
- Summary of the day (4:30 p.m.)

Day 4 – Thursday, August 4, 2022

10:00 a.m. to 1:00 p.m. ET

- Summary of previous day and review remaining open items

2:00 p.m. to 5:00 p.m. ET

- Follow up on any remaining or new open action items
- Technical summary meeting with licensee's audit team (4:00 p.m.)
- Formal exit meeting (5:00 p.m.)

[3] If the afternoon session runs long, the remaining questions will be covered during the morning of Day 4.
[4] If all discussion topics are completed early, may proceed with the technical summary and formal exit meeting on the next day's agenda.

AUDIT QUESTIONS

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT

TSTF-505, REVISION 2

ENERGY NORTHWEST

COLUMBIA GENERATING STATION

DOCKET NO. 50-397

By application dated February 3, 2022 (Reference 1), Energy Northwest (the licensee) submitted a license amendment request (LAR) for Columbia Generating Station (Columbia, CGS). The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, dated July 2, 2018 (Reference 2). The U.S. Nuclear Regulatory Commission (NRC) issued a final revised model safety evaluation (SE) (Reference 3) approving TSTF-505, Revision 2, on November 21, 2018.

The NRC staff determined that the following information is needed to complete its review.

Probabilistic Risk Assessment (PRA) Licensing Branch A (APLA) and C (APLC) Audit Questions

Question 1 (APLA/APLC) - Dispositions of Internal Events PRA Model Assumptions and Sources of Uncertainty

Section 4, item 10 of the NRC final SE (Reference 4) for Nuclear Energy Institute (NEI) 06-09-A, "Risk-Informed Technical Specification Initiative 4b, Risk Managed Technical Specifications (RMTS) Guidelines" (Reference 5), requires that the LAR provide a discussion of how the key assumptions and sources of uncertainty were identified, and how their impact on the RMTS was assessed and dispositioned. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," Final Report (Reference 6) presents guidance on the process of identifying, characterizing, and qualitative screening of model uncertainties.

Several documents on the Columbia portal provide the dispositions for identified key assumptions and sources of uncertainty. Additional information is needed for the NRC staff to determine that the key assumptions and sources of uncertainty have been appropriately assessed for impact on the TSTF-505 application, including, that impacts on the LCOs to be included within the scope of the RMTS program have been considered. Address the following:

a) Section 4 of Energy Northwest Report ENGNW-00554-REPT-004, Revision 0, "CGS Impact of Model Uncertainty to RICT Process" (Reference 7), provides sensitivity results for two sources of uncertainty. However, the results appear to only address changes in overall plant risk, and that the impact to estimated RICTs was not performed.

The NRC staff notes that even small changes in risk can be significant to certain LCO RICTs if the associated structures, systems, and components (SSCs) are driving the change in risk value. Therefore, it is unclear to the NRC staff the impact of these two sources of uncertainty on the proposed LCO RICT calculations.

i. Justify that the two sources of uncertainty (high-pressure core spray and conditional loss of offsite power) do not significantly impact any RICT calculation.

ii. If in response to part i. above, it cannot be determined that the cited assumption has an inconsequential impact on the estimated RICTs, then explain what risk management actions (RMAs) may be considered during a RICT program entry to minimize any potential adverse impact from this uncertainty, and explain how these RMAs are expected to reduce the risk associated with this uncertainty.

b) Item No. 16 in table 5 of Energy Northwest Report ENGNW-00554-REPT-004, regarding the potential need to turn off the low-pressure emergency core cooling system (ECCS) pumps when running on the minimum flow to avoid pump damage states that this failure mode is excluded from the Columba PRA models based on the assumption the pumps can operate in this configuration for hours. It further states that the addition of a human failure event to secure the pump(s) does not impact the model since the hardware success of the low-pressure ECCS pumps are included in the PRA models. The basis for this assumption and the impact of excluding this assumption from the PRA models (internal events PRA, fire PRA (FPRA), and seismic PRA) on the proposed RICT calculations (some of the proposed LCOs are directly related to these SSCs) is unclear to the NRC staff.

i. Justify that the exclusion of the action to turn off the ECCS pump assumption has an inconsequential impact on the RICT calculations. In the response, address the basis for the assumption.

ii. If in response to part i. above, it cannot be determined that the cited assumption has an inconsequential impact on the estimated RICTs, then explain what RMAs may be considered during a RICT program entry to minimize any potential adverse impact from this uncertainty, and explain how these RMAs are expected to reduce the risk associated with this uncertainty

c) Item No. 21 in table 5 of Energy Northwest Report ENGNW-00554-REPT-004, regarding the unverified flood damage heights for the Division 1 and 2 electrical equipment in the vital island (4 kiloVolt (kV) buses SM-7 and -8), states that a higher assumed height is judged to not significantly impact RICT calculations. The NRC staff notes the possibility, given the actual height is unknown, the conservatively appropriate flood damage height may be lower than the one used in the Columbia internal flood PRA. It is unclear to the NRC staff how the assumed flood damage height impacts any of the proposed LCO RICT calculations.

i. Justify that the assumed flood damage height has an inconsequential impact on the RICT calculations. In the response, address the basis for the assumption.

ii. If in response to part i. above, it cannot be determined that the cited assumption has an inconsequential impact on the estimated RICTs, then explain what RMAs may be considered during an RICT program entry to minimize any potential adverse impact from this uncertainty, and explain how these RMAs are expected to reduce the risk associated with this uncertainty.

Question 2 (APLA/APLC) - Feasibility of the FLEX Operator Actions

Section 4, item 10 of the NRC final SE for NEI 06-09-A, requires that the LAR provide a discussion of how the key assumptions and sources of uncertainty were identified, and how their impact on the RMTS was assessed and dispositioned. NUREG-1855 presents guidance on the process of identifying, characterizing, and qualitative screening of model uncertainties.

