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October 4, 2022

GO2-22-096 10 CFR 50.90

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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 SUPPLEMENT TO LICENSE AMENDMENT REQUEST TO ADOPT TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES – RITSTF INITIATIVE 4b"**

Reference:

1. Letter from J. K. Dittmer (Energy Northwest) to U.S. Nuclear Regulatory Commission, "License Amendment Request to Adopt TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b'," dated February 3, 2022 (ADAMS Accession No. ML22034A992)

2. Letter from U.S. Regulatory Commission to Mr. Robert Schuetz (Energy Northwest), "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

Dear Sir or Madam:

In Reference 1, Energy Northwest requested an amendment to the Columbia Generating Station (Columbia) Technical Specifications (TS). The proposed amendment would modify Columbia's TS requirements to permit the use of Risk-Informed Completion Times in accordance with TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b".

In Reference 2, the Nuclear Regulatory Commission (NRC) requested a virtual audit to improve the efficiency of staff reviews. As a result of this regulatory audit that was conducted on August 1-5, 2022, Energy Northwest is supplementing the License Amendment Request (LAR) to support the NRC staff's review of the proposed change.

The attachment to this letter provides information requested by the NRC staff during the regulatory audit to amend Reference 1.

The information contained in the attachment does not affect the Technical Analysis or No Significant Hazards Consideration conclusions contained in the LAR. Additionally, the information provided in this supplement 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.

In accordance with 10 CFR 50.91, Energy Northwest is notifying the State of Washington of this amendment supplement by transmitting a copy of this letter and attachment to the designated State Official.

This letter and its attachment contain no new commitments.

If there are any questions or if additional information is needed, please contact Mr. R.M. Garcia, Licensing Supervisor, at 509-377-8463.

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

Executed this 4 day of October 2022.

Respectfully,

DocuSigned by:
Jeremy Hauger
B3D71514C4434A3...

Jeremy S. Hauger
Vice President, Engineering

Attachment: Audit Question Responses

cc: NRC RIV Regional Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
CD Sonoda – BPA/1399
EFSECutc.wa.gov – EFSEC
E Fordham – WDOH
R Brice – WDOH
L Albin – WDOH

The following information is being provided to amend Reference 1 as a result of the regulatory audit held on August 1-5, 2022 in response to Reference 2.

Audit Question 3 (APLA/APLC) Part a and b – Impact of Seasonal Variations

Response

a.) The Probabilistic Risk Assessment (PRA) model includes extreme cold weather in the Containment Nitrogen and Service Water System A and Service Water System B logic. A basic event is used for the conditional probability of extreme cold weather. The PARAGON model includes a variable for EXTREMECOLD that sets the conditional probability to TRUE. Otherwise, time-averaged data is used.

Other conditions are evaluated to determine the impact on plant equipment availability. Equipment that is determined to be unavailable is set to TRUE and evaluated in the Configuration Risk Management Program (CRMP). This equipment would also be included as unavailable in the Risk-Informed Completion Time calculation. For example, hot summer weather could challenge temperature limits. If temperature limits were unable to be maintained, then supported systems would be considered unavailable.

Severe weather conditions that do not affect plant equipment availability are evaluated qualitatively as High-Risk Evolutions (HRE). An HRE will elevate the Plant Risk Level (risk color) and require risk management actions.

b.) The PARAGON model includes a variable for EXTREMECOLD that is set to TRUE when the outside temperature is less than 10 degrees Fahrenheit.

The CRMP is controlled per station procedure PPM 1.5.14. Step 5.6.3 directs that Operations control room staff evaluate emergent conditions:

> Perform a PARAGON evaluation for emergent work and for questions concerning the current PARAGON evaluation on weekends, backshift, and if the Work Week Manager is not onsite. Notify the Work Week Manager of any plant risk increase to implement the required risk management actions in accordance with Attachment 9.11. Contact the PSA engineer to evaluate elevated risk conditions and develop risk management actions as appropriate.

Emergent conditions are evaluated to determine the impact on plant equipment availability. If it is determined that equipment is unavailable, then the PARAGON schedule is evaluated for risk impact, and risk management actions are implemented as required.

Audit Question 4 (APLA) Part a and b – Performance Monitoring

Response

a.) The Columbia Generating Station (CGS) Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in NUMARC 93-01, as endorsed by Regulatory Guide 1.160.

b.) Audit Question APLA-04 requested answering either part a.) or part b.; therefore, part b.) is not applicable.

Audit Question 5 (APLA/APLC) Parts a through e – In-Scope LCOs and Corresponding PRA Modeling

Response

a.) The Reactor Protection System (RPS) model in the CGS PRA is directly taken from NUREG/CR-5500 without any alteration to data. The RPS is divided into mechanical and electrical functions in the PRA model. The mechanical function (Event CM) failures address valves, rod insertion, and accumulator integrity. NUREG/CR-5500, Volume 3, Table 5 calculates a value of 2.15E-6 per demand for the mechanical failure probabilities (i.e., rod, hydraulic control unit) of the RPS, which is the value used in the PRA model. The electrical portion of the RPS (Event CE) generates the scram signals from the sensors, logical processing of the signal, and the de-energizing of the scram solenoids. NUREG/CR-5500 calculates a value of 3.78E-6 per demand for the electrical failure probabilities (i.e., Channel, Trip System) of the RPS, which is the value used in the PRA model. These values are based on the Fussel-Vesely of the basic event contributions from the cutset solutions. Therefore, the sum of the values of 5.93E-6 is conservative when compared to the total failure probability of 5.8E-6 that is presented in Section 5 of NUREG/CR-5500 for the RPS unavailability.

The value 3.78E-6 that is used in Event CE is the RPS failure probability when no credit is given for operator action to manually scram the RPS. When manual scram is credited, NUREG/CR-5500 calculates a value of 5.20E-7. In the CGS PRA, the 3.78E-6 value is used in all accident sequences, except for small break loss of coolant accident (LOCA) and in safety relief valve (SRV) LOCAs. For those initiators, the 5.20E-7 value is used in Event CE2. For the real time risk model (RTR), only the value representing no manual scram credit will conservatively be used explicitly and Event CE (and the logic described herein) will replace Event CE2).

b.) When compared to the NUREG/CR-5500 model, the CGS RPS point estimate events used in the PRA model are the same.

The NUREG/CR-5500 model uses data from 1984 to 1995, which predates data from NUREG/CR-6928 and its update through 2020. A review of the NUREG/CR-6928 data for RPS systems indicates that the data comes from the RPS System Study from NUREG/CR-5500 for the period of 1986 through 1995. The primary difference is that the NUREG/CR-5500 Volume 3 data is for the boiling water reactor (BWR) RPS systems, while the RPS data reported in NUREG/CR-6928 (2020 update) comes from the combined study of NUREG/CR-5500 Volumes 2, 3, 10, and 11 (i.e., Westinghouse, General Electric, Babcock & Wilcox) that is not differentiated by reactor brand or type in the combined analysis. Therefore, it is recognized that the BWR RPS Data NUREG/CR-5500 Volume 3 remains the industry state of the art data for RPS data for BWRs. Some data types can be compared to newer data from NUREG/CR-6928 for 2006 through 2020 (taken from the Equipment Performance and Information Exchange System and Reliability and Availability Data System). However, these data types showed a small decrease in probabilities (approximately 30%) over the older data used in NUREG/CR-5500 Volume 3. Therefore, using the older data remains acceptable with only a minor conservatism added to the solution value.

Because the data in NUREG/CR-5500 was specifically taken from RPS systems and has not been updated by a more current study, it remains the industry state of the art study and meets the intent of American Society of Mechanical Engineers (ASME) DA-C1 as a generic data source.

c.) The NUREG/CR-5500 model is a simplified model that addresses two RPS automatic functions for initiating the scram signal, high reactor pressure, and low reactor vessel water level (equivalent to CGS Functions 3 and 4). These two functions are appropriate to represent the full range of plant events, because for identified events, it has been shown that at least two functions are challenged for the design bases accidents in the Final Safety Analysis Report (FSAR). Attachment 6 of Reference 1 identifies at least one diverse trip function for each Technical Specification (TS) function. Therefore, it is conservative to assume that there are only two RPS scram functions challenged for any plant event. It is also appropriate for these two functions to represent the reliability of other functions, as there is little variation between the SSCs of each function from a data standpoint. Therefore, the probability of an RPS scram failure, as calculated by NUREG/CR-5500 is a conservative model for the whole range of functions and their SSCs proposed in the risk-informed completion time (RICT) limiting conditions for operation (LCO).

d.) The Columbia RPS model meets the 2009 ASME/ANS PRA standard CC-II requirements. Requirement SY-A7 CC I-II states:

> "Develop detailed systems models, unless (a) sufficient system-level data are available to quantify the system failure probability, or (b) system failure is dominated by operator actions, and omitting the model does not mask contributions to the results of support systems or other dependent-failure

modes. For case (a), USE a single data value only for systems with no equipment or human-action dependencies, and if data exist that sufficiently represent the unreliability or unavailability of the system and account for plant-specific factors that could influence unreliability and unavailability. Examples of systems that have sometimes not been modeled in detail include the scram system, the power-conversion system, instrument air, and the keep-fill systems. JUSTIFY the use of limited (i.e., reduced, or single data value) modeling."

i. The use of the NUREG/CR-5500 model as a point estimate meets the ASME standard requirements because the model can be sufficiently addressed by a point estimate, the model is not dependent on human actions, and the data was system-specific and used state of the art approaches still in practice today. These are consistent with Category II and III ASME supporting requirements, and the results of the model remain conservative when compared to newer data. Also, the ASME standard cites RPS models as an example where a point estimate system model is acceptable.

ii. This question is not applicable because the use of NUREG/CR-5500 was previously justified.

iii. The NUREG/CR-5500 RPS model is being used as a surrogate. The change proposed in the RICT model is to break the existing electrical RPS point failure estimate event down into its four subsystems arranged in the one-out-of-two taken twice logic, consistent with the NUREG/CR-5500 model, but as subsystem (channel) modules versus individual components. Five events will be used to accurately calculate the current RPS electrical event probability (3.78E-6): the four subsystem events and one common cause event. The NUREG/CR-5500 cutset events were categorized, according to the approach in NUREG/CR-5500, into independent failures and common cause failure. The RPS failure probability from independent and common cause failures was then calculated. Independent failure models for each channel were then calculated by reversing the one-out-of-two taken twice logic. The calculated values used in Reference 1 for the independent subsystem event is 2.0E-4, and for the common cause failure event is 3.7E-6. For the RICT program, it is proposed to map any RPS structure, system, component (SSC)-related unavailability to the subsystem that it is associated with. This is conservative, as the entire subsystem will be failed in the PRA model, whereas in the plant, all the other functions would remain available in that subsystem (i.e., relays able to open). The manual pushbutton associated with a scram subsystem would also be applied to fail the modeled subsystem, which meets the requirement to address the manual scram aspect of RPS. The CCF probability is not adjusted, as RICTs will not be entered following a failure if the extent of condition determination identifies common cause contributors for the failed SSC. The approach was compared to results from removing a channel from service in the NUREG/CR-5500 model by failing all K relays associated with each subsystem as shown in Table APLA-Q5.1 below.