Energy Northwest PSA support document, PSA-02-AS-0001, "Accident Sequence Evaluation," Revision 8 (Reference 8), appears to state that the FLEX diesel generators (DGs) need to provide power to the battery chargers within 4 and 2 hours for reactor pressure vessel (RPV) injection. However, Energy Northwest support document, PSA-02-HR-0001, "Human Reliability Assessment, Revision 5 (Reference 9), appears to have an implementation time (T_{sw}) of 303 (5 hours) and 253 (4.2 hours) minutes for DG4 and DG5, respectively, to power the battery chargers (EACHUMN-DG[4/5]-XTIE). The feasibility of some of these actions based on time available and why there is significant disparity between the three operator actions is unclear to the NRC staff. Regarding RPV injection in PSA-02-HR-0001, it appears the T_{sw} for this action (RPVHUMNFLEX-P1] is 149 minutes (2.5 hours). It is unclear if the T_{sw} for RPV injection is accurate.

a) Clarify the implementation times for the FLEX DGs to supply power to the battery chargers and for RPV injection by the FLEX pumps. Include in this discussion, the basis, such as thermo-hydraulic analysis, used to determine these implementation times.

b) Clarify why there are different implementation times for the two FLEX DGs in providing power to the battery chargers. Include in this discussion, the basis for the different implementation times.

c) If justifications cannot be made for any of the implementation times and must be adjusted, propose a mechanism that incorporates the updated implementation times prior to any RICT calculation.

Question 3 (APLA/APLC) - Impact of Seasonal Variations

The Tier 3 requirement of Regulatory Guide (RG) 1.177, Revision 2, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, dated January 2021 (Reference 10) stipulates that a licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. Section 2.3.4 of NEI 06-09-A states, in part, that:

> If the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle…, then the RICT calculation shall either 1) use the more conservative assumption at all time, or 2) be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA.

The LAR does not seem to address whether modeling adjustments are needed to account for seasonal and time of cycle dependencies and what kind of adjustments will be made. Therefore, address the following to clarify the treatment of seasonal and time of cycle variations:

a) Explain how the RICT calculations address changes in PRA data points, basic events, and SSC operability constraints as a result of extreme weather conditions, seasonal variations, other environmental factors, or time of cycle. Also, explain how these adjustments are made in the configuration risk management program (CRMP) model and how this approach is consistent with the guidance in NEI 06-09-A and its associated NRC final SE.

b) Describe the criteria used to determine when PRA adjustments due to extreme weather conditions, seasonal variations, other environmental factors, or time of cycle variations need to be made in the CRMP model and what mechanism initiates these changes.

Question 4 (APLA) - Performance Monitoring

The NRC final SE for NEI 06-09-A, specifies, in accordance with the fifth key safety principle of RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated January 2018 (Reference 11) and RG 1.177, Revision 2, that the impact of the RICT program should be monitored using performance management strategies. Additionally, the final SE for NEI 06-09-A specifies that the LAR should include a description of the monitoring program. Furthermore, NRC staff position C.3.2, "Scope of the Probabilistic Risk Assessment for Technical Specification Change Evaluations," provided in RG 1.177, Revision 2, for meeting the fifth key safety principle, specifies that performance criteria should be established to assess degradation of operational safety over a period of time. The guidance in NEI 06-09-A considers the use of Nuclear Management and Resources Council (NUMARC) 93-01 "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Reference 12), as endorsed by RG 1.160, Revision 4, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated August 2018 (Reference 13), for the implementation of the Maintenance Rule. NUMARC 93-01, Revision 4F, Section 9.0, dated April 2018, contains guidance for the establishment of performance criteria.

The LAR does not address how the licensee's RMTS process captures performance monitoring for the SSCs within-scope of the RMTS program. Therefore, -*either*-

a) Confirm that the Columbia Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in NUMARC 93-01, as endorsed by RG 1.160, -or-

b) Describe the approach or method used by Columbia for SSC performance monitoring, as described in NRC staff position C.3.2 of RG 1.177, Revision 2, for meeting the fifth key safety principle. In the description, include criteria (e.g., qualitative or quantitative), along with the appropriate risk metrics, and explain how the approach and criteria demonstrate the intent to monitor the potential degradation of SSCs in accordance with the NRC final SE for NEI 06-09-A.

Question 5 (APLA/APLC) – In-Scope LCOs and Corresponding PRA Modeling

The NRC final SE for NEI 06-09-A, specifies that the LAR should provide a comparison of the TS Functions to the PRA modeled Functions of the SSCs, subject to those LCO actions, to show that the PRA modeling is consistent with the licensing basis assumptions, or to provide a basis for when there is a difference. LAR enclosure 1, table E1-1, “In-scope TS/LCO Conditions to Corresponding PRA Functions,” the licensee identifies each TS LCO proposed for the RICT program, describes whether the systems and components participating in the TS LCO are implicitly or explicitly modeled in the PRA, and compares the design-basis and PRA success criteria. For certain TS LCO conditions, the table explains that the associated SSCs are not modeled in the PRAs but will be represented using a surrogate event that fails the Function performed by the SSC. For some LCO conditions, the LAR did not provide enough description for NRC staff to conclude that the PRA modeling will be sufficient for each proposed LCO condition.

In table A5-1, “RICT Program PRA Implementation Items,” of attachment 5 to the LAR, the entry regarding reactor protection system (RPS) modeling states that the Columbia RPS failure probability is based on NUREG-CR/5500, “Reliability Study: General Electric Reactor Protection System, 1984-1995,” Volume 3, dated May 1999 (Reference 14), by using a single point estimate event for each RPS sub-system (channel), and the simplified Columbia RPS model generates the exact base probability of NUREG/CR-5500. However, it is stated later that the simplified RPS model provides a more conservative result than the NUREG/CR-5500 model when a Function channel is inoperable or bypassed. Clarity is needed to understand how the Columbia RPS channel modeling is more conservative than NUREG/CR-5500 base probability if the base results in exactly matching the NUREG value. The table entry states the Columbia RPS logic represents RPS failure due to electrical equipment. The NRC staff notes that in section 5 of NUREG/CR-5500, the overall failure probability appears to include operator manual scram, control rod system, and hydraulic control unit system (scram discharge volume and solenoid operated valves (SOVs)). NUREG/CR-5500 provides the following failure rates: (1) section 3.3 states a failure rate of 3.8E-06 for the channel and trip portion of RPS, and (2) section 5 states the mean RPS unavailability as 5.8E-06. Clarity is needed to understand how the Columbia RPS model incorporates all of the necessary SSCs and operator actions to represent the as-built, as-operated plant for the associated proposed RICT TSs.