Table APLA-Q5.1. Comparison of RPS Model Results

Out of Service Case	Proposed RICT Model Result	NUREG/CR-5500 Model Result	Assessment
No Maintenance	3.78E-6	3.77E-6	RICT surrogate model equivalent
All A Channel K Relays (Div 1 impact)	2.04E-4	5.07E-6	RICT surrogate model bounding
All A and B Channel K Relays (Div 1 and Div 2 impact)	4.04E-4	6.37E-6	RICT surrogate model bounding
All A and C Channel K Relays (Div 1 unavailable)	1	9.97E-1	RICT surrogate model equivalent

As demonstrated, the proposed RPS model generates results that are bounding for RICT entry and meets the 2009 ASME/ANS PRA standard CC-II requirements.

This approach is an extension of existing approaches and will be implemented as a matter of model maintenance. This approach does not represent a new approach or model upgrade, as the underlying data is the same as what was previously peer reviewed.

iv. This question is not applicable because the proposed surrogate was previously justified as bounding for RICT entry.

e.) The RPS data used for Event CE in the internal events model and used for the seismic model are the same. Seismic impacts on RPS logic would result in relay chatter, which would open RPS channel circuits and scram the reactor; therefore, seismic events do not degrade reliability of the RPS to scram the reactor. Seismic impacts on the scram function are modeled for control rods, hydraulic control units, and the alternate rod insertion system.

Audit Question 6 (APLA) Parts a through k – Missing Information in Table E1-1

Response

The following cells shaded in grey show updates to Table E1-1, In-Scope TS/LCO Conditions to Corresponding PRA Functions in response to Audit Question 6 (APLA).

a.) TS 3.3.4.1, LCO a.2, "Turbine Governor Valve (TGV) Fast Closure, Trip Oil Pressure – Low"

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) Function a.2. Turbine Governor Valve (TGV) – Fast Closure, Trip Oil Pressure - Low (Four channels)	No	Trip Both Recirculation Pumps	Two Turbine Trip Valve Closure channels in either trip system OR Two Turbine Governor Valve Fast Closure Trip Oil Pressure-Low channels in either trip system	None	See Note 3

b.) TS 3.3.4.2, LCO b, "Reactor Vessel Steam Dome Pressure – High"

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
3.3.4.2.A	Anticipated Transient Without SCRAM Recirculation Pump Trip (ATWS-RPT) Instrumentation - one or more channels inoperable	Function a. RVWL – Low Low Level 2 (Four channels) (See Note 4) Function b. Reactor Vessel Steam Dome (RVSD) Pressure – High (Four channels) (See Note 4)	Yes	Trips Recirculation Pump associated with the trip system	Two RVWL – Low, Low, Level 2 channels in one of two trip systems <u>OR</u> Two RVSD Pressure – High channels in one of two trip systems	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program.

c.) TS 3.3.5.1, Function 1.b, "Drywell Pressure – High"

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
3.3.5.1.B	ECCS Instrumentation – As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	**1. ECCS Actuation Instrumentation for Low Pressure Coolant Injection-A (LPCI) and Low Pressure Core Spray (LPCS) Subsystems**					
		1.a. RVWL – Low Low Low, Level 1 (Two channels)	Yes	Actuate both LPCI A and LPCS	One RVWL Level 1 channel OR One Drywell Pressure - High channel from Two Subsystems	Same as Design Success Criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program.
		1.b. Drywell Pressure – High (Two channels)	Yes	Actuate both LPCI A and LPCS			
		2. ECCS Actuation Instrumentation for LPCI B and LPCI C Subsystems					
		2.a. RVWL – Low Low Low, Level 1 (Two channels)	Yes	Actuate both LPCI B and LPCI C	One RVWL – Level 1 channel OR One Drywell Pressure - High channel from Two Subsystems	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program.
		2.b. Drywell Pressure – High (Two channels)	Yes	Actuate both LPCI B and LPCI C			
		3. ECCS Actuation Instrumentation High Pressure Core Spray (HPCS) System					
		3.a. Reactor Vessel Water Level – Low Low, Level 2 (RVWL2) (Four channels)	Yes	Actuate HPCS	Two RVWL Level 2 differential pressure switches	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program.
		3.b. Drywell Pressure – High (Four channels)	Yes	Actuate HPCS	Two Drywell Pressure - High pressure switches	Same as Design Success Criteria	

d.) TS 3.3.5.1, Function 3.e, "Suppression Pool Water Level – High"

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
3.3.5.1.D	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	**3. ECCS Actuation Instrumentation HPCS System**					
		3.d. Condensate Storage Tank (CST) Level - Low (Two channels) 3.e. Suppression Pool Water Level (SPWL) - High (Two channels)	Yes	Change HPCS suction from CST to Suppression Pool for continued HPCS operation	One of two channels of CST Level – Low OR One channel of SPWL - High	One of four channels of CST Level -Low OR One channel of SPWL- High	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program. Two additional non-TS CST-Level – Low channels are modeled explicitly with the same detail as TS channels.

e.) TS 3.3.5.1, Function 5.3, "Accumulator Backup Compressed Gas System Pressure – Low"

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
3.3.5.1.F	ECCS Instrumentation - As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	**ADS initiation logic and instrumentation functions**					
		4.a. RVWL – Low Low Low, Level 1 (Two channels)	No	Initiate ADS Train A	Two Reactor Vessel Water Level – Low Low Level 1 channels in either of two ADS actuation systems	None ADS Inhibit assumed. Only manual depressuri-zation credited	Failure of Train A ADS SOVs to open is used as an equivalent surrogate for RICT calculation [6]
		5.a. RVWL – Low Low Low, Level 1 (Two channels)	No	Initiate ADS Train B			Failure of Train B ADS SOVs to open is used as an equivalent surrogate for RICT calculation [6]
		4.c. RVWL – Low Level 3 (Permissive) (One channel)	No	ADS Permissive Train A	One-RVWL – Low Level 3 channel in either of two ADS actuation systems	None ADS Inhibit assumed. Only manual depressuri-zation credited	Failure of Train A ADS SOVs to open is used as an equivalent surrogate for RICT calculation [6]
		5.c. RVWL – Low Level 3 (Permissive) (One channel)	No	ADS Permissive Train B			Failure of Train B ADS SOVs to open is used as an equivalent surrogate for RICT calculation [6]
		4.f. Accumulator Backup Compressed Gas System Pressure – Low (Three channels)	No	Align backup nitrogen on low header gas pressure	Two-out-of-three compressed gas header system pressure – Low in either of two ADS actuation systems	One of two nitrogen bottle racks supplying the ADS safety related compressed gas header.	A conservative surrogate of unavailability of the Train A nitrogen supply header will be used.
		5.e. Accumulator Backup Compressed Gas System Pressure – Low (Three channels)	No				A conservative surrogate of unavailability of the Train B nitrogen supply header will be used.

f.) TS 3.3.6.1, Function 5.a, "Residual Heat Removal (RHR) Shutdown Cooling (SDC) System Isolation, Pump Room Area Temperature – High"

g.) TS 3.3.6.1, Function 5.b, "RHR SDC System Isolation, Pump Room Area Ventilation Differential Temperature – High"

h.) TS 3.3.6.1, Function 5.c, "RHR SDC System Isolation, Heat Exchanger Area Temperature – High" (Room 505, 507, 605, and 606 Area)

i.) TS 3.3.6.1, Function 5.d, "RHR SDC System Isolation, Reactor Vessel Water Level – Low, Level 3"

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
3.3.6.1.A	Primary Containment Isolation Instrumentation - one or more channels inoperable	**5. Residual Heat Removal (RHR) Shutdown Cooling (SDC) System Isolation (Function e. and f.)**					
		Functions 5.a through 5.d are for operational mode 3 only and are outside the scope of the RICT program.					
		5.e. Reactor Vessel Pressure – High (Two channels)	No	Automatic Isolation of RHR SDC valves	One RVP – High channel on either isolation system	None	See Note 7.c
		5.f. Manual Initiation (Four channels, two per switch/PB pair)	No	Manual isolation of RHR SDC valves	One switch/PB pair on either isolation system	None	See Note 7.c

j.) 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”

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
3.3.8.1.B	Loss of Power (LOP) instrumentation - As required by Required Action A.1 and referenced in Table 3.3.8.1-1.	**1. Divisions 1 and 2 - 4.16 kV Emergency Bus Undervoltage**					
		1.a.TR-S Loss of Voltage (LOV) – 4.16 kV Basis (Four channels, two per bus	No	LOV sensing capability and time delay to initiate trip of offsite power circuit, start the associated emergency diesel generator (DG) and initiate source transfer to connect to the next available power source on Division 1 or 2 4.16 kV bus.	One LOV channel per bus	None	Function 1.a will be conservatively mapped to modeled relays that fail DG LOV start signal and TR-B transfer, which are affected circuits of the LOV channels. Operable LOV channel will conservatively not be credited for RICT.
		1.b. Loss of Voltage - Time Delay (Four channels, two per bus)	Yes		Two Time Delay channels per bus	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program.

k.) TS 3.3.8.1, Function 1.e, “Divisions 1 and 2 – 4.16 kV Emergency Bus Undervoltage, Degraded Voltage – 4.16 kV Basis”

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
3.3.8.1.C	Loss of Power (LOP) instrumentation - As required by Required Action A.1 and referenced in Table 3.3.8.1-1.	**1. Divisions 1 and 2 - 4.16 kV Emergency Bus Undervoltage (See Note 8)**					
		1.c. TR-B Loss of Voltage - 4.16 kV Basis (Two channels, one per bus)	Yes	Sense LOV on Backup Trans-former and Transfer Bus to DG	One LOV channel per Bus	Same as Design Success Criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program.
		1.d. TR-B Loss of Voltage - Time Delay (Six channels, three per bus)	Yes	Time Delay for power recovery	Three TD channel per bus	Same as Design Success Criteria.	
		1.e. Degraded Voltage (DV) - 4.16 kV Basis (Six channels, three per bus)	Yes	Sense Essential Bus DV and Transfer Bus to DG	Two DV channels per bus	Same as Design Success Criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program.
		1.f. Degraded Voltage - Primary Time Delay (Six channels, three per bus)	Yes	Time Delay for power recovery	Two DV Primary TD Channels per bus	Same as Design Success Criteria.	
		1.g. Degraded Voltage - Secondary Time Delay (Six channels, three per bus)	Yes	Time Delay for power recovery	Three DV Secondary TD Channel per bus	Same as Design Success Criteria.	
		2. Division 3 – 4.16kV Emergency Bus Undervoltage					
		2c. Degraded Voltage - 4.16 kV Basis (Three channels)	Yes	Sense HPCS Bus DV and Transfer to DG	Two DV Channels	Same as Design Success Criteria.	SSCs are modeled consistent with the TS scope and can be directly included in the CRM tool for the RICT program.

Notes for Table E1-1 are listed below for reference. Edits to the Notes section for Table E1-1, In-scope TS/LCO Conditions to Corresponding PRA Functions, are shown in italicized text.

1. Individual RPS instrumentation inputs to the RPS logic system are not modeled in the PRA. The RPS failure probability is based on the NUREG/CR-5500 Volume 3 model (Reference 6). The PRA model uses a single point estimate event to represent RPS failure due to electrical SSC failures. For the RICT calculation, the Reference 6 model cutsets were reviewed and a probability of failure of each of the four sub-systems (channels) was developed. A new simplified RPS model using the four sub-system failure events was developed based on the one-out-of-two taken twice logic. This new RPS model generates the exact base probability of the NUREG/CR-5500 model. For any subsystem with a function considered inoperable or bypassed, the associated subsystem event was failed in this new RPS model to calculate the RICT. This new simplified RPS model was validated to provide more conservative results than the NUREG/CR-5500 model when a function channel is inoperable or bypassed. This new simplified RPS model is used to calculate the values in Table E1-2 with two subsystems out of service (one bypassed and another inoperable), which is allowed for some functions. This RPS model addresses both Condition A and Condition B of TS 3.3.1.1. The CGS PRA CRM program model will be updated to use this new simplified RPS electrical failure logic model prior to RICT program implementation. This is not a model upgrade, but only a model maintenance item and does not introduce any new PRA methods.

2. There are three reactor feedwater system channels of Reactor Vessel Water Level (RVWL) – High used to trip the reactor feed pumps (RFPs) and the main turbine. Failure of the RFP and the main turbine high water level trip functions are not modeled. Failure or unavailability of these trip functions could result in damage to the RCIC turbine, RFP turbines, and main turbine, of which RCIC and the RFPs provide PRA functions. A similar impact from the HPCS discharge valve isolation signal on RVWL high is expected. As a conservative surrogate for the maintenance of any TS 3.3.2.2.A high water level channel or TS 3.3.5.1.C function 3.c. high water level channel, RCIC and the RFPs will be failed. This is conservative because the RCIC and RFPs are assumed failed regardless of a failure of reactor vessel level control and failure of all channels of the trip function.

3. The EOC-RPT instrumentation initiates a recirculation pump trip (RPT) to reduce the peak reactor pressure and power resulting from turbine trip or generator load rejection transients to provide additional margin to the core thermal MCPR Safety Limit. This is not a PRA modeled function. However, EOC-RPT provides another backup to the ATWS-RPT for load reject transients and can be conservatively assessed using a surrogate. Failure of recirculation pump breakers to trip will be used as a conservative surrogate for the RICT calculation. The surrogate is conservative as the breakers are tripped by the EOC-RPT logic. Thus, the RICT calculated for this surrogate is bounding for each channel because one channel out of service does not prevent the trip of the RPT breakers.

4. ATWS-RPT system instrumentation is part of the redundant reactivity control system and has 2 independent trip systems each composed of two channels of each functional input. Each trip system uses a 2-out-of-2 logic for each function. Thus, either two Reactor Water Level - Low Low, Level 2 or two Reactor Vessel Steam Dome Pressure - High signals are needed to trip a trip system. One trip system trips one recirculation pump and the other system trips the other recirculation pump.

5. Instrumentation to open 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.

6. The Fire PRA models individual SOVs and dependencies for each SRV (ADS and non-ADS). For the RICT evaluation in Table E1-2, the Fire PRA logic was also added to the Internal events, Internal flood, and Seismic hazards models for the quantification. Additionally, ADS SRVs are only modeled by common cause failures or their supporting SOVs and supports. For the RICT calculation, individual ADS SRV valve body independent failure to open events were added. This logic will be included in the CRM program models for all hazards prior to RICT program implementation. This is not a model upgrade, but only a model maintenance item and does not introduce new PRA methods.

7. One isolation system is associated with the inner primary containment isolation valves and the other isolation system is associated with the outer primary containment isolation valves with the success criteria being closure of one of the two isolation valves. Where SSCs are modeled consistent with the TS scope, SSC will be directly used for unavailability in the CRM tool for the RICT program. Otherwise, the use of surrogates will be as follows:

    a) For functions 1.e, MS isolation on MS tunnel temperature – high, and 1.f, MS isolation on MS tunnel differential temperature – high, will use the leak detection (LD) monitors for each inoperable channel as a conservative surrogate. The LD monitors are modeled in the PRA, but the individual temperature elements are not modeled in the PRA. Two LD monitors each receive inputs from two MS tunnel temperature channels (one element per channel) and two MS tunnel differential temperature channels (two elements per channel). With one or more elements inoperable, the associated TD monitor(s) will be failed in the PRA. This is conservative because diversity for functions 1.e and 1.f will not be credited.

    b) For RWCU isolation function 4.f, Pump Room Temperature – high channels will use the LD monitors for each inoperable channel as a conservative surrogate. The LD monitors are modeled in the PRA, but the individual temperature elements are not modeled in the PRA. Two LD monitors each receive inputs from two temperature elements channels each. With one or more element inoperable, the associated TD monitor(s) will be failed in the PRA. This is conservative because redundancy of two channels per monitor is not credited for function 4.f. For other unmodeled Function 4 subsets, the RWCU isolation valve will be failed open as a conservative surrogate for the associated instrumentation subsystem, unless the valve is isolated and de-energized. This is conservative because one channel inoperable will not render the associated isolation valve not closable.

c) For functions 5.e. and 5.f, a RHR shutdown cooling suction line isolation MOV will be failed open as a conservative surrogate for the associated instrumentation subsystem, unless the MOV is isolated and de-energized. This is conservative because one channel inoperable will not render the associated isolation valve not closable.

d) For other unmodeled functions (2.a, 2.d, 3 and 6) a large pre-existing containment isolation failure will be used as a conservative surrogate for the LCO condition. See Notes 9 and 10 for the background of the large pre-existing containment failure event and how it will be applied. This approach is conservative because unmodeled functions have been determined not to contribute to LERF.

8. Each 4.16 kV emergency bus has its own independent LOP instrumentation and associated trip logic. The voltage for the Division 1, 2, and 3 buses is monitored at two levels, which can be considered as two different undervoltage functions: loss of voltage and degraded voltage. For Division 1 and 2, the loss of voltage function is monitored by two instruments per bus and the degraded voltage is monitored by three instruments per bus. The degraded voltage signal is generated when a degraded voltage occurs for a specified time interval and also provides a backup for the undervoltage functions.

9. The containment air locks are not explicitly modeled in the CGS PRA. Since the containment airlocks are not modeled, there are no explicit PRA Success Criteria. However, the LCO condition will be modeled using the pre-existing large containment isolation failure as a conservative surrogate in the PRA. The pre-existing large containment failure event probability was derived by the Pacific Northwest Laboratory (PNL) for the NRC (see EPRI Risk Impact Assessment of Extended Integrate Leak Rate Test Intervals, TR-101824) plus the use of NUREG-1493. Columbia is not an outlier in the use of this generic industry accepted data that addresses the operating experience-based probability of containment release pathways being larger than “small”. Because the containment hatch doors have no dependencies, for the LCO condition, it is appropriate to increase the failure probability of the surrogate event in the CRM program (versus setting to logical True) for the RICT calculation. This added probability represents the likelihood of failure of the redundant operable door. A bounding probability was derived from the square root of the pre-existing large isolation failure probability. The RICT in Table E1-2 was calculated using this approach.

10. Where PCIV SSCs are modeled consistent with the TS scope, unavailability can be directly included in the CRM tool for the RICT program. Unmodeled PCIVs were screened in the PRA due to from LERF consideration based on PCIVs being smaller than 2 inches or if the PCIV isolates a closed system inside containment. However, a conservative assessment using a surrogate pre-existing large containment isolation failure will be used to address individual unmodeled PCIV unavailability (See basis for

event in Note 9). Although very conservative, screened penetrations shall be assessed with this surrogate. Where the redundant unisolated operable isolation valve(s) is(are) fail safe, the respective failure probability(ies) shall be added to the surrogate event in the CRM program. If any remaining unmodeled unisolated operable isolation valve is not fail safe, the surrogate shall be set to failed (logical True) for the LCO condition. The RICT values in Table E1-2 were calculated assuming the redundant valve is not fail-safe, and the surrogate event was set to logical True. This approach is conservative because unmodeled PCIV SSCs have been determined not to contribute to LERF.

11. Failure of the drywell spray is used as a conservative surrogate for reactor building to suppression chamber vacuum breakers. For sequences requiring drywell spray, success of vacuum breakers was assumed in the PRA, and the possibility of failure was screened as insignificant. Failure of sprays is modeled to cause containment failure. For the LCO condition, failure of the drywell spray function is a conservative surrogate. This approach is conservative because a single vacuum breaker unavailability will not result in failure of the spray function or containment failure for the accidents for which spray is modeled.

12. If sufficient suppression pool to drywell vacuum breakers fail to open under certain accident sequences, failure of the drywell floor may occur and a loss of LOCA vapor suppression will occur. The drywell floor seal failure event is a conservative surrogate to model vapor suppression bypass. For the LCO condition, the event probability will be increased to that of a bounding common cause factor given a single vacuum breaker is failed. This approach is conservative because a single vacuum breaker unavailability will result in an insignificant increase in the vacuum breaker function unreliability, and the common cause factor is orders of magnitude higher than the calculated risk. Further, if common cause is determined, then the extent of condition determination will result in multiple vacuum breakers inoperable, and a RICT will not be entered.

13. The RHR system contains three separate pump trains, two of which contain heat exchangers for heat removal. The third pump train is for LPCI functions only.