The NRC staff notes that section 5 of NUREG/CR-5500 states that the failure probabilities used were based on U.S. General Electric commercial data from 1984 through 1995, and that the 2009 American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard supporting requirement DA-C1 lists NUREGs that contain failure data from recognized sources. One of those sources, NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” dated February 2007 (Reference 15), contains recent industry data including a 2020 update. How Capability Category (CC)-II technical acceptability is met by using the NUREG/CR-5500 data for this application, or if the Columbia RPS modeling is being implemented as a surrogate for RICT calculations is unclear to the NRC staff.

a) Provide details of the inputs used to determine the Columbia RPS single point estimate failure rate.

b) Provide justification that the Columbia RPS model provides conservative results when compared to the NUREG/CR-5500 model. Include in this discussion the failure probability values used from NUREG/CR-5500.

c) Justify that all of the SSCs associated with the proposed RICT LCOs related to RPS fully represent the functionality of those LCOs. Include in this discussion the apparent disparity of SSCs mentioned in NUREG/CR-5500 to those discussed in Table A5-1 of the LAR, and how the NUREG/CR-5500 failure probability value provides an appropriate comparison for this application.

d) Clarify if the proposed Columbia RPS model meets the 2009 ASME/ANS PRA standard CC-II requirements (answer i and ii below) or is used as a surrogate for RICT calculations (answer iii and iv below).

   i. If the RPS model is to meet the approved PRA standard requirement, then provide justification that the use of NUREG/CR-5500 information meets the associated CC-II requirements, and

   ii. If the use of NUREG/CR-5500 cannot be justified, then propose a mechanism to ensure the Columbia RPS model meets the CC-II requirements of the 2009 ASME/ANS PRA standard prior to any RICT calculation.

   iii. If the RPS model is to be used as a surrogate for the applicable RICT TSs, then provide justification that the use of the surrogate is either conservative or bounding when compared to a CC-II model, and

   iv. If the proposed surrogate cannot be justified as either conservative or bounding, then propose a mechanism to ensure the Columbia RPS model is conservative or bounding prior to any RICT calculation.

e) Section 3.0, “Modeling of Flex,” of enclosure 1 to the LAR states, in part, that the “Internal Events PRA model (Revision 8.0) was used to develop the Seismic PRA model.” Explain the effect of this data on the Seismic PRA.

Question 6 (APLA) – Missing Information from Table E1-1

In LAR enclosure 1, table E1-1, the following TS Functions are missing information or are missing altogether. Therefore, provide the missing information for:

- TS 3.3.4.1, LCO a.2, “Turbine Governor Valve (TGV) Fast Closure, Trip Oil Pressure – Low”
- TS 3.3.4.2, LCO b, “Reactor Vessel Steam Dome Pressure – High”
- TS 3.3.5.1, Function 1.b, “Drywell Pressure – High”
- TS 3.3.5.1, Function 3.e, “Suppression Pool Water Level – High”
- TS 3.3.5.1, Function 5.e, “Accumulator Backup Compressed Gas System Pressure – Low”
- TS 3.3.6.1, Function 5.a, “Residual Heat Removal (RHR) Shutdown Cooling (SDC) System Isolation, Pump Room Area Temperature – High”
- TS 3.3.6.1, Function 5.b, “RHR SDC System Isolation, Pump Room Area Ventilation Differential Temperature – High”
- TS 3.3.6.1, Function 5.c, “RHR SDC System Isolation, Heat Exchanger Area Temperature – High” (Room 505, 507, 605, and 606 Area)
- TS 3.3.6.1, Function 5.d, “RHR SDC System Isolation, Reactor Vessel Water Level-Low, Level 3”

- TS 3.3.8.1, Function 1.b, "Divisions 1 and 2 - 4.16 kV Emergency Bus Undervoltage, TR-S Loss of Voltage – Time Delay"
- TS 3.3.8.1, Function 1.e, "Divisions 1 and 2 – 4.16 kV Emergency Bus Undervoltage, Degraded Voltage – 4.16 kV Basis"

Question 7 (APLA) – Surrogate for TS 3.3.5.1.E

Note 5 in enclosure 1 to the LAR, table E1-1, states that the instrumentation for the minimum flow valves is not modeled explicitly but is modeled as within the valve component boundary. Therefore, the minimum flow valve events are used as instrumentation surrogates for TS 3.3.5.1, Condition E, Functions 1.g, 1.h, 2.g and 3.f for low discharge flow of low-pressure core spray, low-pressure coolant injection A, B and C, and high-pressure core spray pumps, respectively. In addition, table E1-1 states that the PRA success criteria are that the "Minimum flow valve opens."

Typically, the minimum flow instruments in TS 3.3.5.1 are credited to both open and close the minimum flow valves at their designated values, thereby (1) preventing the pumps from overheating and (2) preventing less than full system flow assumed in the accident analysis. Identify the minimum flow valve events used to determine the RICTs for this LCO, and explain how they are bounding.

Question 8 (APLA) – Surrogate for TS 3.3.5.1 Conditions F and G

Enclosure 1 to the LAR, table E1-1, states that TS 3.3.5.1 instrumentation associated with Conditions F and G, Functions 4.a, 4.b, 4.c, 4.d, 4.e, 4.g and 5.a, 5.b, 5.c, 5.d, 5.f are not modeled in the PRA, and there are no PRA success criteria because the automatic depressurization system (ADS) inhibit is assumed and the reactor is manually depressurized. Table E1-1 comments column states that the RICT calculation uses an equivalent surrogate of either Train A/B ADS SOVs failure to open and provides reference to note 6. Note 6 states, in part, that the "ADS SRVs [safety relief valves] are only modeled by common cause failures or their supporting SOVs and supports." Note 6 further states that for "the RICT calculation, individual ADS SRV valve body independent failure to open events were added." It is not clear to the NRC staff what surrogate is being used the ADS SOV failure to open or individual ADS SRV valve body independent failure to open events.

Identify the surrogates used to determine the RICTs for TS 3.3.5.1 Conditions F and G, Functions 4.a, 4.b, 4.c, 4.d, 4.e, 4.g and 5.a, 5.b, 5.c, 5.d, 5.f, and explain how they are bounding.

Question 9 (APLA) – Surrogate for TS 3.3.6.1.A

LAR table E1-1 states that TS 3.3.6.1 instrumentation associated with Condition A, Function 2.e is modeled in the PRA. The PRA success criteria is "[o]ne switch/PB [pushbutton] pair (Only Group 4 Isolation valves contributing to LERF [large early release frequency] are modeled)." Table E1-1 comments column states that "[f]ailure of modeled isolation valves contribute to LERF and bound the effect on unmodeled isolation valves in the other isolation groups." It is not clear whether the switch/PB(s) for groups 2, 3, 4, and 5 are modeled in the PRA and if the PRA success criteria is one switch/PB per group. Therefore, provide the following:

a) Regarding TS 3.3.6.1.A, Function 2.e, manual initiation, clarify how many switch/PB pairs are associated with each group covered by this function, and for each group explain if they are modeled in the PRAs (internal events PRA, FPRA, and seismic PRA) and if so, what are the PRA success criteria.

b) For the switch/PB pairs not modeled in the PRAs identify the surrogate and explain how the surrogate is bounding for the RICT calculation.

Question 10 (APLA) – Surrogate for TS 3.3.8.1.B

Enclosure 1 to the LAR, table E1-1 states that TS 3.3.8.1 instrumentation associated with Condition B, Function 1.a is not modeled in the PRA and there are no PRA success criteria. Table E1-1 comments column states that "Function 1.a will be conservatively mapped to modeled relays that fail DG LOV [loss of voltage] start signal and TR-B transfer, which are affected circuits of the LOV channels. Operable LOV channel will conservatively not be credited for RICT." Provide further clarification on the surrogate modeled in the PRAs (internal events PRA, FPRA, and seismic PRA) and explain how the surrogate is bounding.

Question 11 (APLA) – Success Criteria for TS 3.5.1.E

LAR table E1-1 for TS 3.5.1, Condition E, states that the design success criterion is "Five of seven ADS SRVs," and the PRA success criterion states that "[t]hree of seven ADS SRVs <u>OR</u> three of eleven non-ADS SRVs." Table E1-1 comments column states that the "success criterion is based on MAAP and RETRAN… best estimate analyses." Describe the plant-specific analyses used to support the PRA success criteria, and explain why it adequately captures the configuration risk.

Question 12 (APLA/APLC) – PRA Modeling and Uncertainty of FLEX Equipment and Actions

The NRC memorandum dated May 30, 2017 (Reference 16) provides the NRC's staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 17). The NRC memorandum, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Probabilistic Risk Assessments," dated May 6, 2022 (Reference 18) updated the May 30, 2017, memorandum, to reflect current information.

Regarding uncertainty, section 2.3.4, "PRA Technical Adequacy," of NEI 06-09-A states that PRA modeling uncertainties shall be considered in the application of the PRA base model results to the RICT program and that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation. The guidance in NEI 06-09-A also states that the insights from the sensitivity studies should be used to develop appropriate RMAs, including highlighting risk-significant operator actions, confirming availability and operability of important standby equipment, and assessing the presence of severe or unusual environmental conditions. Uncertainty exists in PRA modeling of FLEX related to the equipment failure probabilities for FLEX equipment used in the model, the corresponding operator actions, and pre-initiator failure probabilities. Therefore, FLEX modeling assumptions can be key assumptions and sources of uncertainty for the RICTs proposed in this application.

In LAR enclosure 1, the licensee indicates that FLEX equipment and actions have been credited in the internal events, fire, and seismic PRAs. Section 3, "Modeling of Flex," of LAR enclosure 1 states that a sensitivity study was performed to measure the risk increase associated with completely removing credit for FLEX strategies from the internal events, fire, and seismic PRAs. However, the results appear to only address changes in overall plant risk, and the impact to estimated RICTs was not performed.

Given the observation above, it is not clear whether the stated sensitivity study, performed to assess the impact of crediting FLEX equipment and actions, is sufficient to conclude that the impact to the RICT program of the uncertainties associated with modeling FLEX is negligible. For this reason, and to understand the credit that will be taken for FLEX equipment and actions in the RICT program, describe LCO-specific sensitivity studies for the internal events, internal flooding, fire, and seismic PRAs that assess impact on the RICT from uncertainties in FLEX equipment failure probabilities and FLEX-independent and FLEX-dependent human error probabilities (HEPs) associated with deploying and staging FLEX portable equipment (e.g., sensitivity studies that remove FLEX credit). As part of the response, include the following information:

1. Justify the FLEX equipment failure probabilities and FLEX-independent and dependent HEP values selected for the sensitivity studies, including justification of why the chosen values constitute bounding realistic estimates (e.g., sensitivity studies that remove FLEX credit).

2. Discuss the bases for the chosen TS LCO conditions in the sensitivity studies. Because the 30-day RICT back-stop condition could mask the impact of this uncertainty in the sensitivity studies, discuss whether the RICTs for plant configurations involving more than one LCO entry (e.g., where the calculated RICTs are less than the 30-day backstop) are significantly impacted by this uncertainty.

3. Provide numerical results on specific selected RICTs and a discussion of the results.

4. Discuss whether the uncertainty associated with FLEX HEPs and/or FLEX equipment failure probabilities are a key source of uncertainty for the RICT program. If this uncertainty is "key," then describe and provide a basis for how this uncertainty will be addressed in the RMTS program (e.g., programmatic changes such as identification of additional RMAs, program restrictions, or the use of bounding analyses which address the impact of the uncertainty). If the programmatic changes include identification of additional RMAs, then (1) describe how these RMAs will be identified prior to the implementation of the RMTS program, consistent with the guidance in section 2.3.4 of NEI 06-09-A; and (2) for those TS LCOs whose RICTs are significantly impacted by this uncertainty, describe specific RMAs that may be considered during a RICT program entry to minimize any potential adverse impact from this uncertainty, and explain how these RMAs are expected to reduce the risk associated with this uncertainty.

Question 13 (APLA) – TS 5.5.16, Proposed Administrative Controls for the RICT Program

In attachment 1 to the LAR, the licensee stated the phrase "used to support this license amendment," which is potentially confusing since there is no indication as to which license amendment is being referred to in this phrase.