14. Each Division 1 and 2 125 VDC subsystem has *an installed redundant standby charger that is normally isolated and can be manually aligned to meet the LCO requirements. The installed redundant spare battery charger is a design feature of the 125 VDC distribution system and is therefore, modeled in the PRA as a recovery action option*.

15. 120/240V 1 Phase power panels PP-7A-F, PP-8A-G, and PP-4A, and 120/208V 3 phase power panels PP-7A-A-A and PP-8-A-A-A are not modeled in the PRA and will use the source motor control center or power panel as a conservative surrogate. This is conservative because additional PRA loads will be impacted by the surrogate being inoperable.

16. 125 VDC distribution panels DP-S1-1E and MC-S1-2D are not modeled in the PRA and will use the source motor control system or power panel as a conservative surrogate.

17. *A qualified offsite circuit meets GDC-17 requirements and consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit AC power from the offsite transmission network to the onsite Class-1E 4.16kV ESF switchgear buses.*

18. *Energization transients of any DC loads that are beyond the capability of the battery charger and normally require the assistance of the battery will not be able to be brought online. To maintain the function of the valves supplied by the 250 VDC system, diverse AC-powered isolation valves are required to be available to maintain the function. This is consistent with TS 3.8.4 Bases.*

19. *Energization transients of any 250 DVC loads that are beyond the capability of the charger may fail the charger. To bound this condition with the 250 VDC battery unavailable, the Basic Event E-C2-1 UNABLE TO MAINTAIN DC LOADS will be set to TRUE.*

20. *Loss of any DC electrical power subsystem does not prevent minimum safety function from being performed. No situation exists where single DC bus failure would prevent plant personnel from achieving a reactor cold shutdown condition.*

Audit Question 12 (APLA/APLC) Parts 1 through 4 – PRA Modeling and Uncertainty of FLEX Equipment and Actions

Response

1) The selected Human Failure Events (HFEs) are those credited in the PRA for initiating the credited Diverse and Flexible Coping Strategies (FLEX) strategies. For the sensitivity study, the HFEs and FLEX equipment were failed (TRUE) so that no credit was given to FLEX for the sensitivity. This is bounding as FLEX is completely removed from credit in the PRA. This is true for all hazards.

2) Sensitivities were selected based on a review of accident sequences and LCOs that would be impacted by FLEX credit. Combinations of LCOs were also selected from these sensitivities that were perceived as sensitive to FLEX credit.

3) Table APLA-Q12.1 below summarizes the impact of FLEX equipment on RICT durations for the chosen sensitivities.

Table APLA-Q12.1. FLEX Sensitivity Results

RICT Configurations	Base RICT (days)	FLEX Not Credited		
		RICT (days)	% Change	RICT Difference (hours)
RCIC	30.00	30.00	0.0%	0
RHR-A	30.00	30.00	0.0%	0
RHR-B	30.00	30.00	0.0%	0
HPCS	18.87	18.67	-1.1%	-4.8
4 kV LOV Div B & DG3	5.94	5.80	-2.4%	-3.4
SW-B & DG3	4.43	4.38	-1.0%	-1.1

No configurations were identified that had a significant change in RICT. A 5% or larger change would be considered for significance review.

4) Neither the FLEX human error probabilities nor FLEX equipment failure probabilities are key sources of uncertainty based on the results of the sensitivities.

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

Response

Energy Northwest agrees that the phrasing contained in Attachment 1 of Reference 1 (page 6 of 15), specifically, “methods approved for use with this program in Amendment No. [###]” provides more clarity than the phrasing in the administrative controls for the RICT program in TS 5.5.16 paragraph e (see Insert 2 of Attachment 2 of Reference 1). Therefore, Energy Northwest proposes to replace paragraph e with the following:

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 approved for use with this program in Amendment No. [###], or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval.

Audit Question 02 (APLB) Parts ii through iv – Deviations from NRC Endorsed Guidance as Source of Modeling Uncertainty

Response

ii.) The CGS Fire PRA includes treatment of sensitive electronics as targets consistent with the guidance in NUREG/CR-6850 and NRC Frequently Asked Question (FAQ) 13-0004. Based on the ignition source walkdowns performed during the development of the Fire PRA, the physical analysis units (PAU) listed below are those in which sensitive electronics are assumed to be present. These PAUs include those housing the main set of control and relay panels supporting the plant. In order to account for sensitive electronics, the scenario development strategy consisted of failing the ignition source and any adjacent panel at the time of ignition or within the zone of influence very early in the event.

- The main control room (Fire Area RC-10): The main control board and all electrical cabinets (including those containing sensitive electronics) within the zone of influence of a source are failed at ignition (at t = 0, no credit for detection or suppression). This PAU was evaluated for hot gas layer conditions in support of main control room abandonment. The fire modeling results suggest that only large fires in the main control board generate hot gas layers descending under the cabinets within 5 to 10 minutes after fire starts. These are conditions that would generate temperatures inside closed cabinets affecting the sensitive electronics. At these times, control room abandonment has been triggered due to smoke descending at operators' height (e.g., visibility conditions, etc.). Therefore, the impact of failure of sensitive electronics is captured in the analysis as control room abandonment due to habitability will force shutdown from outside the main control room without relying on shutdown capabilities within the main control room.

- The High Pressure Core Spray (HPCS) diesel generator room (Fire Area DG-1): This PAU is modeled as a full compartment burn (i.e., all Fire PRA targets failed at time of ignition) with the exception of selected cables that were subjected to detailed fire modeling analysis.

- The diesel rooms (Fire Areas DG-2 and DG-3): These PAUs are modeled as full compartment burn. Therefore, all Fire PRA targets are failed at time of ignition.

- General equipment areas in the reactor buildings R-1-E471 and R-1-E522: These are large areas of the reactor building where detailed fire scenarios have been defined. All electrical cabinets (including those containing sensitive electronics) within the zone of influence of a source are failed at ignition (at t = 0, no credit for detection or suppression). In addition, all equipment targets (i.e., targets other than cables in conduits or cable trays) credited in the Fire PRA are failed in the first two damage states within four minutes from ignition and before a damaging hot gas layer is generated at the height of the ignition sources. Therefore, fire modeling treatment of sensitive electronics in this compartment is conservative relative to the damage criteria in FAQ 13-0004.

- PAUs in the Vital Island, including:
    - Switchgear room #1 (Fire Area RC-14)
    - Division 1 electrical equipment room (Fire Area RC-4)
    - Division 2 electrical equipment room (Fire Area RC-7)
    - Switchgear room #2 (Fire Area RC-8)

  All electrical cabinets in these four PAUs (including those containing sensitive electronics) within the zone of influence of a source are failed at ignition (at t = 0, no credit for detection or suppression). In addition, all equipment (i.e., targets other than cables in conduits or cable trays) targets credited in the Fire PRA are failed in the first two damage states within four minutes from ignition and before a damaging hot gas layer is generated at the height of the ignition sources. At four minutes, the hot gas layer temperature as predicted by the CFAST software (with no credit for Heat Soak model) is less than 65^{o} C. Therefore, sensitive electronics outside the zone of influence are failed within four minutes from ignition and before hot gas layer reaches 65^{o} C. This is conservative relative to the damage criteria in FAQ 13-0004.

The treatment for sensitive electronics described above for each of the PAUs in which sensitive electronics are assumed to be present does not rely on caveats about configurations that can invalidate the technical approach. Specifically: 1.) no credit is taken for thermoset damage criteria for calculating a time to damage for sensitive electronics (i.e., cabinets with PRA targets are assumed damage at ignition), 2.) sensitive electronics are failed at the time of ignition for the cabinets of fire origin, 3.) for PAUs modeled as full compartment burn, sensitive electronics are failed at time zero (i.e., no credit for time to damage), 4.) walkdowns of these fire compartments confirmed that doors to the electrical cabinets in these compartments are closed. Further, the contents of the cabinets in these compartments are not mounted to the door but are instead recessed into the cabinet itself.

iii.) The treatment for sensitive electronics is consistent with FAQ 13-0004.

iv.) The treatment for sensitive electronics is consistent with FAQ 13-0004. No implementation item to replace the current approach with an acceptable approach prior to the implementation of the RICT program is necessary.

Audit Question 1 (EICB) Parts a and b – Manual Scrams and Manual Trips

Response

a.) The following "Manual Scrams" are not modeled in the PRA:

- 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)

The following "Manual Scram" is implicitly modeled for a limited set of accident sequences using Event CE2, ELECTRICAL FAILURE OF SCRAM + MANUAL SCRAM, NUREG/CR-5500. Credit for manual scram is given for small LOCA and Stuck Open Relief Valve sequences only. All other accident sequences do not credit manual scram and use the Event CE, ELECTRICAL FAILURE OF SCRAM SYSTEM NUREG/CR-5500, which does not credit manual scram. For more details regarding the modeling of the RPS functions, see the response to Audit Question APLB-05.

- Function 10, Reactor Mode Switch – Shutdown Position

Below is a discussion on Columbia operations procedures and times associated with the actions evaluated as adequate.

1) Function 1b, Intermediate Range Monitors – Inop [Inoperable] is not credited in any FSAR transient/accident. Therefore, there is no FSAR required time. IRMs are bypassed at significant power [when the mode switch is turned to run (Mode 1)]. In Mode 2, the Average Power Range Monitor Neutron Flux – High (Setdown) provides automatic protection. Plant procedure PPM 1.3.1, Step 4.8.3.d fourth bullet includes the direction for operators to scram prior to exceeding an RPS setpoint. Operators are trained on scram setpoints regularly and the annunciator response procedures include setpoints. Plant procedure PPM 3.3.1 immediate actions describe inserting a manual scram by placing the mode switch in shutdown and depressing the manual scram pushbuttons.

2) Function 2c, Average Power Range Monitors – Neutron Flux – High. FSAR Section 15.1.6 Inadvertent Residual Heat Removal Shutdown Cooling Operation credits operator action to limit power rise after 10 minutes. The operating crews are trained on reactivity control regularly and prior to planned startups and shutdowns. The operating crews use the station developed control rod pull sheets in conjunction with the startup (PPM 3.1.2) and

shutdown (PPM 3.2.1) procedures. Slow positive reactivity additions can be compensated for by inserting control rods either during a startup or shutdown. Additionally, PPM 1.3.1, Step 4.8.3.d fourth bullet includes the direction for operators to scram prior to exceeding an RPS setpoint. Operators are trained on scram setpoints regularly and the annunciator response procedures include setpoints. PPM 3.3.1 immediate actions describe inserting a manual scram by placing the mode switch in shutdown and depressing the manual scram pushbuttons. Additionally, Function 2b, Simulated Thermal Power - High is a diverse instrumentation to Function 2c, Average Power Range Monitors – Neutron Flux – High for the FSAR transient accident Inadvertent Residual Heat Removal Shutdown Cooling Operation in FSAR Section 15.1.6. Therefore, Manual Scram or Manual Trip is not solely relied upon as a diverse means for Function 2c, Average Power Range Monitors – Neutron Flux – High.