The licensee then proposed to modify the statement to read "Methods to assess the risk from extending the Completion Times must be PRA methods approved for use with this program in Amendment No. [###], or other methods…" However, the proposed administrative controls for the RICT program in TS 5.5.16 paragraph "e" of attachment 2 of the LAR seems to be based on the TS markups of TSTF-505, and revised for clarity such that it states, "…used to support this program in Amendment No. [###]…."

The NRC staff recognizes that the final revised model SE for TSTF-505 contains improved phrasing for the administrative controls for the RICT program in TS 5.5.16 paragraph "e," namely the phrasing "approved for use with this program" instead of "used to support this license amendment." In lieu of the original phrasing in TS 5.5.16 paragraph "e," discuss whether the phrases "used to support this program in Amendment No. [###]" or, as discussed in part by the final revised model SE for TSTF-505 and Attachment 1 of the LAR, "methods approved for use with this program in Amendment No. [###]" would provide more clarity for this paragraph.

<u>PRA Licensing Branch B (APLB) Audit Questions</u>

Question 1 (APLB) – Dispositions of FPRA Model Assumptions and Sources of Uncertainty

The NRC final staff SE to NEI 06-09 specifies that the LAR should identify key assumptions and sources of uncertainty and to assess and disposition each as to their impact on the RMTS application. NUREG-1855 presents guidance on the process of identifying, characterizing, and qualitative screening of model uncertainties.

Several documents on the Columbia TSTF-505 online portal provide the dispositions for identified key assumptions and FPRA sources of uncertainty. Additional information is needed for the NRC staff to determine that the key assumptions and FPRA sources of uncertainty have been appropriately assessed for impact on the TSTF-505 application, including the impacts on the LCOs included within the scope of the RMTS program have been considered. Address the following:

a) Section 6.2.1 of the Energy Northwest report FPSA-2-UNC-0001 (Reference 19) provides sensitivity results for systems not credited in the FPRA, such as the control rod drive injection and standby liquid control (SLC) systems. The NRC staff notes the control rod drive injection system is sometimes credited in the design analysis, which could be associated with some of the proposed TS LCOs. Additionally, the NRC staff notes that TS LCO 3.1.7, associated with the SLC system is a proposed LCO for the RICT program. Table E1-1 in enclosure 1 to the LAR, for TS 3.1.7, states that SLC SSCs are modeled. The FPRA modeling status of the SLC system and the possible impact of the excluded systems on any of the proposed RICT calculations is unclear to the NRC staff.

   i. Clarify that the SLC SSCs are included in all the Columbia FPRA models used for RICT calculation.

   ii. If the SLC system is not included in the Columbia FPRA models, then provide justification that its exclusion does not have an inconsequential impact on the RICT calculations.

iii. Alternatively, to part ii. above, propose a mechanism to ensure the SLC system is incorporated in all the Columbia PRA models prior to implementing the RICT program.

iv. Provide justification that the other excluded systems listed in section 6.2.1 do not have an inconsequential impact on the RICT calculations.

v. Alternatively, to part (iv) above, propose a mechanism to ensure any system that can impact an RICT calculation is incorporated in all the Columbia PRA models prior to implementing the RICT program.

b) Section 6.2.3 of the Energy Northwest report FPSA-2-UNC-0001 provides sensitivity results regarding the sources of uncertainty related to FLEX modeling. The results note a 1 percent increase in fire core damage frequency (CDF) risk; however, it does not appear to address the impact on RICT calculations and apparently not addressed in Energy Northwest report ENGNW-00554-REPT-004. The NRC staff notes that even small changes in risk can be significant to certain LCO RICTs if the associated SSC(s) is driving the change in risk value. Therefore, the impact of these two sources of uncertainty on the proposed LCO RICT calculations is unclear to the NRC staff.

i. Justify that the sources of uncertainty related to FLEX modeling has an inconsequential impact on the RICT calculations. In the response, address the basis for the assumption.

ii. If in response to part i. above, it cannot be determined that the cited assumption has an inconsequential impact on the estimated RICTs, then explain what RMAs will be considered to compensate for this uncertainty and the basis for their consideration.

c) Section 6.2.10 of the Energy Northwest report FPSA-2-UNC-0001 provides sensitivity results regarding the joint HEP (JHEP) floor value used in the Columbia FPRA. The results note a 2 percent increase in fire CDF risk; however, it does not appear to address the impact on RICT calculations (and apparently not addressed in Energy Northwest reportENGNW-00554-REPT-004). The NRC staff notes that even small changes in risk can be significant to certain LCO RICTs if the associated SSC(s) is driving the change in risk value. Therefore, the impact of these two sources of uncertainty on the proposed LCO RICT calculations is unclear to the NRC staff.

i. Justify that the sources of uncertainty related to the FPRA JHEP floor value lower than 1E-05 has an inconsequential impact on the RICT calculations. In the response, address the basis for the assumption.

ii. If in response to part i. above, it cannot be determined that the cited assumption has an inconsequential impact on the estimated RICTs, then explain what RMAs will be considered to compensate for this uncertainty and the basis for their consideration.

Question 02 (APLB) – Deviations from NRC Endorsed Guidance as Source of Modeling Uncertainty

RG 1.200, Revision 3 (Reference 20), states “NRC reviewers, … [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application.” The relatively extensive and detailed reviews of FPRA undertaken in support of

LARs to transition to a risk-informed, performance based, National Fire Protection Association 805 (Reference 21) licensing basis determined that implementation of some of the complex FPRA methods often used non-conservative and over-simplified assumptions to apply the method to specific plant configurations. Some of these issues were not always identified in facts and observations by the peer review teams but are considered potential key assumptions by the NRC staff because using more defensible and less simplified assumptions could substantively affect the fire risk and fire risk profile of the plant.

The NRC staff evaluates the acceptability of the PRA for each new risk-informed application, and as discussed in RG 1.174, recognizes that the acceptable technical adequacy of risk analyses necessary to support regulatory decision-making may vary with the relative weight given to the risk assessment element of the decision-making process. The NRC staff notes that the calculated results of the PRA are used directly to calculate a RICT, which subsequently determines how long SSCs (both individual SSCs and multiple, unrelated SSCs) controlled by TSs can remain inoperable. Therefore, the PRA results are given a very high weight in a TSTF-505 application, and the NRC staff requests additional information on the following issues that have been previously identified as potentially key FPRA assumptions.

a) Use of Unacceptable Methods

   The LAR provides the history of the FPRA peer review but does not discuss methods used in the FPRA. Methods may have been used in the FPRA that deviate from guidance in NUREG/CR-6850, "EPRI/NRC-RES [Office of Nuclear Regulatory Research] Fire PRA Methodology for Nuclear Power Facilities" (References 22, 23, and 24), or other acceptable guidance (e.g., frequently asked questions (FAQs), NUREGs, or interim guidance documents).

   i. Identify methods used in the FPRA that deviate from guidance in NUREG/CR-6850 or other acceptable guidance.

   ii. If such deviations exist, then justify their use in the FPRA and impact on the RICT.

   iii. As an alternative to part ii. above, add an implementation item to replace those methods with a method acceptable to NRC prior to the implementation of the RICT program. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

b) Reduced Transient Heat Release Rates

   The key factors used to justify using transient fire reduced heat release rates (HRRs) below those prescribed in NUREG/CR-6850 are discussed in a letter dated June 21, 2012 (Reference 25). If any reduced transient HRRs below the bounding 98 percent HRR of 317 kiloWatt from NUREG/CR-6850 were used, discuss the key factors used to justify the reduced HRRs. Include in this discussion:

   i. Identification of the fire areas where a reduced transient fire HRR is credited, and what reduced HRR value was applied.

   ii. A description for each location where a reduced HRR is credited, and a description of the administrative controls that justify the reduced HRR, including, how location-specific attributes and considerations are addressed.

Include a discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance. Also, include discussion of the personnel traffic that would be expected through each location.

iii. The results of a review of records related to compliance with the transient combustible and hot work controls.

c) Treatment of Sensitive Electronics

A memorandum dated December 3, 2013 (FAQ 13-0004) (Reference 26), provides supplemental guidance for application of the damage criteria provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850, Volume 2, for solid-state and sensitive electronics.

i. Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in FAQ 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).

ii. If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the RICT calculations.

iii. As an alternative to part ii. above, add an implementation item to replace the current approach with an acceptable approach prior to the implementation of the RICT program. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

d) Obstructed Plume Model

NUREG-2178, "Refining And Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: "Peak Heat Release Rates and Effect of Obstructed Plume" (Reference 27), contains refined peak HRRs compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction. Additionally, NUREG-2178 provides guidance that indicates that the obstructed plume model is not applicable to cabinets in which the fire is assumed to be located at elevations of less than one-half of the cabinet.

i. If obstructed plume modeling was used, then indicate whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet.

ii. Justify any modeling in which the base of an obstructed plume is located at less than one half of the cabinet's height.

iii. As an alternative to part ii. above, add an implementation item to remove credit for the obstructed plume model in the FPRA prior to the implementation of the RICT program.

e) Well-Sealed Motor Control Center Cabinets

Guidance in FAQ 08-0042 from Supplement 1 of NUREG/CR-6850 applies to electrical cabinets below 440 volts (V). With respect to Bin 15, as discussed in NUREG/CR-6850, Volume 2, Chapter 6, it clarifies the meaning of “robustly or well-sealed.” Thus, for cabinets of 440V or less, fires from well-sealed cabinets do not propagate outside the cabinet. For cabinets of 440V and higher, the original guidance in Chapter 6 remains and requires that Bin 15 panels, which house circuit voltages of 440V or greater, are counted because an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires).

A memorandum, “Close-Out of Fire Probabilistic Risk Assessment Frequently Asked Question 14-0009 on Treatment of Well-Sealed MCC Electrical Panels Greater than 440V,” dated April 29, 2015 (FAQ 14-0009) (Reference 28) provides the technique for evaluating fire damage from motor control center (MCC) cabinets having a voltage greater than 440V. Therefore, propagation of fire outside the ignition source panel must be evaluated for all MCC cabinets that house circuits of 440V or greater.

i. Describe how fire propagation outside of well-sealed MCC cabinets greater than 440V is evaluated.

ii. If well-sealed cabinets less than 440V are included in the Bin 15 count of ignition sources, provide justification for using this approach as this is contrary to the guidance.

f) Influence Factors for Transient Fires

NUREG/CR-6850 Section 6 and a memorandum, “Close-Out of National Fire Protection Association 805 Frequently Asked Question 12-0064 on Hot Work/Transient Fire Frequency Influence Factors,” dated January 17, 2013 (FAQ 12-0064) (Reference 29) describe the process for assigning influence factors for hot work and transient fires. Provide the following regarding application of this guidance:

i. Indicate whether the methodology used to calculate hot work and transient fire frequencies applies influencing factors using NUREG/CR-6850 guidance or FAQ 12-0064 guidance.

ii. Indicate whether administrative controls are used to reduce transient fire frequency, and if so, describe and justify these controls.

iii. Indicate whether Columbia has any combustible control violations, and discuss Columbia’s treatment of these violations for the assignment of transient fire frequency influence factors. For those cases where Columbia has violations and has assigned an influence factor of 1 (Low) or less, indicate the value of the influence factors Columbia has assigned and provide justification.

iv. If Columbia has assigned an influencing factor of “0” to maintenance, occupancy, or storage, or hot work for any fire physical analysis units, provide justification.

v. If a weighting factor of “50” was not used in any fire physical analysis unit, provide a sensitivity study that assigns weighting factors of “50” per the guidance in FAQ 12-0064.

g) Fire Scenario Treatment of the Main Control Board (MCB)

Traditionally, the cabinets on front face of the MCB have been referred to as the MCB for purposes of FPRA. Appendix L of NUREG/CR-6850 provides a refined approach for developing and evaluating MCB fire scenarios. A memorandum, “Close-Out of Fire Probabilistic Risk Assessment Frequently Asked Question 14-0008 on Main Control Board Treatment,” dated July 22, 2014 (FAQ 14-0008) (Reference 30) clarifies the definition of the MCB and effectively provides guidance for when to include the cabinets on the back side of the MCB as part of the MCB for FPRA. It is important to distinguish between MCB and non-MCB cabinets because misinterpretation of the configuration of these cabinets can lead to incomplete or incorrect fire scenario development. FAQ 14-0008 also provides several alternatives to NUREG/CR-6850 for using Appendix L to treat partitions in an MCB enclosure. Therefore, address the following:

i. Briefly describe the MCB configuration and use the guidance in FAQ 14-0008 to determine whether cabinets on the rear side of the MCB are a part of the MCB. Provide justification using the FAQ guidance.