3) Function 2d, Average Power Range Monitor – Inop is not credited in any FSAR transient/accident. Therefore, there is no FSAR required time. Plant procedure PPM 1.3.1, Step 4.8.3.d fourth bullet includes the direction for operators to scram prior to exceeding an RPS setpoint. Operators are trained on scram setpoints regularly and the annunciator response procedures include setpoints. Plant procedure PPM 3.3.1 immediate actions describe inserting a manual scram by placing the mode switch in shutdown and depressing the manual scram pushbuttons.

4) Function 2e, Average Power Range Monitors – 2-Out-of-4 Voter is implicitly assumed in the accident and transient analyses. Failure of this function would fail the following functions:

   - 2a, Neutron Flux – High (Setdown)
   - 2b, Simulated Thermal Power – High
   - 2c, Neutron Flux – High
   - 2d, Inop
   - 2f, OPRM Upscale

   Functions 2a, Neutron Flux – High (Setdown), 2b, Simulated Thermal Power – High, 2d, Inop, and 2f, OPRM upscale are not credited in any FSAR transient/accidents.

   Function 2c, Neutron Flux- High is credited in FSAR Section 15.1.6 Inadvertent Residual Heat Removal Shutdown Cooling Operation, 15.4.9 Control Rod Drop Accident, and 15.2.1 Pressure Regulator Failure – Close. Failure of Function 2e, Average Power Range Monitors – 2-Out-of-4 Voter does not impact the diverse Instrumentation listed in Attachment 6 for Function 2c, Neutron Flux-High except for the Inadvertent Residual Heat Removal Shutdown Cooling Operation, which is discussed above.

Furthermore, procedure PPM 1.3.1, Step 4.8.3.d fourth bullet includes the direction for operators to scram prior to exceeding an RPS setpoint. Operators are trained on scram setpoints regularly and the annunciator response procedures include setpoints. PPM 3.3.1 immediate actions describe inserting a manual scram by placing the mode switch in shutdown and depressing the manual scram pushbuttons.

5) Function 5, Main Steam Isolation Valve – Closure. After further review Function 4, Reactor Vessel Water Level – Low, Level 3 is a diverse Instrumentation to Function 5 for FSAR Section 15.6.4 Steam System Piping Break Outside Containment. Therefore, Manual Scram or Manual Trip is not solely relied upon as a diverse means for Function 5, Main Steam Isolation Valve – Closure. Furthermore, procedure PPM 1.3.1, Step 4.8.3.d fourth bullet includes the direction for operators to scram prior to exceeding an RPS setpoint. Operators are trained on scram setpoints regularly and the annunciator response procedures include setpoints. PPM 3.3.1 immediate actions describe inserting a manual scram by placing the mode switch in shutdown and depressing the manual scram pushbuttons.

6) Function 10, Reactor Mode Switch – Shutdown Position is not credited in the FSAR transient accident. Therefore, there is no FSAR required time. Procedure PPM 1.3.1, Step 4.8.3.d fourth bullet includes the direction for operators to scram prior to exceeding an RPS setpoint. Operators are trained on scram setpoints regularly and the annunciator response procedures include setpoints. PPM 3.3.1 immediate actions describe inserting a manual scram by placing the mode switch in shutdown and depressing the manual scram pushbuttons.

b.) Function 3c, Reactor Vessel Water Level – High, Level 8 is listed to maintain continuity of functions to match the TS table.

Audit Questions 1.a through 10.b.2 (EEEB) – Electrical Engineering Branch (EEEB) Audit Questions

Response

Revisions to Table E1-1, In-Scope TS/LCO Conditions to Corresponding PRA Functions, are shown following the responses to the follow-on questions submitted by the NRC staff.

1.a) Remove the information on diesel generators; keep note but move to notes section with a footer to reference the note. Keep the information on the offsite circuit.

1.b) Function should not be called out in Column 3, only in Column 5.

1.c) Yes.

2.a) Reference note has been removed.

2.b) Just Column 5 should address the function of SSCs.

2.c) The suggested edit has been accepted by CGS and incorporated into the table.

3.a) Reference note has been removed.

3.b) Just Column 5 should address the function of SSCs.

3.c) The suggested edit has been accepted by CGS and incorporated into the table.

4.a) The suggested edit has been accepted by CGS and incorporated into the table.

4.b) Just Column 5 should address the function of SSCs.

4.c) The suggested edit has been accepted by CGS and incorporated into the table.

5.a) Just Column 5 should address the function of SSCs.

5.b) Note 14 has been revised to address redundant chargers.

6.a) No reference notes are needed for this TS. The HPCS charger does not have a spare or redundant charger.

6.b) Just Column 5 should address the function of SSCs.

6.c) The suggested edit has been accepted by CGS and incorporated into the table.

7.a) Just Column 5 should address the function of SSCs.

7.b) The suggested edit has been accepted by CGS and incorporated into the table.

8.a) Just Column 5 should address the function of SSCs.

8.b) Column 6 should only address minimum remaining required power sources; however, the chargers cannot complete the design function under all conditions. Note 18 has been added to provide further clarification.

9.a) Just Column 5 should address the function of SSCs.

9.b) The suggested edit has been accepted by CGS and incorporated into the table.

10.a) Just Column 5 should address the function of SSCs.

10.b.1) The suggested edit has been accepted by CGS and incorporated into the table.

10.b.2) The suggested edit has been accepted by CGS and incorporated into the table.

The following pages include updated Table E1-1 content to reflect the changes noted above. This information replaces the applicable portions of Table E1-1 in Reference 1.

Table E1-1. In-scope TS/LCO Conditions to Corresponding PRA Functions

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
3.8.1.A	One offsite circuit inoperable	There are two qualified offsite power circuits [17] (meeting GDC-17 provisions) that provide AC power from the BPA transmission network to the CGS onsite Class-1E electric power systems (1) one is from TR-S (230 kV as the Preferred AC power source) for Divisions 1, 2, and 3, and (2) the other is from TR-B (115 kV as the Backup AC power source) for Divisions 1 and 2.	Yes	Each qualified offsite circuit supplies power to onsite Class 1E AC distribution system when unit's main generator (MG) is unavailable. The Class 1E AC distribution system supplies electrical power to three divisional load groups, Divisions 1, 2, and 3, with each division powered by an independent Class 1E 4.16 kV ESF switchgear bus) – Division 1 (SM-7), Division 2 (SM-8), and Division 3 (SM-4).	One qualified offsite circuit supplying power to two Class 1E 4.16 kV ESF switchgear buses.	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program.

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
3.8.1.B	One required DG inoperable	Three onsite emergency power DGs providing AC power to ESF divisions with each DG connected to a separate Class-1E 4.16 kV ESF switchgear bus where: (i) SM-7 is supplied by DG1, Division 1 (LPCS/LPCI DG), (ii) SM-8 is supplied by DG2, Division 2 (LPCI DG) and, (iii) SM-4 is supplied by DG3, Division 3 (HPCS DG), when offsite power to any of these 4.16 kV ESF buses is not available.	Yes	Onsite emergency DG AC power sources supply ESF divisions when connected to their respective Class 1E 4.16 kV ESF switchgear bus when offsite power to the 4.16 kV ESF bus is not available.	At a minimum, two onsite emergency DG AC power sources operable to supply associated Class-1E 4.16 kV ESF switchgear buses.	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program.

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
3.8.1.C	Two offsite circuits inoperable	AC power from the preferred 230kV power source at TR-S and the Backup 115 kV power source at TR-B to the respective 4.16 kV ESF switchgear buses is not operable; this constitutes, a loss of offsite power for this LCO. DG1 can supply AC power to Division 1 ESF loads (LPCS/LPCI DG), DG2 can supply AC power to Division 2 ESF loads (LPCI DG) and, DG3 can supply AC power to Division 3 loads (HPCS DG).	Yes	With two offsite circuits inoperable there are three onsite emergency DG power sources available to supply AC power to ESF loads (arranged in division load groups) either: Division 1 (LPCS/LPCI) from DG1, Division 2 (two LPCI systems) from DG2, and Division 3 (HPCS) from DG3.	At a minimum, two onsite emergency DG AC power sources operable to supply associated Class-1E 4.16 kV ESF switchgear buses	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program.

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
3.8.1.D	One offsite circuit inoperable AND one required DG inoperable	There are two qualified offsite power circuits [17] that provide AC power from the BPA transmission network to the CGS onsite Class-1E electric power systems (1) one is from TR-S (230 kV as the Preferred AC power source) for Divisions 1, 2, and 3, and (2) the other is from TR-B (115 kV as the Backup AC power source) for Divisions 1 and 2 Three onsite emergency power DGs providing AC power to ESF divisions with each DG connected to a separate Class-1E 4.16 kV ESF switchgear bus where: (i) SM-7 is supplied by DG1, Division 1 (LPCS/LPCI DG), (ii) SM-8 is supplied by DG2, Division 2 (LPCI DG) and, (iii) SM-4 is supplied by DG3, Division 3 (HPCS DG), when offsite power to any of these 4.16 kV ESF buses is not available.	Yes	Each qualified offsite circuit [17] supplies power to onsite Class 1E AC distribution system when unit's main generator (MG) is unavailable. Onsite emergency DG AC power sources supply ESF systems when connected to their respective Class 1E 4.16 kV ESF switchgear bus when offsite power to the 4.16 kV ESF bus is not available	One qualified offsite circuit [17] OR two onsite emergency DG AC power sources operable to supply two Class-1E 4.16 kV ESF switchgear buses	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program.

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
3.8.4.A	One required Division 1 or 2 125 VDC battery charger inoperable	The installed spare charger in each division can be placed into service via plant procedure should the operating battery charger in either Division 1 or Division 2 become unavailable. [14] The 125 VDC electrical power system consists of three independent Class-1E DC electrical power subsystems, Divisions 1, 2 and 3. Each redundant subsystem for Divisions 1 and 2 consists of a station battery (E-B1-1 or E-B1-2), associated two full capacity battery charger(s) (E-C1-1A & -1B, or E-C1-2A & -2B (one in service at a time) and an installed spare) and all the associated control equipment and interconnecting cabling for the 125 VDC electrical distribution buses and their loads.	Yes	Each 125 VDC battery charger supplies power to Division 1 or Division 2 125 VDC electric power distribution loads with the associated station battery floating on that DC subsystem. Each 125 VDC subsystem supplies DC control and motive power to auxiliary distribution loads including control and switching during all modes of operation to ensure the availability of the required DC power to support shut down of the reactor and to maintain it in a safe condition after an AOO or DBA. Note: The Division 3 125 VDC subsystem has a separate LCO 3.8.4.B.	A Division 1 or 2 125 VDC operable battery charger to supply its subsystem's 125 VDC loads. [14] [20]	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program.

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
3.8.4.B	One required Division 3 125 VDC battery charger inoperable.	The 125 VDC electrical power system consists of three independent Class-1E DC electrical power subsystems, Divisions 1, 2 and 3. The subsystem for Division 3 consists of a station battery (HPCS-B1-DG3), associated battery charger (HPCS-C1-1), and all the associated control equipment and interconnecting cabling for its 125 VDC electrical distribution buses and their loads.	Yes	The 125 VDC battery supplies power to Division 3 125 VDC electric power distribution loads when its battery charger is unavailable. This 125 VDC subsystem supplies DC control and motive power to HPCS auxiliary distribution loads including control and switching during all modes of operation to ensure the availability of the required DC power to support shut down of the reactor and to maintain it in a safe condition after an AOO or DBA. Note: The Division 1 and 2 125 VDC subsystems have a separate LCO 3.8.4.A.	A Division 3 125 VDC operable battery to supply its subsystem's 125 VDC loads. [20]	The battery is only credited for loss of offsite power and SBO sequences	PRA success criteria is more conservative, as the battery is not credited for non SBO or non-loss of offsite power sequences.

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
3.8.4.C	One required Division 1 250 VDC battery charger inoperable	Division 1 250 VDC subsystem consists of a 250 VDC battery (E-B2-1), associated battery charger (E-C2-1), and all the associated control equipment and interconnecting cabling for 250 VDC electrical distribution bus and its loads.	Yes	The 250 VDC battery supplies the 250 VDC subsystem loads including those for reactor core isolation cooling (RCIC), residual heat removal (RHR), etc., when the battery charger is unavailable to ensure the availability of the required power to support shut down of the reactor and to maintain it in a safe condition after an AOO or a postulated DBA.	A Division 1 250 VDC operable battery to supply its subsystem's 250 VDC loads. [20]	The battery is only credited for loss of offsite power and SBO sequences	PRA success criteria is more conservative, as the battery is not credited for non SBO or non-loss of offsite power sequences.

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
3.8.4.D	One required Division 1 or 2 125 VDC battery inoperable	The 125 VDC electrical power system consists of three independent Class-1E DC electrical power subsystems, Divisions 1, 2 and 3. Each redundant subsystem for Divisions 1 and 2 consists of a station battery (E-B1-1 or E-B1-2), associated battery charger(s) (E-C1-1A & -1B or E-C1-2A & -2B [one in service at a time] and an installed spare), and all the associated control equipment and interconnecting cabling for the 125 VDC electrical distribution buses and their loads.	Yes	Each 125 VDC battery supplies power to its subsystem (Division 1 or 2) 125 VDC loads when the battery chargers for either subsystem are unavailable. Each 125 VDC subsystem supplies DC control and motive power to auxiliary distribution loads including control and switching during all modes of operation to ensure the availability of the required DC power to support shut down of the reactor and to maintain it in a safe condition after an AOO or DBA. Note: The Division 3 125 VDC subsystem has a separate LCO 3.8.4.E.	One Division 1 or 2 battery is available to supply its subsystem's 125 VDC loads. [18] [20]	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program.

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
3.8.4.E	One required Division 3 125 VDC battery inoperable.	The 125 VDC electrical power system consists of three independent Class-1E DC electrical power subsystems, Divisions 1, 2 and 3. The subsystem for Division 3 consists of a station battery (HPCS-B1-DG3), associated battery charger (HPCS-C1-1), and all the associated control equipment and interconnecting cabling for its 125 VDC electrical distribution buses and their loads.	Yes	The 125 VDC battery charger supplies power to Division 3 125 VDC electric power distribution loads with the associated station battery floating on that DC subsystem. This 125 VDC subsystem supplies DC control and motive power to HPCS auxiliary distribution loads including control and switching during all modes of operation to ensure the availability of the required DC power to support shut down of the reactor and to maintain it in a safe condition after an AOO or DBA. Note: The Division 1 and 2 125 VDC subsystems have a separate LCO 3.8.4.D.	Division 3 125 VDC subsystem's battery charger to supply its required 125 VDC loads. [18] [20]	Same as Design Success Criteria	PRA success criteria is more limiting depending on the accident sequence due to battery depletion and battery charger capability.

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
3.8.4.F	One required Division 1 250 VDC battery inoperable.	Division 1 250 VDC subsystem consists of a 250 VDC battery (E-B2-1), associated battery charger (E-C2-1), and all the associated control equipment and interconnecting cabling for 250 VDC electrical distribution bus and its loads.	Yes	The 250 VDC battery charger supplies the 250 VDC subsystem loads including those for reactor core isolation cooling (RCIC), residual heat removal (RHR), etc., when the battery is unavailable to ensure the availability of the required power to support shut down of the reactor and to maintain it in a safe condition after an AOO or a postulated DBA.	A Division 1 250 VDC operable battery charger to supply its subsystem's 250 VDC loads. [18] [20]	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program. [19]
3.8.4.G	Division 1 or 2 125 VDC electrical power subsystem inoperable for reasons other than Condition A or D.	The 125 VDC electrical power system consists of three independent Class-1E DC electrical power subsystems, Divisions 1, 2 and 3 with each having a battery, battery chargers, and 125 VDC buses with this LCO focusing on buses E-DP-S1/1 (Division 1) and E-DP-S1/2 (Division 2) and associated motor control centers and distribution panels.	Yes	The 125 VDC buses, motor control centers, and distribution panels deliver power from their battery or battery chargers to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA.	One redundant 125 VDC subsystem (Division 1 or 2) with the requisite 125 VDC bus and its associated motor control centers and distribution panels capable of delivering power to its required Division 1 or 2 125 VDC loads. [20]	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program.

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
3.8.7.A	Division 1 or 2 AC electrical power distribution subsystem inoperable	Division 1 and Division 2 AC electrical distribution subsystems consist of 4.16 kV ESF AC switchgear buses for Division 1 (SM-7) and Division 2 (SM-8) and associated 480 VAC load centers, motor control centers, and distribution panels.	Yes	The required AC electrical power distribution subsystems ensure the availability of AC electrical power for the plant systems required to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA.	Division 1 or Division 2 AC electrical distribution subsystem consisting of its 4.16 kV ESF AC switchgear bus and associated 480 VAC load centers, motor control centers, and distribution panels capable of delivering power to their required loads.	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program (for unmodeled distribution panels). [15]
3.8.7.B	Division 1 or 2 125 VDC electrical power distribution subsystem inoperable	The 125 VDC electrical power system consists of three independent Class-1E DC electrical power subsystems, Divisions 1, 2 or 3. This LCO focuses on an entire Division 1 or 2 125 VDC subsystem being inoperable for any reason including loss of its power sources battery and battery chargers, or/and their 125 VDC buses E-DP-S1/1 (Division 1) and E-DP-S1/2 (Division 2) and associated motor control centers and distribution panels.	Yes	The required DC electrical distribution subsystems ensure the availability of DC electrical power for the plant systems required to shut down the reactor and maintain it in a safe condition after an AOO or a postulated DBA.	One redundant 125 VDC subsystem (Division 1 or 2) with the requisite 125 VDC bus and its associated motor control centers and distribution panels capable of delivering power to its required Division 1 or 2 125 VDC loads.	Same as Design Success Criteria	SSCs are modeled consistent with the TS scope and so can be directly included in the CRM tool for the RICT program (for unmodeled distribution panels). [16]

Edits to the Notes section for Table E1-1, In-scope TS/LCO Conditions to Corresponding PRA Functions, are shown below.

Edited Note 14: Each Division 1 and 2 125 VDC subsystem has an installed redundant standby charger that is normally isolated and can be manually aligned to meet the LCO requirements. The installed redundant spare battery charger is a design feature of the 125 VDC distribution system and is therefore, modeled in the PRA as a recovery action option.

New Note 17: A qualified offsite circuit meets GDC-17 requirements and consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit AC power from the offsite transmission network to the onsite Class-1E 4.16 kV ESF switchgear buses.

New Note 18: Energization transients of any DC loads that are beyond the capability of the battery charger and normally require the assistance of the battery will not be able to be brought online. To maintain the function of the valves supplied by the 250 VDC system, diverse AC-powered isolation valves are required to be available to maintain the function. This is consistent with TS 3.8.4 Bases.

New Note 19: Energization transients of any 250 VDC loads that are beyond the capability of the charger may fail the charger. To bound this condition with the 250 VDC battery unavailable, the Basic Event E-C2-1 UNABLE TO MAINTAIN DC LOADS will be set to TRUE.

New Note 20: Loss of any DC electrical power subsystem does not prevent minimum safety function from being performed. No situation exists where single DC bus failure would prevent plant personnel from achieving a reactor cold shutdown condition.

<u>Audit Question (APLC - Seismic) Parts a, b, c and d – NEW Q1</u>

<u>Response</u>

a.) A Request for Information for the Sequoyah Nuclear Plant TSTF-505 application (ADAMS Accession No. ML22182A390) was reviewed to determine if a similar error in performing seismic uncertainty analysis could have occurred at CGS. The Sequoyah error was related to failing to reload the reliability database in the cutset after setting the reliability database to use the parametric sampling equations for seismic interval initiators and fragility basic events, which are used in the Electric Power Research Institute (EPRI) UNCERT software. CGS parametric uncertainty analysis was free of this error.

    During the process of evaluating the mean values for this response, it was discovered that the EPRI UNCERT software only loads active cutsets in the solution file to calculate the mean Seismic Core Damage Frequency (SCDF)/Seismic Large Early Release Frequency (SLERF). Preparation of the cutset file to be used in UNCERT changes the seismic basic events initial value from the mean value to the median value, a variable in the distribution formula, causing many cutsets values to fall below the initial cutset truncation of 1E-12. When below truncation, the cutset editor marks the cutsets as inactive. Because UNCERT does not load those inactive cutsets, the full solution is not processed. Setting the cutset solution truncation to zero in the cutset editor, reactivates all cutsets so all cutsets are loaded and processed by UNCERT. Setting the truncation limit to zero is only done to run UNCERT. This feature of UNCERT was previously unknown. The truncation limit set during quantification was not changed and remains 1.0E-12. After resetting the cutsets file truncation to zero, preliminary results showed an increase in mean results by an unexpected amount. Table APLC-Q1.1 below summarizes these results.