ii. If the cabinets on the rear side of the MCB are part of a single integral MCB enclosure using the definition in FAQ 14-0008, then confirm that the guidance in FAQ 14-0008 was used to develop fire scenarios in the MCB and determine the frequency of those scenarios

iii. If the cabinets on the rear side of the MCB are part of a single integral MCB enclosure, and the guidance in FAQ 14-0008 was not used to develop fire scenarios involving the MCB, then provide a description of how the fire scenarios for the backside cabinets are developed, and an explanation of how the treatment aligns with NRC accepted guidance.

iv. If in response to part iii. above, the current treatment of the MCB and those cabinets on the rear side of the MCB cannot be justified using NRC accepted guidance, then justify that the treatment has no impact on the RICT calculations. Alternatively, propose a mechanism that ensures that the FPRA is updated to treat the MCB enclosure consistent with the guidance in FAQ 14-0008, prior to implementation of the RICT program.

<u>Instrumentation and Controls Branch (EICB) Audit Question</u>

Question 1 (EICB) – Manual Scrams and Manual Trips

Energy Northwest’s LAR is a risk-informed request to modify Columbia’s TS consistent with the approach approved in TSTF-505. In the model safety evaluation for TSTF-505, section 3.1.2.3 “Evaluation of Instrumentation and Control Systems,” the NRC clarifies the basis of the NRC staff’s SE is to consider “a number of potential plant conditions allowed by the new TSs” and to consider “what redundant or diverse means were available to assist the licensee in responding to various plant conditions.” TSTF-505 states that at least one redundant or diverse means (e.g., other automatic features or manual action) to accomplish the safety functions (e.g., reactor trip, safety injection, or containment isolation) remain available during the use of the RICT.

In addition, RG 1.174, Revision 3, states the licensee should understand that “when the proposed licensing basis change necessitates reliance on programmatic activities as compensatory measures, the licensee should justify that this reliance is not excessive (i.e., not overly reliant).”

In the table in attachment 6, “Evaluation of Instrumentation and Control Systems,” of the LAR, the licensee identifies the diverse means for each affected instrumentation and control function under each postulated accident. A number of diverse means are identified solely as “Manual scram” or “Manual Trip”.

a) Please confirm that these “Manual scrams” or “Manual Trips” identified in the table in attachment 6 are modeled in the PRA, defined in Columbia’s operations procedures to which operators are trained, and describe how the times associated with these actions are evaluated as adequate.

   Based on a review of the table in attachment 6 that covers TS table section 3.3.1.1-1, the impacted functions are:

   - Function 1b. Intermediate Range Monitors – Inop [Inoperable]
   - Function 2c. Average Power Range Monitors – Neutron Flux – high
   - Function 2d. Average Power Range Monitor - Inop
   - Function 2e. Average Power Range Monitors – 2-Out-of-4 Voter
   - Function 5 Main Steam Isolation Valve Closure, Final Safety Analysis Report (FSAR) section 15.6.4, “Steam System Piping Break Outside Containment” (Reference 31)
   - Function 10 Reactor Mode Switch – Shutdown Position

b) Additionally based on a review of the table in attachment 6 that covers, TS Table 3.3.5.1-1, for 3. “High Pressure Core Spray (HPCS) System,” the NRC staff noticed that Function 3c. “Reactor Vessel Water Level – High, Level 8 is listed with “Manual Trip” being the only diverse instrumentation available to complete the function. However, there is no “Transient Accident” associated with this function. Therefore, the NRC staff requests clarification to better understand why this function was included in the table (e.g., to maintain continuity of functions so they match the TS table) when the associated table in attachment 6 was created.

<u>**Electrical Engineering Branch (EEEB) Audit Questions**</u>

Question 1 (EEEB) – TS LCO 3.8.1, Condition A

For LCO 3.8.1, Condition A, identify: 1) in columns 3, 5, and 6 of LAR enclosure 1, table E1-1, all affected SSCs and their purpose more clearly; the offsite sources, both preferred and backup, that supply three divisions, as stated on TS Bases page B 3.8.1-1 and the FSAR section 8.3.1.1.1 (page 8.3-2) (Reference 33); 2) in column 6, the minimum alternating current (AC) sources available if one offsite circuit is unavailable during a design-basis accident (DBA); and 3) whether offsite circuit means “qualified offsite circuit,” as defined on TS Bases page B 3.8.1-2.

Question 2 (EEEB) – TS LCO 3.8.1, Condition B

For LCO 3.8.1, Condition B, identify in columns 3, 5, and 6 of LAR enclosure 1, table E1-1 all affected SSCs and their purpose more clearly; and in column 6, the minimum AC sources available if one emergency DG is unavailable during a DBA with a loss of offsite power.

Question 3 (EEEB) – TS LCO 3.8.1, Condition C

For LCO 3.8.1, Condition C, identify in columns 3 and 6 of LAR enclosure 1, table E1-1 whether offsite circuit means “qualified offsite circuit” and if so, the minimum AC sources to address a DBA in column 6 given that the result is determined by LCO 3.8.1 Conditions A and B.

Question 4 (EEEB) – TS LCO 3.8.1, Condition D

For LCO 3.8.1, Condition D, identify in columns 3 and 6 of LAR enclosure 1, table E1-1 whether offsite circuit means “qualified offsite circuit” and if so, the minimum AC sources to address a DBA in column 6.

Question 5 (EEEB) – TS LCO 3.8.4, Condition A

For LCO 3.8.4, Condition A, identify in column 3 of LAR enclosure 1, table E1-1 whether standby charger for affected divisions should be included as SSCs since placed into operation for loss of primary charger.

Question 6 (EEEB) – TS LCO 3.8.4, Condition B

For LCO 3.8.4, Condition B, identify in columns 3 and 5 of LAR enclosure 1, table E1-1 all affected SSCs and their purpose more clearly; and in column 6, if battery charger is unavailable, what the direct current (DC) power source is.

Question 7 (EEEB) – TS LCO 3.8.4, Condition C

For LCO 3.8.4, Condition C, identify in columns 3 and 5 of LAR enclosure 1, table E1-1 all affected SSCs and their purpose more clearly; and in column 6, if battery charger is unavailable, what the DC power source is.

Question 8 (EEEB) – TS LCO 3.8.4, Conditions D, E, and F

For LCO 3.8.4, Conditions D, E, and F, identify what is meant by “abnormal operation” in column 5 of LAR enclosure 1, table E1-1 since all divisions operate with their batteries in float with chargers supplying divisional loads.

Question 9 (EEEB) – TS LCO 3.8.4, Condition G

For LCO 3.8.4, Condition G, identify in columns 3 and 5 of LAR enclosure 1, table E1-1, all affected SSCs and their purpose more clearly, and in column 6, the divisions affected by one DC electrical power distribution subsystem.

Question 10 (EEEB) – TS LCO 3.8.7, Conditions A and B

For LCO 3.8.7, Conditions A and B, identify in columns 3 and 5 of LAR enclosure 1, table E1-1, all affected SSCs and their purpose more clearly, and in column 6, the divisions affected.

Question 11 (EEEB) - PRA

Provide a discussion of or demonstrate at a high level the fault tree(s) and/or event tree(s) that capture the electrical sources associated with TS LCOs 3.8.1, 3.8.4, and 3.8.7.

Technical Specifications Branch (STSB) Audit Questions

Question 1 (STSB) – TS LCO 3.3.4.1

LAR enclosure 1, table E1-1, shows the image depicted below. For TS LCO 3.3.4.1, Condition A, is entered when one or more required channels are inoperable. The NRC staff notes that TS LCO 3.3.4.1, Condition B, is entered when "One or more Functions with EOC-RPT [end of cycle recirculation pump trip] capability not maintained AND MCPR [Minimum Critical Power Ratio] limit for inoperable EOC-RPT not made applicable".

Based on a literal interpretation of the LCO, it appears that as soon as one valve/channel (for the turbine throttle valve (TTV) or TGV) is inoperable, the plant would enter a 72-hour completion time for Condition A and may also be required to simultaneously enter a 2-hour completion time as indicated in Condition B, which appears to contradict or at least severely curtail the time allotted for the Condition A completion time of 72 hours. Based on a description of how the TTV and TGV systems operate, with two out of two logic (provided in the TS bases), whenever one valve of either system is inoperable, the Function also appears to become disabled.

a) Describe the actions the operator would take and what CONDITIONS would be entered when one channel (valve) is inoperable for either the TTV or TGV systems, and the progression that would occur if or when the MCPR is (or is not) implemented.

b) Explain the evolution through the CONDITIONS listed in the TS LCO, using the method explained in a) above, and;

c) Explain how the language in the TS Bases for this TS LCO and associated CONDITIONS and ACTIONS required align with the language in the TS for 3.3.4.1.

CGS TS	CGS TS Description	SSCs Covered by TS LCO Condition	Modeled in PRA	Function Covered by TS LCO Condition	Design Success Criteria	PRA Success Criteria	Comments
3.3.4.1.A	End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation - One or more required channels inoperable.	Function a.1 Turbine Throttle Valve (TTV) – Closure (Four channels)	No	Trip Both Recirculation Pumps	Two Turbine Trip Valve Closure channels in either trip system <u>OR</u> Two Turbine Governor Valve Fast Closure Trip Oil Pressure-Low channels in either trip system	None	See Note 3
		Function a.2. Turbine Governor Valve (TGV) – Fast Closure, Trip Oil Pressure - Low (Four channels)					

Question 2 (STSB) – Editorial Issues

The NRC staff identified the following editorial issues in attachment 3 (bases) of the LAR. Confirm and correct.

- TS 3.3.2.2.A.1 is missing “Time” from the proposed note for the Risk Informed Completion Time Program.
- TS 3.6.1.6.C.1 is missing “Program.” from the proposed note for the Risk Informed Completion Time Program.

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2. Technical Specifications Task Force, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,” Traveler TSTF-505, Revision 2, dated July 2, 2018 (ML18183A493).

3. Cusumano, V. G., U.S. Nuclear Regulatory Commission, letter to Technical Specifications Task Force, “Final Revised Model Safety Evaluation of Traveler TSTF-505, Revision 2, ‘Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,’” dated November 21, 2018 (ML18269A041).

4. Golder, J. M., U.S. Nuclear Regulatory Commission, letter to B. Bradley, Nuclear Energy Institute, “Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, “Risk-Informed Technical Specification Initiative 4B, Risk Managed Technical Specifications (RMTS) Guidelines” (TAC No. MD4995),”dated May 17, 2007 (ML071200238).

5. Nuclear Energy Institute, “Risk Informed Technical Specifications Initiative 4b Risk Managed Technical Specifications (RMTS) Guidelines,” NEI 06-09-A, Revision 0, dated November 2006 (ML12286A322).

6. U.S. Nuclear Regulatory Commission, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking, Final Report,” NUREG-1855, dated March 2017 (ML17062A466).

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12. Nuclear Energy Institute, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01, Revision 4F, dated April 2018, (ML18120A069).

13. U.S. Nuclear Regulatory Commission, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Regulatory Guide 1.160, Revision 4, dated August 2018 (ML18220B281).

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15. Idaho National Laboratory, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," NUREG/CR-6928, dated February 2007 (ML070650650).

16. Reisi-Fard, M., U.S. Nuclear Regulatory Commission, memorandum to J. G., Giitter, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis," dated May 30, 2017 (ML17031A269).

17. U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 2, dated March 2009 (ML090410014).

18. Zoulis, A. M., memorandum to M. Franovich, "Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Probabilistic Risk Assessments," dated May 6, 2022 (ML22014A084).

19. Energy Northwest, "CGS Fire PRA Uncertainty and Sensitivity Analysis Notebook," Report FPSA-2-UNC-0001, Revision 1.

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SUBJECT: COLUMBIA GENERATING STATION − REGULATORY AUDIT AGENDA AND QUESTIONS FOR LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4B" (EPID L-2022-LLA-0023) DATED JULY 6, 2022

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