Table APLC-Q1.1. Results from Differences in Truncation in UNCERT Program

Risk Measure	Point Estimate	Mean with Truncation at 1E-12	Calculated Mean with Truncation at Zero
SCDF	1.73E-05	1.81E-05	2.64E-05**
SLERF	5.16E-06	4.30E-06	8.35E-06**

**Mean value calculated at ACUBE limit within UNCERT. See Tables APLC-Q1.2 and APLC-Q1.3.

The advanced cutset upper bound evaluation (ACUBE) software uses an approach to remove the conservatism associated with the min-cut upper bound (MCUB) approximation. This approach is very computer memory intensive. Initial investigation found that the mean solution in UNCERT is sensitive to the number of cutsets processed by ACUBE, and the tool is not capable of processing the full solution with ACUBE. Therefore, the number of cutsets in a cutset solution that are evaluated by ACUBE in UNCERT are relatively small compared to the total cutsets in the solution. Table APLC-Q1.2 below shows the impact of ACUBE used in the point estimate solution and the mean solution.

Table APLC-Q1.2. Impact of ACUBE Cutsets on Point Estimate and Calculated Mean SLERF

# Cutsets Processed in ACUBE (of 73,431 total)	SLERF Point Estimate (PE)*	UNCERT Calculated Mean SLERF§	% Higher of Mean to PE	% Higher of Mean to Lowest PE
0	8.47E-06	9.47E-06	12%	84%
5000	7.91E-06	9.19E-06	16%	78%
10000	7.53E-06	8.74E-06	16%	69%
15000	7.30E-06	8.35E-06	14%	62%
20000	6.93E-06	Software limitation prevents more refined solution.		
30000	5.71E-06			
73431	5.16E-06**			

* Point estimate calculated from a combined cutset file for all but the last case
** Point estimate calculated for individual initiator bin cutset files and summed
§ Results based on the average mean from several Monte-Carlo simulations.

With a capability of running UNCERT with only 15,000 ACUBE cutsets in the SLERF combined cutset file, the SLERF mean approximation is 62% higher than the most refined point estimate approximation (5.16E-06). This is overestimated, as comparing the point estimate to the mean estimate with the same number of cutsets processed by ACUBE demonstrates the average increase is only 15%.

If the software were not limited, the mean estimate is expected to be conservatively limited to a 20% increase to the point estimate (i.e., ~1.03E-06).

A similar process was used to evaluate the core damage frequency (CDF) solution. The CDF solution that has all ground motions bins combined into a single cutset can only be partially processed by the ACUBE program in UNCERT. Table APLC-Q1.3 below shows the summary of the evaluation.

Table APLC-Q1.3. Impact of ACUBE Cutsets on Point Estimate and Calculated Mean SCDF

# Cutsets Processed in ACUBE (of 39,805 total)	SCDF Point Estimate (PE)*	UNCERT Calculated Mean SCDF§	% Higher of Mean to PE	% Higher of Mean to Lowest PE
0	2.29E-05	2.86E-05	25%	65%
2000	2.18E-05	2.74E-05	26%	58%
4000	2.14E-05	2.71E-05	26%	57%
6000	2.07E-05	2.64E-05	28%	53%
8000	1.91E-05	Software limitation prevents more refined solution.		
39805	1.73E-05**			

* Point estimate calculated from a combined cutset file for all but the last case
** Point estimate calculated for individual initiator bin cutset files and summed
§ Results based on the average mean from several Monte-Carlo simulations.

With a capability of running UNCERT with only 6,000 ACUBE cutsets in the SCDF combined cutset file, the SCDF mean approximation is 53% higher than the most refined point estimate approximation (1.73E-05). This is overestimated, as comparing the point estimate to the mean estimate with the same number of cutsets processed by ACUBE demonstrates the average increase is only 26%.

If the software were not limited, the mean estimate is expected to be conservatively limited to a 35% increase to the point estimate (i.e., ~6.06E-06).

Regulatory Guide (RG) 1.174, in conjunction with NUREG-1855, describes a process for addressing the state of knowledge correlation (SOKC) when the probability distribution of the large early relief (LERF) cutsets cannot be fully calculated. This process is to identify those cutsets that contain multiple events with the same state of knowledge data and calculate the impact of the distribution propagation on these cutsets. To accomplish this, the SLERF cutsets solution was reviewed, and out of 73,431 cutsets, 136 were found with multiple basic events using the same data state of knowledge. Two failure modes impacted the SOKC for these cutsets: seismic-induced motor control center (MCC) fires, and diesel run failures following seismic events. These cutsets were extracted from the solution and were assessed in the UNCERT program to calculate the increase in SLERF. Table APLC-Q1.4 below shows the results from extracting these cutsets (with a limited SOKC).

Table APLC-Q1.4 Probability Distribution Assessment from SOKC Cutsets

SOKC Cutsets Point Estimate SLERF	SOKC Cutsets Calculated Mean SLERF	Increase in SLERF due to SOKC
1.50E-08	3.65E-08	2.16E-08

The total mean SLERF can be estimated based on the Table APLC-Q1.2 results as the sum of the point estimate value and a conservative 20% mean adjustment (which bounds the contribution for SOKC cutsets in Table APLC-Q1.4), as follows:

Mean SLERF = 5.16E-06 + 1.03E-06 = 6.19E-06/year

The total mean SCDF can be estimated based on Table APLC-Q1.3 results as the sum of the point estimate value and a conservative 35% mean adjustment as follows:

Mean SCDF = 1.73E-05 + 6.06E-06 = 2.34E-05/year

b.) CGS finds there are limited SOKC events in the seismic probabilistic risk assessment (SPRA). The following discussion supports this conclusion.

The potential for the SOKC to impact the SPRA is very limited due to the nature of the component grouping supporting the fragility development. Three areas of the SPRA are considered: hazard frequency, component fragility, and events propagated from the internal events model.

The seismic hazard frequency is taken from a single source and the possible spectra is segmented to arrive at the ground motion intensity bin frequency based on the defined bin acceleration range. Each intensity bin considers a unique region of the hazard curve over a specific range of accelerations. Therefore, each bin interval is independent of the other bin intervals and no SOKC exists.

Component fragilities can provide a source for SOKC impact when multiple component groups utilize the same fragility development. For CGS, this type of relationship does not occur because all component groups are independent of each other with respect to the fragility development as applied to the group. The component grouping is location-specific, and assumes complete correlation between grouped components, which results in single point failures representing the failure of the component group There is no SOKC contribution required for this type of modeling since the correlation is complete between the grouped components and the fragility values are uniquely assessed based on grouped component location.

Another source SOKC is identified in Response a.) that is non-fragility based seismic failures (MCC fire) using the same data type that were added to the seismic model for completeness.

The last contribution involves relationships within the internal events model (random failures) that are propagated to the SPRA (diesel fail to run during seismic) and is identified in Response a.).

c.) This question was asked as an alternative to parts a.) and b.) and is therefore, not applicable.

d.) The total CDF and LERF mean values (internal events/flooding, fire, and seismic) meet the RG 1.174 threshold requirements of 1E-04 per year for CDF and 1E-05 per year for LERF. The seismic CDF and LERF mean values represent the estimated bounding values presented in Response a.). See Table APLC-Q1.5 below for a summary of all hazard mean CDF and LERF.

Table APLC-Q1.5. Summary of Total All Hazard Mean CDF and LERF

HAZARD	Point Estimate (per year)		Calculated Mean (per year)	
	CDF	LERF	CDF	LERF
Internal Events/Internal Flood	2.36E-06	1.60E-07	2.61E-06	1.62E-07
Internal Fire	4.06E-05	3.34E-06	4.28E-05	3.48E-06
Seismic	1.73E-05	5.16E-06	2.34E-05	6.19E-06
TOTAL	**6.03E-05**	**8.66E-06**	**6.88E-05**	**9.83E-06**

Audit Question (APLC - Seismic) – NEW Q2

Response

For the CGS Quantification of RICT duration in Table E1-2 of Reference 1, SSC seismically-induced failure probabilities for ground motion bins greater than 3.1g were set to TRUE to enable quantification at the required truncations within a reasonable time frame. Table APLC-Q2.1 below shows the conditional core damage probabilities (CCDP) and conditional large early release probabilities (CLERP) for each of the seismic acceleration bins and calculates the maximum RICT difference if the balance of CCDP or CLERP is assumed to apply to delta risk from an LCO configuration for seismic bins G30 through G46. For this analysis, the change in a 30-day RICT provides the largest RICT difference.

Table APLC-Q2.1. Sensitivity of Seismic Bin Initiators Set to Initiator Cutsets

Group	Description	IE Freq-uency	ZM* CDF (/rcry)	ZM* CCDP	Available CDF Impact IEF*(1-CCDP)*	ZM* LERF (/rcry)	ZM* CLERP	Available LERF Impact *IEF(1-CLERP)*
%G01	Seismic IE (0.125g - <0.3g)	1.69E-3	3.91E-9	2.31E-6		2.28E-11	1.35E-8	
%G02	Seismic IE (0.3g to <0.4g)	2.30E-4	9.44E-9	4.10E-5		1.56E-10	6.80E-7	
%G03	Seismic IE (0.4g to <0.5g)	1.11E-4	3.68E-8	3.31E-4		5.25E-10	4.73E-6	
%G04	Seismic IE (0.5g to <0.6g)	6.64E-5	1.01E-7	1.51E-3		2.94E-9	4.43E-5	
%G05	Seismic IE (0.6g to <0.7g)	4.30E-5	1.75E-7	4.06E-3		7.05E-9	1.64E-4	
%G06	Seismic IE (0.7g to <0.8g)	2.88E-5	2.26E-7	7.86E-3		9.40E-9	3.26E-4	
%G07	Seismic IE (0.8g to <0.9g)	1.98E-5	2.77E-7	1.40E-2		1.18E-8	5.98E-4	
%G08	Seismic IE (0.9g to <1g)	1.38E-5	9.76E-7	7.07E-2		8.24E-8	5.97E-3	
%G09	Seismic IE (1g to <1.1g)	9.81E-6	1.01E-6	1.03E-1		9.23E-8	9.41E-3	
%G10	Seismic IE (1.1g to <1.2g)	7.16E-6	1.04E-6	1.45E-1		1.02E-7	1.42E-2	
%G11	Seismic IE (1.2g to <1.3g)	5.38E-6	1.04E-6	1.94E-1		1.11E-7	2.06E-2	
%G12	Seismic IE (1.3g to <1.4g)	4.15E-6	1.05E-6	2.53E-1		1.20E-7	2.88E-2	
%G13	Seismic IE (1.4g to <1.5g)	3.30E-6	1.16E-6	3.52E-1		1.32E-7	4.01E-2	
%G14	Seismic IE (1.5g to <1.6g)	2.68E-6	1.16E-6	4.33E-1		1.44E-7	5.36E-2	
%G15	Seismic IE (1.6g to <1.7g)	2.20E-6	1.13E-6	5.16E-1		1.64E-7	7.47E-2	
%G16	Seismic IE (1.7g to <1.8g)	1.81E-6	1.07E-6	5.93E-1		1.70E-7	9.40E-2	
%G17	Seismic IE (1.8g to <1.9g)	1.50E-6	1.01E-6	6.74E-1		2.07E-7	1.38E-1	
%G18	Seismic IE (1.9g to <2g)	1.25E-6	8.62E-7	6.90E-1		2.24E-7	1.79E-1	
%G19	Seismic IE (2g to <2.1g)	1.04E-6	7.90E-7	7.59E-1		2.34E-7	2.25E-1	
%G20	Seismic IE (2.1g to <2.2g)	8.72E-7	7.15E-7	8.20E-1		2.36E-7	2.70E-1	
%G21	Seismic IE (2.2g to <2.3g)	7.36E-7	6.36E-7	8.64E-1		2.47E-7	3.36E-1	
%G22	Seismic IE (2.3g to <2.4g)	6.25E-7	5.60E-7	8.96E-1		2.39E-7	3.83E-1	
%G23	Seismic IE (2.4g to <2.5g)	5.34E-7	4.95E-7	9.28E-1		2.33E-7	4.37E-1	
%G24	Seismic IE (2.5g to <2.6g)	4.59E-7	4.36E-7	9.49E-1		2.26E-7	4.93E-1	
%G25	Seismic IE (2.6g to <2.7g)	3.96E-7	3.75E-7	9.46E-1		2.11E-7	5.34E-1	
%G26	Seismic IE (2.7g to <2.8g)	3.44E-7	3.30E-7	9.59E-1		2.02E-7	5.88E-1	
%G27	Seismic IE (2.8g to <2.9g)	3.00E-7	2.92E-7	9.73E-1		1.92E-7	6.40E-1	
%G28	Seismic IE (2.9g to <3g)	2.63E-7	2.57E-7	9.79E-1		1.80E-7	6.86E-1	
%G29	Seismic IE (3g to <3.1g)	2.32E-7	2.30E-7	9.91E-1		1.69E-7	7.28E-1	
%G30	Seismic IE (3.1g to <3.2g)	2.05E-7	2.03E-7	9.92E-1	1.55E-9	1.57E-7	7.68E-1	4.75E-8
%G31	Seismic IE (3.2g to <3.3g)	1.82E-7	1.82E-7	9.98E-1	3.91E-10	1.46E-7	8.02E-1	3.60E-8
%G32	Seismic IE (3.3g to <3.4g)	1.61E-7	1.61E-7	1.00E+0	8.05E-11	1.34E-7	8.35E-1	2.65E-8
%G33	Seismic IE (3.4g to <3.5g)	1.44E-7	1.44E-7	9.98E-1	3.22E-10	1.23E-7	8.54E-1	2.10E-8
%G34	Seismic IE (3.5g to <3.6g)	1.28E-7	1.28E-7	9.97E-1	4.14E-10	1.12E-7	8.77E-1	1.57E-8
%G35	Seismic IE (3.6g to <3.7g)	1.15E-7	1.15E-7	1.00E+0	5.75E-11	1.04E-7	9.01E-1	1.14E-8
%G36	Seismic IE (3.7g to <3.8g)	1.03E-7	1.03E-7	1.00E+0	1.15E-11	9.47E-8	9.20E-1	8.29E-9
%G37	Seismic IE (3.8g to <3.9g)	9.22E-8	9.22E-8	1.00E+0	1.61E-11	8.62E-8	9.35E-1	5.99E-9

Group	Description	IE Freq-uency	ZM* CDF (/rcry)	ZM* CCDP	Available CDF Impact IEF*(1-CCDP)*	ZM* LERF (/rcry)	ZM* CLERP	Available LERF Impact *IEF(1-CLERP)*
%G38	Seismic IE (3.9g to <4g)	8.30E-8	8.30E-8	1.00E+0	1.15E-11	7.87E-8	9.49E-1	4.26E-9
%G39	Seismic IE (4g to <4.1g)	7.48E-8	7.48E-8	1.00E+0	0.00E+0	7.17E-8	9.59E-1	3.08E-9
%G40	Seismic IE (4.1g to <4.2g)	6.76E-8	6.76E-8	1.00E+0	1.38E-11	6.45E-8	9.54E-1	3.12E-9
%G41	Seismic IE (4.2g to <4.3g)	6.12E-8	6.11E-8	9.99E-1	5.06E-11	5.90E-8	9.63E-1	2.23E-9
%G42	Seismic IE (4.3g to <4.4g)	5.55E-8	5.55E-8	1.00E+0	0.00E+0	5.39E-8	9.71E-1	1.59E-9
%G43	Seismic IE (4.4g to <4.5g)	5.05E-8	5.05E-8	9.99E-1	4.02E-11	4.93E-8	9.76E-1	1.19E-9
%G44	Seismic IE (4.5g to <4.6g)	4.60E-8	4.60E-8	1.00E+0	2.30E-11	4.52E-8	9.82E-1	8.28E-10
%G45	Seismic IE (4.6g to <4.7g)	4.20E-8	4.20E-8	9.99E-1	4.60E-11	4.14E-8	9.85E-1	6.21E-10
%G46	Seismic IE (>4.7g)	5.48E-7	5.48E-7	1.00E+0	0.00E+0	5.47E-7	9.98E-1	8.74E-10
		SUM	**1.96E-5**		**3.03E-9**	**5.92E-6**		**1.90E-7**
Bin G30-G46 Maximum Potential Change in 30 day RICT (days)					**0.0007**		**(days)**	**0.462**
				(hours)	**0.018**		**(hours)**	**11.09**
				Maximum Potential RICT Error	**0.002%**			**1.54%**

*ZM – Zero Maintenance

Based on these results, the impact on RICTs from setting seismically-induced SSC failure probabilities to TRUE for high acceleration (>3.1g) ground motion bins is minimal. Although LERF shows a 1.5% change, in most LCO cases, CDF controls the RICT durations.

In order to achieve the desired quantification speed for the all-hazard one-top RTR model while maintaining risk insights from RICT calculations, CGS plans to also set all fragilities that have a probability of 0.9 or greater to TRUE. This approach has been tested and found to provide acceptable results. This step reduces the reliance on ACUBE, therefore CGS retains the option to use the RTR model with or without the use of ACUBE. A sensitivity of all the RICT calculations performed as input to Table E1-2 of Reference 1 was performed to show that RICT durations are not sensitive to this approach. The configurations provided in the following Table APLC-Q2.2 are those that showed a non-zero change in RICT duration. Three results columns are provided as follows:

- Base LAR Case: Fragilities G30-G46 TRUE, ACUBE used. This is the baseline that sensitivities are compared to.

- Sensitivity 1: Base LAR Case plus Fragilities with Prob. >0.9 TRUE, ACUBE used. This demonstrates minimal difference from the Base LAR Case.

- Sensitivity 2: Base LAR Case plus Fragilities with Prob. >0.9 TRUE, ACUBE not used. This demonstrates minimal difference from the Base LAR Case when ACUBE is not used.

Table APLC-Q2.2 below shows those LCOs evaluated that showed a change greater than zero in either sensitivity.

Table APLC-Q2.2. Sensitivity of Seismic Fragilities with Probabilities >0.9 TRUE

LCO	LCO Components	RICT Durations Sensitivities Compared to Base				
		Base	Sen 1		Sen 2	
		Days	Days	%Change	Days	%Change
CDF Results						
3.5.1.B	HPCS	18.91	19.17	1.4%	19.11	1.0%
3.5.1.C	LPCI-A & LPCS	13.17	13.15	-0.1%	13.18	0.1%
3.5.1.C	HPCS & LPCI-A	5.50	5.54	0.7%	5.52	0.3%
3.5.1.C	HPCS & LPCI-B	8.10	8.14	0.6%	8.13	0.5%
3.7.1.B	SW Pump A	11.82	11.80	-0.2%	11.82	0.0%
3.8.1.D	TR-S & DG3	3.97	3.98	0.3%	3.98	0.2%
3.8.1.D	TR-B & DG3	8.46	8.52	0.7%	8.50	0.5%
3.8.4.B	HPCS Battery Charger	20.12	20.18	0.3%	20.28	0.8%
LERF Results						
3.3.5.1.G	Div. II ADS Instr. & HPCS	3.74	3.76	0.5%	3.75	0.4%
3.5.1.C	HPCS & LPCI-A	22.54	23.23	3.1%	23.12	2.6%
3.5.1.C	HPCS & LPCI-B	23.92	24.63	3.0%	24.51	2.5%
3.8.1.D	TR-S & DG3	7.16	7.22	0.9%	7.21	0.7%
3.8.1.D	TR-B & DG3	19.13	19.57	2.3%	19.50	1.9%
3.8.4.D	125 VDC Battery E-B1-1	22.63	22.95	1.4%	22.87	1.1%
3.8.4.G	DC BUS S1-1	11.23	11.32	0.8%	11.29	0.5%
3.8.4.G	DC BUS S1-2	8.37	8.37	0.1%	8.38	0.1%
3.8.7.B	DC Panel DP-S1/1A	13.64	13.76	0.9%	13.73	0.6%
3.3.6.1.A	Large Containment Failure Surrogate	14.90	15.01	0.8%	15.02	0.8%

The results above demonstrate that setting additional fragilities with a probability of >0.9 to TRUE and quantifying with or without ACUBE has minimal effect on the RICT duration. The largest changes are in the LERF results (1.4% to 3.1%) and are considered insignificant (<5%). In addition, for those combination LCOs, CDF controls the RICT duration, which is not as sensitive to the approach and has much shorter completion time.

CGS will continue to quantify the SPRA model of record in the same conventional manner. The described SPRA simplifications are planned only for the RTR model.

CGS will see the following major benefits from the described approach in the RTR SPRA model:

- Improved quantification speed and reliability,
- Improved usefulness of the RTR model (shorter run times), and
- No loss of decision-making capabilities.